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PICKERING NGS PERIODIC SAFETY REVIEW 2 (PSR2) BASIS DOCUMENT

Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document

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Revision Summary

Revision Number	Date	Comments
R000	2016-01-27	Initial issue.
R001	2016-06-01	- Documentation freeze date changed to January 15, 2016. Revised throughout, including affected documentation changes. - Sections 1.2 and 2.4. PSR2 validity period extended to the end of 2028, to align with the anticipated Power Reactor Operating Licence period. The period 2025-2028 is termed the "Licence Period Bounding case". Clarity provided that management of the activities and program changes associated with the transition to safe storage and safe storage periods will be addressed outside of the PSR process. - Section 2.2. "Define Review Tasks" added to Figure 1. "Ranking of Global Issues with identified actions" added to Figure 1. - Section 2.4. Clarified PSR1 / PSR2 interface and process for application of PSR1 findings to PSR2. - Section 2.5. Added that the scope of PSR2 is restricted to the facilities regulated under the Pickering NGS Power Reactor Operating Licence. - Section 2.6.1.2. Revised to clarify the process used to derive the PSR2 Review Tasks. - Section 3.2.1. Added footnote to clarify that Review Task compliance assessments are generally independent of PSR1 findings. Added "Plant Condition Assessments" as a source of information for Safety Factor compliance assessments. - Sections 3.2.1 and 3.2.3. Stated that effectiveness reviews would be documented for OPG programs used to demonstrate compliance with the PSR2 Assessment Basis. - Sections 3.2.2 and 3.2.3. Clarified description of review types, Compliances and Gaps. - Section 3.2.3. Noted that separate reports will be produced to document reviews of the Laws, Regulation, Codes and Standards in the PSR2 Assessment Basis, and also to document the derivation of the Safety Factor Review Tasks. Noted that Safety Factor reports will be submitted to CNSC staff for review. - Section 3.3.2. Added an element "Ranking of Global Issues with identified actions". - Section 3.3.3, Section on Prioritization. • Removed reference to ranking. • Clarified the origin and derivation of the information in Appendices E and F. • Added former Table E3. - Section 3.3.3. Added new section on ranking of Global Issues with identified actions. - Section 3.3.4. Noted that the Global Assessment Report will include a ranked list of those Global Issues with identified actions, with rationale for the ranking. - Section 4. Clarified wording relating to engagement of external contractors. Aligned PSR2 deliverable submission schedule with OPG-CNSC Protocol and added R1 of PSR2 Basis Document. - Added new Appendix A. PSR1 / PSR2 interface. - Previous Appendix B deleted (mapping of Safety Factor areas to CNSC Safety and Control Areas, comparison of PSR2 vs Darlington Safety Factor areas). Previous Appendix A (SSCs within scope of PSR2) moved to become new Appendix B. - Appendix C. Introduction revised to clarify the process used to derive the PSR2 Review Tasks. Review Tasks revised to ensure that IAEA SSG-25 and CNSC REGDOC-2.3.3 review elements are all addressed. Rewording of Objective for Safety Factors 2 and 4. - Appendix D, Section D.1.0. Section 2.10 reworded as an exclusion.

		- Appendix D, Section D.1.0. Section 3.2 inclusion rule clarified. - Appendix D, Section D.1.0. Section 3.3 added to clarify that informative / non-mandatory sections of documents in the PSR2 Assessment Basis are not included. - Appendix D, Table D1. Column identifying applicable Safety Factors deleted. - Appendix D, Table D1. Changes to listed documents, to modern versions, and to review types as a result of freeze date change and new information identified to date, as follows: • Added documents – CSA N288.7, CSA N290.8 • Modern version revised – CSA N286.7, CSA N290.14, CSA N291, NFPA 20, NFPA 24, CNSC REGDOC-2.3.2, ASME B31.1 • Review type revised – CSA N290.15, CNSC REGDOC-2.10.1 - Appendix E. • Revised Appendix E and Table E2 titles, for clarity. • Former Table E3 deleted (moved into Section 3.3.3). - Appendix F. Revised Appendix F title, for clarity.
R002	2016-06-29	- The term “Licence Period Bounding case” is no longer used to refer to the time period 2025-2028. Revised throughout the document. - Section 1.2. Clarified text concerning the planning basis, for PSR2 purposes, of operation of the Pickering NGS units. - Section 2.4. Clarified text concerning the PSR2 validity period. - Section 3.2.2. Definition of Incremental Review revised. - Section 3.2.3. Definition of Compliance and Gap, for Incremental Reviews, revised. - Sections 3.3.3 and 3.3.4. Added text to indicate that development of the Integrated Implementation Plan will be concurrent with Global Assessment phase activities. - Appendix D, Table D1. Review type for REGDOC 2.5.2 and REGDOC 2.2.2 changed from High Level to Incremental, at the request of CNSC staff.

1.0 INTRODUCTION

1.1 Purpose

This document is the Periodic Safety Review (PSR) Basis Document for the Pickering NGS Periodic Safety Review 2 (PSR2). It defines the scope and methodology of the conduct of PSR2. It addresses:

- The approach to conducting PSR2, including the proposed operating strategy of the facility,
- The PSR2 Assessment Basis, including applicable modern versions of Laws, Regulations, Codes and Standards,
- The methodology for conducting the elements of PSR2:
    - The Safety Factors to be reviewed and the strategy for the reviews,
    - The process for categorizing, prioritizing, tracking and resolving Gaps arising from the Safety Factor reviews,
    - Conduct of the Global Assessment,
    - The methodology for preparing the Integrated Implementation Plan,
- The major milestones, including the freeze date for document revisions,
- The project management and quality management processes to be followed in carrying out PSR2.

PSR2 is being performed in support of the evaluation of extended operation of the Pickering NGS units, beyond the year 2020, which is in accordance with the recent announcement by the government of the Province of Ontario. The CNSC has indicated [1] that OPG should perform a “subsequent PSR” should OPG decide to operate Pickering NGS units beyond 2020.

As noted, PSR2 is a subsequent PSR, an update building on the review basis of earlier OPG PSR work and other associated assessments (termed here “PSR1”). Specifically, PSR1 consists of:

- The Pickering B Integrated Safety Review (ISR), performed in support of refurbishment and continued operation (for another 30 years) of the Pickering 5-8 units [2],
- Pickering 1, 4 integrated safety assessments performed during the Pickering A Return to Service (PARTS) work (further described in Section 2.3 of this document), in support of approval to restart Units 1 and 4, and
- The Darlington ISR, performed in support of refurbishment and continued operation of the Darlington units [3] [4] (programmatic parts applicable to Pickering).

1.2 Operating Strategy of the Facility

Pickering NGS is located on the shore of Lake Ontario in the City of Pickering, in the Regional Municipality of Durham. The station has 6 operating nuclear reactors and 2 reactors (Units 2 and 3) that have been removed from operations and placed in the safe storage and surveillance state. Pickering NGS was built in two phases. Construction of Units 1 through 4 started in 1966, with Unit 1 becoming operational in 1971 and Unit 4 becoming operational in 1973. The construction of Units 5 through 8 started in 1974 and the units entered operations between 1983 and 1986.

Currently, the Pickering 5-8 units are approved to operate to 247,000 Effective Full Power Hours [5]. This operation limit is expected to be reached on some units in 2020. The current planning basis for Pickering NGS is an assumption of operation of Pickering NGS units until the end of 2024. To align with the anticipated expiry date of the next Power Reactor Operating Licence, for the purposes of PSR2 the period of operation of Pickering NGS units is extended until the end of 2028. Some Structures, Systems and Components (SSCs) will continue to be operated while the station is placed in the safe storage and surveillance state. OPG will make the final decision regarding the permanent shut down dates for the six reactors following the performance of a technical evaluation that will include PSR2, and will communicate it to the CNSC by June 30, 2017 as required by the current Power Reactor Operating Licence.

1.3 Assurance of Safe Operation

The safety of Pickering NGS is regularly and thoroughly assessed through several processes that are part of the current licensing framework. The near-term safety of the plant is validated and assured by these processes. In addition, OPG applies routine comprehensive safety assessment and improvement programs that deal with specific safety issues, significant events and changes in standards and operating practices as they arise. These programs allow assessment of safety and plant operation to be improved on a continuous basis and they can be correlated to all of the Safety Factors reviewed in this PSR2. They include programs that ensure safe operations, effective configuration management, equipment reliability, life cycle management, aging management, periodic inspection and maintenance. Programs are also in place in the area of organization management and safety culture that focus on safety-related behaviours and accountability.

The ongoing programs and the previously completed safety assessments are extensive and include safety reviews conducted for Pickering NGS Units 1 and 4, and Units 5 - 8, assessments that are part of the licence renewal process, as well as other internal and external assessments and audits.

1.4 Statement of Current Licensing Basis

CNSC REGDOC-2.3.3, Periodic Safety Reviews [6], requires a statement of the current licensing basis of the facility. Licensing basis is defined in CNSC document INFO-0795, Licensing Basis Objective and Definition [7] as follows:

> *The Licensing Basis for a regulated facility or activity is a set of requirements and documents comprising:*
>
> *(i) the regulatory requirements set out in the applicable laws and regulations*

(ii) the conditions and safety and control measures described in the facility's or activity's licence and the documents directly referenced in that licence

(iii) the safety and control measures described in the licence application and the documents needed to support that licence application.

The following demonstrates how this definition of Licensing Basis is applied for Pickering NGS:

Item (i): The primary applicable Act and Regulations are the Nuclear Safety and Control Act and the Regulations made under this Act. Other Laws and Regulations applicable to the facility are listed on the CNSC website.

Item (ii): This item refers to licence conditions and documents directly referenced in the Power Reactor Operating Licence (PROL). The Pickering NGS PROL in effect at the time of the PSR2 documentation freeze date (defined in Section 2.4 of this report) was PROL 48.02/2018 [8]. The Canadian Standards Association (CSA) and CNSC documents directly referenced in the PROL are listed in the associated revision of the Licence Conditions Handbook (LCH), LCH-PNGS-R004 [9], in LCH Tables C.1 and C.2, respectively.

Item (iii): CNSC document INFO-0795 clarifies that documents needed to support the safety and control measures described in the licence application are those documents which demonstrate that (a) the applicant is qualified to carry out the licensed activities, and (b) appropriate provisions are in place to protect worker and public health and safety, to protect the environment, and to maintain national security and measures required to implement international obligations to which Canada has agreed.

Appendix D of the LCH [9] consists of a list of the key OPG documents that describe OPG's safety and control measures, taken from OPG's licence application for Pickering NGS. These are the OPG documents that require written notification to the CNSC of document revisions.

The OPG documents associated with Item (iii) above will be used to support the reviews in PSR2. Additional OPG documents will also be used, as required, to support the reviews.

2.0 PSR2 SCOPE

2.1 PSR2 Objectives

CNSC REGDOC-2.3.3 [6] identifies the objectives of a PSR as follows:

The objectives of a PSR are to determine:

1. *The extent to which the facility conforms to modern codes, standards and practices*
2. *The extent to which the licensing basis remains valid for the next licensing period*

3. *The adequacy and effectiveness of the programs and the structures, systems and components (SSCs) in place to ensure plant safety until the next PSR or, where appropriate, until the end of commercial operations.*
4. *The improvements to be implemented to resolve any gaps identified in the review and timelines for their implementation.*

Through the PSR2 work, OPG will confirm that the design, condition and operation of Pickering NGS supports continued safe operation for the period of PSR2, and will determine reasonable and practical enhancements that may be made.

2.2 Elements of PSR2

CNSC REGDOC-2.3.3 [6] provides regulatory expectations for the conduct and content of a PSR. This REGDOC incorporates requirements from the International Atomic Energy Agency's (IAEA) Safety Standards Series, Specific Safety Guide No. SSG-25, *Periodic Safety Review for Nuclear Power Plants* [10].

In accordance with REGDOC-2.3.3, the elements of PSR2 will consist of the following four phases:

1. Preparation of a PSR2 Basis Document (this document).
2. Conduct of Safety Factor reviews and identification of Compliances and Gaps.
3. Analysis of the Gaps and the potential safety enhancements for Pickering NGS in the Global Assessment process.
4. Preparation of a plan for the implementation of safety enhancements (Integrated Implementation Plan).

The elements of PSR2 are shown in a process flowchart in Figure 1. The steps associated with this process flowchart are further described in Section 3 of this document.

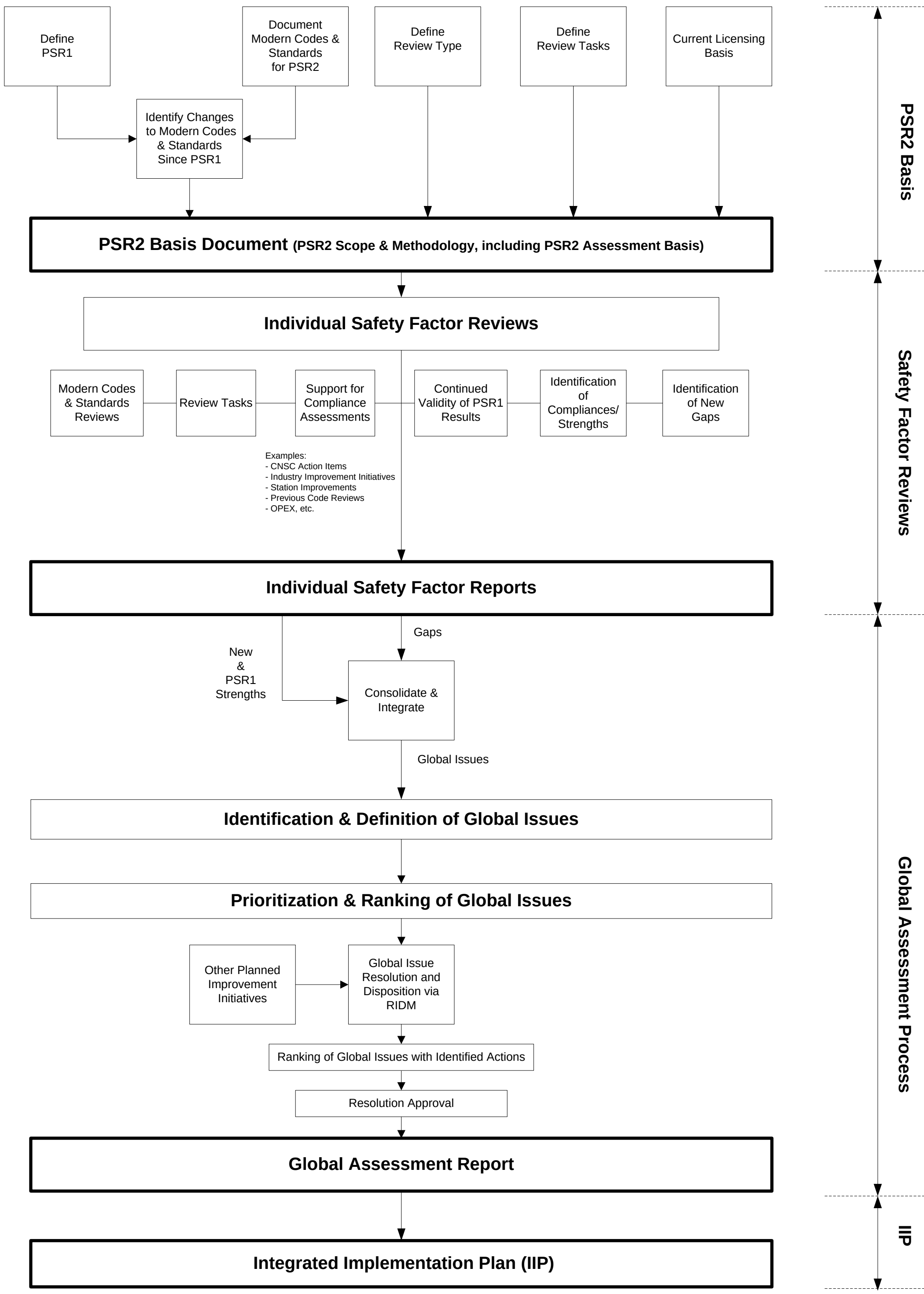

Figure 1 – Pickering NGS PSR2 Process Flowchart

2.3 Initial Pickering NGS PSR – PSR1 Description

PSR2 is a subsequent PSR, as defined in CNSC REGDOC-2.3.3 [6] and IAEA SSG-25 [10]. PSR2 is an update, building on the review basis of earlier OPG PSR work and other associated assessments (termed here “PSR1”). PSR1 consists of a combination of earlier reviews of OPG nuclear stations. These reviews are described below.

Pickering 5-8:

The Pickering B Integrated Safety Review (ISR), which included a comprehensive review of Codes and Standards, was completed in 2009, to support refurbishment and continued operation of the Pickering 5-8 units [2]. At that time, OPG was considering the option of refurbishment of the reactor components in each unit in order to continue to operate the Pickering 5 to 8 reactors for another 30 years. OPG later decided not to proceed with the refurbishment, for economic reasons, and decided to pursue instead the alternative of extended operations of all of the units to the end of 2020 without the replacement of the reactor components. In support of this approach, safety enhancements were identified (based on the results of the ISR) in the context of an operation timeframe extending to approximately 2025. These actions were documented in the Continued Operations Plan (COP), and progress has been reported to the CNSC on a regular basis. All of the COP actions, with one exception still in progress, have been completed [11].

Pickering 1,4:

Pickering 1,4 integrated safety assessments were performed during the Pickering A Return to Service (PARTS) work, in support of approval to restart Units 1 and 4 following the shutdown to enable implementation of an enhancement to the shutdown systems. (The reactor components in these units had previously been refurbished in the late 1980’s and early 1990’s. Units 2 and 3 had been placed in the state of safe storage with surveillance.) Based on the results of these safety assessments, Pickering Units 1 and 4 were restarted. These integrated safety assessments, termed Systematic Review of Safety, include the following key documentation:

- A review of Pickering A design against then-current codes and standards, including:
    - AECL report, 44RS-00531-ASD-001 R04, “Review of Pickering A Design Against Current Codes and Standards (C1-007-03-01-007)”, November 8, 2000 [12], which compared the Pickering A plant design with then current codes, standards and requirements to identify and recommend design changes to be considered by OPG;
    - AECB letter, C. B. Parsons to R. J. Strickert, “Pickering NGS-A Return to Service Project, Request for Approval of Code Effective Dates”, April 6, 2000, NA44-CORR-00531-00125 [13], and OPG report, NA44-REP-00584.1-10001 R00, “Pickering A ASME Codes Reconciliation”, August 2000 [14], which confirmed that the PARTS project conformed to code effective date requirements for various system design, equipment and materials. The impact of various code changes on the design basis was also assessed;

  - The Pickering A Basis for Return To Service, including AECL report 44RS-00531-AB-001 R01, "Methodology for Review of Pickering A Design Against Current Regulations and Standards (C1-007-03-01-002)", November 8, 2000 [15], which further outlined the approach followed for the review of Pickering A against selected codes, standards, and regulations.

- Other PARTS documentation, including OPG letter R.J. Strickert to J.S.C. Tong "Pickering A - Updated Basis for Return to Service Document" April 20, 2001, NA44-CORR-00531-00381 [16], in which 41 standards were reviewed to ascertain "direct and immediate effect on installed Design Features". This letter pertained primarily to design support analysis, quality assurance, and operations aspects.

Darlington:

The Darlington ISR was completed in 2011 [3], with a code refresh review [4] in December 2013. The Darlington ISR was performed in support of refurbishment and continued operation of the Darlington units. Extensive reviews (primarily clause-by-clause reviews) of Codes and Standards were completed. Much of the compliance assessment and evaluation of Safety Factor health for the Darlington ISR is based on programs and practices that apply across OPG's nuclear operations. As a result, Darlington ISR programmatic conclusions are applicable to the Pickering PSR2 for situations where Pickering is confirmed to follow the same nuclear programs and practices that were assessed for Darlington.

2.4 PSR2 Strategy

The scope of the Pickering NGS PSR2 is to conduct a review of Pickering NGS that meets the elements of CNSC REGDOC-2.3.3 [6] and IAEA SSG-25 [10].

PSR2 is a subsequent PSR, building on the review basis of PSR1. CNSC REGDOC-2.3.3 and IAEA SSG-25 identify that subsequent PSRs should focus on changes in requirements, facility conditions, operating experience and new information, rather than repeating activities conducted in previous reviews. As such, PSR2 will be forward looking, focusing on changes to requirements since the last applicable assessment, confirmation that the condition of Pickering NGS supports the additional years of operation, and new operating experience since the last assessments. PSR2 will seek to identify incremental enhancements to safety above those that have already been implemented or that are already committed.

Appendix A depicts the interface between the elements of PSR1 and PSR2, and the process by which PSR1 findings are to be applied to PSR2. PSR2 will include consideration and confirmation that the findings of PSR1 remain valid, as applicable, for the operation period. This includes assessment of PSR1 conclusions against implications resulting from extended operation. In particular, Pickering PSR1 results are applicable to PSR2 if there was a PSR1 gap that is still open, or if a closed PSR1 gap could be affected by extended operation. If so these gaps are carried forward into PSR2 for consideration in the Global Assessment. (When references to PSR1 are made, the source document will be identified and the relevant text from that source document will be summarized in the context of PSR2.)

With respect to the Darlington ISR, much of the evaluation of Safety Factor health is based on programs and practices that apply across OPG's nuclear operations. As a result, Darlington ISR programmatic conclusions are applicable to Pickering PSR2 where Pickering is confirmed to follow the same nuclear programs and practices that were assessed for Darlington. Darlington PSR1 results are applicable to Pickering PSR2 if there are Darlington PSR1 gaps that are found to be relevant to Pickering PSR2. An effectiveness review (effectiveness at Pickering NGS) of OPG programs used to demonstrate compliance with the PSR2 Assessment Basis will be conducted, using recent audit and self-assessment results.

Documentation Freeze Date

To establish a basis for conducting the review, and in order to reflect the interrelationships of OPG internal documents and external documents (e.g. Laws, Regulations, Codes, and Standards), a date of January 15, 2016 was chosen as the freeze date for PSR2. Where an external document was issued in January 2016 but the effective date is unknown, it will be considered to have been issued prior to the freeze date.

PSR2 Period of Validity

PSR2 is valid for the period up to the expiry of the next Power Reactor Operating Licence for Pickering NGS, which is anticipated to be the end of 2028.

The current planning basis for Pickering NGS is an assumption of operation of Pickering NGS units until the end of 2024. To align with the anticipated expiry date of the next Power Reactor Operating Licence, the planning basis for PSR2 extends the period of operation of Pickering NGS units until the end of 2028. Within this extended planning basis, individual units would only continue operating based on the ability to demonstrate assurance of continued safe operation for each unit through continued safe operation of critical systems and components.

PSR2 is focussed on operating units. When units are permanently shut down prior to 2028, some Structures, Systems and Components will continue to be operated while those units are placed in the safe storage and surveillance state (which will include defueling of the units), and Condition Assessments are in progress to address these requirements. During and approaching this transition period, some OPG program elements will change to align with new station conditions and new requirements. These program changes (and resulting activities) will be developed and approved following OPG's change management processes, and will include regulatory involvement prior to implementation (including CNSC notification or approval) as required. Description and management of the activities and program changes associated with the transition period (approaching and during) and safe storage period will be addressed outside of the PSR process.

2.5 Structures, Systems and Components within the Scope of the PSR2 Review

The scope of PSR2 is restricted to the facilities regulated under the Pickering NGS Power Reactor Operating Licence. The Structures, Systems and Components (SSC) within the scope of the PSR2 review will encompass the Pickering Safety Related Systems [17], with a focus on the Pickering Systems Important to Safety (SIS) [18] [19] and the Safe Operating Envelope (SOE) systems [20]. The SSCs within the scope of the PSR2 review are detailed in Appendix B.

Facilities regulated under other operating licences (Pickering Waste Management Facility) are not considered with the Pickering PSR2 scope.

2.6 PSR2 Assessment Basis

The PSR2 Assessment Basis consists of:

(1) Safety Factor Review Tasks derived from CNSC REGDOC-2.3.3 and IAEA SSG-25, taking into consideration the Review Tasks used in the Pickering B and Darlington Integrated Safety Reviews.

(2) A set of modern Laws, Regulations, Codes and Standards, determined to be applicable for PSR2 review.

These components of the PSR2 Assessment Basis are described further below.

2.6.1 Safety Factor Review Tasks

2.6.1.1 Safety Factors

Safety Factors cover all aspects important to the safety of an operating nuclear power plant. There are 15 Safety Factors used in the PSR2 review; 14 are identified in IAEA SSG-25, grouped into five subject areas, and one additional Safety Factor (Radiation Protection) is identified in CNSC REGDOC-2.3.3. These 15 Safety Factors are shown in Table 1.

Subject Area	Safety Factor	
The Plant	1	Plant Design
	2	Actual Condition of Structures, Systems and Components Important to Safety
	3	Equipment Qualification (environmental and seismic)
	4	Aging
Safety Analysis	5	Deterministic Safety Analysis
	6	Probabilistic Safety Assessment
	7	Hazard Analysis
Performance and Feedback from Operating Experience	8	Safety Performance
	9	Use of Experience from other Plants and Research Findings
Management	10	Organization, the Management System and Safety Culture
	11	Procedures
	12	Human Factors
	13	Emergency Planning
Environment	14	Radiological Impact on the Environment
Radiation Protection	15	Radiation Protection

Table 1 – Pickering NGS PSR2 Safety Factors

2.6.1.2 Safety Factor Review Tasks

The PSR2 Safety Factor Objectives and Review Tasks are shown in Appendix C. Review Tasks for Safety Factors 1 to 14 are an interpretation of requirements and guidance set out in IAEA SSG-25 [10]. The Review Tasks have been chosen to fulfil the requirements of IAEA SSG-25 while maximizing consistency with Review Tasks used in previous ISRs. CNSC REGDOC-2.3.3 encompasses all of the PSR Safety Factors recommended by the IAEA in SSG-25 and expands upon it by adding Safety Factor 15, Radiation Protection. As a result, PSR2 Review Tasks for Safety Factor 15 are derived from REGDOC-2.3.3 Appendix A.

In IAEA SSG-25, Safety Factors are described under headings of “Objectives”, “Scope and Tasks”, and “Methodology”. All of these headings contain information that is pertinent to the definition of Review Tasks, although the “Scope and Tasks” section is most directly related to Review Tasks. Review Tasks were generated to interpret the requirements in SSG-25 and also to retain alignment to the extent practicable with earlier Review Tasks from the Darlington and Pickering B ISRs [2] [3] [4]. (These previous ISRs were completed in accordance with CNSC RD-360, Life Extension of Nuclear Power Plants [21] and IAEA NS-G-2.10, Periodic Safety Reviews of Nuclear Power Plants [22], which have since been superseded.) By keeping the structure and grouping of similar SSG-25 Review Tasks as close as possible to previously used Review Tasks, the review effort can more effectively focus on safety significant differences including changes in requirements, plant conditions, operating experience and new information. This focus is consistent with the intent of a subsequent PSR as described in REGDOC-2.3.3.

Review Tasks were confirmed to address the requirements from the "Scope and Tasks" sections of SSG-25 and, if not, the Review Tasks were expanded or new Review Tasks were added. In cases where "Methodology" clauses from SSG-25 provide unique information, they were converted into Review Tasks or Review Tasks were expanded to incorporate the methodology statements. A final check was performed for each Safety Factor to confirm that Review Tasks align with the SSG-25 stated "Objectives" and, if not, the Review Tasks were expanded as required. Review Tasks for Safety Factor 15, Radiation Protection, were derived from Appendix A of REGDOC-2.3.3.

In SSG-25, some of the Safety Factor descriptions include discussion of reviews of codes and standards, and effectiveness reviews of programs that are used to demonstrate compliance with Review Tasks or codes and standards. In such cases, these discussions were not translated into PSR2 Review Tasks, since these reviews are completed separately for each Safety Factor, and thus having a stand-alone Review Task for this work would be redundant. See Section 3.2.1 for additional detail.

2.6.2 Modern Laws, Regulations, Codes and Standards Applicable to PSR2

The process to identify the modern Laws, Regulations, Codes and Standards that are applicable to the PSR2 Assessment Basis involved first creating a broad list from multiple sources (potential candidate Laws, Regulations, Codes and Standards) and then filtering it to identify those that are most significant, and that are applicable to the PSR2 scope. The identification and selection criteria are detailed in Appendix D.

The result of the identification and selection process was a set of modern Laws, Regulations, Codes and Standards that became part of the PSR2 Assessment Basis. These documents are listed in Table D1 in Appendix D. This table also identifies the modern version / date of the Law, Regulation, Code or Standard, and the type of review that will be completed in PSR2. The type of review is explained in greater detail in Section 3.2.2.

For the purpose of the performance of PSR2, OPG has defined the freeze date for modern Laws, Regulations, Codes and Standards to be January 15, 2016 (refer to Section 2.4).

3.0 PSR2 METHODOLOGY

3.1 PSR2 Process Overview

Figure 1 shows the general process overview of PSR2. The PSR2 Basis Document (this report) defines the PSR2 Assessment Basis, scope and methodology. Safety Factor reviews are conducted in accordance with the PSR2 Assessment Basis, and identify Gaps and Compliances. The Compliances (and groups of Compliances) are taken into the Global Assessment for consideration as Strengths. The Gaps are consolidated into Global Issues in the Global Assessment process. The Global Issues are prioritized, assessed and ranked, and proposed resolutions (enhancements) undergo reviews and approvals. The Global Assessment outputs are documented in a Global Assessment Report. The Integrated Implementation Plan documents the proposed enhancements from the Global Assessment Report, and provides implementation timelines for the enhancements. Further details on all of these steps are provided in the sections that follow.

3.2 Safety Factor Reviews

3.2.1 Safety Factor Review Process

Safety Factor reviews will be conducted for the 15 Safety Factors included in PSR2. Each review will address the associated Review Tasks for the Safety Factor (Appendix C), and will conduct reviews against applicable modern Laws, Regulations, Codes and Standards from the PSR2 Assessment Basis (Appendix D). An effectiveness review of OPG programs used to demonstrate compliance with the PSR2 Assessment Basis will be conducted, using recent audit and self-assessment results. A Safety Factor report will be completed for each of the Safety Factors.

Safety Factor compliance assessments will incorporate information from:

- OPG programs and procedures listed in the LCH, and any other programs and procedures which support the compliance arguments;
- Plant Condition Assessments (for Safety Factor 2);
- Commitments previously made to the CNSC, open CNSC action items, and exemptions granted by CNSC since the current operating licence was issued (safety significant issues, per the Pickering LCH) to determine if there are any impacts associated with Pickering operation past 2020. Fukushima actions will be included as appropriate as commitments or actions;
- Previously identified ISR gaps related to each Safety Factor and the status of OPG's improvement plans or other dispositions to address these;
- Assessments and reviews performed since the PSR1 documents were completed.

As a subsequent PSR, the PSR2 Safety Factor reviews will focus on changes in requirements (Laws, Regulations, Codes and Standards), updated plant conditions, operating experience and information from research, rather than repeating the activities of previous reviews. The methodology for performing the Safety Factor reviews is designed to take full advantage of the safety assessments and Law, Regulation, Code and Standard compliance work previously completed by OPG, and to focus effort on areas where incremental impacts may exist. This approach is in accordance with the guidance provided by the CNSC in REGDOC-2.3.3 that the effort required to undertake a subsequent PSR for a station will require considerably less effort, subject to confirmation that previous conclusions remain valid.

The approach for performing the PSR2 Safety Factor reviews will be in accordance with the following:

- Fully utilize existing Darlington ISR results to the extent that the programmatic aspects can be confirmed to be applicable to Pickering[1,2] ;
- Focus attention on findings from the 2009 Pickering B ISR and Pickering A Return to Service assessments that may no longer be applicable for operation beyond 2020, so that their impact can be assessed [3];
- Focus attention on requirements that are new or that have changed in relation to the requirements that were used as the basis for PSR1 ISRs, so that their impact on Pickering NGS can be assessed (e.g., new or revised Laws, Regulations, Codes and Standards, as well as incremental guidance contained in IAEA SSG-25).
- Apply information provided in Pickering PROL renewal applications. Documents specifically listed in the PROL have generally been in place for many years, and therefore have been subject to prior assessments as well as ongoing compliance verification activities by both OPG in the form of audits and self-assessments, and by the CNSC in the form of inspections and audits. This includes information on Pickering Units 1,4 performance that is useful in PSR Safety Factor reviews.
- Consider information related to OPG programs and processes for Pickering 5-8 in NK30-PLAN-00531-00001 “Pickering 5-8 Continued Operations Plan” [23], as well as OPG programs and processes common to both Pickering 1,4 and 5-8 in P-PLAN-09314-00001 “Pickering Sustainable Operations Plan” [24].

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1 Areas where the Darlington ISR assessment cannot be judged to be applicable or sufficient to address Pickering will be identified. Account will be taken of any relevant feedback that the CNSC may have provided in relation to the Darlington ISR assessment for the Safety Factor.

2 In general, compliance assessments for PSR2 Review Tasks will be prepared independently of PSR1 findings. That is, compliance for each Review Task will be demonstrated via discussion of current OPG Governance, Programs, Policies, Procedures, Audits and Self-Assessments as required, and will only mention the findings of PSR1 if there are pertinent PSR1 gaps which need to be addressed within the context of PSR2.

3 Past integrated safety assessments of Pickering Units 1 and 4 will be reviewed. Any differences between Pickering Units 1,4 and 5-8, as well as any relevant feedback that the CNSC may have provided in relation to the Pickering B ISR assessment for each Safety Factor, will be taken into account. Previous Pickering B ISR Safety Factor Report content will be identified as being valid in relation to Pickering Units 1,4 where applicable.

3.2.2 Reviews of Modern Laws, Regulations, Codes and Standards

CNSC REGDOC-2.3.3 requires that the PSR Basis document identify:

- Which modern Laws, Regulations, Codes and Standards will be used in the Safety Factor reviews, including their effective dates, and the criteria for their selection – provided in Appendix D;
- The PSR documentation freeze date beyond which documentation changes and new information will not be considered – January 15, 2016; and
- The type of review to be performed (Clause-by-Clause, High Level, or alternative) – discussed in this section, below. PSR2 will include an alternative review type of "Incremental".

It is important that the methodology for PSR2 be focused on addressing aspects of the review that are likely to have material impact in terms of identifying enhancements that will be reasonable and practicable to implement during the remaining commercial life of Pickering NGS.

PSR2 will conduct reviews against a baseline of the PSR1 work. It is important to note, though, that OPG conducts regular reviews of new and revised Codes and Standards, so a large amount of information is already available to assist in the Safety Factor compliance reviews. In OPG letter W.M.Elliott to P.A Webster and M. Santini, "Design Codes and Standards Effective Dates for OPG Nuclear Fleet", April 30, 2012, N-CORR-00531-05661 [25], OPG stated:

> "...OPG commits to completing a code-over-code review (i.e., review of changes) of subsequent editions, addendum and/or updates of the Codes and Standards listed in [*Attachment 1 of the referenced document*]. Key emerging issues due to major changes in the codes will be addressed immediately, or as agreed with the CNSC on a case-by-case basis. Otherwise, OPG will confirm in a letter to the CNSC that these reviews have been completed and there are no significant technical issues..."

As a result, many of the updated codes and standards issued since PSR1 have already had gap assessments performed, to varying degrees of detail, which will be utilized and cited in the present Pickering PSR2.

As a subsequent PSR, PSR2 will focus on changes in requirements, plant conditions, operating experience and new information, rather than repeating the activities of previous reviews. Since PSR2 is an update of previous ISRs, it will incorporate reviews of Laws, Regulations, Codes and Standards that have occurred as new versions have been issued. Therefore, clause-by-clause reviews of the majority of applicable Laws, Regulations, Codes and Standards have already been completed and there is little value in repeating that process. If clause-by-clause reviews were to be undertaken in PSR2, a major portion of the review effort would be consumed by repackaging existing information that remains largely applicable and, therefore, is not contributing to the identification of new insights and enhancements. A more constructive approach is therefore planned that maximizes the value and usefulness of the work by focusing attention where it is most beneficial, i.e., on identifying new issues. The primary objective for this work, which is to identify safety significant enhancements that may be implemented during the limited remaining life of the station, will be achieved using this process and is expected to result in the same (safety significant) Global Issues being identified as would result from a clause-by-clause assessment.

Since this assessment is a subsequent PSR, the focus will be on identifying safety significant differences between what was previously assessed and what is now different within the current Pickering PSR2 Assessment Basis. In general, these differences relate to:

- More recent (new or revised) versions than what was previously assessed;[4]
- Safety significant differences between Pickering and Darlington, if Darlington is the basis for the earlier assessment; and
- Safety significant differences between Pickering Units 1,4 and Units 5-8.

In most cases Law, Regulation, Code or Standard reviews will be incremental in nature and performed by topic or subject matter for revised requirements. The rationale for this is that new or updated requirements that need to be included in PSR2 will be predominantly replacements for other Laws, Regulations, Codes or Standards that were previously assessed, and specify requirements that can be readily mapped to existing OPG programs to demonstrate compliance.

As noted earlier in this section, CNSC REGDOC-2.3.3 identifies three types of review that are appropriate for a PSR – Clause-by-Clause, High Level, and Alternative. For the reasons given above that align with the goals of a subsequent PSR, OPG plans to include an alternate review type called “Incremental”, as defined below. Thus, the following three review types will be applied for PSR2:

- Clause-by-Clause review: New Laws, Regulations, Codes and Standards referenced in Pickering PROL 48.02/2018 (listed in Appendix C of the Licence Conditions Handbook) will be subjected to a clause-by-clause type review. In a clause-by-clause review, conformance with individual clauses is demonstrated by supporting evidence stating whether the requirements stipulated in the requirement document are met;
- High Level review: New Laws, Regulations, Codes and Standards not referenced in Pickering PROL 48.02/2018 but which are in the PSR2 Assessment Basis will be subject to a high level review. In a high level review, the degree of conformance with clauses or groups of clauses in the Law, Regulation, Code or Standard is demonstrated by supporting evidence stating whether the intent of the requirements stipulated in the requirement document is met; and

[4] “New” refers to a Law, Regulation, Code or Standard that was not previously considered in the context of earlier assessments. “Revised” refers to an updated version of a Law, Regulation, Code or Standard that was previously considered in the context of earlier assessments. Where a document has a new number/type, but addresses the same topic from the same organization, it is a “revised”, not “new”, document (e.g., if a REGDOC replaces a CNSC G or RD document).

- Incremental review: For Laws, Regulations, Codes and Standards that have been reviewed in PSR1 but have had revisions since the last review, a topical review will be performed of the changes.

Reports will be produced to document the reviews of Laws, Regulations, Codes and Standards in the PSR2 Assessment Basis. These reports will provide more details of the three review types and how the reviews will be conducted.

Modern Laws, Regulations, Codes and Standards in the PSR2 Assessment Basis will generally receive incremental reviews since PSR2 is an update of PSR1, so assessments and clause-by-clause or high level reviews for the majority of the Laws, Regulations, Codes and Standards in the PSR2 Assessment Basis have already been completed. In addition, implementation plans exist for many of the codes and standards not addressed in PSR1 and therefore an incremental review will be applied to these documents. Table D1 identifies the review type to be applied to each of the Laws, Regulations, Codes and Standards in the PSR2 Assessment Basis. Following further assessment of past work, the review type of a listed modern Law, Regulation, Code or Standard may be changed from Clause-by-Clause or High Level to Incremental.

3.2.3 Safety Factor Results and Reports

The Safety Factor reviews will identify Compliances and Gaps with respect to the review elements in the PSR2 Assessment Basis. These are further defined as:

- <u>Compliance</u>:
  - For Clause-by-Clause reviews of modern Laws, Regulations, Codes and Standards, Compliance indicates that the safety requirement is met.
  - Where a High Level review has been performed, Compliance indicates that the intent of the safety requirement is met.
  - Where an Incremental review has been performed, Compliance indicates that the change in the safety requirement, per the topical review, is met.
  - For reviews of Safety Factor Review Tasks, Compliance indicates that either the safety requirement or the intent of the Review Task is met.
- <u>Gap</u>:
  - For Clause-by-Clause reviews of modern Laws, Regulations, Codes and Standards, a Gap indicates that the safety requirement is not met.
  - Where a High Level review has been performed, a Gap indicates that the intent of the safety requirement is not met.
  - Where an Incremental review has been performed, a Gap indicates that the change in the safety requirement, per the topical review, is not met.
  - For reviews of Safety Factor Review Tasks, a Gap indicates that the intent of the Review Task is not met.

Compliances that are equivalent to or surpass PSR2 Assessment Basis requirements or practices will be forwarded into the Global Assessment process for consideration as Strengths. Gaps will be evaluated by the Global Assessment methodology to identify Global Issues and, with justification, Acceptable Deviations.

The results of the Safety Factor reviews will be documented in Safety Factor reports, which will be submitted to CNSC staff for review. These reports will include:

- The scope of the review,
- Applicable elements of the PSR2 Assessment Basis (Review Tasks and applicable Laws, Regulations, Codes and Standards),
- Review methodology,
- Assessment of compliance with Review Tasks,
- Effectiveness review of OPG programs supporting compliance assessments,
- Review findings (Compliances and Gaps),
- Impacts on other Safety Factor reviews,
- Overall assessment of the Safety Factor.

Separate reports will be produced to document (a) the reviews of Laws, Regulations, Codes and Standards in the PSR2 Assessment Basis, and (b) the derivation of the Safety Factor Review Tasks from IAEA SSG-25 and CNSC REGDOC-2.3.3. The Safety Factor reports will draw on the information in these reports.

3.3 Global Assessment

The objective of the Global Assessment is to provide an overall assessment of the safety of the plant, and to arrive at a judgement of the plant's suitability for continued operation on the basis of a balanced view of the results from the reviews of the separate Safety Factors [6] [10]. This judgement takes into account the safety enhancements identified in the Global Assessment (plant and process modifications), strengths and residual Global Issues / Acceptable Deviations (as defined in Section 3.3.3), impact of aggregate effects of the results, and consideration of existing planned safety enhancements and recent overall station safety performance.

3.3.1 Global Assessment Team

The Global Assessment will be conducted by an interdisciplinary team, with appropriate expertise in operations, design and safety at the plant, including appropriate participants from the Safety Factor reviews, and members who are independent from the Safety Factor review teams, consistent with the requirements in [10].

3.3.2 Global Assessment Process

The Global Assessment Process consists of the following elements:

1. Identification and consolidation of Strengths and Gaps from the Safety Factor Reports.
2. Identification of Global Issues.
3. Assessment of interfaces between the various Safety Factors, Aggregate Impact of Global Issues.
4. Prioritization of Global Issues.
5. Development of Resolutions / Dispositions of Global Issues (and Gaps).
6. Consideration of defence-in-depth and aggregate impact of residual Global Issues / Acceptable Deviations.
7. Ranking of Global Issues with identified actions.
8. Senior Management Scope Review Board approval of proposed modifications for the purposes of PSR2.
9. Assessment of overall acceptability of operation of the plant over the period considered in PSR2.
10. Preparation of the Global Assessment Report to summarize the assessments, and document the Global Assessment.

These elements are described further in Section 3.3.3.

3.3.3 Global Assessment Logistics

Details of the elements involved in executing the Global Assessment are provided below.

Identification and Consolidation of Strengths and Gaps from the Safety Factor Reports:

The Strengths and Gaps from the 15 individual Safety Factor Reports will be consolidated and grouped by topic area to support the Global Assessment. Recommendations from the Component Condition Assessments conducted in support of Safety Factor 2 will also be considered as part of this review.

Identification of Global Issues:

The consolidation of Gaps into Global Issues will provide a means to assemble Gaps of a common nature, facilitating the assessment of safety impact and identifying and assessing practical and effective resolutions. The Global Issues will be tabularized, tracking sources of the issues, to facilitate further review and assessment.

Assessment of the Interfaces Between the Various Safety Factors, Aggregate Impact of Global Issues:

With the assembly of Global Issues and Strengths, and considering the recommendations from the Component Condition Assessments, the aggregate impact of the Global Issues will be assessed. In this way, the interaction between issues will be identified. New Global Issues may be identified as part of this consolidation review. This will support the prioritization and ranking of Global Issues as described below.

Prioritization of Global Issues

PSR2 Global Issues will be prioritized with respect to their importance to Nuclear Safety, using the tables provided in Appendices E and F to determine the Safety Significance level associated with each Global Issue. This will support the resolution evaluation method and the outcome of the resolution process. This methodology is consistent with OPG prioritization processes [26] used in previous Integrated Safety Reviews and industry practice.

The Safety Significance level will consider deterministic (Appendix E) and probabilistic (Appendix F) safety analysis impact, as appropriate. The assignment of Safety Significance values for prioritization in Appendices E and F was derived based on OPG experience and takes into account the priority values from the OPG guidelines for evaluating and prioritizing Safety Report Issues, the COG Benefit-Cost Analysis processes, and the OPG Station Condition Record categorization process. Probability levels selected for delineation between categories are based on significance and engineering judgement, and are as used in previous Integrated Safety Reviews. These values account for overall safety impact and align, where appropriate, with requirements and limits in relevant safety standards.

Safety Significance will be derived primarily through the thresholds defined in Appendix E Table E1 and per the risk assessment criteria in Appendix F. Appendix E Table E2 thresholds will be used as an additional means to evaluate nuclear Safety Significance (where appropriate) and for the assessment of the significance of Global Issues without a direct nuclear safety impact. If a Global Issue is evaluated using multiple tables from Appendices E and F, the most safety significant result will be reported for the Global Issue for carrying forward.

The relationship between Safety Significance Level and Impact on Nuclear Safety is shown in the table below:

Safety Significance Level	Impact on Nuclear Safety
1	High
2	Medium
3	Low
4	Very Low

Development of Resolutions / Dispositions of Global Issues (and Gaps)

Resolution options will be developed and assessed using risk informed decision making techniques. The development of the resolution will utilize the following strategy:

- In assessing potential dispositions, defence-in-depth [27] elements will be considered.
- In developing the resolutions, consideration of overall safety significance will guide the resolution process.
- For Global Issue resolution – the process will be:
  - Evaluate the Global Issue to understand safety basis, and intent of requirement.
  - Consider possible options for resolution/mitigation. Consider safety significance and defence-in-depth elements.
  - Evaluate options with respect to effectiveness, cost, schedule, practicality. For potential plant modifications, this may require an evaluation of the safety impact, both deterministic and probabilistic. If it is not practicable to fully resolve a Global Issue, other mitigation options will be considered for enhancements.
  - Practicality of a proposed resolution will be evaluated in terms of cost, resources, schedule, and considered in relation to the overall safety impact.
  - Some proposed resolutions will be dependent on whether plant operation is assumed to continue into the 2025-2028 time period. These proposed resolutions will be distinguished as such.
  - Propose recommended resolution/mitigation.
  - Document the decision making process.
- Items of High or Medium Impact on Nuclear Safety (Safety Significance levels 1 and 2) will require more in-depth analysis to fully understand the issue and potential impact, and to develop the proposed recommended resolution/mitigation. This may require deterministic and/or probabilistic assessments to measure nuclear safety impact of modifications and more detailed evaluation of the cost/practicality of proposed resolutions. Insights from available site Probabilistic Safety Analyses may be used in evaluating the benefit/practicality of potential options. This will be done concurrent with the development of the Integrated Implementation Plan.
- Items of Very Low Impact on Nuclear Safety (Safety Significance level 4) will generally be deemed as Acceptable Deviations within the context of PSR2 (with the rationale provided), and while these items will not be tracked beyond the Global Assessment, they will be shared with the accountable organizations for consideration as potential enhancement initiatives for their future work program

planning purposes. This will allow the organizations to prioritize the initiatives as part of their integrated programs to ensure the focus is on the right overall priorities. A similar treatment will be applied for items of Low Impact on Nuclear Safety (Safety Significance level 3) for which a practicable solution is not readily evident.

- Proposed resolutions will be categorized as i) Programmatic (changes to procedures and programs), ii) Engineering (plant modifications), or iii) Analytical (e.g., safety analysis, hazard analysis), to facilitate binning of potential work. In some cases, the proposed resolutions will entail work from more than one of these categories.

- In some cases, the development of resolutions/dispositions to the Global Issues will be part of an OPG or industry initiative underway or planned. Or, the resolution and development of options may require more detailed analysis and assessment, extending beyond the timelines for submission of PSR2. In these instances, the status of the initiative and plans will be included in the disposition. The work will be included in the Global Assessment to facilitate continued tracking.

- The results of previous Global Assessments for OPG stations will be considered in the review.

- If in the assessment it is determined that a Global Issue / Gap has been closed, due to work done in the interim or for other reasons, the rationale will be documented and the Global Issue / Gap will be set to Resolved and Closed.

- At the recommendation of the senior management team, an alternate process / resolution may be utilized for a particular Global Issue / Gap.

Consideration of Defence-in-Depth and Aggregate Impact of Residual Global Issues / Acceptable Deviations:

An important element of the development of proposed recommendations will be to assess the overall defence-in-depth and aggregate impact of the residual Global Issues / Acceptable Deviations. After evaluating a range of resolutions for Global Issues, and determining a recommended resolution to be selected, the impact on defence-in-depth [27], considering both deterministic and probabilistic elements, will be evaluated to assess the aggregate impact on overall safety. It may be necessary to refine the proposed resolutions based on the results of this review. This overall assessment will be an important element in supporting the enhancement plans and the planned operational strategy over the period of PSR2.

Ranking of Global Issues with Identified Actions

All Global Issues whose resolution involves identified actions will be ranked from 1 through N, where N is the total number, in accordance with overall safety significance. This will be based on engineering judgement applied by the assigned Global Assessment team. The ranking process will consider factors such as the priority previously determined (Safety Significance Level), the contribution to defence-in-depth, the significance of the source (e.g., the type of document that generated the gap(s) leading to the Global Issue), and the degree of non-compliance with the PSR2 Assessment Basis. The ranking process will also account for the extent of impact on multiple safety factors or areas.

Senior Management Scope Review Board Approval of Proposed Modifications for the Purposes of PSR2:

The enhancements identified in the PSR2 Global Assessment Report, with their priority and safety basis, will be presented to the OPG Senior Management Scope Review Board for approval. This review will ensure alignment with the resolutions proposed, their basis and context, and will be the means to obtain concurrence that the proposed enhancements are practicable and effective. This will also allow the senior management team to consider potential realignment of overall priorities based on the insights from PSR2. Consistent with OPG Project Management processes, additional approval gates will be required as the resolution development continues towards full implementation.

Assessment of Overall Acceptability of Operation of the Plant Over the Period Considered in PSR2:

As a final step in the assessment process, the team will assess the overall acceptability of operation of the plant over the period considered in PSR2. This will entail a review of the results of the Safety Factor Reviews, a consideration of enhancements planned (both newly identified in PSR2 and from other station plans), and a consideration of plant performance and initiatives underway.

Preparation of the Global Assessment Report

Preparation of the Global Assessment Report will be conducted to summarize the assessments and document the Global Assessment, as detailed in Section 3.3.4 below.

3.3.4 Global Assessment Report

The results of the Global Assessment will be documented in a Global Assessment Report, presenting the results, assessing the overall defence-in-depth of the plant, and documenting the conclusions, corrective actions, and enhancements to be considered. The Global Assessment Report will include a ranked list of those Global Issues with identified actions, with rationale for the ranking. This will be done concurrent with the development of the Integrated Implementation Plan.

Residual Global Issues and Acceptable Deviations will be noted in the report, summarizing the assessed aggregate impact on safe operations. These items will be conveyed to the accountable organizations for their consideration as potential enhancement initiatives for their future work program planning purposes. These initiatives will be weighed against other important program and plant modifications as part of the base and project work within these organizations. These items will not be tracked further beyond the Global Assessment Report or carried forward into the Integrated Implementation Plan.

The Global Assessment Report will include a statement of OPG's assessment of the overall acceptability of operation of the plant [6]. Reviews and approval of the report will be conducted as required under the OPG Management System. The Global Assessment Report will be submitted to CNSC staff for review.

3.4 Integrated Implementation Plan

The proposed enhancements resulting from the Global Assessment will be documented in the Integrated Implementation Plan (IIP). The IIP will provide the proposed timeline for the implementation of the enhancements and it will also document and confirm the resulting enhancement.

The enhancements summarized in the IIP will be mapped to the CNSC Safety and Control Areas (per Appendix B of CNSC REGDOC-2.3.3 [6]) so as to facilitate the CNSC's review.

3.4.1 Integrated Implementation Plan Logistics

The IIP listing of enhancements will include those resulting from the Global Assessment Report, including both new modifications proposed as part of the resolution of Global Issues, and also considering the existing planned station process and physical modifications that were integral to the overall assessment of safety. The enhancements arising from resolutions dependent on an assumption of plant operation continuing into the 2025-2028 time period will be specifically identified as such.

A review will be conducted with program owners and appropriate managers to derive plans for implementation based on priority and resources. These plans will be developed with due consideration of the other important initiatives underway at Pickering NGS as part of the continued implementation of the "Passion for Excellence" vision.

The initiatives will be tabularized with owners assigned and planned implementation dates. Existing initiatives integral to the overall assessment of safety during the Global Assessment will also be included in this listing. The listing will include the priority and the basis for the priority. The implementation of the initiatives will be tracked and reported.

The IIP will be presented to OPG senior management to obtain support for the initiatives and plans. As the IIP will be based on initial conceptual consideration of the resolution plans (or range of plans), a change management process will be implemented to manage evolution of the resolution details and implementation schedules. Senior management approval for any proposed changes to resolution scope and/or completion timeframes will be required, and documented, consistent with OPG Project Management processes.

The Integrated Implementation Plan will be tracked and progress will be regularly reported throughout the implementation period.

3.4.2 Integrated Implementation Plan Report

The Integrated Implementation Plan Report will be structured to allow a reader to understand the implementation plan and the basis for the plan. The report will begin with a summary of work completed in the Safety Factor Reports and the Global Assessment Report.

The tabularized IIP will be included in the report to facilitate a full understanding of the related safety enhancement initiatives, their priority, and safety basis. These will include the new initiatives that came from the Safety Factor Reviews and the Global Assessment, and the existing initiatives that were integral to the overall assessment of safety.

To facilitate the CNSC review of the Integrated Implementation Plan, the plan will be presented in a manner aligned with the CNSC Safety and Control Areas [6].

The report will also summarize the implementation tracking and reporting process, and the change management process for the IIP. The processes will allow tracking of initiatives to completion or resolution in an auditable manner, consistent with OPG's Management System.

In closing, the report will again reiterate the overall assessment of safety. It will summarize the strengths and the enhancement plans with attention to defence-in-depth. It will affirm the safety of the plant, and it will state that the enhancements identified through the review, combined with the existing station programs and other enhancements planned, will reinforce the continued safe operation of the station through the period of PSR2.

The Integrated Implementation Plan Report will be submitted to CNSC staff for acceptance, per CNSC REGDOC-2.3.3 [6]. The IIP will be a component of OPG's PROL renewal application, should OPG make the decision to pursue extended operation of Pickering NGS beyond 2020.

4.0 PSR2 GOVERNANCE

<u>Quality Management</u>

PSR2 work will be conducted under OPG's quality management program (compliant with CSA N286-05). Where external contractors are engaged in performing portions of PSR2, they will either work under OPG's quality program, or under a quality program that has been accepted by OPG as meeting the quality requirements for the contracted work scope. It will be confirmed that these contractors have appropriate prior experience in order to conduct the OPG PSR2 work. This approach is consistent with OPG's management system.

Teams of reviewers composed of qualified and experienced experts in the subject matter of the reviews have been established. The results of the reviews will be evaluated by OPG to confirm that the reviewers had access to all of the information relevant to the performance of the review. OPG will accept the contracted deliverables for OPG use.

Project Management

OPG project management principles will be applied in performing PSR2.

The key target dates for completion, and submission to the CNSC, of PSR2 deliverables are:

Date	Item
January 2016	PSR2 Basis Document R0
June 2016	PSR2 Basis Document R1
June 2016	Safety Factor review reports 2 (R0), 3 & 4
September 2016	Remainder of the Safety Factor review reports (including R1 of the Safety Factor 2 report)
December 2016	Global Assessment Report
April 2017	Integrated Implementation Plan R0
August 2017	Integrated Implementation Plan R1

These dates are aligned with the agreed upon strategy and schedule in the OPG-CNSC Protocol for PSR2 conduct [32]. The completion / submission date for Revision 1 of the IIP is dependent on the timing of receipt and review of CNSC review comments on Revision 0 of the IIP.

Project Communications

Effective communications practices will be used throughout the performance of PSR2. Interfaces between the work of different external contractors and OPG staff will be managed to ensure consistency between related deliverables and accuracy of information.

Urgent issues will be communicated to the project manager promptly, including any emerging observations that suggest a non-compliance with the current PROL.

Communication with the CNSC will take place through Nuclear Regulatory Affairs and Stakeholder Relations, consistent with established OPG processes.

5.0 ACRONYMS

AECB	Atomic Energy Control Board
AECL	Atomic Energy of Canada Limited
CNSC	Canadian Nuclear Safety Commission

COP	Continued Operations Plan
CSA	Canadian Standards Association
EPRI	Electric Power Research Institute
IAEA	International Atomic Energy Agency
IIP	Integrated Implementation Plan
ISR	Integrated Safety Review
LCH	Licence Conditions Handbook
PARTS	Pickering A Return to Service
PROL	Power Reactor Operating Licence
PSR	Periodic Safety Review
REGDOC	CNSC Regulatory Document
SIS	Systems Important to Safety
SOE	Safe Operating Envelope
SSC	Structures, Systems and Components
SSG	Specific Safety Guide
US-NRC	United States - Nuclear Regulatory Commission

6.0 REFERENCES

[1] CNSC Letter, M. Santini to B. McGee, “Pickering NGS: CNSC Staff Assessment of 2014 COP, SOP, and CALs”, June 18, 2015, P-CORR-00531-04493, e-Doc 4782433.

[2] OPG Report, “Pickering NGS-B Integrated Safety Review – Final ISR Report”, NK30-REP-03680-00015.

[3] OPG Report, “Darlington NGS Integrated Safety Review – Final ISR Report”, NK38-REP-03680-10104.

[4] OPG Report, “Darlington NGS Integrated Safety Review Emerging Issues Report”, NK38-REP-03680-10207.

[5] CNSC, Summary Record of Proceedings – Application to Request a Removal of the Hold Point for the Pickering Nuclear Generating Station, June 2014

[6] CNSC Regulatory Document REGDOC-2.3.3, Periodic Safety Reviews, April 2015.

[7] CNSC Document INFO-0795, Licensing Basis Objective and Definition, January 2010.

[8] CNSC, Nuclear Power Reactor Operating Licence, Pickering Nuclear Generating Station, PROL 48.02/2018, effective date December 18, 2015.

[9] CNSC, Licence Conditions Handbook, LCH-PNGS, R004, effective date December 22, 2015.

[10] IAEA Safety Standards Series, Specific Safety Guide No. SSG-25, “Periodic Safety Review for Nuclear Power Plants”, 2013.

[11] OPG Letter, B. McGee to M. Santini, “Pickering 5-8 Continued Operations Plan – 2015 Final Update”, December 15, 2015, P-CORR-00531-04470.

[12] AECL Report, “Review of Pickering A Design Against Current Codes and Standards (C1-007-03-01-007)”, 44RS-00531-ASD-001 R04, November 8, 2000.

[13] AECB Letter, C. B. Parsons to R. J. Strickert, “Pickering NGS-A Return to Service Project, Request for Approval of Code Effective Dates”, April 6, 2000, NA44-CORR-00531-00125.

[14] OPG Report, “Pickering A ASME Codes Reconciliation”, NA44-REP-00584.1-10001 R00, August 2000.

[15] AECL Report, “Methodology for Review of Pickering A Design Against Current Regulations and Standards (C1-007-03-01-002)”, 44RS-00531-AB-001 R01, November 8, 2000.

[16] OPG Letter, R.J. Strickert to J.S.C. Tong “Pickering A - Updated Basis for Return to Service Document”, April 20, 2001, NA44-CORR-00531-00381.

[17] OPG List, “Pickering A and B List of Safety Related Systems”, P-LIST-06937-00001, 2011.

[18] OPG Report, “Pickering A Systems Important to Safety”, NA44-REP-03611-00004 R1, 2008.

[19] OPG Report, “Pickering B Systems Important to Safety”, NK30-REP-03611-00024, 2014.

[20] OPG Instruction, “Preparation of Safe Operating Envelope Compliance Tables”, N-INS-03602-10001 R1, 2015.

[21] CNSC Regulatory Document RD-360, Life Extension of Nuclear Power Plants.

[22] IAEA Safety Standards Series, Safety Guide No. NS-G-2.10, Periodic Safety Reviews of Nuclear Power Plants.

[23] OPG Report, “Pickering 5-8 Continued Operations Plan”, NK30-PLAN-00531-00001.

[24] OPG Report, “Pickering Sustainable Operations Plan”, P-PLAN-09314-00001.

[25] OPG Letter, W.M.Elliott to P.A Webster and M. Santini, “Design Codes and Standards Effective Dates for OPG Nuclear Fleet”, April 30, 2012, N-CORR-00531-05661.

[26] OPG Instruction, “Nuclear Refurbishment Issue Prioritization Process – Darlington”, N-INS-00770-10005 R2, 2014.

[27] IAEA International Safety Advisory Group, “Defence in Depth in Nuclear Safety”, INSAG-10, 1996.

[28] OPG Standard, “Reliability Monitoring and Reporting of Systems Important to Safety”, N-STD-RA-0033 R2, 2013.

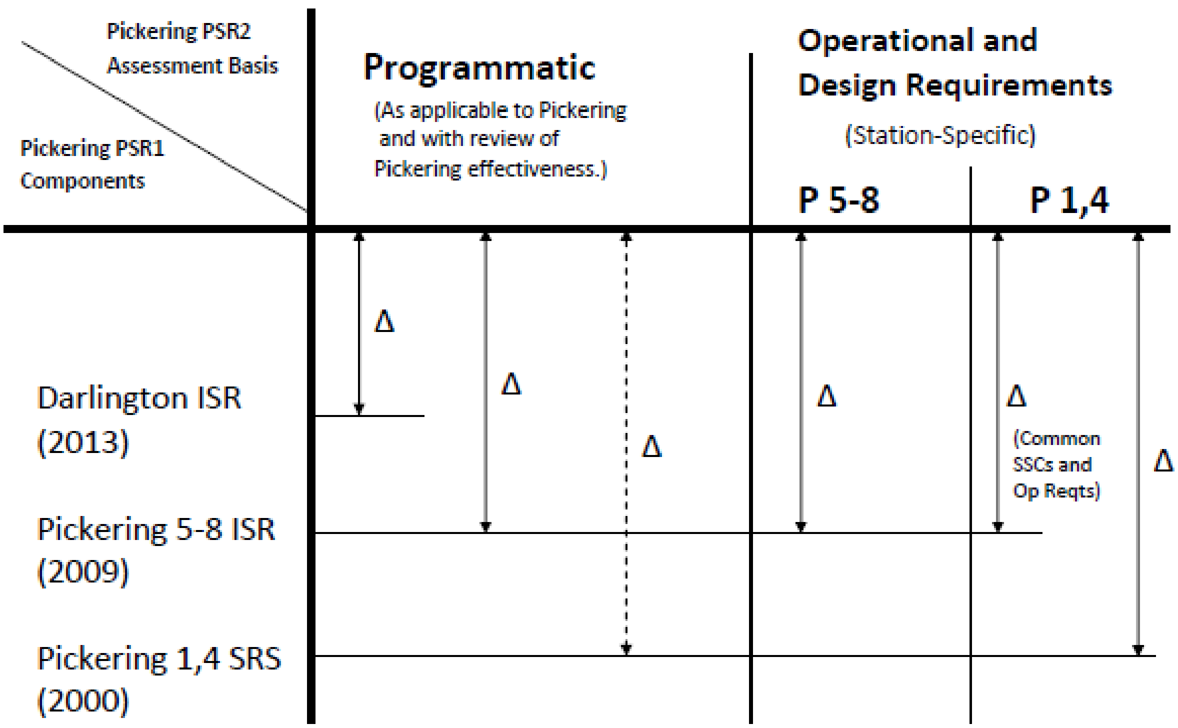
[29] CNSC Regulatory Document, “Reliability Programs for Nuclear Power Plants”, RD/GD-98, 2012.

[30] OPG Standard, “Safe Operating Envelope”, N-STD-MP-0016 R2, 2012.

[31] OPG Procedure, “Nuclear Refurbishment Integrated Safety Review – Darlington”, N-PROC-LE-0005 R03.

[32] Protocol, “OPG-CNSC Protocol for the Conduct of a Periodic Safety Review in Support of Pickering NGS Licence Renewal”, e-Doc 4927127, P-CORR-00531-04725, February 2016.

Appendix A: PSR1 / PSR2 Interface



The diagram is a schematic depiction of the primary interfaces between the elements of PSR1 and PSR2, and the process by which PSR1 findings are to be applied to PSR2. PSR2 is a subsequent PSR, and builds on the assessments of PSR1 to the extent that they are applicable. The focus is on changes between the PSR1 and PSR2 Assessment Bases and applicable results.

Programmatic aspects of the Pickering 5-8 and Darlington ISRs may be (or remain) applicable to Pickering NGS, and will be updated to reflect the latest OPG Governance where required (with a review of the effectiveness at Pickering for the related OPG Programs). Requirements related to station-specific aspects of the PSR2 Assessment Basis will draw on aspects of the Pickering 5-8 and Pickering 1,4 elements of PSR1. In addition, some station-specific gaps from the Darlington ISR will be applied to Pickering NGS if it is determined that they also apply to Pickering NGS.

Appendix B: Structures, Systems and Components within the Scope of PSR2

The scope of the Structures, Systems and Components (SSC) within the PSR2 review will encompass the Pickering Safety Related Systems [17] with a focus on Pickering Systems Important to Safety (SIS) [18], [19], and Safe Operating Envelope (SOE) Systems [20].

OPG defines Safety Related Systems as those systems, and the components and structures thereof, which, by virtue of failure to perform in accordance with the design intent, have the potential to impact on the radiological safety of the public or plant personnel from operation of the nuclear power plant [17]. Safety Related Systems are associated with the provision of safety related functions including regulation (and shutdown and start up) (control), and cooling of the reactor core (cool), and limiting the release of radioactive material and exposure of plant personnel and/or the public during normal operation and accident conditions (contain).

Systems Important to Safety are defined following OPG governance [28], considering the risk importance of systems and utilizing expert review panels to select these systems. The identification of SIS is consistent with the requirements of CNSC RD/GD-98 [29].

Safe Operating Envelope systems are identified per OPG governance [30], and include systems, and their associated critical components and structures, for which operational safety requirements are specified to conform with the Safety Report and hence, the Safety Analysis. OPG has prepared formal Operational Safety Requirements (OSR) documents in support of the SOE systems. The SOE systems and associated OSRs are listed in OPG governance [20] and are also listed in the Licence Conditions Handbook [9].

The Pickering NGS 1,4 SOE/SIS systems are listed in Table B1 below. The Pickering NGS 5-8 SOE/SIS systems are listed in Table B2 below.

Critical Structures (e.g., Reactor Buildings, Pressure Relief Duct, Vacuum Building) will be considered in the review.

Additional systems may be considered in the reviews conducted as part of PSR2 if required to demonstrate compliance with specific modern codes and standards.

B.1.0 TABLE B1 – PICKERING NGS 1,4 SOE / SIS SYSTEMS

#	Pickering NGS 1,4 Systems*	SOE**	SIS***
1.	Emergency Coolant Injection System	√	√ (Also includes associated recovery system (with Moderator Pumps and Moderator Room Active Drainage Sump Pumps))
2.	Shutdown System A	√	√
3.	Shutdown System E	√	√
4.	Negative Pressure Containment Systems	√	√
5.	Powerhouse Emergency Venting System	√	√
6.	Reactor Regulating System	√	
7.	Service Water Systems	√	√ (Limited to Emergency Low and High Pressure Service Water)
8.	Moderator System	√	
9.	Electrical Power System	√	√ (Limited to Standby Class III power and Class III 600V Interstation Transfer Bus, Emergency Transfer Scheme and Class III 600V Motor Control Centres 54130-MCC-18 and MCC-19)
10.	Emergency Boiler Water Supply	√	
11.	Heat Transport System++	√	√ (Limited to Heat Transport Pressure and Inventory Control System and Heat Transport D_2O Recovery System)
12.	Shutdown Cooling System	√	√
13.	Boiler Emergency Cooling System	√	√
14.	Feedwater System	√	√
15.	Main Steam Supply System	√	
16.	Fuel and Reactor Physics	√	
17.	Annulus Gas System	√	
18.	Fuel Handling System & Irradiated Fuel Bays	√	
19.	Critical Safety Parameter Monitoring Instrumentation	√	
20.	Shield Cooling System	√	
21.	Interstation Transfer Bus	√	
22.	Powerhouse Environmental Protection System	√	
23.	Critical Structures (e.g., Reactor Buildings, Pressure Relief Duct and Vacuum Building)		Included in Review – See +Note below

* Includes required critical components and structures.
**Specific USIs are provided in associated system OSRs listed in [20] and [17]. Also includes elements of specified support systems (e.g., Instrument Air) where required for credited design basis functions. (This is generally reflected in criticality coding.)
***Specific USIs and required functional elements are detailed in [18] and [17].
+Note: Critical Structures supporting SOE/SIS operation will also be included in the review.
++Fuel Channels and Feeders will also be included in this review.

B.2.0 TABLE B2 – PICKERING NGS 5-8 SOE / SIS SYSTEMS

#	Pickering 5-8 Systems*	SOE**	SIS***
1.	Emergency Coolant Injection System	√	√
2.	Shutdown System One	√	√
3.	Shutdown System Two	√	√
4.	Negative Pressure Containment Systems	√	√
5.	Powerhouse Emergency Venting System	√	√
6.	Reactor Regulating System	√	
7.	Service Water Systems	√	√ (Limited to Class III Service Water (Low and High Pressure))
8.	Moderator Systems	√	
9.	Group 1 Electrical Power System	√	√ (Limited to Standby Class III Power / Class II Power and also includes Class II UPS Room Ventilation)
10.	Emergency Water Supply System	√	√ (Limited to Emergency Water Supply to Boilers, Heat Transport, and Moderator)
11.	Heat Transport System++	√	
12.	Shutdown Cooling System	√	√
13.	Boiler Emergency Cooling System	√	
14.	Feedwater System	√	√ (Limited to Auxiliary Boiler Feedwater and Auxiliary Condensate Systems)
15.	Main Steam Supply System	√	
16.	Fuel and Reactor Physics	√	
17.	Emergency Power Supply	√	√
18.	Fuel Handling & Irradiated Fuel Bays	√	
19.	HPECI Power Supplies	√	
20.	Annulus Gas System	√	
21.	Critical Safety Parameter Monitoring Instrumentation	√	
22.	Shield Cooling System	√	
23.	Critical Structures (e.g. Reactor Buildings, Pressure Relief Duct and Vacuum Building)		Included in Review – See +Note below

* Includes required critical components and structures
**Specific USIs are provided in associated system OSRs listed in [20] and [17]. Also includes elements of specified support systems (e.g., Instrument Air) where required for credited design basis functions. (This is generally reflected in criticality coding.)
***Specific USIs and required functional elements are detailed in [19] and [17].
+Note: Critical Structures supporting SOE/SIS operation will also be included in the review.
++Fuel Channels and Feeders will also be included in this review.

Appendix C: Safety Factor Review Tasks

Objectives and Review Tasks for the 15 Safety Factor reviews are shown in the tables that follow.

Review Tasks for Safety Factors 1 to 14 are an interpretation of requirements and guidance set out in IAEA SSG-25 [10]. The Review Tasks have been chosen to fulfil the requirements of IAEA SSG-25 while maximizing consistency with Review Tasks used in previous ISRs. CNSC REGDOC-2.3.3 encompasses all of the PSR Safety Factors recommended by the IAEA in SSG-25 and expands upon it by adding Safety Factor 15, Radiation Protection. As a result, PSR2 Review Tasks for Safety Factor 15 are derived from REGDOC-2.3.3 Appendix A.

Additional detail regarding the derivation and application of Review Tasks is provided in Sections 2.6.1.2 and 3.2.1.

Safety Factor 1 – Plant Design	
Objective	The objective of the review of plant design is to determine the adequacy of the design of the nuclear power plant and its documentation by assessment against the current licensing basis and national and international standards, requirements and practices.
Review Tasks	The following Review Tasks are intended to support the Objective above: *Note: The Safety Factor Review Tasks and Process Sections of this report (Sections 2.6.1.2 and 3.2.1) provide information on the derivation and application of Review Tasks for PSR2.* (1) Confirm that a detailed description of the plant design, documenting the Design Basis, supported by layout, systems and equipment drawings exists. (2) Assess the adequacy of design documentation. (3) Identify the SSCs important to safety (Appendix B of this document). (4) Review the application of defence in depth. This includes an examination of: - The degree of independence of the levels of defence in depth; - The adequacy of delivery of preventive and mitigatory safety functions; - Redundancy, separation and diversity of SSCs important to safety; - Defence in depth in the design of structures (for example: • The degree of independence of the levels of defence in depth; • The adequacy of delivery of preventive and mitigatory safety functions; • Redundancy, separation and diversity of SSCs important to safety; • Review of the integrity of fuel, cooling circuit and containment building). (5) Confirm that the human–machine interface is considered in the design of the control room and other workstations, that analysis of human information requirements and task workload is performed, and that there is linkage to the Probabilistic Safety Assessment, Deterministic Safety Analyses and Hazard Analysis. This review should include a discussion of how guidance such as U.S. NRC NUREG-0700 Revision 2, "Human-System Interface Design Review Guidelines" and NUREG-0711 Revision 2, "Human Factors Engineering Program Review Model" identified in CNSC REGDOC-2.5.2 are relevant to the design of Pickering U1,4 and U5-8. *Note: The above activity will be dealt with in the Plant Design Safety Factor (rather than Human Factors), as it is the only human factors activity that deals with plant design.* (6) Assess the adequacy of the arrangements for providing radiological protection. (7) Where the plant has undergone a significant number of modifications over its lifetime or in the period since PSR1, examine the cumulative effects of all modifications on the design. (8) Confirm that the plant SSCs are compliant with the design specifications and consistent with the design documentation.

Safety Factor 2 – Actual Condition of SSCs Important to Safety	
Objective	The objective of the review of this Safety Factor is to determine the actual condition of SSCs important to safety and so to consider whether they are capable and adequate to meet design requirements, throughout the period of PSR2. In addition, the review should verify that the condition of SSCs important to safety is properly documented, as well as reviewing the ongoing maintenance, surveillance and in-service inspection programmes, as applicable.
Review Tasks	The following Review Tasks are intended to support the Objectives above: *Note: The Safety Factor Review Tasks and Process Sections of this report (Sections 2.6.1.2 and 3.2.1) provide information on the derivation and application of Review Tasks for PSR2.* *Note: The SSCs within the scope of PSR2 (SSCs important to safety) are defined in Appendix B of this document.* (1) Assess and document present conditions of the SSCs important to safety and confirm appropriate measures to address any significant existing or anticipated aging degradation. Any major difference between operating units with respect to aging degradation mechanisms, present condition, or recommended actions shall also be presented. (2) Confirm resources and facilities (on and off site) are available for ongoing plant maintenance. (3) After determining the actual condition of the SSCs important to safety, each of these SSCs will be assessed against the current design basis to confirm that design basis assumptions have not been significantly challenged and will remain that way throughout the period of PSR2. (4) Review the condition and operation of spent fuel storage facilities and their effect on the spent fuel storage strategy for Pickering NGS. (5) Assess dependence on obsolescent equipment for which no direct substitute is available. (6) Assess dependence on essential services and/or supplies external to the plant.

Safety Factor 3 – Equipment Qualification (Environmental and Seismic)	
Objective	The objective of the review of equipment qualification is to determine whether plant equipment important to safety has been properly qualified (including for environmental conditions) and whether this qualification is being maintained through an adequate program of maintenance, inspection and testing that provides confidence in the delivery of safety functions throughout the period of PSR2.
Review Tasks	The following Review Tasks are intended to support the Objective above: *Note: The Safety Factor Review Tasks and Process Sections of this report (Sections 2.6.1.2 and 3.2.1) provide information on the derivation and application of Review Tasks for PSR2.* (1) Confirm there exists a suite of Engineering programs or processes to ensure equipment qualification requirements are met and documented. (2) Confirm equipment qualification has been adequately established for all service conditions expected during normal operation, anticipated operational occurrences and accident conditions. These service conditions are subdivided into environmental conditions and operational conditions. Environmental conditions include ambient temperature, pressure, humidity/steam, radiation, water/chemical sprays, fluid submergence, fire and seismic vibration. Operational conditions include process related conditions such as vibration, load cycling, electrical loading parameters, electromagnetic interference, mechanical loads and process fluid conditions. (3) Perform an objective confirmation that the installed equipment is qualified to perform its Design Basis function for its operational life and that effective programs exist to monitor for timely maintenance or replacement, as required. (4) Confirm existence of a process for ensuring compliance with these programs and of documented previous qualification measures taken to ensure qualification throughout the equipment’s installed life (i.e. prescribed testing, calibration, maintenance, and parts replacement). (5) Confirm existence of a surveillance program and a feedback procedure to ensure aging degradation of qualified equipment remains insignificant. (6) Confirm existence of monitoring of actual environmental conditions and identification of ‘hot spots’ of high activity or temperature. (7) Confirm existence of an assessment that determines the effects of equipment failures on equipment qualification and appropriate corrective actions and/or safety improvements to maintain equipment qualification. (8) Confirm there is protection and adequate separation of qualified equipment from adverse environmental conditions. (9) Confirm physical condition and functionality capability of qualified equipment has been checked by walkdowns. (10) Confirm that changes to equipment classification have occurred, as required, as a result of major design modifications made since PSR1.

Safety Factor 4 – Aging	
Objective	The objective of the review of aging is to determine whether aging aspects affecting SSCs important to safety are being effectively managed and whether an effective aging management program is in place so that all required safety functions will be delivered throughout the period of PSR2.
Review Tasks	The following Review Tasks are intended to support the Objective above: *Note: The Safety Factor Review Tasks and Process Sections of this report (Sections 2.6.1.2 and 3.2.1) provide information on the derivation and application of Review Tasks for PSR2.* (1) Confirm there is a documented method and criteria for identifying safety related SSCs covered by the Aging Management Program. (2) Ensure there is an effective Aging Management Program and dedicated organization with clearly defined roles and responsibilities with sufficient resources to continually assess aging effects in safety related SSCs. (3) Establish a list of SSCs covered by the aging management program and records that provide information in support of the management of aging. (4) Evaluate and document impact of potential aging degradation of safety related SSCs. (5) Confirm or develop understanding of dominant aging mechanisms of safety related SSCs. (6) Confirm existence of predictive maintenance program. (7) Ensure existence of programs for timely detection and mitigation of aging mechanisms and/or aging effects of any SSCs important to safety, including obsolescence of technology used in the plant or obsolescence of services or supplies external to the plant. (8) Establish acceptance criteria and required safety margin for safety related SSCs for the period of PSR2 through reliability and risk assessments. (9) Confirm adequacy of management of the effects of aging on those parts of the plant that will be required for safety when the reactor has ceased operation, for example the spent fuel storage facilities. (10) Confirm the models used to predict the evolution and advancement of aging degradation are properly supported in accordance with current accepted practices pertaining to aging degradation.

Safety Factor 5 – Deterministic Safety Analysis	
Objective	The objective of the review is to determine to what extent the existing deterministic safety analysis is complete and remains valid when the following aspects have been taken into account: • The actual plant design, including all modifications of SSCs since the last update of the safety analysis report or PSR1; • Current operating modes and fuel management; • The actual condition of SSCs important to safety and their predicted state at the end of the period covered by PSR2; • The use of modern, validated computer codes; • Current deterministic methods; • Current safety standards and knowledge (including research and development outcomes); • The existence and adequacy of safety margins.
Review Tasks	The following Review Tasks are intended to support the Objective above: *Note: The Safety Factor Review Tasks and Process Sections of this report (Sections 2.6.1.2 and 3.2.1) provide information on the derivation and application of Review Tasks for PSR2.* (1) Confirm the existence of current deterministic safety analyses and the assumptions used to perform these analyses. (2) Evaluate the documentation and processes for defining, implementing, and maintaining the Safe Operating Envelope. (3) Perform assessment of OPG's Deterministic Safety Analysis to determine if the postulated events, event sequences and event combinations covered by the existing analysis are sufficient when compared against those for a modern nuclear power plant in accordance with the methodology in CNSC REGDOC-2.4.1, "Deterministic Safety Analysis". (4) Review adequacy of the documented guidelines for Deterministic Safety Analysis. (5) Evaluate the supporting analyses for design extension conditions to confirm that the arrangements aimed at preventing or mitigating severe core damage meet regulatory requirements. (6) Confirm that the impact of equipment failures and human errors, as well as the adequacy of engineering and administrative measures to prevent and mitigate accidents, have been analyzed and documented. (7) Confirm that the capabilities of the plant in its current state, and where relevant with account taken of planned safety improvements, have been demonstrated to be within regulatory requirements and expectations for both normal operation and accident conditions. In addition, confirm that plans are in place to ensure that forecast operational conditions of the plant will meet acceptance criteria for the design basis, including adequacy of safety margins, throughout the period of PSR2.

Safety Factor 6 - Probabilistic Safety Assessment (PSA)	
Objective	The objectives of the review of the PSA are to determine: • The extent to which the existing PSA study remains valid as a representative model of the plant; • Whether the results of the PSA show that the risks are sufficiently low and well balanced for all postulated initiating events and operational states; • Whether the scope (which should include all operational states and identified internal and external hazards), methodologies and extent (i.e. Level 1, 2 or 3) of the PSA are in accordance with current national and international standards and good practices; • Whether the existing scope and application of PSA are sufficient.
Review Tasks	The following Review Tasks are intended to support the Objective above: *Note: The Safety Factor Review Tasks and Process Sections of this report (Sections 2.6.1.2 and 3.2.1) provide information on the derivation and application of Review Tasks for PSR2.* (1) Confirm existence of a PSA and the assumptions used, the fault schedule, the representations of operator actions and common cause events, the modelled plant configuration and consistency with other aspects of the safety case. (2) Confirm existence of processes to assess the impact of changes in plant design, operation, and plant specific failure data and update the PSA to reflect the current plant status as required. (3) Confirm there are guidelines to account for operator actions, common cause events, cross-link effects, redundancy, and diversity. (4) Confirm that the accident management programs for accident conditions (design basis accident conditions and design extension conditions) are consistent with PSA models and results. (5) Confirm that the results of the PSA show that risks are sufficiently low and well balanced for all postulated initiating events and operational states, and meet relevant probabilistic safety criteria. (6) Review the extent to which hazards are represented in the PSA to verify that omissions are based on site specific justifications and that these omissions do not weaken the overall risk assessment for the plant.

Safety Factor 7 – Hazard Analysis	
Objective	The objective of the review of hazard analysis is to determine the adequacy of protection of the plant against internal and external hazards, with account taken of the plant design, site characteristics, the actual condition of the SSCs important to safety and their predicted state at the end of the period covered by PSR2, and current analytical methods, safety standards and knowledge.
Review Tasks	The following Review Tasks are intended to support the Objective above: *Note: The Safety Factor Review Tasks and Process Sections of this report (Sections 2.6.1.2 and 3.2.1) provide information on the derivation and application of Review Tasks for PSR2.* (1) Perform an assessment of the existing Deterministic and Probabilistic analyses to confirm existence of hazard analyses for hazards listed below. The following hazards are to be included in the assessment: (i) Internal Hazards: - Fire, Pipe whip, Steam release, Toxic gas, Flooding, Missiles, Spray, Explosion. (ii) External Hazards: - Changes in site characteristics, High winds (Tornado), Seismic, Toxic gas, Flooding, Extreme temperatures, Aircraft crash, Explosions. (2) Confirm that the analyses and/or methods take into account the plant design and the condition of SSCs important to safety (both at present and predicted for the end of the period covered by PSR2). (3) For each relevant hazard, verify, by means of current analytical techniques and data, that the frequency of occurrence and/or the consequences of the hazard are sufficiently low so that either no specific protective measures are necessary, or the preventive and mitigatory measures in place are adequate.

Safety Factor 8 – Safety Performance	
Objective	The objective of the review of safety performance is to determine whether the plant's safety performance indicators and records of operating experience, including the evaluation of root causes of plant events, indicate any need for safety improvements.
Review Tasks	The following Review Tasks are intended to support the Objective above: *Note: The Safety Factor Review Tasks and Process Sections of this report (Sections 2.6.1.2 and 3.2.1) provide information on the derivation and application of Review Tasks for PSR2.* (1) Confirm existence of a system for identifying, classifying and recording safety related incidents and operating experience including: • Safety related incidents, low level events and near misses; • Safety related operational data; • Maintenance, inspection and testing; • Replacements of SSCs important to safety owing to failure or obsolescence; • Modifications, either temporary or permanent, to SSCs important to safety; • Unavailability of safety systems; • Radiation doses (to workers, including contractors); • Off-site contamination and radiation levels; • Discharges of radioactive effluents; • Generation of radioactive waste; (2) Confirm that safety related incidents are investigated using root cause analysis and that lessons learned from investigation of these incidents are fed back into the conduct of Operations and Maintenance. (3) Confirm that the results of the root cause analysis are used to minimize the chances of the same incident reoccurring. (4) Confirm that information from trend analysis of safety related incidents is fed back into the conduct of Operations and/or Maintenance. (5) Confirm there is an adequate set of performance indicators that provides a systematic and comprehensive method to record, trend and analyze safety related data including the major system parameters, and maintenance and inspection records. Performance indicators may include: - Frequency of unplanned trips while the reactor is critical - Satisfactory performance of safety system tests within required limits - Special Safety System unavailability - Reliability of Systems Important to Safety - Collective annual radiation dose of plant staff - Amount of gaseous and liquid radioactive release relative to permitted limits - Heavy water escape and loss rates - Fuel reliability - Chemistry index - Volume of Low Level radioactive waste - Change control index - Maintenance backlog - Training - Environment Index - Non-radioactive effluents, including hazardous substances - Non-radioactive wastes - Spills.

	(6) Confirm that for cases where performance indicators show an unsatisfactory trend, corrective action is taken. (7) Review the adequacy of: - Records of the integrity of physical barriers for the containment of radioactive material. - Records of radiation doses to persons on the site. - Records of data from off-site radiation monitoring and records of the quantities of radioactive effluents. - Records of non-radioactive effluents, including hazardous substances. - Records of radioactive and non-radioactive waste. - Records of spills. - Records of other environmental impacts. (8) Consider the effects of any changes in operation at the plant on safety performance. In particular, confirm that current indicators and other safety performance methods continue to be relevant in the context of current and future operations, and confirm that only relevant data and records are used.

Safety Factor 9 – Use of Experience from Other NPPs and Research Findings	
Objective	The objective of the review of experience from other plants and research findings is to determine whether there is adequate feedback of relevant experience from other nuclear power plants and from the findings of research and whether this is used to introduce reasonable and practicable safety improvements at the plant or in the operating organization.
Review Tasks	The following Review Tasks are intended to support the Objective above: *Note: The Safety Factor Review Tasks and Process Sections of this report (Sections 2.6.1.2 and 3.2.1) provide information on the derivation and application of Review Tasks for PSR2.* (1) Confirm existence and adequacy of a program for the sending and receiving of experience relevant to safety to and from other nuclear power plants and relevant nonnuclear plants. (“Other nuclear power plants” specifically include the IAEA, OECD/NEA, WANO, INPO as well as CANDU Owners Group and experience within OPG at Darlington.) (2) Confirm existence of a program for receiving of information on the findings of relevant research programs. (3) Confirm there is a process for assessing the significance of operating experience from other plants and incorporating the lessons learned into improving safety performance at the station. (4) Confirm that there is a process for assessing the significance of research findings and technology developments and for incorporating relevant improvements into the station's design and operation. (5) Review adequacy and effectiveness of the feedback arrangements and timely implementation of assessment findings. (Assess program audit results.) (6) List the major OPEX events and resulting plant changes that have resulted since PSR1 was completed.

Safety Factor 10 – Organization, Management System and Safety Culture	
Objective	The objective of the review of this safety factor is to determine whether the organization, management system and safety culture are adequate and effective for ensuring the safe operation of the plant.
Review Tasks	The following Review Tasks are intended to support the Objective above: *Note: The Safety Factor Review Tasks and Process Sections of this report (Sections 2.6.1.2 and 3.2.1) provide information on the derivation and application of Review Tasks for PSR2.* (1) Review organization and administrative procedures to ensure they play a significant role in defining safety culture and evaluate the adequacy of safety culture indicators. (2) Establish existence of a safety policy to ensure that safety takes precedence over production where a conflict between these two requirements exists. (3) Identify the method for setting performance targets and confirm that these targets are regularly and systematically reviewed. Confirm that appropriate actions are initiated if safety targets are not met. (4) Confirm that the published Nuclear organization clearly defines the roles and responsibilities of all individuals and work groups who are involved in activities that could influence the safe operation of the station. Ensure that this organization is understood and that adequate and effective procedures are in place to ensure the availability of these resources and control changes to this organization. (5) Establish that mechanisms for maintaining configuration control of the plant and its documentation are effective and up-to-date. (6) Confirm that there are formal arrangements for employing external technical, maintenance or other specialist staff, and confirm that the contracting procedures ensure that contract employees are qualified to do the work assigned to them. (7) Confirm that there is an approved Quality Assurance program and that regular Quality Assurance audits are conducted involving both internal and independent assessors. (8) Confirm that a program for self-assessment and continuous improvement has been adequately and effectively implemented including feedback of experience relating to organizational and management failures. (9) Confirm there is a system to ensure that comprehensive, easily retrievable, and auditable records exist of baseline design information, and operational and maintenance history. (10) Confirm there is an effective framework in place to support the management of regulatory affairs. (11) Confirm that the organization and management system include the processes and supporting information that explain how work is to be specified, prepared, reviewed, performed, recorded, assessed and improved. (12) Confirm there is control of purchasing of equipment and services where this affects plant safety. (13) Confirm there are comprehensive communication policies in place. (14) Confirm that a questioning attitude exists and conservative decision making is undertaken in the organization. (15) Verify that there is a process in place for prioritization of safety issues, with realistic objectives and timescales that ensures that these issues receive proper resources.

Safety Factor 11 – Procedures	
Objective	The objective of the review of procedures is to determine whether the operating organization's processes for managing, implementing and adhering to operating and working procedures and for maintaining compliance with operational limits and conditions and regulatory requirements are adequate and effective and ensure plant safety.
Review Tasks	The following Review Tasks are intended to support the Objective above: *Note: The Safety Factor Review Tasks and Process Sections of this report (Sections 2.6.1.2 and 3.2.1) provide information on the derivation and application of Review Tasks for PSR2.* *The review of this safety factor will focus on those procedures that have the highest safety significance.* (1) Determine if there is a process for the development, approval, and documenting of all safety related procedures. (2) Confirm there is a formal process for modifying procedures affecting safety, including adequate arrangements for tracking changes. (3) Confirm there is a program for assessing procedures and performance audits to determine if there is regular review and maintenance of these procedures. (4) Confirm that self-assessments are performed to ensure that the procedures are followed. (5) Establish that there is a means for assessing the adequacy of safety related procedures in comparison with industry good practices. (6) Confirm that there are operating procedures that apply comprehensively to normal, abnormal and emergency conditions (including anticipated operational occurrences, design basis accident conditions, post-accident conditions, and design extension conditions). (7) Confirm there is a means for assuring the clarity of procedures taking into account human factors. (8) Evaluate processes to update procedures to allow for changes in the assumptions made and/or the limits and conditions arising from the safety analysis, plant design and operating experience. (9) Verify that the analysis and justification of the accident management procedures are documented. (10) Verify that an appropriate process is in place for the categorization of procedures in accordance with their significance to safety. (11) Examine whether there is adequate involvement in the development of procedures by the staff who will use them. (12) Evaluate the distribution process for the control, copying and removal of obsolete versions of procedures, so that only the last approved edition is used. (13) Evaluate audits, self-assessments, safety performance and events to determine whether there is adequate understanding and acceptance of these procedures by managers and staff.

Safety Factor 12 – Human Factors	
Objective	The objective of the review of this safety factor is to evaluate the various human factors that may affect the safe operation of the nuclear power plant and to seek to identify improvements that are reasonable and practicable.
Review Tasks	The following Review Tasks are intended to support the Objective above: *Note: The Safety Factor Review Tasks and Process Sections of this report (Sections 2.6.1.2 and 3.2.1) provide information on the derivation and application of Review Tasks for PSR2.* (1) Confirm that there are procedures to ensure that a minimum number of qualified staff, appropriate to the operating state of the plant, is available at all times. (2) Confirm that adequate staff training facilities, training staff and training programs exist. (3) Confirm that the method of selecting staff for new positions and for promotions involves systematic and validated staff selection methods and a method for succession planning. (4) Confirm that there are appropriate programs for initial, refresher, and upgrade training. For operating staff, this should include the use of simulators. (5) Establish that there is training in safety culture, including for management staff, that includes work supervision practices and internal communication practices and expectations. (6) Confirm there are fitness for duty guidelines relating to hours of work, health and substance abuse. (7) Confirm that the human–machine interface is considered in the design of the control room and other workstations, that analysis of human information requirements and task workload is performed, and that there is linkage to the Probabilistic Safety Assessment, Deterministic Safety Analyses, and Hazard Analysis. This review should include a discussion of how guidance such as U.S. NRC NUREG-0700 Revision 2, "Human-System Interface Design Review Guidelines" and NUREG-0711 Revision 2, "Human Factors Engineering Program Review Model" identified in CNSC REGDOC-2.5.2 are relevant to the design of Pickering U1,4 and U5-8. *Note: This Review Task will be dealt with in the Plant Design Safety Factor as it is the only human factors activity that deals with plant design.* (8) Confirm the style and clarity of procedures provides an appropriate level of detailed guidance for staff through a review of plant events identifying inadequate procedures as a contributing cause.

Safety Factor 13 – Emergency Planning	
Objective	The objective of the review of emergency planning is to determine: (a) whether the operating organization has in place adequate plans, staff, facilities and equipment for dealing with emergencies; and (b) whether the operating organization's arrangements have been adequately coordinated with the arrangements of local and national authorities and are regularly exercised.
Review Tasks	The following Review Tasks are intended to support the Objective above: *Note: The Safety Factor Review Tasks and Process Sections of this report (Sections 2.6.1.2 and 3.2.1) provide information on the derivation and application of Review Tasks for PSR2.* (1) Confirm the full range of accidents and radiation emergencies have been identified and studied. (2) Confirm the appropriate response and mitigation strategies have been developed and have taken account of major changes at site and around the site (industrial, commercial, residential development). (3) Confirm that the station organization includes dedicated Emergency Response personnel on duty at the plant at all times, to handle accidents and emergencies. (4) Assess the adequacy of the training program for emergency response personnel including training, emergency exercises and qualification records. (5) Confirm there is a process for notification of staff that will be brought in to assist in the management of the response in the longer term. (6) Determine that there is a classification of accidents to guide the type of response. (7) Confirm there is a mechanism for notifying and informing relevant off-site organizations such as the police, fire departments, hospitals, ambulance services, regulatory bodies, local authorities, government, public welfare authorities and the news media. (8) Confirm the availability of sufficient communications equipment at the plant and at the off-site Emergency Centre to permit effective communications with Emergency Response Teams, both on and off site. (9) Assess adequacy of the emergency response procedures and training and exercises for all site staff. Confirm that integrated and partial emergency exercises have been conducted to check satisfactory function of the emergency organization and its equipment. (10) Confirm that the adequacy of on-site equipment and facilities for emergencies and offsite emergency facilities or locations, including walkdowns of relevant areas on and off the site. (11) Confirm development or existence of a program for Severe Accident Management.

Safety Factor 14 – Radiological Impact on the Environment	
Objective	The objective of the review of this safety factor is to determine whether the operating organization has an adequate and effective program for monitoring the radiological impact of the plant on the environment, which ensures that emissions are properly controlled and are as low as reasonably achievable.
Review Tasks	The following Review Tasks are intended to support the Objective above: *Note: The Safety Factor Review Tasks and Process Sections of this report (Sections 2.6.1.2 and 3.2.1) provide information on the derivation and application of Review Tasks for PSR2.* (1) Confirm there are procedures in place to ensure that permitted release limits of radiological substances are not exceeded and, if they are, that appropriate corrective action is taken to minimize the possibility of limits being exceeded in the future. (2) Confirm records of radiological effluent release are maintained in accordance with regulatory requirements. (3) Confirm that a program exists to define the requirements for alarm systems to respond to unplanned effluent releases from on-site facilities. (4) Confirm the environmental data recorded by the station is published and is available on request to the general public. (5) Review the environmental data recorded by the station and compare with the values measured before the plant was put into operation. (6) Confirm there is a process to address changes in the use of land external to the site with respect to the impact on public safety from facility releases. (7) Confirm that the monitoring program is appropriate and sufficiently comprehensive. In particular, confirm that the radiological impact of the plant on the environment is not significant compared with that due to other sources of radiation.

Safety Factor 15 – Radiation Protection	
Objective	The objective of this safety factor is to confirm that Radiation Protection has been adequately accounted for in the design and operation of the reactor facility, that Radiation Protection provisions (including design and equipment) provide adequate protection of persons from the harmful effects of radiation, and ensure that contamination and radiation exposures and doses to persons are monitored and controlled, and maintained as low as reasonably achievable (ALARA).
Review Tasks	The following Review Tasks are intended to support the Objective above: *Note: The Safety Factor Review Tasks and Process Sections of this report (Sections 2.6.1.2 and 3.2.1) provide information on the derivation and application of Review Tasks for PSR2.* (1) Confirm the adequacy of the reactor design features for Radiation Protection. (2) Confirm the adequacy of the Radiation Protection equipment and instrumentation for radiation monitoring. (3) Confirm that adequate provisions are in place to address Radiation Protection of the public and workers during nuclear emergencies. (4) Confirm that the Radiation Protection provisions have been improved as the result of external operating experience. (5) The review will demonstrate that the ALARA principle has been incorporated in any modifications of the reactor design and operational programs and arrangements.

Appendix D: Modern Laws, Regulations, Codes and Standards Applicable to PSR2

The process to identify the modern Laws, Regulations, Codes and Standards that are applicable to the PSR2 Assessment Basis involved first creating a broad list from multiple sources (potential candidate Laws, Regulations, Codes and Standards) and then filtering it to identify those that are most significant, and that are applicable to the PSR2 scope. The identification and selection criteria are detailed below.

The result of the identification and selection process was a set of modern Laws, Regulations, Codes and Standards that becomes part of the PSR2 Assessment Basis. These documents are listed in Table D1. This table also identifies the modern version/date of the Law, Regulation, Code or Standard, and the type of review that will be done in PSR2. The type of review is explained in greater detail in Section 3.2.2 of this report.

For the purposes of the performance of PSR2, OPG has defined the freeze date for modern Laws, Regulations, Codes and Standards to be January 15, 2016. Where a document was issued in January 2016 but the effective date is unknown, it will be considered to have been issued prior to the freeze date.

D.1.0 PROCESS TO DETERMINE THE MODERN LAWS, REGULATIONS, CODES AND STANDARDS APPLICABLE TO PSR2

1. Identify potential candidate Laws, Regulations, Codes and Standards:

   1.1. List all Laws, Regulations, Codes and Standards that are identified in the Licence Conditions Handbook (LCH) and group the identified Laws, Regulations, Codes and Standards into those that are:

   - Directly referenced in the Power Reactor Operating Licence (PROL) (listed in LCH Appendix C).
   - Identified in the LCH as guidance or criteria (LCH Appendix E1).
   - Identified in the LCH as referenced in the LCH (LCH Appendix E2).
   - Referenced in the main body of the LCH but not listed in the LCH Appendices.

   1.2. Identify Laws, Regulations, Codes and Standards listed by the CNSC which have been published on or prior to the freeze date (CNSC Acts and Regulations webpage, Section 2.0 – Safety and Control Areas, http://www.nuclearsafety.gc.ca/eng/acts-and-regulations/regulatory-documents/index.cfm#R2).

   1.3. Confirm that the Laws, Regulations, Codes and Standards listed by the CNSC as noted in Step 1.2 are up-to-date by cross-referencing to the CNSC Regulatory Framework webpage (Category 2 – Safety and Control Areas, http://nuclearsafety.gc.ca/eng/acts-and-regulations/regulatory-framework/regulatory-framework-plan-table.cfm).

1.4. Identify Laws, Regulations, Codes and Standards from the Darlington Integrated Safety Review (ISR) basis (basis document OPG Procedure N-PROC-LE-0005 R03 [31] and Code Refresh report NK38-REP-03680-10207 R00 [4]). The Darlington ISR basis is considered to be an important listing of currently applicable Laws, Regulations, Codes and Standards as it resulted from recent and extensive discussion between OPG and the CNSC.

The foregoing represents a comprehensive list of all possible Laws, Regulations, Codes and Standards that could apply to the Pickering PSR2 Assessment Basis, but that are not yet filtered for significance or value.

2. Determine from this candidate list of Laws, Regulations, Codes and Standards those that should be excluded from the PSR2 Assessment Basis.

The following criteria were applied to remove Laws, Regulations, Codes and Standards from the candidate list:

2.1. If the Law, Regulation, Code or Standard is identified by more than one source, duplication in the foregoing list is removed.

2.2. If the Law, Regulation, Code or Standard is not in the Darlington ISR basis, not on the CNSC website listings as currently applicable, and not one of the PROL Laws, Regulations, Codes and Standards, then it is not part of the PSR2 Assessment Basis.

2.3. If the Law, Regulation, Code or Standard is not listed in the PROL and is in only one of the Darlington ISR basis or the CNSC website listings as currently applicable, then it was assessed for significance on a case-by-case basis and was excluded from the PSR2 Assessment Basis where it covered an area that is not significant to nuclear safety.

2.4. Laws, Regulations, Codes and Standards that are uniquely related to the CNSC Safety Control Areas for “Waste Management”, “Packaging and Transport”, “Security” and “Safeguards and Non-proliferation” are not part of the PSR2 Assessment Basis since they are not addressed in the Safety Factors (REGDOC-2.3.3). Requirements related to Construction, Decommissioning, “Ontario nuclear funds agreement”, and “public engagement and information” are not included in the PSR2 Assessment Basis.

2.5. Since the portions of the Pickering Nuclear site that are subject to PSR2 are federally regulated, the National Building Code of Canada and the National Fire Code of Canada are in the PSR2 Assessment Basis, and the Ontario Building Code and the Ontario Fire Code are not included in the PSR2 Assessment Basis.

2.6. Codes and Standards defined as guidelines are not part of the PSR2 Assessment Basis unless they are cited as mandatory within a PROL Code or Standard. (See Step 3.2 below.)

2.7. IAEA Codes and Standards are generally not part of the PSR2 Assessment Basis since they will have been used as references in the development of the Codes and Standards listed by the CNSC as being current. However if an IAEA Code or Standard is cited as mandatory within a PROL Code or Standard, it will be added to the PSR2 Assessment Basis. (See Step 3.2 below.)

2.8. US-NRC, and EPRI Codes and Standards are not part of the PSR2 Assessment Basis since they will have been used as references in the development of the Codes and Standards listed by the CNSC as being current.

2.9. CNSC Policy documents are excluded from the PSR2 Assessment Basis since they are for internal CNSC application.

2.10. Laws and Regulations are not considered to be part of the PSR2 Assessment Basis if they are not listed in the PROL (LCH Appendix C).

Following the exclusions in Steps 2.1 to 2.10, the remaining Laws, Regulations, Codes and Standards represent those that will form part of the PSR2 Assessment Basis, subject to the amendments noted below in Steps 3 and 4:

3. Additional considerations applied to the Laws, Regulations, Codes and Standards for use in the PSR Assessment Basis:

3.1. The version of the Law, Regulation, Code or Standard as of January 15, 2016 (the PSR documentation freeze date) is used for the PSR2 Assessment Basis. (Where a document was issued in January 2016 but the effective date is unknown, it will be considered to have been issued prior to the freeze date.) This includes where the Laws, Regulations, Codes and Standards that are on the CNSC website listings as currently applicable supersede those identified in the Darlington ISR basis. In those cases, Laws, Regulations, Codes and Standards from the CNSC website list are considered to be part of the PSR2 Assessment Basis. Where a document has a new number/type, but addresses the same topic from the same organization, the document will be treated as a revision within the PSR Assessment Basis (e.g., if a REGDOC replaces a CNSC G or RD document).

3.2. If a sub-tier Code or Standard is called up or cited as mandatory by a PROL Law, Regulation, Code or Standard (or a more recent version of a PROL Law, Regulation, Code or Standard as identified in Step 3.1), and either the sub-tier Code or Standard was not already assessed as part of PSR1 or the sub-tier Code or Standard has been updated since PSR1, then the applicable parts of the sub-tier Code or Standard are included in the PSR2 Assessment Basis if they are determined to be safety significant.

3.3. Informative / Non-mandatory sections or appendices of documents in the PSR2 Assessment Basis are not included.

4. An expert level review was performed of the Laws, Regulations, Codes and Standards removed from the initial potential candidate list generated in Step 1, to confirm that the safety significant Laws, Regulations, Codes and Standards are all included in the PSR2 Assessment Basis. The review also confirmed that the Laws, Regulations, Codes and Standards that were selected for the PSR2 Assessment Basis are consistent with the methodology approach to be taken for a subsequent PSR as defined in CNSC REGDOC-2.3.3.

D.2.0 TABLE D1 - LAWS, REGULATIONS, CODES AND STANDARDS APPLICABLE TO PSR2

#	Document Number	Document Title	Modern Version for PSR2	Type of Review
Documents Referenced in Pickering PROL 48.02/2018				
1	CSA N286	Management System Requirements for Nuclear Power Plants	N286-12	Incremental
2	CSA N290.15	Requirements for the Safe Operating Envelope of Nuclear Power Plants	N290.15-10	Incremental
3	CSA N286.7	Quality Assurance Of Analytical, Scientific And Design Computer Programs For Nuclear Power Plants	N286.7-16	Incremental
4	CSA N285.0	General Requirements For Pressure-Retaining Systems And Components in CANDU Nuclear Power Plants	N285.0-12	Incremental
5	CSA N290.13	Environmental Qualification of Equipment for CANDU Nuclear Power Plants	N290.13-05	Incremental
6	CSA N285.4	Periodic Inspection Of CANDU Nuclear Power Plant Components	N285.4-14	Incremental
7	CSA N285.5	Periodic Inspection Of CANDU Nuclear Power Plant Containment Components	N285.5-13	Incremental
8	CSA N287.7	In-Service Examination and Testing Requirements for Concrete Containment Structures For CANDU Nuclear Power Plant Components	N287.7-08	Incremental
9	CSA N288.1	Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities	N288.1-14	Incremental
10	CSA N288.4	Environmental Monitoring Program at Class I Nuclear Facilities and Uranium Mines and Mills	N288.4-10	Incremental
11	CSA N293	Fire Protection for CANDU Nuclear Power Plants	N293-12	Incremental
12	CNSC RD-204	Certification of Persons Working at Nuclear Power Plants	2008	Incremental
13	CNSC REGDOC 3.1.1	Reporting Requirements for Nuclear Power Plants	2014	Incremental

#	Document Number	Document Title	Modern Version for PSR2	Type of Review
14	CNSC REGDOC 2.4.1	Deterministic Safety Analysis	2014	Incremental
15	CNSC REGDOC 2.4.2	Probabilistic Safety Assessment (PSA) for Nuclear Power Plants	2014	Incremental
16	CNSC RD/GD-210*	Maintenance Programs for Nuclear Power Plants	2012	Incremental
17	CNSC RD/GD-98	Reliability Programs for Nuclear Power Plants	2012	Incremental
18	CNSC REGDOC 2.6.3*	Aging Management	2014	Incremental
19	CNSC REGDOC 2.9.1*	Environmental Protection: Policies, Programs and Procedures	2013	Incremental
20	CNSC REGDOC 2.10.1*	Nuclear Emergency Preparedness and Response	2014	Incremental
Additional Documents				
21	CSA N287.1	General Requirements for Concrete Containment Structures for Nuclear Power Plants	N287.1-14	Incremental
22	CSA N287.2	Material requirements for Concrete Containment Structures for CANDU Nuclear Power Plants	N287.2-08	Incremental
23	CSA N287.3	Design Requirements for Concrete Containment Structures for Nuclear Power Plants	N287.3-14	Incremental
24	CSA N287.5	Examination and Testing Requirements for Concrete Containment Structures for Nuclear Power Plants	N287.5-11	Incremental
25	CSA N289.1	General Requirements for Seismic Design and Qualification of CANDU Nuclear Power Plants	N289.1-08	Incremental
26	CSA N289.2	Ground Motion Determination for Seismic Qualification of Nuclear Power Plants	N289.2-10	Incremental
27	CSA N289.3	Design Procedures for Seismic Qualification of Nuclear Power Plants	N289.3-10	Incremental
28	CSA N289.4	Testing Procedures for Seismic Qualification of Nuclear Power Plants Structures, Systems, and Components	N289.4-12	Incremental
29	CSA N289.5	Seismic Instrumentation Requirements for Nuclear Power Plants and Nuclear Facilities	N289.5-12	Incremental

#	Document Number	Document Title	Modern Version for PSR2	Type of Review
30	CSA N290.0	General Requirements for Safety Systems of Nuclear Power Plants	N290.0-11	Incremental
31	CSA N290.1	Requirements for the Shutdown Systems of Nuclear Power Plants	N290.1-13	Incremental
32	CSA N290.2	General Requirements for Emergency Core Cooling Systems for Nuclear Power Plants	N290.2-11	Incremental
33	CSA N290.3	Requirements for Containment System of Nuclear Power Plants	N290.3-11	Incremental
34	CSA N290.4	Requirements for Reactor Control Systems of Nuclear Power Plants	N290.4-11	Incremental
35	CSA N290.5	Requirements for Electrical Power and Instrument Air Systems of CANDU Nuclear Power Plants	N290.5-06	Incremental
36	CSA N290.6	Requirements for Monitoring and Display of Nuclear Power Plant Safety Functions in the Event of an Accident	N290.6-09	Incremental
37	CSA N290.11	Requirements for Reactor Heat Removal Capability During Outage of Nuclear Power Plants	N290.11-13	High Level
38	CSA N290.14	Qualification of Pre-Developed Software for Use in Safety-related Instrumentation and Control Applications in Nuclear Power Plants	N290.14-15	High Level
39	CSA N291	Requirements for Safety-related Structures for CANDU Nuclear Power Plants	N291-15	Incremental
40	CSA N285.6 Series	Material Standards for Reactor Components for CANDU Nuclear Power Plants	N285.6 Series-12	Incremental
41	ASME B31.1	Power Piping	B31.1-2014	Incremental
42	ASME BVPC	Boiler and Pressure Vessel Code	BPVC 2015	Incremental
43	CSA B51	Boiler, Pressure Vessel, and Pressure Piping Code	B51-14	Incremental
44	CSA N285.8	Technical Requirements for In-Service Evaluation of Zirconium Alloy Pressure Tubes in CANDU Reactors	N285.8-15	Incremental
45	CNSC G-323	Ensuring Presence of Sufficiently Qualified Staff at Class I Nuclear Facilities-Minimum Shift Complement	2007	Incremental
46	CNSC G-278	Human Factors Verification and Validation Plans	2003	Incremental

#	Document Number	Document Title	Modern Version for PSR2	Type of Review
47	CNSC G-276	Human Factors Engineering Program Plans	2003	Incremental
48	CNSC G-129	Keeping Radiation Exposures and Doses “As Low As Reasonably Achievable (ALARA)”	2004	Incremental
49	CNSC G-228	Developing and Using Action Levels	2001	Incremental
50	S.C.1997, C.9	Nuclear Safety and Control Act (NSCA) and its associated Regulations	Amended in February 2015	Incremental
51	SOR/2000-202	The General Nuclear Safety and Control Regulations	Amended in June 2015	Incremental
52	SOR/2000-203	The Radiation Protection Regulations	Amended in June 2015	Incremental
53	CSA N1600	General Requirements for Nuclear Emergency Management Programs	N1600-14	High Level
54	CSA N288.6	Environment Risk Assessments at Class I Nuclear Facilities and Uranium Mines and Mills	N288.6-12	High Level
55	CSA N288.5	Effluent Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills	N288.5-11	High Level
56	NFPA 20	Standard for the Installation of Stationary Pumps for Fire Protection	NFPA-20 (2016)	Incremental
57	NFPA 24	Standard for the Installation of Private Fire Service Mains and Their Appurtenances	NFPA-24 (2016)	Incremental
58	CNSC REGDOC 2.5.2	Design of Reactor Facilities: Nuclear Power Plants	2014	Incremental
59	CNSC REGDOC 2.2.2	Personnel Training	2014	Incremental
60	CNSC REGDOC 2.2.3	Personnel Certification: Radiation Safety Officers	2014	High Level
61	CNSC REGDOC 2.3.2	Accident Management, Version 2	2015	Incremental
62	CNSC REGDOC 2.3.3	Periodic Safety Reviews	2015	High Level
63	CSA N286.7.1	Guideline for the Application of N286.7-99, Quality Assurance of Analytical,	N286.7.1-09	High Level

#	Document Number	Document Title	Modern Version for PSR2	Type of Review
		Scientific, and Design Computer Programs for Nuclear Power Plants		
64	CSA N290.12	Human Factors in Design for Nuclear Power Plants	N290.12-14	High Level
65	CNSC G-144	Trip Parameter Acceptance Criteria for the Safety Analysis of CANDU Nuclear Power Plants	2006	Incremental
66	CNSC G-149	Computer Programs Used in Design and Safety Analyses of Nuclear Power Plants and Research Reactors	2000	Incremental
67	CNSC R-77	Overpressure Protection Requirements for Primary Heat Transport Systems in CANDU Power Reactors Fitted with Two Shutdown Systems	1987	Incremental
68	CSA N288.2	Guidelines for Calculating Radiological Consequences to the Public from a Release of Airborne Radioactive Material for Nuclear Reactor Accidents	N288.2-14	Incremental
69	CSA N288.3.4	Performance Testing of Nuclear Air-Cleaning Systems at Nuclear Facilities	N288.3.4-13	High Level
70	CSA N290.7	Cyber-Security for Nuclear Power Plants and Small Reactor Facilities	N290.7-14	High Level
71	-	National Building Code of Canada	NBC 2010	Incremental
72	-	National Fire Code of Canada	NFC 2010	Incremental
73	CSA N288.7	Groundwater Protection Programs at Class I Nuclear Facilities and Uranium Mines and Mills	N288.7-15	High Level
74	CSA N290.8	Technical Specification Requirements for Nuclear Power Plant Components	N290.8-15	High Level

* Superseding documents to those already in PROL 48.02/2018.

Appendix E: PSR2 Issue Prioritization – Deterministic Considerations

E.1.0 TABLE E1 – DEFENCE IN DEPTH

Safety Significance Level	Safety Significance Criteria						
	Impact on Protection Against Process Failures		**Impact on Protection Against Design Basis Accidents**		**Impact on Initiating Events / Challenges to Safety Systems & Personnel**		**Impact on Operational Performance & Safety Culture**
1	A barrier or safety function is seriously degraded by the issue.	**or**	Primary safety function is inadequate; one or more levels of protection are lost so that the primary safety function capability is disabled for certain design basis (DB) accident sequences.	**or**	The issue causes a new initiating event or an increase of the frequency of certain initiating events and challenges to safety systems and personnel, leading to a major impact on risk such that immediate corrective measures are necessary to reduce the risk.	**or**	The level of operational performance and safety culture is unacceptable.
2	A barrier or a safety function which protects against anticipated serious process failures is degraded by the issue.	**or**	Primary safety function is adequate; one or more levels of protection are significantly affected by the issue so that the primary safety function capability to protect the barrier(s) is questionable for certain DB accident sequences.	**or**	The issue causes a new initiating event or an increase of the frequency of certain initiating events and challenges to safety systems and personnel, leading to a significant impact on risk such that interim corrective measures are usually necessary in the short term.	**or**	The level of operational performance and safety culture is inadequate.
3	A (safety) function is affected by the issue but the effect does not impair the capability of safety provisions to terminate an anticipated serious process failure.	**or**	Safety function affected by the issue is robust; the issue does not affect the safety function capability for more than one level of protection so that the capability to protect the safety barrier(s) is not impaired for the majority of DB accident sequences.	**or**	The issue causes a new initiating event or an increase of the frequency of certain initiating events and challenges to safety systems and personnel leading to a small impact on risk such that interim corrective measures may be considered and implemented within a specified time schedule if shown to be reasonably practicable.	**or**	The level of operational performance and safety culture warrants improvements.
4	No (safety) function is affected by the issue.	**and**	The issue does not impair capability to protect the safety barriers for any DB accident sequences.	**and**	The issue does not cause new initiating events and has no impact on frequency of known initiating events.	**and**	The issue has no impact on the level of operational performance and safety culture.

E.2.0 TABLE E2 – SAFETY SIGNIFICANCE LEVELS (IN PARTICULAR FOR GLOBAL ISSUES WITHOUT A DIRECT NUCLEAR SAFETY IMPACT)

Safety Significance Level	Safety Significance Criteria		
1	Issue or condition causes a major reduction in margin of safety to public or to station personnel.	**and/ or**	Issue or condition has major impact on environment or production or other business deliverables.
2	Issue or condition causes some reduction in margin of safety to public or to station personnel.	**and/ or**	Issue or condition has some impact on environment or production or other business deliverables.
3	Issue or condition is not significant by itself but, has potential to be more significant or may be precursor to a more significant issue or condition.		
4	Issue or condition adverse to quality that may help identify areas that need more attention when it is later reviewed along with other issues / conditions as part of the PSR2 Global Assessment.	**and/ or**	Issue or condition is not significant by itself but may need more attention when it is later reviewed along with other issues / conditions as part of the PSR2 Global Assessment.

Appendix F: PSR2 Issue Prioritization – Probabilistic Considerations

F.1.0 TABLE F1 – REACTOR SAFETY – CORE DAMAGE FREQUENCY

Estimated reduction in core damage frequency resulting from resolution of issue or condition:	Safety Significance Level
> 10^{-5} / reactor-year	1
10^{-6} to 10^{-5} / reactor-year	2
10^{-7} to 10^{-6} / reactor-year	3
< 10^{-7} / reactor-year	4

F.2.0 TABLE F2 – REACTOR SAFETY – DEFENCE IN DEPTH

	Initiating Event Frequency			
Event / Sequence / Condition:	> 10^{-2}/y	> 10^{-3}/y	> 10^{-4}/y	> 10^{-5}/y
Could challenge effectiveness of a system or function whose failure could lead to breach of multiple safety barriers	1	1	2	3
Could challenge a derived acceptance criterion or effectiveness of a system whose failure could lead to breach of a safety barrier	1	2	3	4
Could lead to partial loss of safety margin on a primary parameter or reduction in reliability of a SIS system	2	3	4	4
Could lead to partial loss of safety on a secondary parameter or reduction in reliability of back-up systems	3	4	4	4

Note – Numbers 1 through 4 are Safety Significance levels.

F.3.0 TABLE F3 – PUBLIC RADIATION SAFETY

Event / Sequence with potential for ***change*** in public dose: Indiv. (Δ mSv) **or** Pop. (Δ Sv)	Initiating Event Frequency > 10^{-2}/y	> 10^{-3}/y	> 10^{-4}/y	> 10^{-5}/y
> 100 **or** > 1000	1	1	1	2 1 [if > 250 mSv indiv.]
> 10 **or** > 100	1	1	2	3
> 1 **or** > 0	1	2	3	4
> 0.1 **or** > 1	2	3	4	4

Note – Numbers 1 through 4 are Safety Significance levels.

F.4.0 TABLE F4 – PLANT OPERABILITY

Condition which may require or lead to:	Probability ~ 1	~0.1	~0.01	< 0.001
Extended period of plant shutdown or power de-rating	1	2	3	4
Outage to correct or change to Operating Policies and Principles (OP&P) limit leading to significant increase in complexity of plant operation	2	3	4	4
Change to OP&P limit without major impact on plant operation	3	4	4	4
Some loss of operating margin	4	4	4	4

Note – Numbers 1 through 4 are Safety Significance levels.

F.5.0 TABLE F5 – OCCUPATIONAL RADIATION SAFETY

Condition likely to cause actual or potential exposure:			Probability			
Indiv. (mSv)	**or**	**Incremental Collective (mSv/y)**	**~ 1**	**~0.1**	**~0.01**	**< 0.001**
> 20	**or**	> 1000	1	2	3	4
> 2	**or**	> 100	2	3	4	4
		> 10	3	4	4	4
		> 1	4	4	4	4

Note – Numbers 1 through 4 are Safety Significance levels.

F.6.0 TABLE F6 – EMERGENCY PREPAREDNESS

Condition which could result in:	Probability			
	~ 1	**~0.1**	**~0.01**	**< 0.001**
Spurious declaration of a General Emergency	1	2	3	4
Release sufficient to require emergency response (e.g. liquid emission)	2	3	4	4
Major change to Emergency Preparedness procedures	3	4	4	4
Significant change to Emergency Preparedness procedures	4	4	4	4

Note – Numbers 1 through 4 are Safety Significance levels.

F.7.0 TABLE F7 – ENVIRONMENT

Condition which could result in:	Probability			
	~ 1	~0.1	~0.01	< 0.001
Widespread contamination of soil / groundwater requiring remedial action	1	2	3	4
Contamination of lake water due to liquid release in excess of emission limits (chemical or radiological)	2	3	4	4
Localized contamination of soil / groundwater requiring remedial action	3	4	4	4
Failure or bypass of effluent monitoring	4	4	4	4

Note – Numbers 1 through 4 are Safety Significance levels.

F.8.0 TABLE F8 – LIKELIHOOD ASSESSMENT

Likelihood	Probability
Consequence or outcome expected to occur	~1.0
Consequence or outcome not expected to occur but possible	~0.1
Consequence or outcome highly unlikely	~0.01
Consequence or outcome not predicted by analysis but not impossible	< 0.001

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ONTARIO POWER GENERATION	
ACCEPTED	✓
ACCEPTED AS NOTED	
REVISE AND RESUBMIT	
Signature	**Date** 03 MAR 2017
Name: Mike Ruffolo, Manager	
Dept: Pickering Engineering - Aging Management & Strategic Initiatives	
OPG Proprietary	
Doc No.: P-REP-03680-00008	**Rev: 000**
This acceptance does not relieve the contractor from responsibility for errors or omissions or from any obligations or liability under this contract.	

amec foster wheeler

Pickering NGS PSR2 Safety Factor 1 Report: Plant Design

PS112/RP/007 R01

March 3, 2017

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Revision Summary – For Amec Foster Wheeler Report PS112/RP/007

Rev	Date	Author	Comments
R00	August 3, 2016	R. Jayasundera	Initial issue for OPG review and comment.
R01	March 3, 2017	R. Jayasundera, A. Johnstone, R. Ross	Updated report addressing OPG comments on R00 Report.

EXECUTIVE SUMMARY

Ontario Power Generation (OPG) is performing a Periodic Safety Review (PSR) to support the possibility of continued operation of Pickering Nuclear Generating Station (NGS) beyond 2020. The PSR (referred to as "PSR2") is a subsequent PSR building on the review basis of earlier OPG Integrated Safety Reviews and other associated assessments. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

Part of PSR2 involves the preparation of Safety Factor reports for each of fifteen major topic areas. Safety Factor reports consist of:

- Assessing compliance against "Review Tasks" identified in the PSR2 Basis Document [1]. These Review Tasks are derived from Canadian Nuclear Safety Commission (CNSC) REGDOC-2.3.3, "Periodic Safety Reviews" [2] and International Atomic Energy Agency (IAEA) SSG-25, "Periodic Safety Review for Nuclear Power Plants" [3];

- Documenting assessments against applicable modern Laws, Regulations, Codes and Standards (L/R/C/Ss) as defined in Reference [1]; and

- Reviewing the effectiveness of applicable OPG Nuclear Programs that support the above assessments, through review of audit and self-assessment results.

The PSR2 review of Safety Factor 1, *Plant Design* is presented in this report. OPG Governance, Programs, Policies, Procedures, Instructions and Guidelines related to Safety Factor 1 were reviewed for the eight PSR2 Review Tasks specified in Section 4.1 of this report. L/R/C/S and OPG Nuclear Program effectiveness reviews for Safety Factor 1 were prepared per Sections 4.2 and 4.3, respectively. Per Section 4.4, the PSR2 assessment includes a review of previously identified PSR1 gaps related to Safety Factor 1 (to ascertain the implications of extending Pickering NGS operation beyond 2020), as well as a review of the R04 Pickering Licence Conditions Handbook [4] for any impacts of Pickering NGS operation beyond 2020 on: a) OPG commitments previously made to the CNSC, b) open CNSC action items, and c) exemptions granted by the CNSC (all related to Safety Factor 1).

The results of the review of Safety Factor 1 are discussed in Section 5.0 of this report. The review has confirmed, by assessment against the current licensing basis and applicable standards, requirements and practices, that the design of Pickering NGS and its documentation is adequate. As discussed in Section 5.0, the review identified 36 gaps that will need to be addressed further as part of the PSR2 Global Assessment process.

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1.0 INTRODUCTION

Ontario Power Generation (OPG) is performing a Periodic Safety Review (PSR) to support the possibility of continued operation of Pickering Nuclear Generating Station (NGS) beyond 2020.[1] A comprehensive Integrated Safety Review (ISR) was completed for Pickering Units 5 through 8 in 2009 in support of refurbishment and continued operation. Pickering Units 1,4 integrated safety assessments were also performed for Pickering A Return to Service (PARTS) in support of approval to restart Units 1 and 4. In addition to these Pickering-specific studies, the 2013 Darlington ISR performed extensive code and standard reviews that were updated in relation to the versions that were assessed in the 2009 Pickering B ISR.[2] These previous ISRs are considered to constitute the first PSR completed for Pickering (referred to as "PSR1"). The current PSR (referred to as "PSR2") is a subsequent PSR building on the basis of earlier OPG integrated safety assessments through review of the various studies, assessments and licence renewals performed since PSR1. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

PSR2 will support and complement the licence renewal application for Pickering NGS going forward. Fifteen Safety Factors will be assessed as part of the PSR. The purpose of Safety Factor reviews is to confirm that the design, condition and operation of Pickering Units 1,4 and 5-8 (as well as common systems) will support continued safe operation for the period of PSR2 by:

- Assessing compliance against "Review Tasks" identified in the PSR2 Basis Document [1], which were derived from Canadian Nuclear Safety Commission (CNSC) REGDOC-2.3.3 [2] and International Atomic Energy Agency (IAEA) SSG-25 [3];

- Documenting assessments against applicable modern Laws, Regulations, Codes and Standards (L/R/C/Ss) (as defined in Reference [1]); and

- Reviewing the effectiveness of applicable OPG Nuclear Programs that support the above assessments, through review of audit and self-assessment results.

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[1] Currently, Pickering Units 5-8 are approved to operate to 247,000 Effective Full Power Hours. This operation limit is expected to be reached on some units in 2020. For the purposes of PSR2, OPG assumes operation of Pickering NGS for up to eight additional years, from 2020 until 2028. OPG will make a decision regarding the permanent shut down dates for the six reactors following the performance of a technical evaluation that will include PSR2, and will communicate it to the CNSC as required by the current Power Reactor Operating Licence (PROL).

[2] Much of the compliance assessment and evaluation of Safety Factor health for the Darlington ISR is based on programs and practices that apply across OPG's Nuclear operations. As a result, where Pickering is confirmed to follow the same Nuclear programs and practices as were assessed for Darlington, the Darlington ISR programmatic conclusions are applicable to Pickering. As discussed in Section 1.0, an effectiveness review (effectiveness at Pickering NGS) of OPG programs used to demonstrate compliance with the PSR2 Assessment Basis will be conducted using recent audit and self-assessment results.

As outlined in IAEA SSG-25 [3], the objective of the review of Plant Design Safety Factor 1 is to: "determine the adequacy of the design of the nuclear power plant and its documentation by assessment against the current licensing basis and national and international standards, requirements and practices". REGDOC-2.3.3 [2] requires that: "The licensee shall conduct a PSR in accordance with this regulatory document for the period until the next PSR or, if applicable, until the end of commercial operation of the plant."

This report documents the results of the review of Safety Factor 1 for Pickering PSR2. The report is based on the OPG Governance, Programs, data, and material available up to January 15, 2016 which is the freeze date for PSR2.

2.0 SCOPE OF REVIEW

2.1 Review Task Assessments

The Pickering PSR2 Safety Factor 1 Review Tasks are defined in Reference [1]. Details of the derivation of these Review Tasks from CNSC REGDOC-2.3.3 [2] and IAEA SSG-25 [3] are shown in Reference [5]. The Safety Factor 1 Review Tasks are:

1) Confirm that a detailed description of the plant design, documenting the Design Basis, supported by layout, systems and equipment drawings exists.

2) Assess the adequacy of design documentation.

3) Identify the Structures, Systems and Components (SSCs) important to safety (Appendix B of the PSR2 Basis Document).

4) Review the application of defence in depth. This includes an examination of:

   - The degree of independence of the levels of defence in depth;

   - The adequacy of delivery of preventive and mitigatory safety functions;

   - Redundancy, separation and diversity of SSCs important to safety;

   - Defence in depth in the design of structures (for example, review of integrity of fuel, cooling circuit and containment building[3]).

5) Confirm that the human–machine interface is considered in the design of the control room and other workstations, that analysis of human information requirements and task workload is performed, and that there is linkage to the Probabilistic Safety Assessment, Deterministic Safety Analyses and Hazard Analysis. This review should include a discussion of how guidance such as U.S. NRC NUREG-0700 Revision 2, "Human-System Interface Design Review Guidelines" and NUREG-0711 Revision 2, "Human Factors Engineering Program Review Model" identified in CNSC REGDOC-2.5.2 are relevant to the design of Pickering U1,4 and U5-8. (Note: In PSR1, a similar Review Task was addressed in the Human Factors Safety Factor. As it is the only human

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[3] Note, an editorial error (duplicated text) was made in the Review Task 4 wording documented in the Pickering NGS PSR2 Basis document [1]. The Review Task 4 wording presented in this Safety Factor report has been modified to correct this issue. (Note: The Review Task wording continues to align with IAEA SSG-25 [3] guidance).

factors activity that deals with plant design it is being assessed as a PSR2 Plant Design Review Task.)

6) Assess the adequacy of the arrangements for providing radiological protection.

7) Where the plant has undergone a significant number of modifications over its lifetime or in the period since PSR1, examine the cumulative effects of all modifications on the design.

8) Confirm that the plant SSCs are compliant with the design specifications and consistent with the design documentation.

The methodology for the reviews is discussed in Section 3.1. Review Task findings are summarized in Section 4.1 of this report.

2.2 L/R/C/S Reviews

The applicable Laws, Regulations, Codes and Standards relevant to the Plant Design Safety Factor are identified in Reference [1] and are listed in Table 1 below. Table 1 also identifies the modern version and date of each L/R/C/S to be considered, the Safety Factor(s) to which each document is applicable, and the type of review that will be completed in PSR2.

All of the Safety Factor 1 L/R/C/S reviews are high level or incremental in nature. The definitions of High Level Review and Incremental Review are as follows:

- High Level: New L/R/C/Ss not referenced in Pickering PROL 48.02/2018 but which are in the PSR2 Assessment Basis will be subject to a high level review. In a high level review, the degree of conformance with clauses or groups of clauses in the L/R/C/S is demonstrated by supporting evidence stating whether the intent of the requirements stipulated in the requirement document is met; and

- Incremental Review: For L/R/C/Ss that have been reviewed in PSR1 but have had revisions since the last review, a topical review will be performed of the changes.

The methodology for the reviews is discussed in Section 3.2. A detailed assessment for each L/R/C/S is provided in References [6] [7], and [8]. Associated findings are summarized in Section 4.2 of this report.

Table 1: L/R/C/Ss Reviewed for Plant Design Safety Factor 1

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
L/R/C/Ss Referenced in Pickering NGS PROL 48.02/2018						
1	CSA N286.7	Quality Assurance Of Analytical, Scientific And Design Computer Programs	N286.7-16	1, 5, 6, 7, 10	Incremental	N286.7 addressed as part of Pickering B and Darlington ISRs.
2	CSA N285.0	General Requirements For Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants	N285.0-12	1	Incremental	N285.0 addressed as part of Pickering B and Darlington ISRs and PARTS.
3	CSA N285.4	Periodic Inspection of CANDU Nuclear Power Plant Components	N285.4-14	1, 2, 4	Incremental	N285.4 addressed as part of Pickering B and Darlington ISRs.
4	CSA N285.5	Periodic Inspection of CANDU Nuclear Power Plant Containment Components	N285.5-13	1, 2, 3, 4	Incremental	N285.5 addressed as part of Pickering B and Darlington ISRs.
5	CSA N293	Fire Protection for Nuclear Power Plants	N293-12	1, 7, 13	Incremental	N293 addressed as part of Pickering B and Darlington ISRs and PARTS.
Additional L/R/C/Ss						
6	CSA N287.1	General Requirements for Concrete Containment Structures for Nuclear Power Plants	N287.1-14	1	Incremental	N287.1 addressed as part of Pickering B and Darlington ISRs and PARTS.
7	CSA N287.2	Material Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants	N287.2-08	1, 2, 3, 4	Incremental	N287.2 addressed as part of Pickering B and Darlington ISRs and PARTS.
8	CSA N287.3	Design Requirements for Concrete Containment Structures for Nuclear Power Plants	N287.3-14	1	Incremental	N287.3 addressed as part of Pickering B and Darlington ISRs and PARTS.
9	CSA N287.5	Examination and Testing Requirements for Concrete Containment Structures for Nuclear Power Plants	N287.5-11	1, 2	Incremental	N287.5 addressed as part of Darlington ISR.
10	CSA N289.1	General Requirements for Seismic Design and Qualification of CANDU Nuclear Power Plants	N289.1-08	1, 3	Incremental	N289.1 addressed as part of Pickering B and Darlington ISRs.

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
11	CSA N289.2	Ground Motion Determination for Seismic Qualification of Nuclear Power Plants	N289.2-10	1, 3	Incremental	N289.2 addressed as part of Pickering B and Darlington ISRs.
12	CSA N289.3	Design Procedures for Seismic Qualification of Nuclear Power Plants	N289.3-10	1, 3	Incremental	N289.3 addressed as part of Pickering B and Darlington ISRs.
13	CSA N289.4	Testing Procedures for Seismic Qualification of Nuclear Power Plants Structures, Systems, and Components	N289.4-12	1, 3	Incremental	N289.4 addressed as part of Pickering B and Darlington ISRs.
14	CSA N289.5	Seismic Instrumentation Requirements for Nuclear Power Plants and Nuclear Facilities	N289.5-12	1, 3	Incremental	N289.5 addressed as part of Pickering B and Darlington ISRs.
15	CSA N290.0	General Requirements for Safety Systems of Nuclear Power Plants	N290.0-11	1	Incremental	N290.0 addressed as part of Darlington ISR.
16	CSA N290.1	Requirements for the Shutdown Systems of Nuclear Power Plants	N290.1-13	1	Incremental	N290.1 addressed as part of Pickering B and Darlington ISRs and PARTS.
17	CSA N290.2	Requirements for Emergency Core Cooling Systems of Nuclear Power Plants	N290.2-11	1	Incremental	N290.2 addressed as part of Darlington ISR. CNSC R-9 (precursor to N290.2) addressed as part of Pickering B and Darlington ISRs and PARTS.
18	CSA N290.3	Requirements for the Containment System of Nuclear Power Plants	N290.3-11	1	Incremental	N290.3 addressed as part of Darlington ISR. CNSC R-7 (precursor to N290.3) addressed as part of Pickering B and Darlington ISRs and PARTS.
19	CSA N290.4	Requirements for Reactor Control Systems of Nuclear Power Plants	N290.4-11	1	Incremental	N290.4 addressed as part of Pickering B and Darlington ISRs and PARTS.
20	CSA N290.5	Requirements for Electrical Power and Instrument Air Systems of CANDU Nuclear Power Plants	N290.5-06	1	Incremental	N290.5 addressed as part of Pickering B and Darlington ISRs and PARTS.

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
21	CSA N290.6	Requirements for Monitoring and Display of Nuclear Power Plant Safety Functions in the Event of an Accident	N290.6-09	1	Incremental	N290.6 addressed as part of Pickering B and Darlington ISRs and PARTS.
22	CSA N290.11	Requirements for Reactor Heat Removal Capability During Outage of Nuclear Power Plants	N290.11-13	1	High Level	N290.11 not addressed as part of Pickering B or Darlington ISRs.
23	CSA N290.14	Qualification of Digital Hardware and Software for Use in Instrumentation and Control Applications for Nuclear Power Plants	N290.14-15	1	High Level	N290.14 not addressed as part of Pickering B or Darlington ISRs.
24	CSA N291	Requirements for Safety-related Structures for Nuclear Power Plants	N291-15	1, 2, 4	Incremental	N291 addressed as part of Darlington ISR.
25	CSA N285.6 Series-12	Material Standards for Reactor Components for CANDU Nuclear Power Plants	N285.6 Series-12	1	Incremental	N285.6 addressed as part of Pickering B and Darlington ISRs and PARTS.
26	ASME B31.1	Power Piping	B31.1-14	1	Incremental	B31.1 addressed as part of Darlington ISR and PARTS.
27	ASME BPVC	Boiler and Pressure Vessel Code	BPVC 2015	1	Incremental	BPVC addressed as part of Pickering B and Darlington ISRs and PARTS.
28	CSA B51	Boiler, Pressure Vessel, and Pressure Piping Code	B51-14	1	Incremental	B51 addressed as part of Pickering B and Darlington ISRs and PARTS.
29	CNSC G-278	Human Factors Verification and Validation Plans	2003	1, 12	Incremental	G-278 addressed as part of Pickering B and Darlington ISRs.
30	CNSC G-276	Human Factors Engineering Program Plans	2003	1, 12	Incremental	G-276 addressed as part of Pickering B and Darlington ISRs.
31	NFPA 20	Standard for the Installation of Stationary Pumps for Fire Protection	NFPA-20 (2016)	1	Incremental	NFPA 20 addressed as part of Darlington ISR.
32	NFPA 24	Standard for the Installation of Private Fire Service Mains and Their Appurtenances	NFPA-24 (2016)	1	Incremental	NFPA 24 addressed as part of Darlington ISR.

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
33	CNSC REGDOC-2.5.2	Design of Reactor Facilities: Nuclear Power Plants	2014	1, 5, 6, 7	Incremental	RD-337 and NS-R-1 (precursors to REGDOC-2.5.2) addressed as part of Darlington ISR. NS-R-1 also addressed as part of Pickering B ISR.
34	CNSC REGDOC-2.3.2	Accident Management, Version 2	2015	1, 5, 6, 7, 8, 10	Incremental	REGDOC-2.3.2 addressed as part of Darlington ISR.
35	CSA N286.7.1	Guideline for the Application of N286.7-99, Quality Assurance of Analytical, Scientific, and Design Computer Programs for Nuclear Power Plants	N286.7.1-09	1, 5, 6, 7, 10	N/A [4]	N286.7.1 not addressed as part of Pickering B or Darlington ISRs.
36	CSA N290.12	Human Factors in Design for Nuclear Power Plants	N290.12-14	1, 12	Incremental [5]	N290.12 not addressed as part of Pickering B or Darlington ISRs. OPG has completed a gap analysis against mandatory requirements of N290.12.
37	CNSC G-149	Computer Programs Used in Design and Safety Analyses of Nuclear Power Plants and Research Reactors	2000	1, 5, 6, 7	Incremental	G-149 addressed as part of Pickering B and Darlington ISRs.
38	CNSC R-77	Overpressure Protection Requirements for Primary Heat Transport Systems in CANDU Power Reactors Fitted with Two Shutdown Systems	1987	1	Incremental	R-77 addressed as part of Pickering B and Darlington ISRs and PARTS.

[4] The N286.7.1 guide has been amalgamated into the new (-16) edition of the N286.7 Standard. The N286.7 CSA Impact Statement states [9]: "The CSA N286.7.1 guide will no longer be maintained after this new edition of N286.7 is issued. Any relevant guidance has been put into the new edition of N286.7." As a result, only the review of N286.7-16 has been prepared for PSR2.

[5] Per CNSC's request in P-CORR-03680-0607223, "Pickering PSR2 – Change to Review Type for CSA N290.12" [10], the Review Type for CSA N290.12-14 was changed from High Level to Incremental.

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
39	CSA N290.7	Cyber-Security for Nuclear Power Plants and Small Reactor Facilities	N290.7-14	1	High Level [6]	N290.7 not addressed as part of Pickering B or Darlington ISRs.
40	NBCC	National Building Code of Canada	NBCC 2010	1	Incremental	NBCC addressed as part of Pickering B and Darlington ISRs and PARTS.
41	NFCC	National Fire Code of Canada	NFCC 2010	1	Incremental	NFCC addressed as part of Pickering B and Darlington ISRs and PARTS.
42	CSA N290.8	Technical Specification Requirements for Nuclear Power Plant Components	N290.8-15	1	High Level	N290.8 not addressed as part of Pickering B or Darlington ISRs.

2.3 OPG Program Effectiveness Reviews

The OPG Nuclear Programs (N-PROGs) reviewed for Safety Factor 1 are listed in Table 2 below.[7] The methodology for the effectiveness reviews is discussed in Section 3.3. The assessment results of each of the N-PROGs in Table 2 are provided in Appendix B, and findings are summarized in Section 4.3.

Table 2: OPG Programs Reviewed for Safety Factor 1

Document Number	Document Title
N-PROG-MP-0007 [11]	Conduct of Engineering
N-PROG-MP-0006 [12]	Software
N-PROG-MP-0005 [13]	Configuration Management
N-PROG-MP-0009 [14]	Design Management

6 As summarized in Section 4.2, a gap analysis for N290.7-14 has been completed by OPG and satisfies the intent of this PSR2 High Level Review. For reasons of security and confidentiality, the findings of this gap analysis will not be discussed in PSR2.

7 The list of Nuclear Programs to be assessed for effectiveness for PSR2 was derived from review of current OPG Governance. Although there may be content in Nuclear Programs that is applicable to multiple Safety Factors, N-PROG reviews are only provided in one Safety Factor report and are not duplicated.

2.4 Additional Reviews

The PSR2 Safety Factor 1 report includes a review of the R04 Pickering Licence Conditions Handbook (LCH) [4] for any impacts of Pickering NGS operation beyond 2020 on the following (all related to Safety Factor 1):

- OPG commitments previously made to the CNSC;
- Open CNSC action items; and
- Exemptions granted by the CNSC.

The PSR2 assessment includes identification and review of previously identified PSR1 gaps related to Safety Factor 1 to ascertain the implications of extending Pickering NGS operation beyond 2020. The methodology for these reviews is described in Section 3.4. Any PSR2 gaps identified as a result of programmatic Darlington PSR1 gaps related to Safety Factor 1 are discussed in Section 4.4 of this report. The review of Pickering PSR1 gaps previously identified in the Pickering Units 5-8 Continued Operations Plan (COP) [15] is provided in a separate PSR2 COP Review Report.

In addition, Fukushima Action Items (FAIs) were reviewed to identify implications of extending operation beyond 2020 (if any). This review is presented in a separate PSR2 FAI Review Report.

Any PSR2 gaps identified as a result of the Safety Factor 1 review which are relevant to other Safety Factors are discussed in Section 4.4 of this report.

3.0 METHODOLOGY

The sub-sections below summarize the methodology used to assess Review Tasks, L/R/C/Ss, and Nuclear Program effectiveness for the Plant Design Safety Factor.

3.1 Review Tasks

As discussed earlier, the Safety Factor Review Tasks are derived from CNSC REGDOC-2.3.3 [2] and IAEA SSG-25 [3], taking into consideration the Review Tasks used in the Pickering B and Darlington ISRs (as derived in [5]).

For each Safety Factor 1 Review Task identified in Section 2.1, a confirmation of the existence of applicable OPG Governance, Programs, Policies and Procedures (as well as Instructions and Guidelines, as applicable) was performed. Compliance against Review Tasks is also assessed by reference to applicable Condition Assessments, safety analyses and operating experience, as required.

The Review Task assessments identify Compliances and Gaps as defined below:

- Compliance: Compliance indicates that either the safety requirement or the intent of the Review Task is met.
- Gap: A Gap indicates that the intent of the Review Task is not met.

3.2 L/R/C/S Reviews

The process to identify the modern L/R/C/Ss that are applicable to the PSR2 Assessment Basis involved first creating a broad list from multiple sources (potential candidate L/R/C/Ss) and then filtering it to identify those that are most significant and that are applicable to the PSR2 scope. The identification and selection criteria are detailed in Reference [1]. The result of the identification and selection process was a set of modern L/R/C/Ss that became part of the "PSR2 Assessment Basis".

PSR2 is focused on the extension of Pickering NGS operations beyond 2020, and will conduct reviews against a baseline of past PSR1 work. As a subsequent PSR, PSR2 focuses on changes in requirements, plant conditions, operating experience and new information. Since PSR2 is an update of previous ISRs, it incorporates reviews of L/R/C/Ss that have occurred as new versions have been issued. Since this assessment is a subsequent PSR, the focus is on identifying differences between what was previously assessed and what is now different within the current Pickering PSR2 Assessment Basis. In general, these differences relate to:

- More recent (new or revised) L/R/C/S versions than what was previously assessed as part of PSR1;[8]

- Safety significant differences between Pickering and Darlington, if the Darlington ISR is the basis for the earlier assessment;

- Implications of extending Pickering NGS operation beyond 2020; and

- Safety significant differences between Pickering Units 1,4 and Units 5-8.

As described in Reference [1], L/R/C/S review types are clause-by-clause, high level or incremental. Most of the L/R/C/Ss in the PSR2 Assessment Basis receive incremental reviews since PSR2 is an update of previous PSR1 assessments and clause-by-clause or high level reviews for the majority of the L/R/C/Ss in the PSR2 Assessment Basis have already been completed. Implementation plans (including gap analyses or code-over-code reviews) also exist for the latest editions of many L/R/C/Ss. As a result, incremental review is also used in circumstances where a L/R/C/S in the PSR2 Assessment Basis was not assessed in previous PSR1 reviews but an implementation plan currently exists for compliance.

The PSR2 incremental reviews in this report include an assessment of the intent of recent changes to the L/R/C/Ss on a topic or subject-matter basis where there is potential to impact nuclear safety. Incremental reviews provide:

- A summary of the purpose of the L/R/C/S;

- Pertinent background information about the current revision of the L/R/C/S that is being considered;

- Identification of which Safety Factor(s) are applicable to the current revision of the L/R/C/S;

- A description of which version(s) of the L/R/C/S were assessed for PSR1 (i.e., Darlington ISR (for programmatic content), Pickering B ISR and PARTS code reviews);

- Identification of whether the current version of the L/R/C/S is an update of a previous version of the L/R/C/S that was assessed in PSR1 (and if so, a description of the major changes in the latest revision is provided as discussed below);

[8] "New" refers to a code or standard that was not previously considered in the context of earlier assessments. "Revised" refers to an updated version of a code or standard that was previously considered in the context of earlier assessments. Where a document has a new number/type, but addresses the same topic from the same organization, it is a "revised", not "new", document (e.g., if a REGDOC replaces a CNSC G or RD document).

- An assessment of the applicability of PSR1 assessment findings (gaps and conclusions), including the implications of extending Pickering NGS operation beyond 2020 if any;

- An assessment of the applicability of assessment findings that address more recent (post-PSR1) editions of the L/R/C/S, including any implementation or transition plans that are already committed to by OPG; and

- Where PSR1 and post-PSR1 assessments are not sufficient to address changes in the latest edition of the L/R/C/S, an assessment of the changes from the previously assessed edition of the L/R/C/S (including identification of any safety significant PSR2 gaps which result).

High Level reviews provide the same information as above, where applicable, in a similar format. However, given that High Level L/R/C/Ss generally have not received past assessment during PSR1, the incremental review content is augmented by a high level, section-by-section assessment of the degree of conformance of Pickering NGS with the L/R/C/S (demonstrating, with supporting evidence, whether the intent of the requirements stipulated in the document are met).

There are currently no L/R/C/S clause-by-clause reviews identified in the PSR2 Assessment Basis.

The Safety Factor 1 L/R/C/S reviews identify Compliances and Gaps as defined below:[9]

- <u>Compliance</u>:

    - Where an Incremental review has been performed, Compliance indicates that the change in the safety requirement, per the topical review, is met.

    - Where a High Level review has been performed, Compliance indicates that the intent of the safety requirement is met.

    - For Clause-by-Clause reviews of modern L/R/C/Ss, Compliance indicates that the safety requirement is met. (Note: No Clause-by-Clause reviews were performed as part of Safety Factor 1.)

[9] Safety Factor assessments for Review Tasks and L/R/C/S reviews make use of: a) OPG Governance, Programs, Policies and Procedures which support the assessment arguments, b) Commitments previously made to the CNSC, open CNSC action items, and exemptions granted by the CNSC (all related to the Safety Factor under review), as identified in the R04 Pickering LCH [4], c) Identification of previously identified Pickering-specific or programmatic PSR1 gaps related to the Safety Factor under review and the status of OPG's improvement plan(s) or other dispositions to address these, and d) Assessments and reviews performed since the PSR1 documents were completed.

- Gap:
    - Where an Incremental review has been performed, a Gap indicates that the change in the safety requirement, per the topical review, is not met.
    - Where a High Level review has been performed, a Gap indicates that the intent of the safety requirement is not met.
    - For Clause-by-Clause reviews of modern L/R/C/Ss, a Gap indicates that the safety requirement is not met. (Note: No Clause-by-Clause reviews were performed as part of Safety Factor 1.)

The reviews assume that use of the word:

- "Shall" is used in an L/R/C/S to express a requirement, i.e., a provision that the licensee is obliged to satisfy in order to comply with the standard;
- "Should" is used to express a recommendation or that which is advised but not required;
- "May" is used to express an option or that which is permissible within the limits of the standard; and
- "Can" is used to express possibility or capability.

3.3 OPG Program Effectiveness Reviews

As discussed earlier, effectiveness reviews (effectiveness at Pickering NGS) of OPG programs used to demonstrate compliance with the PSR2 Assessment Basis were conducted, using recent applicable audit and self-assessment results:

- OPG Nuclear Oversight independent performance-based Program audits (typically performed in 1 to 5 year cycles) and self-assessments. This includes review of associated Station Condition Records (SCRs) and Action Requests to determine the status of any resulting corrective actions; and
- CNSC "Type I" and "Type II" inspections of the effectiveness and performance of OPG programs, where discussed in OPG audits or self-assessments.

There are many audits and self-assessments that are performed to assess the effectiveness of important aspects of each program. A sample of audits and self-assessments has been summarized for each program in order to demonstrate that program effectiveness is being assessed on an ongoing basis. The focus of these reviews was on effectiveness of the programs at Pickering NGS, where specific information is available. Results from these audits and self-assessments will be considered in the Global Assessment process. It is noted that audits and self-assessments are, by their nature, self-critical and are used to drive excellence in performance. As a result, the broad review scope of program audits focuses on

identifying improvement opportunities rather than presenting a balanced picture of program performance.

Program effectiveness is also monitored and addressed through the Fleetview Program Health and Performance Reporting process [16]. This process involves direct oversight by the Chief Nuclear Officer, and includes participation by the Nuclear Executive Committee members. Programs are reviewed, senior oversight is provided, and improvement plans are generated.

The list of Nuclear Programs to be assessed for each Safety Factor was derived from review of current OPG Governance, and has used the most recent version of these documents as of the PSR2 freeze date of January 15, 2016.

3.4 Additional Reviews

A review of the R04 Pickering LCH [4] was performed to determine if there are any impacts associated with operation of the Pickering Units past 2020 on the following (all related to Safety Factor 1):

- Commitments previously made to the CNSC;
- Open CNSC action items; and
- Exemptions granted by the CNSC.

The PSR2 assessment includes identification and review of previously identified Pickering-specific or programmatic PSR1 gaps related to Safety Factor 1 (as identified in the Darlington ISR Integrated Implementation Plan (IIP) [17] and Pickering Units 5-8 Continued Operations Plan [15]) to ascertain the status of OPG's improvement plan(s) or other dispositions to address these and the implications of extending operation beyond 2020 (if any).[10]

Fukushima Action Items were reviewed to identify implications of extending operation beyond 2020 (if any). The methodology for this review is provided in a separate PSR2 FAI Review Report.

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[10] PSR2 includes consideration and confirmation that the findings of PSR1 remain valid, as applicable, for the operation period. This includes assessment of PSR1 conclusions against implications resulting from extended operation. In particular, Pickering PSR1 results are applicable to PSR2 if there was a PSR1 gap that is still open, or if a closed PSR1 gap could be affected by extended operation. If so these gaps are carried forward into PSR2 for consideration in the Global Assessment. (When references to PSR1 are made, the source document is identified and the relevant text from that source document is summarized in the context of PSR2.) With respect to the Darlington ISR, much of the evaluation of Safety Factor health is based on programs and practices that apply across OPG's nuclear operations. As a result, Darlington ISR programmatic conclusions are applicable to Pickering PSR2 where Pickering is confirmed to follow the same nuclear programs and practices that were assessed for Darlington. Darlington PSR1 results are applicable to Pickering PSR2 if there are Darlington PSR1 gaps that are found to be relevant to Pickering PSR2.

Any PSR2 gaps identified as a result of the Safety Factor 1 review which are relevant to other Safety Factors are also discussed.

4.0 REVIEW FINDINGS

4.1 Review Tasks

The sub-sections below provide an assessment of the adequacy of applicable OPG Governance, Programs, Policies and Procedures (as well as Instructions and Guidelines, as applicable) in demonstrating compliance against the Safety Factor 1 Review Tasks.

4.1.1 Review Task #1: Documentation of Plant Design

Confirm that a detailed description of the plant design, documenting the Design Basis, supported by layout, systems and equipment drawings exists.

The plant design basis is documented in an extensive set of documents and drawings defined in N-LIST-01300-10000, "Bounded Document Set" [18]. The Bounded Document Set lists the types of documents that are maintained when modifying the plant or when modifying other Bounded Document Set documents and provides for a consistent set of configuration managed documentation. For example, Appendix A of N-LIST-01300-10000 [18] (Bounded Document Set Listing) lists the set of documentation or data that:

- Represents the physical plant;
- Represents the design (design input or output);
- Ensures the physical plant is operated consistently within the design envelope (including training);
- Establishes acceptability or suitability of detailed design and physical entity; and
- Is controlled to ensure that the physical plant is consistent with the paper plant and its operation and maintenance.

The specific documents describing the plant design basis include:

- History Dockets;
- Design Manuals;
- Design Requirements (including software);
- Design Drawings;
- Software Release Notices;
- Flow Diagrams;

- Operational Flow Sheets (system drawings);
- Layout Drawings (Plant, Mechanical, Electrical);
- Equipment Drawings and Lists;
- Piping Design Specifications;
- Bills of Materials;
- System Classification Lists;
- Electrical Wiring Drawings;
- Technical Specifications
- Technical Basis Documents;
- Operational Safety Requirements; and
- Safety Report.

N-PROG-MP-0009, "Design Management" [14] provides the framework for the establishment, maintenance and compliance with the design basis for Pickering NGS. The Design Management program provides assurance that design and procedure changes are prepared, reviewed, approved, documented and implemented in accordance with approved procedures, applicable regulatory requirements, standards and industry practices (Note: Review Task #2 provides additional details on Design Management governance).

Conclusion:

The conclusion of this Review Task assessment is that a detailed description of the plant design, documenting the Design Basis, supported by layout, systems and equipment drawings exists. The intent of Review Task #1 is met and therefore Pickering NGS is compliant.

4.1.2 Review Task #2: Adequacy of Design Documentation

Assess the adequacy of design documentation.

<u>Configuration Management</u>

Configuration Management (CM) refers to the industry-accepted process to ensure design documentation is prepared and consistent with the plant design basis and matches the physical plant. The Pickering NGS CM Program has been established in accordance with the requirements of N-PROG-MP-0005, "Configuration Management" [13] and the objectives of this program are as follows:

- Assure the physical configuration matches the configuration documents for all states, including normal operation, upset, post-accident and emergency states;
- Ensure configuration information is accurate, consistent and readily accessible;
- Establish clear configuration control scope, responsibilities, authorities, and interfaces among organizations;
- Manage proposed changes effectively by:
    - Confirming physical configuration or configuration information changes conform to the design and licensing basis, by ensuring required regulatory and licensing reviews, approvals and safety evaluations are completed.
    - Reviewing impacts so that related configuration information is maintained consistent with the change.
    - Ensuring changes to the design and licensing basis receive appropriate verification and approvals before the change is made.
    - Ensuring change processes work in accordance and consistently with each other for design, procurement, construction, installation, commissioning, operation and maintenance, including surveillance, training and testing.
- Set requirements for:
    - Identification, control and management of configuration information;
    - Relationships among configuration items;
    - Change control;
    - Communication and training; and
    - Program review, performance monitoring and continuous improvement.

<u>Engineering Change Control and Design Management</u>

N-PROG-MP-0009, "Design Management" [14] specifies requirements for the following:

- Management of prescribed activities appropriate for execution and control of required design, design support and documentation for nuclear facilities;
- Processes for creating or modifying documentation required controlling the design basis and design outputs;

- Minimum set of documentation that identifies and describes the design basis, design output and design process; and
- Procurement Engineering processes ensuring implementation and maintenance of the physical nuclear facilities meet the design basis requirements.

An essential part of Design Management is the maintenance of accurate documentation. OPG-PROG-0001, "Information Management" [19], establishes a set of standards and procedures for the management of information throughout the plant's life-cycle, regardless of media. OPG-PROC-0179, "Nuclear Quality Assurance Records" [20] (an implementing procedure of OPG-PROG-0001 [19]) defines the process to ensure documents related to design are retained, stored, controlled and are traceable and retrievable. Also, Controlled Document changes are reviewed to ensure that the change does not impact the design basis or require a change to an SSC to maintain alignment between the documented and physical configuration. This review is required for engineering documents (normally those documents described in N-LIST-01300-10000, "Bounded Document Set" [18]). Document changes that do not affect the design basis (e.g., addition or revision of information for clarity), are processed in accordance with the governing processes applicable to the document type. Document changes in support of a modification process (i.e., changes to engineering documentation that reflect a change to the design basis or a physical change to SSCs) are listed in the Affected Document List of the specific Engineering Change in Asset Suite and processed in accordance with the procedure governing the specific document type. Hence, assurances are provided that any changes to controlled documents do not impact the design basis and maintain alignment with the physical configuration of the plant.

N-PROG-MP-0009 [14] requires that design changes be initiated, implemented and tracked in accordance with N-PROG-MP-0001, "Engineering Change Control" [21]. The primary objective of the Engineering Change Control (ECC) Program, is to ensure that all modifications to plant SSCs, including software and station engineered tooling, are planned, designed, installed, commissioned, decommissioned, placed into service or removed from service within the safe operating envelope, design basis and plant licensing conditions. It defines a systematic process and methodology for controlling design modifications for plant SSCs to meet the requirements of Canadian Standards Association (CSA) N285.0, "General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants" and CSA N286, "Management System Requirements for Nuclear Power Plants". The interrelationship between ECC and Design Management is shown in Figure 1 (as documented in Figure 1 of N-PROG-MP-0009 [14]).

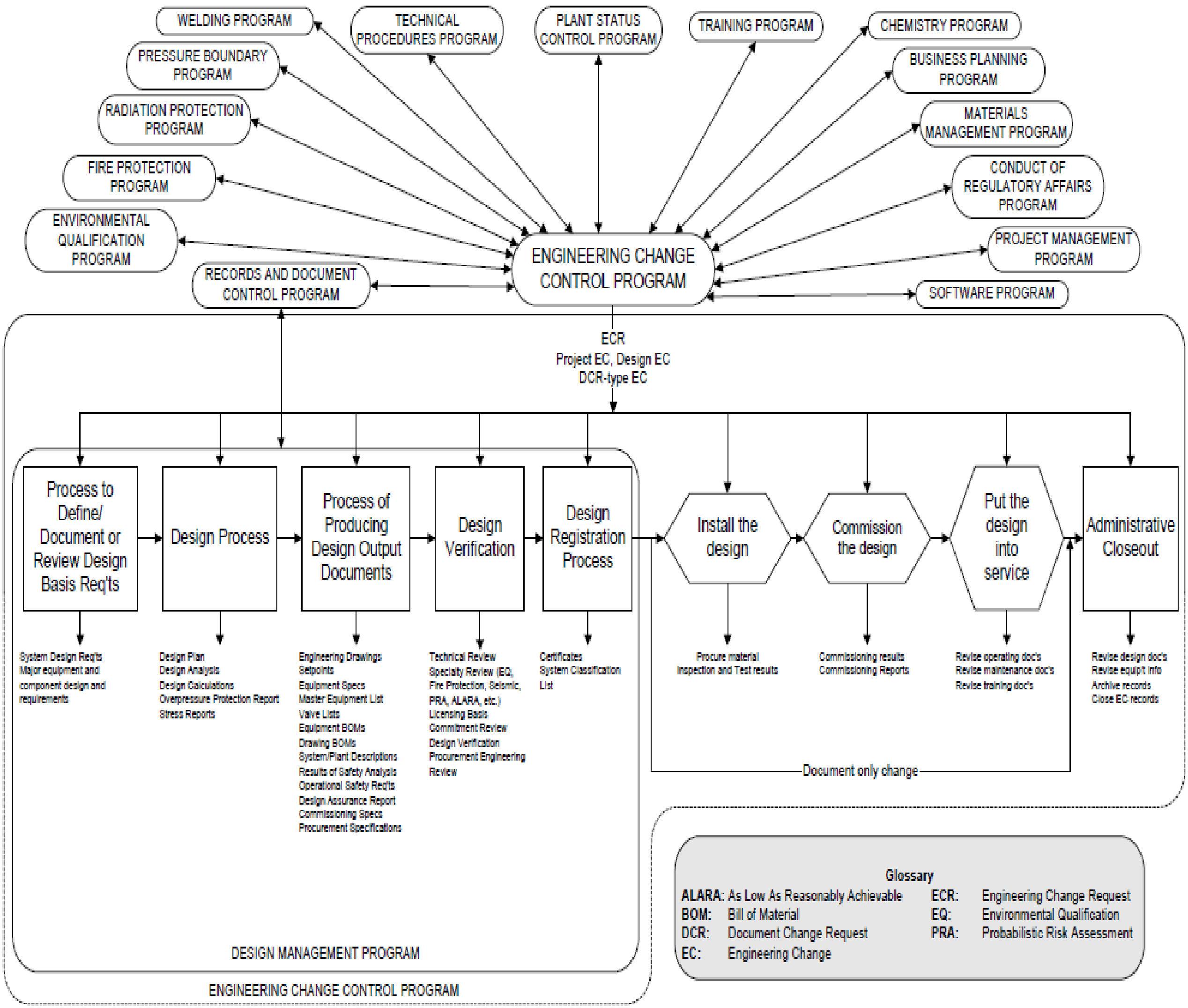

Figure 1: Engineering Change Control – Design Management Interrelationship

N-PROC-MP-0090, "Modification Process" [22] defines the process to be followed for all changes to the design basis, including modifications to, removal of, or abandonment of SSCs, software and engineered tooling designs (N-PROC-MP-0090 [22], and receives its authority from N-PROG-MP-0001, "Engineering Change Control" [21]). N-GUID-01130-10000, "Modifications for Beyond Design Basis Accidents" [23], provides guidance related to the design, modification, procurement, maintenance, testing and operation of SSCs for mitigating Beyond Design Basis Accidents (BDBAs). A primary input to the modification process is defined in N-PROC-MP-0083, "Constructability, Operability, Maintainability, and Safety (COMS)" [24] and N-FORM-10480, "COMS Checklist" [25]. N-PROC-MP-0083 [24] provides direction on:

- The identification of stakeholders from departments involved with or impacted by the modification;

- Determination of which stakeholders comprise the COMS team;
- Identification of user-centred issues and risks related to the COMS of a modification; and
- Addressing the issues and risks appropriately.

N-FORM-10480 [25] is a repository of questions to assist in determining that all appropriate issues have been identified during the design phase. The use of N-PROC-MP-0083 [24] and N-FORM-10480 [25] ensures that stakeholder and subject matter expert input is considered and that risks impacting the safety of the plant and personnel are adequately identified and addressed.

Conclusion:

The conclusion of this Review Task assessment is that design documentation is adequate. The intent of Review Task #2 is met and therefore Pickering NGS is compliant.

4.1.3 Review Task #3: List of SSCs Important to Safety

Identify the SSCs important to safety (Appendix B of the PSR2 Basis Document).

The scope of SSCs within the PSR2 review encompasses the Pickering Safety Related Systems, with a focus on Pickering Systems Important to Safety (SIS) and Safe Operating Envelope (SOE) Systems. OPG defines Safety Related Systems as those systems, and the components and structures thereof, which, by virtue of failure to perform in accordance with the design intent, have the potential to impact on radiological safety of the public or plant personnel from the operation of the nuclear power plant (P-LIST-06937-00001 R00, "Pickering A and B List of Safety Related Systems" [26]).

SIS are defined per N-STD-RA-0033, "Reliability Monitoring and Reporting of Systems Important to Safety" [27] which considers the risk importance of systems and utilizes expert review panels to select these systems. The identification of SIS is consistent with the requirements of CNSC Regulatory Document RD/GD-98, "Reliability Programs for Nuclear Power Plants" [28].

SOE systems are identified per N-STD-MP-0016, "Safe Operating Envelope" [29] which identify systems and their associated critical components and structures, for which operational safety requirements are specified to conform with the Safety Report.

As detailed in the Appendix B of the PSR2 Basis Document [1] and derived from References [30], [31], the SSCs important to safety for Pickering Units 1,4 within the scope of PSR2 are as follows:

- Emergency Coolant Injection System (ECIS);

- Shutdown System A (SDSA);
- Shutdown System Enhancement (SDSE);
- Negative Pressure Containment System (NPCS);
- Powerhouse Emergency Venting System (PEVS);
- Reactor Regulating System (RRS);
- Service Water Systems;
- Moderator System;
- Electrical Power System;
- Emergency Boiler Water Supply;
- Heat Transport System (Pressure and Inventory Control and D_2O Recovery);
- Shutdown Cooling System;
- Boiler Emergency Cooling System;
- Feedwater System;
- Main Steam Supply System;
- Fuel and Reactor Physics;
- Annulus Gas System;
- Fuel Handling System and Irradiated Fuel Bays;
- Critical Safety Parameter Monitoring Instrumentation;
- Shield Cooling System;
- Interstation Transfer Bus;
- Powerhouse Environmental Protection System; and
- Critical Structures.

As detailed in the Appendix B of the PSR2 Basis Document [1] and derived from References [31], [32], the SSCs important to safety for Pickering Units 5-8 within the scope of PSR2 are as follows:

- ECIS;

- Shutdown System 1 (SDS1);
- Shutdown System 2 (SDS2);
- NPCS;
- PEVS;
- RRS;
- Service Water Systems;
- Moderator Systems;
- Group 1 Electrical Power System;
- Emergency Water Supply System;
- Heat Transport System;
- Shutdown Cooling System;
- Boiler Emergency Cooling System;
- Feedwater System;
- Main Steam Supply System;
- Fuel and Reactor Physics;
- Emergency Power Supply;
- Fuel Handling and Irradiated Fuel Bays;
- High Pressure Emergency Coolant Injection Power Supplies;
- Annulus Gas System;
- Critical Safety Parameter Monitoring Instrumentation;
- Shield Cooling System; and
- Critical Structures

Conclusion:

The conclusion of this Review Task assessment is that SSCs important to safety have been identified. The intent of Review Task #3 is met and therefore Pickering NGS is compliant.

4.1.4 Review Task #4: Design for Defence in Depth

> *Review the application of defence in depth. This includes an examination of:*
>
> - *The degree of independence of the levels of defence in depth;*
> - *The adequacy of delivery of preventive and mitigatory safety functions;*
> - *Redundancy, separation and diversity of SSCs important to safety;*
> - *Defence in depth in the design of structures (for example, review of the integrity of fuel, cooling circuit and containment building[11])*

<u>General</u>

As outlined in Part 2 of the Pickering 1,4 and 5-8 Safety Reports [33], [34], defence in depth is represented by a number of barriers between radioactive materials and the general public. The barriers (which include designed structures) in place to prevent radioactivity from escaping to the environment include:

1. The UO_2 fuel pellets, which bind the majority of radioactive fission products within a solid matrix;

2. The fuel sheath, which contains the fission products not retained in the fuel matrix;

3. The heat transport system boundary, which contains any leakage from the fuel sheath;

4. The containment structure, which contains any release from the heat transport system; and

5. The exclusion zone surrounding the facility, which provides for dilution of any release from containment.

The first three barriers prevent radioactive release accidents. So long as they are intact, very little radioactive material will escape into containment. Containment and the exclusion zone come into play to mitigate doses when all of the first three barriers are breached (e.g., following a loss of coolant accident with fuel failures). Based on protecting these barriers, the fundamental principles that guided the design of CANDU reactors in Canada can be categorized as:

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[11] Note, an editorial error (duplicated text) was made in the Review Task 4 wording documented in the Pickering NGS PSR2 Basis document [1]. The Review Task 4 wording presented in this Safety Factor report has been modified to correct this issue. (Note: The Review Task wording continues to align with IAEA SSG-25 [3] guidance).

- Accident Prevention
  - Build high quality and reliability into systems to minimize the stresses on the first three barriers to prevent accidents from occurring; and
  - Anticipate component and system failures and build in defences to protect the first three barriers to prevent such failures from developing into an accident.
- Accident Mitigation
  - Anticipate a large range of accident faults, including very low probability failure of piping systems, and build in defences to mitigate the consequences of such accidents. These defences include the last two barriers to release.

Inherent to this approach is the requirement to postulate a range of process equipment failures that would impair one or more of the barriers, and then to establish that resultant releases of radioactive material will not result in radiation doses above allowable limits. To meet this requirement, a number of safety-related functions (as distinct from process functions associated with routine power production) are provided. These functions are performed principally by the special safety systems:

- The shutdown system (SDSA/SDSE for Pickering Units 1,4 and SDS1/SDS2 for Pickering Units 5-8);
- ECIS; and
- NPCS.

The special safety systems are independent of the process systems, such that a process system impairment will have minimal (if any) impact upon the effective functioning of a special safety system.

For Pickering Units 5-8, redundant equipment and circuits are separated to ensure the safety of the station following a common mode event (e.g., local fires). The station systems (both safety and process) have been divided into two groups (Group 1 and Group 2) and designed and located to provide maximum separation between these two groups. Each group is capable, independently of the other group, of safely shutting down the reactor, cooling the fuel, and providing the operator with indication of system conditions. Pickering Units 1,4 were not designed with Group 1 and Group 2 systems. However, systems have been either qualified or retrofitted to function as required for a given common mode event by ensuring effective separation and diversity. For example, the Inter-Station Transfer Bus and Emergency Boiler Water Supply systems have specifically been installed to mitigate high energy pipe failures in the powerhouse. Steam protected rooms and barriers have been installed to protect critical equipment and staff. For seismic events, systems including power (Class

I/II/III), water and control have been hardened to provide assurance that they will remain operational.

Alignment with the Five Levels of Defence in Depth

Five levels of defence in depth are defined in REGDOC-2.4.1, "Deterministic Safety Analysis" [35]. Based on the information provided above, alignment of Pickering NGS against these five levels can be summarized as follows:

- Level 1 – Prevent deviations from normal operation and prevent failures of SSCs.

  The first level of defence requires a high quality in the design and construction of the plant with barriers to prevent the occurrence of abnormal operating conditions. This is particularly important for the physical barriers surrounding the radioactive material in the fuel. Safe, conservative operation of the plant by qualified staff and a continued focus on preventive maintenance ensures reliable functionality of plant equipment under normal operation and therefore prevents process upsets and failures.

- Level 2 – Detect and intercept deviations from normal operation in order to prevent process upsets from escalating to accident conditions.

  The second level of defence is the provision of barriers to prevent process upsets from progressing to accidents. The Pickering NGS plant design possesses a number of strong features regarding Level 2 Defence in Depth. For example:

  - Automatic reactor control features detect and respond to abnormal conditions before these conditions progress to the point that the next level of barriers are required to act.

  - A large number of safety related and process system tests are completed routinely to detect problems regarding plant equipment.

  - A well-established framework of operating procedures is in place to respond to equipment malfunctions in a timely manner thereby ensuring that the plant stays within its well-defined safe operating envelope.

- Level 3 – Minimize the consequences of accidents.

The third level of defence consists of the barriers to minimize the consequences of accidents should they occur by providing inherent safety features, fail-safe design, additional equipment (including Emergency Mitigating Equipment (EME)), and mitigating procedures. The Pickering Units 1,4 and 5-8 Probabilistic Safety Assessments (PSA) [36], [37], [38], [39], demonstrate that the overall plant design has a Core Damage Frequency and Large Release Frequency within the specified safety limits, indicating robustness in the design, and reliable equipment that is capable of responding effectively to accident scenarios.

- Level 4 – Ensure that radioactive releases caused by severe accidents are kept as low as practicable.

  The fourth level of defence includes those barriers to control severe plant conditions. Significant progress in the Severe Accident Management Guidance program implementation has resulted in Pickering NGS strengthening its capability to respond to low probability Severe Accidents. Implementation of lessons learned from the Fukushima event, and installation of additional hydrogen mitigation equipment for BDBAs has added further capability to this defence in depth level. N-BDB-03600-00002, "OPG Emergency Mitigating Equipment for Beyond Design Basis Accidents: Technical Basis Document" [40], summarizes the analyses that have been done for each step of the Severe Accident progression, and the modifications undertaken by OPG to improve defence in depth.

- Level 5 – Mitigate the radiological consequences of potential releases of radioactive materials that may result from accident conditions.

  The fifth level of defence is associated with the management and mitigation of radiological off-site consequences should an accident occur. In the event of a nuclear plant accident, OPG is prepared with the necessary staff, equipment and procedures to support the Province in managing and mitigating off-site radiological consequences as required by the Provincial Nuclear Emergency Response Plan [41]. Also, the Consolidated Nuclear Emergency Plan (N-PROG-RA-0001 [42]) documents the concepts, roles and resources required by OPG Nuclear to implement and maintain its emergency response capability to protect the public, employees and the environment in the event of a nuclear emergency. It provides a framework for interaction with external authorities and defines OPG commitments under the Provincial Nuclear Emergency Response Plan. As a result of the events at Fukushima, OPG has conducted a review of its emergency response program in order to address lessons learned and identify further areas for enhancement. Note: Safety Factor 13 Report, Emergency Planning, provides details of Emergency Planning for Pickering NGS.

Conclusion:

The conclusion of this Review Task assessment is that defence in depth has been applied for the Pickering NGS design. The intent of Review Task #4 is met and therefore Pickering NGS is compliant.

4.1.5 Review Task #5: Design for Human-Machine Interfaces

> *Confirm that the human–machine interface is considered in the design of the control room and other workstations, that analysis of human information requirements and task workload is performed, and that there is linkage to the Probabilistic Safety Assessment, Deterministic Safety Analyses and Hazard Analysis. This review should include a discussion of how guidance such as U.S. NRC NUREG-0700 Revision 2, "Human-System Interface Design Review Guidelines" and NUREG-0711 Revision 2, "Human Factors Engineering Program Review Model" identified in CNSC REGDOC-2.5.2 are relevant to the design of Pickering U1,4 and U5-8. (Note: In PSR1, a similar Review Task was addressed in the Human Factors Safety Factor. As it is the only human factors activity that deals with plant design it is being assessed as a PSR2 Plant Design Review Task.)*

Human Factors Engineering (HFE) evaluates the role of humans in human-machine systems and how systems can be designed to work well with people, particularly in terms of safety and efficiency. Pickering NGS was originally designed to the standards of the day and designers relied on best design practices in addition to incorporation of operations and maintenance experience (e.g., for Pickering Units 1-4, experience was incorporated from Douglas Point and Nuclear Power Demonstration, while Pickering Units 5-8 incorporated experience from Pickering Units 1-4 and Bruce NGS A). Input obtained from OPG HFE subject matter experts is that Ontario Hydro design practice in this area recognized the need for focus on the operator interfaces in the control centre(s), and recognition of the integration of the control centre and the related human systems interfaces. Project records from the time Pickering 5-8 was being built show that HFE principles were being considered and applied in the design. Operation of the Pickering NGS design over the years has demonstrated that the plant layout and facilities provide a safe working environment. Operating experience and improvements have been incorporated into the processes and design to improve the human-machine interfaces in many areas (e.g., Control Room annunciation upgrades as a result of changes to computer hardware and operator interface [43]). Additionally, training and qualification processes (and certification processes for control room staff) for Operations positions ensure that the staff are competent to carry out functions assigned to them. The simulator is used extensively for initial training and qualification, as well as for refresher/requalification training. For example, per N-INS-08920-10002, "Simulator-Based Initial Certification Examinations for Shift Personnel" [44], Simulator Exercise Guides are used as part of training for certified staff.

Since 2000, HFE has been explicitly considered for all design changes at Pickering NGS, resulting in continuous improvements to the human-machine interfaces throughout the plant. In terms of design changes to the control room and other work stations, the following governance exists[12]:

Pickering Units 1,4

- 44RS-06700-HFP-004, "Human Factors Design Guideline: Maintenance, Inspection, and Testing [45]: This design guide addresses the human factors aspects of equipment and components that field personnel maintain, test or inspect.

- 44RS-66000-HFP-001, "Human Factors Minor Change Design Guideline: Pickering 'A' Main Control Room Unit Panels and Field Control Panels" [46]: This design guide provides Human Factors guidance for detailed design of human-system interface changes to panel-mounted operator instrumentation on main control room unit panels and field control panels.

- 44RS-66000-HFP-002, "Human Factors Minor Change Design Guideline: Pickering 'A' CRT-Based Displays" [47]: This design guide provides basic display guidelines to be used in making minor modifications to Cathode Ray Tube (CRT) based displays within the main control room at Pickering Units 1,4.

- NA44-MAN-60300-00001, "Annunciation Design Guide Pickering A, Units 1 and 4" [48]: This manual provides human-system interface design guidance to assist designers, human factor specialists and system engineers in implementing changes involving annunciation, which includes local field annunciation and main control room annunciation.

Pickering Units 5-8

- N-MAN-06700-10000, "Human Factors Design Guideline: Maintenance, Inspection and Testing [49]: This design guide addresses the human factors aspects of equipment and components that field personnel maintain, test or inspect (note, this document is a conversion of 44RS-06700-HFP-004 [45] to an OPG Nuclear Manual format and is applicable to both Pickering 1,4 and Pickering 5-8).

- NK30-MAN-06700-00001, "Human Factors Engineering Design Guideline for Main Control Room Unit Panels and Field Control Panels" [50]: This manual provides human factors guidance for detailed design of human-system interface changes to panel-mounted operator instrumentation on main control room unit panels and field control panels.

[12] In order to improve the human machine interface of the Pickering Units 1,4 Digital Control Computer (DCC) interface, the Pickering A Control Room Enhancement (PACE) was initiated. The PACE is part of the DCC system and forms an interface between control room operators and the DCCs.

- NK30-MAN-06700-00002, "Human Factors Engineering Design Guideline for CRT Displays" [51]: This manual provides basic display guidelines to be used in making minor modifications to CRT-based displays within the main control room.

- NK30-MAN-60300-00001, "Annunciation Design Guide Pickering B, Units 5-8" [52]: This manual provides human-system interface design guidance to assist designers, human factors specialists and system engineers in implementing changes involving annunciation which includes local field annunciation and main control room annunciation.

N-PROG-MP-0001, "Engineering Change Control" [21] applies to changes to SSCs which include the physical plant, software, and station engineered tooling, all of which can impact the human-machine interface. More specifically, N-PROC-MP-0090, "Modification Process" [22] (which receives its authority from N-PROG-MP-0001 [21]) provides the detailed steps and procedures required to meet all of the high-level requirements of the ECC program, which include the preparation of N-FORM-10959, "Design Scoping Checklist" [53]. Section 2.9 of N-FORM-10959 [53] contains a listing of high level HFE questions that are designed to identify whether the proposed modification has an HFE impact.

When required, Human Factors Engineering Program Plans (HFEPP) are prepared in accordance with N-MAN-06700-10002, "Guide for OPG Human Factors Engineering Process" [54], which describes OPG's HFE processes and approach to the conduct of HFE activities and OPG's expectations for performing HFE activities. Per Section 2.2.1 of N-MAN-06700-10002 [54], the HFEPP meets the requirements of the following CNSC guides:

- G-276, "Human Factors Engineering Program Plans" [55]; and

- G-278, "Human Factors Verification and Validation Plans" [56].

CNSC REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants" [57] refers to U.S. NRC NUREG-0711, "Human Factors Engineering Program Review Model" [58] in terms of guidance for the development of HFEPPs. Per Section 2.2.1 of N-MAN-06700-10002 [54], the HFEPP is expected to meet the intent of NUREG-0711 [58] (Section 2.4 Review Criteria). Also, Figure 2 of N-MAN-06700-10002 [54] identifies the NUREG-0711 [58] HFE Program Elements (i.e., Planning and Analysis, Design, Verification and Validation, Implementation and Operation). CNSC REGDOC-2.5.2 [57] also refers to U.S. NRC NUREG-0700, "Human-System Interface Design Review Guidelines" [59] for guidance relating to the design of human-system interfaces. As part of the Pickering B ISR, OPG agreed to complete a high level review against NUREG-0700 [59] if refurbishment was pursued [60]. This is not a gap for PSR2 since refurbishment is not being considered as part of Pickering NGS operation beyond 2020. As discussed earlier, Pickering NGS was originally designed to the standards and best design practices of the day. Many years of operation of Pickering NGS have demonstrated that the plant layout and facilities provide a safe working environment. Improvements based on operation and maintenance experience have been incorporated into processes and the design to improve the human-machine interface. In addition, training and qualification processes (and certification processes for control room staff) for Operations positions ensure that the staff are competent to carry out functions assigned to them.

In terms of analysis of human information requirements and task workload, Task Analysis is performed as part of the HFEPP for modifications with a significant HFE impact. Task Analysis identifies the specific tasks needed to accomplish human actions and the information, control and task support required to complete those tasks. The following two reports provide recent examples of modifications for which Task Analysis has been performed:

- NA44-REP-41170-00006, "Human Factors Engineering Summary Report – Pickering NGS Units 1,4 Turbine Governor System Upgrade" [61]; and
- NK30-REP-41220-10002, "Human Factors Engineering Summary Report – PB Main Generator Automatic Voltage Regulator (AVR) Replacement Project" [62].

The Human-Machine interface linkages to Deterministic Safety Analysis and Probabilistic Safety Assessment are demonstrated as follows:

- Tables 1-2 to 1-11 and Tables S.1-1 to S.1-10 in Part 3 of the Pickering 1,4 and 5-8 Safety Report respectively [63], [64], summarize all required operator action credits.
- The PSA includes human interaction events in the fault tree model for significant human interface related events which could lead to an accident. Examples of such human interaction events include [65]:
  - Failure to perform a required task;
  - Performing an incorrect operation; or
  - Failure to detect an alarmed component failure.

The Human-Machine interface linkage to Hazards analysis is established by the fact that hazards analysis contains consequential failures that are built into the PSA, which in turn captures human interaction events as discussed above.

In addition, the Minimum Shift Complement (MSC) is also linked to the plant design and safety analyses. The MSC is the minimum number of qualified workers required to be present at the plant at all times to respond to all credible events including for the most resource-intensive conditions, to ensure the safe operation and maintenance of the Plant. CNSC Regulatory Guide G-323, "Ensuring the Presence of Sufficient Qualified Staff at Class I Nuclear Facilities – Minimum Shift Complement" [66] describes the CNSC recommended approach for defining the MSC. OPG Instruction P-INS-09100-00003, "Pickering Minimum Shift Complement" [67] is in compliance with G-323 [66]. Analysis to determine the MSC requires consideration of the most resource-intensive initiating events and credible failures considered in the Safety Report and PSA. OPG Instruction N-INS-03490-10003, "Minimum Shift Complement Resources, Qualifications and Procedures Required for Responding to Resource Limiting Events" [68], requires that changes (e.g., design modifications, procedure changes) not be implemented without appropriate consideration and analysis of the MSC.

Conclusion:

The conclusion of this Review Task assessment is that the human-machine interface is considered in the design of the control room and other work stations, that analysis of human information requirements and task workload is performed and that there is a linkage to the PSA, Deterministic Safety Analysis and Hazard Analysis. Guidance such as U.S. NRC NUREG-0700 Revision 2, "Human –System Interface Design Review Guidelines" and NUREG-0711 Revision 2, "Human Factors Engineering Program Review Model", were considered in this assessment. The intent of Review Task #5 is met and therefore Pickering NGS is compliant.

4.1.6 Review Task #6: Design for Radiological Protection

Assess the adequacy of the arrangements for providing radiological protection.

N-PROG-RA-0013, "Radiation Protection" [69] implements a series of standards and procedures for the conduct of activities within the station, in order to achieve the following objectives:

1. Controlling occupational and public exposure:
    - Keeping individual doses below regulatory limits.
    - Avoiding unplanned exposures.
    - Keeping collective doses As Low As Reasonably Achievable (ALARA), social and economic factors taken into account.
2. Preventing the uncontrolled release of contamination or radioactive materials from the nuclear sites through the movement of people and materials.
3. Demonstrating the achievement of 1) and 2) through monitoring.

An important principle of the Radiation Protection program is the control of exposures. Guidelines for general dose rates were established during the design phase of Pickering NGS (for locations such as accessible areas or areas that are only accessible during shutdowns) that would be consistent with the occupancy requirements. Good engineering practice was followed during the initial design of the station such that the layout and operation of facility SSCs and processes are consistent with the established guidelines and contribute to maintaining occupational radiation exposures ALARA. For example, specific design features at Pickering NGS to control radiation dose include the use of radiological zones, the provision of area radiation monitoring equipment, and the use of shielding to control radiation exposures.

When making design changes, engineers maintain or improve upon designs that reduce occupational exposures throughout the lifecycle of the station. N-STD-RA-0018, "Controlling Exposure As Low As Reasonably Achievable" [70] describes elements of the managed system to keep occupational collective dose ALARA, social and economic factors taken into account. Per Section 1.6.2 of N-STD-RA-0018 [70], ALARA principles are to be applied to any changes to the facility design and Engineering must ensure proposed changes to radiological systems are reviewed by the facility Radiation Protection Department in accordance with N-PROG-MP-0001, "Engineering Change Control" [21] and N-PROC-MP-0083, "Constructability, Operability, Maintainability, and Safety (COMS)" [24]. This ensures that radiological safety requirements are identified and addressed as required.

In addition to engineering changes, Radiation Protection staff review changes to the use of space in radiological zones in accordance with N-PROC-RA-0054, "Control of Space Allocation for Transient Material and Extended Storage of Material within the Site" [71]. This procedure prescribes the administrative requirements regarding control of space allocation, transient materials, extended storage of material, and re-locatable structures.

Deposition of small irradiated particles on system pipework during system operation may result in radiological hot spots causing both local and general radiation fields to increase. The impact of these hot spots may be reduced by removing them or applying shielding. Hot spots are shielded in accordance with N-PROC-MA-0060, "Control of Temporary Shielding" [72]. This procedure outlines the processes and controls for requesting, evaluating, approving, installing and removing temporary shielding.

Certain areas of the station are subject to high radiation fields as a result of normal reactor operation, irradiated fuel transfer or equipment operation. Inadvertent entry to these areas is prevented through the use of locked access points. When work is required in these areas, workers use procedures and physical controls to ensure the access hazards are not present or, if present, are strictly controlled. Procedures for accessing areas impacted by station operations (including fuelling activities) for Pickering NGS are implemented via P-INS-09071-00002, "Access Control" [73]. This instruction defines radiation protection requirements that all individuals must follow when entering Access Control Areas. Note: provisions for radiation protection are further detailed in Safety Factor 15 Report, Radiation Protection.

Conclusion:

The conclusion of this Review Task assessment is that the arrangements for providing radiological protection are adequate. The intent of Review Task #6 is met and therefore Pickering NGS is compliant.

4.1.7 Review Task #7: Cumulative Effects of Design Modifications

> *Where the plant has undergone a significant number of modifications over its lifetime or in the period since PSR1, examine the cumulative effects of all modifications on the design.*

The cumulative effects of a design modification are addressed through the ECC Program (N-PROG-MP-0001 [21]) as each new design modification considers the cumulative effects of both the existing design (including previous design changes) and the planned design change. The ECC program ensures that each new design modification remains within the SOE, design basis and licensing basis and also improves or maintains operability, maintainability, radiological and conventional safety, regulatory compliance and production.

A listing of some of the major safety design modifications at Pickering NGS since PSR1 (as outlined in Part 2 of the Pickering A and B Safety Reports [33], [34] and the 2012 Power Reactor Operating Licence Renewal Application [74]) are provided below:

- Fukushima Project related modifications, including:
    - Installation of Passive Autocatalytic Recombiners on all units;
    - Addition of EME including portable diesel pumps and diesel generators;
    - Enhancements to water makeup/cooling capability for the Irradiated Fuel Bays; and
    - Additional flood barriers installed around the Pickering A Standby Generator Fuel Forwarding Pump house.
- Seismic Monitoring System Upgrades;
- ECI Strainer Capacity increase;
- Enhancements to improve Unit 1 and 4 ECI recovery availability;
- Airlock related Design Improvements;
- Fuel Handling Equipment Reliability Improvement;
- Improvements to Inter-Station Transfer Bus;
- Unit 1 and 4 installation of enhancement to the Shutdown Systems (i.e., SDSE and SDSA) as part of the PARTS; and
- Units 2 and 3 safe storage.

Examples of the cumulative effects of these plant design modifications are as follows:

- Equipment Qualification - the above mentioned design modifications have impacted equipment classification (i.e., Environmental/Seismic Qualification status). For example, the airlocks in Pickering Units 1,4 were upgraded to be Environmentally and Seismically Qualified; as part of Units 2 and 3 safe storage, the Unit 2 Class 1 batteries were replaced with new batteries in Seismically Qualified racks.

- Probabilistic Safety Assessment – The Pickering NGS 1,4 and 5-8 Level 1 At-Power Internal Events Risk Assessment [36], [37], have been updated to ensure that the PSA is consistent with the current station design and operation, which includes EME implementation. Likewise, the Level 2 PSA for Pickering NGS 1,4 and 5-8 [38], [39], have incorporated the risk benefits gained from the Fukushima enhancements (e.g., BDBA procedures/guides and EME).

- Deterministic Safety Analysis – To provide additional shutdown system trip parameter coverage for Pickering Units 1,4 (in particular for large loss of coolant accidents), additional trip parameters were added to the Shutdown System. These additional trip parameters (heat transport high/low pressure, neutron overpower, high neutron log rate and manual trip) are designated as SDSE and are independent from the original trip parameters (SDSA). The primary purpose of SDSE is to significantly reduce the probability of a failure to shut down following an initiating event (i.e., large loss of coolant accident (LOCA), fast loss of reactivity control).

Conclusion:

The conclusion of this Review Task assessment is that Pickering NGS has undergone significant modifications over its lifetime and the cumulative effects of all modifications on the design have been assessed. The intent of Review Task #7 is met and therefore Pickering NGS is compliant.

4.1.8 Review Task #8: Compliance with Design Specifications

Confirm that the plant SSCs are compliant with the design specifications and consistent with the design documentation.

The Pickering NGS CM Program (N-PROG-MP-0005 [13]) ensures design documentation is prepared and consistent with the plant design basis and matches the physical plant. Change control programs, policies, and procedures maintain the alignment. N-PROG-MP-0001, "Engineering Change Control" [21] ensures that all modifications to plant SSCs, including software and station engineered tooling, are planned, designed, procured, installed, commissioned, decommissioned, placed into service or removed from service within the safe operating envelope, design basis (including design specifications) and plant licensing conditions. For example, the ECC program ensures that approved modifications are:

- Evaluated based on the risk level, to determine the scope of work activities, stakeholders, resources, documentation updates required to reflect the modification and materials required to complete the modifications;
- Designed in accordance with relevant codes and standards;
- Installed in accordance with the approved design and installation requirements; and
- Commissioned and tested in accordance with acceptance criteria as specified in commissioning specifications.

The SOE is the set of limits and conditions within which the plant shall be operated to ensure conformance with the Safety Report and to ensure that the Safety Report conclusions remain valid. N-STD-MP-0016, "Safe Operating Envelope" [29] provides requirements for defining, implementing and maintaining the SOE. The specific objectives of the SOE are to establish the following:

- Thorough and current record of safety credits and operating limits in the form of Operational Safety Requirements and associated Instrument Uncertainty Calculation reports. Safe Operating Limits and Conditions of Operability (SOE Limits) are captured in station operating documentation, which provide plant operators with the information required to ensure safe operation of the plant in conformance with the requirements of the Safety Analysis.
- A compliance framework whereby plant operation within the requirements established as part of the SOE is verified on a regular basis and appropriate corrective actions are initiated upon discovery of plant operation outside of the SOE.
- Infrastructure by which the SOE is integrated with other relevant business processes and maintained current over the life of the station.

To ensure effective monitoring (e.g., ensuring SSC compliance with design requirements and specifications), maintenance and enhancement of system performance and reliability, N-PROC-MA-0024, "System Performance Monitoring" [75], provides a consistent and comprehensive process for System Engineers.

Pickering NGS has processes in place to ensure design documentation non-compliances are promptly identified and corrected. N-PROC-RA-0022, "Processing Station Condition Records" [76] provides a consistent reporting and evaluation process for identifying adverse conditions at OPG Nuclear. Upon identifying an adverse condition that directly impacts the ability of the station to operate safely, or one that represents an actual or potential operability concern, or one that represents a condition reportable under the operating licence, a Station Condition Record (SCR) is initiated to document the conditions. A corrective action plan is then developed to correct the adverse condition, as required. All actions are tracked to completion under the OPG Action Tracking system. N-PROG-RA-0003, "Corrective Action" [77] establishes the processes to ensure deficiencies, non-conformances, weaknesses, etc., identified in the SCR, are promptly corrected or dispositioned.

Conclusion:

The conclusion of this Review Task assessment is that plant SSCs are compliant with the design specifications and consistent with the design documentation. The intent of Review Task #8 is met and therefore Pickering NGS is compliant.

4.2 L/R/C/S Reviews

As per Section 2.2 of this report, detailed reviews for 42 L/R/C/Ss with content applicable to Safety Factor 1 are provided in References [6], [7] and [8]. Associated findings applicable to Safety Factor 1 are summarized in Table 3 below.

Table 3: PSR2 L/R/C/S Review Results for Safety Factor 1

L/C/R/S Reviewed	PSR2 L/R/C/S Review Results for Safety Factor 1
CSA N286.7-16, "Quality Assurance of Analytical, Scientific and Design Computer Programs"	There are no PSR2 gaps for CSA N286.7-16. Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with CSA N286.7-16.
CSA N285.0-12, "General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants"	There are two PSR2 CSA N285.0-12 (including Updates No. 1 and No. 2) gaps which relate to Safety Factor 1: 1. Clause A.2.3.1 of CSA N285.0-06 identifies that for Shutdown Systems, pressure-retaining portions shall be classified as Class 1, except for three listed exceptions. It was identified during the Pickering B ISR that a limited number of Liquid Injection Shutdown System (LISS) components, which should have been Class 1, were purchased and installed as Class 3. In follow-up, OPG proposed four actions to address the deficiency. When refurbishment was not pursued, a code classification concession was accepted for continued operations. This code classification concession and the four actions identified in the Pickering B ISR gap resolution need to be reconsidered in the context of operation of Pickering NGS beyond 2020. Therefore, this has been identified as a PSR2 gap (**Pickering PSR2 Gap SF1-1**). 2. The PARTS review against CSA-N285.0-95 identified two Acceptable Deviations relating to Clause 7.0 requiring confirmation that the allowable cycles for fatigue would not be exceeded. For Pickering Units 1,4 and Units 5-8 operation beyond 2020, further confirmation is required that the allowable cycles for fatigue will continue to bound current service limits for extended operation. Therefore, this has been identified as a PSR2 gap (**Pickering PSR2 Gap SF1-2**).
CSA N285.4-14, "Periodic Inspection of CANDU Nuclear Power Plant Components"	For Safety Factor 1, there are no PSR2 gaps for CSA N285.4-14.

L/C/R/S Reviewed	PSR2 L/R/C/S Review Results for Safety Factor 1
CSA N285.5-13, "Periodic Inspection of CANDU Nuclear Power Plant Containment Components"	For Safety Factor 1, there are no PSR2 gaps for CSA N285.5-13.
CSA N293-12, "Fire Protection for Nuclear Power Plants"	There are three PSR2 gaps for CSA N293-12 which relate to Safety Factor 1: 1. Clause 7.2.1.10.1 states: "A display and control centre shall be located in the MCR… capable of providing detailed information on the location and nature of the signal. In addition, the panel operator shall be able to control the fire alarm system without having to leave his or her station." Pickering 014 Display Annunciation Station 014-67140-WS2342 in the Emergency Operating Centre is capable of providing annunciation only, and there is no Display Annunciation Station in the Pickering 014 MCR (although there is limited annunciation). Therefore, this has been identified as a PSR2 gap (**Pickering PSR2 Gap SF1-3**). 2. Clause 7.2.1.13 of CSA N293-12 states: "Electrical conductors that are installed in service spaces containing other combustible materials and that are used in connection with fire alarm systems and emergency equipment, including fire alarm cables… shall be capable of performing their intended functions for not less than 1 hour after the start of a fire." Modifications to the Fire Protection System meet the requirements of CAN/ULC-S524 which mandates a 1 hour fire rating as described in Section 2.5 of NA44-DM-71400.2-00001 R001, Section A.2 of NA44-DM-71400-00002 R000 and Section 2 of NK30-DM-71400-00001 R006. This is achieved by the use of Edwards System Technology (EST) that connects the fire alarm control panels via a data communication link with dual redundant circuit wiring paths. However, existing Pyrotronics fire alarm control panels are not similarly connected and, hence, may be susceptible to loss of alarm signal due to spot burning of a cable. While measures such as lack of combustible material in service spaces, combustible transient material control practices, and inherent protection afforded by Pickering NGS cable routing practices used in the Fire Protection systems mitigate the lack of such a feature, it could not be confirmed based on existing documentation that all essential fire alarm cables are capable of performing their intended functions for not less than 1 hour after the start of a fire to meet the requirement of N293-12 sub-clause 7.2.1.13. As a result, this has been identified as a PSR2 gap (**Pickering PSR2 Gap SF1-4**). 3. Clause 7.3.2.2 (d) of CSA N293-12 states: "At a minimum, the fire protection water pumping system shall consist of at least one diesel-engine-driven fire pump and one electric-motor-driven fire pump set, with each pump set being capable of providing, the flow rate and pressure specified in Item (a)". This Clause is met at Pickering Units 1,4 with the provision of diesel-driven firewater pumps, backed up by supplies from the High Pressure Service Water (HPSW) system (as noted in the Pickering A Safety Report NA44-SR-01320-00001 R015, Section 11.5.1.1). It is not met at Pickering Units 5-8, where the Fire Protection System is comprised of the HPSW supplies from the four units only. As a result, Pickering Units 5-8 does not comply with Clause 7.3.2.2 (d) of CSA N293-12 and this has been identified as a PSR2 gap (**Pickering PSR2 Gap SF1-5**).

L/C/R/S Reviewed	PSR2 L/R/C/S Review Results for Safety Factor 1
CSA N287.1-14, "General Requirements for Concrete Containment Structures for Nuclear Power Plants"	There are no PSR2 gaps for CSA N287.1-14. Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with CSA N287.1-14.
CSA N287.2-08, "Material Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants"	There are no PSR2 gaps for CSA N287.2-08. Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with N287.2-08.
CSA N287.3-14, "Design Requirements for Concrete Containment Structures for Nuclear Power Plants"	There are no PSR2 gaps for CSA N287.3-14. Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with CSA N287.3-14.
CSA N287.5-11, "Examination and Testing Requirements for Concrete Containment Structures for Nuclear Power Plants"	There is one PSR2 CSA N287.5-11 gap which relates to Safety Factor 1: 1. The Concrete Containment Structures (CCSs) at Pickering A and B were built and tested to meet the 1965 and 1970 National Building Code of Canada requirements, respectively, prior to the initial issuance of CSA N287.5. No assessments exist which demonstrate that the requirements in effect during construction of Pickering NGS CCSs comply with the requirements of CSA N287.5. Ongoing confirmation that the Pickering NGS CCSs remain fit for service is demonstrated via periodic and in-service inspections conducted in accordance with the requirements of CSA N285.5 and N287.7, and the resultant inspection reports attest to the quality of the design. In addition, the ECC process ensures that any design changes made to the Pickering CCSs will comply with N287.5 going forward, as applicable. The original Pickering construction included requirements for tests and quality control procedures which generally meet the intent of N287.5. Furthermore, retroactive application of N287.5 to the as-built design of CCSs cannot be practically achieved without rebuilding them. Nevertheless, there is a PSR2 gap for Pickering NGS given that compliance with the specific requirements of N287.5 has not been demonstrated (**Pickering PSR2 Gap SF1-6**).
CSA N289.1-08, "General Requirements for Seismic Design and Qualification of CANDU Nuclear Power Plants"	There are no PSR2 gaps for CSA N289.1-08. Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with N289.1-08.
CSA N289.2-10, "Ground Motion Determination for Seismic Qualification of Nuclear Power Plants"	There are no PSR2 gaps for CSA N289.2-10. Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with N289.2-10.
CSA N289.3-10, "Design Procedures for Seismic Qualification of Nuclear Power Plants"	For Safety Factor 1, there are no PSR2 gaps for CSA N289.3-10.

L/C/R/S Reviewed	PSR2 L/R/C/S Review Results for Safety Factor 1
CSA N289.4-12, "Testing Procedures for Seismic Qualification of Nuclear Power Plants Structures, Systems, and Components"	For Safety Factor 1, there are no PSR2 gaps for CSA N289.4-12.
CSA N289.5-12, "Seismic Instrumentation Requirements for Nuclear Power Plants and Nuclear Facilities"	For Safety Factor 1, there are no PSR2 gaps for CSA N289.5-12.
CSA N290.0-11, "General Requirements for Safety Systems of Nuclear Power Plants"	There are three PSR2 N290.0-11 gaps which relate to Safety Factor 1: 1. The Darlington ISR identified a gap against Clause 4.14.10 of N290.0-11 as a result of the lack of design standards related to HFE or HFE activities being formally documented when the control rooms were originally designed and constructed. Pickering NGS has many years of successful Special Safety System (SSS) operation and the absence of formal HFE in the original design is not expected to have any nuclear safety significance relating to SSSs. However, the Darlington gap is also applicable to Pickering NGS and is therefore identified as a PSR2 gap (**Pickering PSR2 Gap SF1-7**). 2. Clause 4.2 of N290.0-11 requires that Plant States be grouped into several categories, including Anticipated Operational Occurrences (AOOs). This is consistent with clauses of REGDOC-2.4.1 and REGDOC-2.5.2 related to identification and classification of initiating events. Since AOOs have not been identified and analyzed in the current Pickering Safety Reports, the requirements and credits attributed to the Special Safety Systems for AOOs, if any, cannot be readily ascertained. This issue has therefore been identified as a PSR2 gap (**Pickering PSR2 Gap SF1-8**). It is being addressed as part of REGDOC-2.4.1 implementation. 3. Clause 4.13 of N290.0-11 identifies requirements to address dynamic piping effects. OPG is currently in the process of completing the High Energy Line Break Assessment (HELBA) for Pickering NGS. Preliminary results show that there would be no consequential damage caused by the rupture of high energy pipes inside containment to safety related equipment, beyond that already accounted for in the Safety Reports. The final HELBA reports for Pickering Units 5-8 have been completed, while Pickering Units 1,4 are expected to be completed in 2017. Since this work has not been completed for Pickering 1,4, this is identified as a PSR2 gap (**Pickering PSR2 Gap SF1-9**).
CSA N290.1-13, "Requirements for the Shutdown Systems of Nuclear Power Plants"	There is one PSR2 CSA N290.1-13 gap which relates to Safety Factor 1: 1. Clause 4.1.8.2 of CSA N290.1-13 is for a new plant and requires remote tripping and monitoring capability for both Shutdown Systems. Pickering Units 1,4 only have one Shutdown System with tripping capability from separate logic (SDSA and SDSE). Remote tripping capability is available for Pickering 5-8 SDS2 and Pickering 1,4 SDSE. However, Pickering Units 5-8 and 1,4 do not have remote tripping and monitoring capability for SDS1 or SDSA respectively.

L/C/R/S Reviewed	PSR2 L/R/C/S Review Results for Safety Factor 1
	Therefore, this has been identified as a PSR2 gap (**Pickering PSR2 Gap SF1-10**).
CSA N290.2-11, "Requirements for Emergency Core Cooling Systems of Nuclear Power Plants"	There are two PSR2 CSA N290.2-11 gaps which relate to Safety Factor 1: 1. Clause 5.2.1.2 of CSA N290.2-11 requires that ECIS design requirements be based on the assumption that the least effective of the Shutdown Systems has operated successfully. The Pickering Units 5-8 Safety Report analysis does address this requirement and the requirement is also contained in the Pickering Units 5-8 Design Requirements. However, this requirement cannot be met for Pickering Units 1,4 since there is only one Shutdown System (albeit with tripping capability from separate SDSA and SDSE logic). Therefore, this has been identified as a PSR2 gap (**Pickering PSR2 Gap SF1-11**). 2. Clause 5.14.11 of CSA N290.2-11 requires instrumentation to be available to monitor post-accident effectiveness and to determine the extent of plugging of ECIS debris interceptors (strainers). While relative health of a strainer can be inferred by a combination of ECIS recovery pump performance and reactor building water level, there is no direct correlation between these conditions and debris loading available. Therefore, this has been identified as a PSR2 gap (which is applicable to both Pickering Units 5-8 and 1,4) (**Pickering PSR2 Gap SF1-12**).
CSA N290.3-11, "Requirements for the Containment System of Nuclear Power Plants"	There is one PSR2 CSA N290.3-11 gap which relates to Safety Factor 1: 1. Per CSA N290.3-11, a Containment Energy Management System (EMS) and Radionuclide Management System (RMS) are required to protect containment and minimize radiological releases for BDBAs. The Pickering EMS and RMS use the Filtered Air Discharge System (FADS) and Reactor Building Air Cooling Units (ACUs). Enhancements to the AC power supplies to these systems and related loads are being provided by Phase 2 EME, which is not yet fully implemented. This PSR2 gap has been identified to track the implementation of Phase 2 EME such that it can be used to support the EMS and RMS (**Pickering PSR2 Gap SF1-13**).
CSA N290.4-11, "Requirements for Reactor Control Systems of Nuclear Power Plants"	There is one PSR2 CSA N290.4-11 gap which relates to Safety Factor 1: 1. Clause 4.2 and Clause 5.19 of CSA N290.4-11 require the capability of the RRS to be assessed to deal with AOOs, by preventing them from escalating into Design Basis Accidents (DBAs) that would require Shutdown System action. In general, the setback function (and stepback in Pickering Units 5-8) addresses this requirement; however, AOOs have not been identified and analyzed in the current Pickering Safety Reports. Therefore, this has been identified as a PSR2 gap (**Pickering PSR2 Gap SF1-14**). It is being addressed as part of REGDOC-2.4.1 implementation. Note: There are also additional clauses which refer to requirements of RRS during AOOs (Clauses 5.6.2, 5.19, 5.16.1); however, for convenience, all issues related to AOO requirements for RRS in N290.4-11 are captured under this one PSR2 gap.

L/C/R/S Reviewed	PSR2 L/R/C/S Review Results for Safety Factor 1
CSA N290.5-06 (R2011) including Update No. 1, "Requirements for Electrical Power and Instrument Air Systems of CANDU Nuclear Power Plants"	There is one PSR2 CSA N290.5-06 (R2011) including Update No. 1 gap which relates to Safety Factor 1: 1. A gap exists for the Pickering Units 1,4 and 5-8 Instrument Air and Electrical Systems on Clauses 7.1 and 7.4.2 of N290.5-06 (R2011) including Update No. 1 dealing with requirements for AOOs. These clauses introduce the requirement for components to be qualified to perform their required functions during normal operation and AOOs. Only the portion of this clause on AOOs is pertinent to nuclear safety. It is likely that AOOs, due to their nature, do not result in a challenge to the qualification of systems, including Instrument Air and Electrical systems. However, AOOs have not been identified and analyzed in the current Pickering Safety Reports. This issue has therefore been identified as a PSR2 gap (**Pickering PSR2 Gap SF1-15**). It is being addressed as part of REGDOC-2.4.1 implementation.
CSA N290.6-09, "Requirements for Monitoring and Display of Nuclear Power Plant Safety Functions in the Event of an Accident"	There are no PSR2 gaps for CSA N290.6-09. Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with CSA N290.6-09.
CSA N290.11-13, "Requirements for Reactor Heat Removal Capability During Outage of Nuclear Power Plants"	There are three PSR2 CSA N290.11-13 gaps which relate to Safety Factor 1 (Plant Design): 1. The CSA N290.11-13 Clause 5.1.1 to 5.1.5 requirement for back-up heat sinks to mitigate the conditions following an AOO is not specified in governance/procedures. Loss of a division of power, a single component failure, etc., which are likely to be in the set of AOOs, are accounted for in the specification of heat sinks. However, AOOs have not been identified and analyzed in the current Pickering Safety Reports. This issue is therefore a PSR2 gap (**Pickering PSR2 Gap SF1-16**). It is being addressed as part of REGDOC-2.4.1 implementation. 2. Clause 5.6.1 of CSA N290.11-13 requires design reliability to be established for outage heat sinks. Although some emergency heat sinks (e.g., Emergency Boiler Water Supply and Emergency Water Supply) have design reliability requirements, design reliability requirements have not been established for all normal and back-up heat sinks used at Pickering. Reliability of all outage heat sinks (including those without explicit targets) is managed under the Risk & Reliability Program (both through unavailability models as well as through Probabilistic Safety Assessment), hence reactor safety impact is assessed and monitored. However, there is a PSR2 gap with respect to establishment of design reliability requirements for Pickering Units 1,4 and 5-8 outage heat sinks (**Pickering PSR2 Gap SF1-17**). 3. Clause 5.6.1 of CSA N290.11-13 requires that the designed reliability for process heat sinks be consistent with AOO frequency limits, such that an emergency heat sink does not need to be used for an AOO. AOOs have not been identified and analyzed in the current Pickering Safety Reports. This issue is therefore a PSR2 gap (**Pickering PSR2 Gap SF1-18**) and is being addressed as part of REGDOC-2.4.1 implementation.

L/C/R/S Reviewed	PSR2 L/R/C/S Review Results for Safety Factor 1
CSA N290.14-15, "Qualification of Digital Hardware and Software for Use in Instrumentation and Control Applications for Nuclear Power Plants"	There is one PSR2 CSA N290.14-15 gap relating to Safety Factor 1 (Plant Design): 1. Correspondence with the CNSC identifies all of the software application qualifications for software Categories 1, 2 and 3 from January 1, 2007 to the time of the correspondence (June 2016). However, an evaluation of legacy Real-Time Process Computing applications with respect to the requirements of N290.14-15 for Categories 1, 2 and 3 software has not been performed. Therefore, this has been identified as a PSR2 gap (**Pickering PSR2 Gap SF1-19**).
CSA N291-15, "Requirements for Safety-related Structures for Nuclear Power Plants"	There are three PSR2 CSA N291-15 gaps which relate to Safety Factor 1 (Plant Design): 1. Clause 6.5.2.2 of CSA N291-15 imposes new requirements for bolted connections in members that are part of the seismic load resisting system. Pickering NGS structures were not explicitly designed to meet these requirements and this is therefore identified as a PSR2 gap (**Pickering PSR2 Gap SF1-20**). 2. Clause 9 of CSA N291-15 contains new requirements related to aging management (including design provisions to account for aging) that are not in CSA N291-08 and that may have significance for operation of Pickering beyond 2020. Pickering structures were not explicitly designed to meet these requirements and this is therefore identified as a PSR2 gap (**Pickering PSR2 Gap SF1-21**). 3. Clauses 6.1.1(b) and 6.9.2.1.4 of CSA N291-15 state requirements for aspects of the design that are specifically based on the plant service life. Pickering structures were not explicitly designed or assessed in relation to the requirements of these clauses for operation beyond 2020. This is identified as a PSR2 gap (**Pickering PSR2 Gap SF1-22**). There is also one PSR2 gap for CSA N291 related to submission of Periodic Inspection Plans and Life Cycle Management Plans for a number of safety-significant civil structures to address fitness for service "to end of mission time" (which will need to be extended for Pickering operation beyond 2020). The gap is related to Safety Factor 4 (Aging). This issue is identified as a PSR2 gap in OPG Report P-REP-03680-00024 R000, "Pickering 5-8 Continued Operations Plan Review in Support of PNGS Periodic Safety Review 2 (PSR2)" [78]. Therefore, a duplicate gap has not been created under CSA N291-15.
CSA N285.6 Series-12, "Material Standards for Reactor Components for CANDU Nuclear Power Plants"	There are no PSR2 gaps for CSA N285.6 Series-12. Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with CSA N285.6 Series-12.
ASME B31.1-14, "Power Piping"	There are no PSR2 gaps for ASME B31.1-14. Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with ASME B31.1-14.
ASME BPVC (2015), "Boiler and Pressure Vessel Code"	There are no PSR2 gaps for ASME BPVC (2015). Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with ASME BPVC (2015).

L/C/R/S Reviewed	PSR2 L/R/C/S Review Results for Safety Factor 1
CSA B51-14 (including Update No. 1), "Boiler, Pressure Vessel, and Pressure Piping Code"	There are no PSR2 gaps for CSA B51-14 (including Update No. 1). Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with CSA B51-14 (including Update No. 1).
CNSC G-278 (2003), "Human Factors Verification and Validation Plans"	There are no PSR2 gaps for CNSC G-278 (2003). Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with CNSC G-278 (2003).
CNSC G-276 (2003), "Human Factors Engineering Program Plans"	There are no PSR2 gaps for CNSC G-276 (2003). Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with CNSC G-276 (2003).
NFPA 20 (2016), "Standard for the Installation of Stationary Pumps for Fire Protection"	There are no PSR2 gaps for NFPA 20 (2016). Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with NFPA 20 (2016).
NFPA 24 (2016), "Standard for the Installation of Private Fire Service Mains and Their Appurtenances"	There are two PSR2 gaps for NFPA 24 (2016) which relate to Safety Factor 1 (Plant Design): 1. For OPG Report NK30-REP-71400-10001 R001, "Fire Protection Code Compliance Review Pickering Nuclear Generating Station B", there is an outstanding issue (Deviation # 13301) which relates to NFPA 24 1970 Section 3601: "Yard post indicator valves at PNGS B are not secured in the open position as required by code" (and which applies to Pickering Units 1,4 as well as Units 5-8). Work to resolve this deviation is currently in progress with locks installed on the majority of the affected valves. Based on OPG List P-LIST-71400-00001 R000, there are a number of SSCs in the yard which directly support plant operation and which are defined as being "related to nuclear safety". As a result, fire water supply to these SSCs is a credited safety function. Deviation # 13301 is not yet complete. Therefore, this has been identified as a PSR2 gap (**Pickering PSR2 Gap SF1-23**). 2. For Pickering Units 5-8 the baseline for NFPA 24 compliance is the 1970 version of the standard. Pickering Units 1,4 have not been previously assessed against NFPA 24. Although recent changes to the 2013 and 2016 versions of NFPA 24 will be addressed in any firewater system design changes going forward (as a result of Code-over-Code reviews performed for NFPA 24), compliance has not been formally documented for Pickering Units 1,4 or Units 5-8 against the most recent versions of NFPA 24. Furthermore, there have been a large number of significant changes to NFPA 24 since 1970, including the 2002 edition which "represented a complete revision of NFPA 24". Since Pickering NGS has not demonstrated compliance with the 2016 version of NFPA 24, this has been identified as a PSR2 gap (**Pickering PSR2 Gap SF1-24**). It is noted that OPG is proactively replacing portions of the firewater piping in accordance with NFPA 24, under the Pickering A Firewater Pipe Replacement Project 13-80069.

L/C/R/S Reviewed	PSR2 L/R/C/S Review Results for Safety Factor 1
CNSC REGDOC-2.5.2 (2014), "Design of Reactor Facilities: Nuclear Power Plants"	There are eight PSR2 CNSC REGDOC-2.5.2 (2014) gaps which relate to Safety Factor 1 (Plant Design): 1. Containment Leak Tightness for Design Extension Conditions (DECs): Clauses 7.3 and 8.6.12 of REGDOC-2.5.2 require containment to provide a leak tight barrier following DECs with severe core damage for a period sufficient to implement off-site emergency measures. REGDOC-2.5.2 guidance suggests this period be at least 24 hours. Such a requirement does not exist in BDBA/Severe Accident (SA) mitigation, so this represents a PSR2 gap (**Pickering PSR2 Gap SF1-25**). 2. On-Demand Reliability of Safety Systems: Clause 7.6 of REGDOC-2.5.2 requires all SSCs important to safety to meet an on-demand failure rate of <10^{-3} yrs/yr. This requirement is not met for several systems including Pickering 1,4 ECI and is therefore identified as a PSR2 gap (**Pickering PSR2 Gap SF1-26**). 3. Sharing of Safety Systems and Turbine Hall: Clause 7.6.5 of REGDOC-2.5.2 has a new requirement that sharing of safety systems and the turbine generator building not be permitted. Pickering units share ECI and NPC, as well as the turbine hall; therefore, this has been identified as a PSR2 gap (**Pickering PSR2 Gap SF1-27**). 4. Allowable Times for Crediting On-Site Operator Actions: Clauses 7.10 and 8.10.4 of REGDOC-2.5.2 establish new time limits for crediting operator actions, i.e., 30 minutes for MCR actions and 1 hour for field actions. Pickering NGS has not demonstrated that deterministic safety analysis consequences are acceptable if MCR and field action are not credited for these times respectively. Therefore, this has been identified as a PSR2 gap (**Pickering PSR2 Gap SF1-28**). 5. Seismic Qualification and Design: Clause 7.13.1 of REGDOC-2.5.2 requires that Beyond Design Basis (BDB) Earthquake seismic margin be a factor of 1.67 beyond that required for the new plant Design Basis Earthquake (DBE). Fragility evaluations were completed for seismic mitigating SSCs, however, based on available information it could not be confirmed that the new plant BDB Earthquake margin of 1.67 would be achieved. Therefore, this has been identified as a PSR2 gap (**Pickering PSR2 Gap SF1-29**). 6. Human Factors in Design: Clauses 7.21 and 8.10.1 of REGDOC-2.5.2 introduce new requirements for the systematic application of HFE principles to plant design. Many years of safe and reliable operating experience indicate that the design and processes for integration of human interactions with the plant were and remain robust. However, Pickering plant design predates the current requirements for incorporating HFE into the design and the existing plant has not been systematically demonstrated to meet the requirements for a new plant. Therefore, this has been identified as a PSR2 gap (**Pickering PSR2 Gap SF1-30**). 7. Detection/Isolation of ECI Heat Exchanger Tube Leak: Clause 8.5 of REGDOC-2.5.2 requires ECI recovery heat exchanger tube leak detection capability. Pickering Units 5-8 ECI recovery heat exchangers do not have leak detection

L/C/R/S Reviewed	PSR2 L/R/C/S Review Results for Safety Factor 1
	capability on the cooling water side. Therefore, this has been identified as a PSR2 gap (**Pickering PSR2 Gap SF1-31**). 8. Safety Parameter Display System Qualification for DECs: Clause 8.10.1.1 of REGDOC-2.5.2 requires the MCR to contain a Safety Parameter Display System (SPDS) that presents sufficient information on safety-critical parameters for the diagnosis and mitigation of DBAs and DECs. The SPDSs are to be qualified for DEC and have parameters available in both the MCR and Secondary Control Areas (SCA), per Clause 8.10.2. Pickering SPDSs are not Review Level Condition (RLC) qualified or available in all locations. As part of the Fukushima follow-up, instrumentation to support critical parameters required to function for DECs has been evaluated for survivability. The instrument loops associated with these parameters have been identified for use in Critical Safety Parameter Monitoring (CSPM) and BDBA procedures. However, the indications from these loops are not in one central location and, in some cases, require field action (e.g., power) to obtain data. This does not fully satisfy the requirements to have these parameters available from a SPDS in the MCR and SCA. Therefore, this has been identified as a PSR2 gap relating to the new plant requirement to have SPDS that is DEC qualified and with parameters available in the MCR and SCA (**Pickering PSR2 Gap SF1-32**).
CNSC REGDOC-2.3.2 (2015), "Accident Management, Version 2"	There is one PSR2 CNSC REGDOC-2.3.2 (2015) gap which relates to Safety Factor 1 (Plant Design): 1. Full provision of Complementary Design Features for containment integrity as required by Clause 4.2.1 of REGDOC-2.3.2 will be addressed with the completion of Phase 2 Emergency Mitigating Equipment. This work is currently scheduled to be fully implemented by the end of 2017. Since this work is still in progress, it has been identified as a PSR2 gap (**Pickering PSR2 Gap SF1-33**).
CSA N286.7.1-09, "Guideline for the Application of N286.7-99, Quality Assurance of Analytical, Scientific, and Design Computer Programs for Nuclear Power Plants"	The N286.7.1 guide has been amalgamated into the new (-16) edition of the N286.7 Standard. The N286.7 CSA Impact Statement states [9]: "The CSA N286.7.1 guide will no longer be maintained after this new edition of N286.7 is issued. Any relevant guidance has been put into the new edition of N286.7." As a result, only the review of N286.7-16 has been prepared for PSR2.
CSA N290.12-14, "Human Factors in Design for Nuclear Power Plants"	There are no PSR2 gaps for CSA N290.12-14. Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with CSA N290.12-14.
CNSC G-149 (2000), "Computer Programs Used in Design and Safety Analyses of Nuclear Power Plants and Research Reactors"	There are no PSR2 gaps for CNSC G-149 (2000). Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with G-149 (2000).

L/C/R/S Reviewed	PSR2 L/R/C/S Review Results for Safety Factor 1
CNSC R-77 (1987), "Overpressure Protection Requirements for Primary Heat Transport Systems in CANDU Power Reactors Fitted with Two Shutdown Systems"	There are no PSR2 gaps for CNSC R-77 (1987). Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with CNSC R-77 (1987).
CSA N290.7-14, "Cyber-Security for Nuclear Power Plants and Small Reactor Facilities"	The gap analysis, N-REP-69000-10003 R000, "Gap Analysis Between CSA N290.7-14 Cyber Security Requirements for Nuclear Power Plants and Small Reactor Facilities" [79] and implementation plan for N290.7-14 was accepted by the CNSC. For reasons of security and confidentiality, the findings of the gap analysis for N290.7-14 will not be discussed in PSR2.
NBCC (2010), "National Building Code of Canada"	There are no PSR2 gaps for NBCC (2010). Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with NBCC (2010).
NFCC (2010), "National Fire Code of Canada"	There is one PSR2 gap for NFCC (2010), related to piping for flammable or combustible liquids at building entrances. The gap is related to Safety Factor 1 (Plant Design). This issue is identified as a PSR2 gap in OPG Report P-REP-03680-00024 R000, "Pickering 5-8 Continued Operations Plan Review in Support of PNGS Periodic Safety Review 2" [78]. Therefore, a duplicate gap under NFCC (2010) has not been created.
CSA N290.8-15, "Technical Specification Requirements for Nuclear Power Plant Components"	There is one PSR2 CSA N290.8-15 gap which relates to Safety Factor 1 (Plant Design): 1. Clause 4.7 of CSA N290.8-15 mandates that the technical specification requires the supplier to identify and describe all digital items included in their equipment. In the event that the use of digital items is identified by OPG in advance of issuing a Request for Proposal (RFP) or Request for Quotation (RFQ), existing OPG procedures are adequate for ensuring that requirements related to digital items are documented in the technical specification. However, a requirement for a supplier to self-identify whether their product contains any digital items is not reflected in OPG governing documents. This has therefore been identified as a PSR2 gap (**Pickering PSR2 Gap SF1-34**).

4.3 OPG Program Effectiveness Reviews

The OPG Nuclear Programs reviewed for Safety Factor 1 are identified in Table 2, and details of the associated effectiveness reviews for each of the N-PROGs are provided in Appendix B.

4.4 Additional Review Findings

As discussed in Section 3.4, the PSR2 Safety Factor 1 assessment also included a review of commitments previously made to the CNSC, open CNSC action items, and exemptions granted by the CNSC, as identified in the R04 Pickering LCH [4], to determine if there are any impacts associated with operation of the Pickering Units past 2020. The review also included identification and review of previously identified programmatic Darlington PSR1 gaps related to Safety Factor 1 to determine impacts associated with operation of the Pickering Units past 2020.

Review of the Darlington IIP [17] for gaps that may need to be reassessed in the context of Pickering PSR2 for operation past 2020, identified the following gap:

- **Gap SF1-35** – Darlington Gap IIP-OI 063 was identified based on the requirement to replace single wall fuel oil piping with double wall piping if degraded piping is found. AR# 28175307 was initiated which required revision of N-PROC-MA-0088, "Buried Piping Program Requirements" to use a graded approach for the replacement of single walled piping with double walled material in instances of leakage. AR# 28175307 currently has corrective actions in place and is expected to be completed by Q1 2020. This issue is also applicable to Pickering NGS and is therefore a gap for Pickering PSR2.

Review of the Pickering LCH [4] identified a concession in Section 6.2 that will need to be considered in the context of continued operation past 2020 for PSR2 Safety Factor 1. The concession is related to exemption from requiring a Canadian Registration Number (CRN) for certain fittings and components associated with fire protection systems. This is identified as a PSR2 gap (**Pickering PSR2 Gap SF1-36**).

Findings from the review of previously identified PSR1 gaps in the Pickering Units 5-8 Continued Operations Plan [15] are provided in a separate PSR2 COP Review Report. Findings from the review of Fukushima Action Items are provided in a separate PSR2 FAI Review Report. Results from the Continued Operations Plan and Fukushima Action Items reviews will be considered in the Global Assessment process.

The following PSR2 gaps identified in this Safety Factor 1 report are relevant to other Safety Factors:

- Gaps SF1-2, SF1-21 and SF1-22 are relevant to Safety Factor 4 (Aging);
- Gaps SF1-8, SF1-14, SF1-15, SF1-16, SF1-18 and SF1-28 are relevant to Safety Factor 5 (Deterministic Safety Analysis);
- Gaps SF1-17, SF1-18 and SF1-26 are relevant to Safety Factor 6 (Probabilistic Safety Assessment);
- Gap SF1-9 is relevant to Safety Factor 7 (Hazard Analysis);
- Gaps SF1-7 and SF1-30 are relevant to Safety Factor 12 (Human Factors); and
- Gaps SF1-13, SF1-25 and SF1-33 are relevant to Safety Factor 13 (Emergency Planning).

5.0 RESULTS AND CONCLUSIONS

OPG Governance, Programs, Policies, Procedures, Instructions and Guidelines related to Safety Factor 1 were reviewed for the eight PSR2 Review Tasks in Section 4.1 of this report and resulted in no Pickering PSR2 gaps. L/R/C/S and OPG Nuclear Program effectiveness reviews for Safety Factor 1 were prepared per Sections 4.2 and 4.3, respectively, and resulted in PSR2 Gaps SF1-1 to SF1-34 below. Per Section 4.4, this report also included identification and review of previously identified programmatic Darlington PSR1 gaps related to Safety Factor 1 (to ascertain the implications of extending Pickering NGS operation beyond 2020), as well as a review of the R04 Pickering LCH [4] for any impacts of Pickering NGS operation beyond 2020 on a) OPG commitments previously made to the CNSC, b) open CNSC action items, and c) exemptions granted by the CNSC (all related to Safety Factor 1), which resulted in PSR2 Gaps SF1-35 and SF1-36.

The 36 PSR2 gaps that will need to be addressed as part of Pickering PSR2 are:

- **<u>Gap SF1-1</u>**: Clause A.2.3.1 of CSA N285.0-06 identifies that for Shutdown Systems, pressure-retaining portions shall be classified as Class 1, except for three listed exceptions. It was identified during the Pickering B Integrated Safety Review (ISR) that a limited number of Liquid Injection Shutdown System (LISS) components, which should have been Class 1, were purchased and installed as Class 3. In follow-up, OPG proposed four actions to address the deficiency. When refurbishment was not pursued, a code classification concession was accepted for continued operations. This code classification concession and the four actions identified in the Pickering B ISR gap resolution need to be reconsidered in the context of operation of Pickering NGS beyond 2020. Therefore, this has been identified as a PSR2 gap.

- **<u>Gap SF1-2</u>**: The Pickering A Return to Service review against CSA-N285.0-95 identified two Acceptable Deviations relating to Clause 7.0 requiring confirmation that the allowable cycles for fatigue would not be exceeded. For Pickering Units 1,4 and Units 5-8 operation beyond 2020, further confirmation is required that the allowable cycles for fatigue will continue to bound current service limits for extended operation. Therefore, this has been identified as a PSR2 gap.

- **<u>Gap SF1-3</u>**: Clause 7.2.1.10.1 of CSA N293-12 states: "A display and control centre shall be located in the MCR [Main Control Room]… capable of providing detailed information on the location and nature of the signal. In addition, the panel operator shall be able to control the fire alarm system without having to leave his or her station." Pickering 014 Display Annunciation Station 014-67140-WS2342 in the Emergency Operating Centre is capable of providing annunciation only, and there is no Display Annunciation Station in the Pickering 014 MCR (although there is limited annunciation). Therefore, this has been identified as a PSR2 gap.

- **Gap SF1-4**: Clause 7.2.1.13 of CSA N293-12 states: "Electrical conductors that are installed in service spaces containing other combustible materials and that are used in connection with fire alarm systems and emergency equipment, including fire alarm cables… shall be capable of performing their intended functions for not less than 1 hour after the start of a fire." Modifications to the Fire Protection System meet the requirements of CAN/ULC-S524 which mandates a 1 hour fire rating as described in Section 2.5 of NA44-DM-71400.2-00001 R001, Section A.2 of NA44-DM-71400-00002 R000 and Section 2 of NK30-DM-71400-00001 R006. This is achieved by the use of Edwards System Technology (EST) that connects the fire alarm control panels via a data communication link with dual redundant circuit wiring paths. However, existing Pyrotronics fire alarm control panels are not similarly connected and, hence, may be susceptible to loss of alarm signal due to spot burning of a cable. While measures such as lack of combustible material in service spaces, combustible transient material control practices, and inherent protection afforded by Pickering NGS cable routing practices used in the Fire Protection systems mitigate the lack of such a feature, it could not be confirmed based on existing documentation that all essential fire alarm cables are capable of performing their intended functions for not less than 1 hour after the start of a fire to meet the requirement of N293-12 sub-clause 7.2.1.13. As a result, this has been identified as a PSR2 gap.

- **Gap SF1-5**: Clause 7.3.2.2 (d) of CSA N293-12 states: "At a minimum, the fire protection water pumping system shall consist of at least one diesel-engine-driven fire pump and one electric-motor-driven fire pump set, with each pump set being capable of providing, the flow rate and pressure specified in Item (a)". This Clause is met at Pickering Units 1,4 with the provision of diesel-driven firewater pumps, backed up by supplies from the High Pressure Service Water (HPSW) system (as noted in the Pickering A Safety Report NA44-SR-01320-00001 R015, Section 11.5.1.1). It is not met at Pickering Units 5-8, where the Fire Protection System is comprised of the HPSW supplies from the four units only. As a result, Pickering Units 5-8 does not comply with Clause 7.3.2.2 (d) of CSA N293-12 and this has been identified as a PSR2 gap.

- **Gap SF1-6**: The Concrete Containment Structures (CCSs) at Pickering A and B were built and tested to meet the 1965 and 1970 National Building Code of Canada requirements, respectively, prior to the initial issuance of CSA N287.5. No assessments exist which demonstrate that the requirements in effect during construction of Pickering NGS CCSs comply with the requirements of CSA N287.5. Ongoing confirmation that the Pickering NGS CCSs remain fit for service is demonstrated via periodic and in-service inspections conducted in accordance with the requirements of CSA N285.5 and N287.7, and the resultant inspection reports attest to the quality of the design. In addition, the Engineering Change Control (ECC) process ensures that any design changes made to the Pickering CCSs will comply with N287.5 going forward, as applicable.

The original Pickering construction included requirements for tests and quality control procedures which generally meet the intent of N287.5. Furthermore, retroactive application of N287.5 to the as-built design of CCSs cannot be practically achieved without rebuilding them. Nevertheless, there is a PSR2 gap for Pickering NGS given that compliance with the specific requirements of N287.5 has not been demonstrated.

- **Gap SF1-7**: The Darlington Integrated Safety Review (ISR) identified a gap against Clause 4.14.10 of N290.0-11 as a result of the lack of design standards related to Human Factors Engineering (HFE) or HFE activities being formally documented when the control rooms were originally designed and constructed. Pickering NGS has many years of successful Special Safety System (SSS) operation and the absence of formal HFE in the original design is not expected to have any nuclear safety significance relating to SSSs. However, the Darlington gap is also applicable to Pickering NGS and is therefore identified as a PSR2 gap.

- **Gap SF1-8**: Clause 4.2 of N290.0-11 requires that Plant States be grouped into several categories, including Anticipated Operational Occurrences (AOOs). This is consistent with clauses of REGDOC-2.4.1 and REGDOC-2.5.2 related to identification and classification of initiating events. Since AOOs have not been identified and analyzed in the current Pickering Safety Reports, the requirements and credits attributed to the Special Safety Systems for AOOs, if any, cannot be readily ascertained. This issue has therefore been identified as a PSR2 gap. It is being addressed as part of REGDOC-2.4.1 implementation.

- **Gap SF1-9**: Clause 4.13 of N290.0-11 identifies requirements to address dynamic piping effects. OPG is currently in the process of completing the High Energy Line Break Assessment (HELBA) for Pickering NGS. Preliminary results show that there would be no consequential damage caused by the rupture of high energy pipes inside containment to safety related equipment, beyond that already accounted for in the Safety Reports. The final HELBA reports for Pickering Units 5-8 have been completed, while Pickering Units 1,4 are expected to be completed in 2017. Since this work has not been completed for Pickering 1,4, this is identified as a PSR2 gap.

- **Gap SF1-10**: Clause 4.1.8.2 of CSA N290.1-13 is for a new plant and requires remote tripping and monitoring capability for both Shutdown Systems. Pickering Units 1,4 only have one Shutdown System with tripping capability from separate logic (SDSA and SDSE). Remote tripping capability is available for Pickering 5-8 SDS2 and Pickering 1,4 SDSE. However, Pickering Units 5-8 and 1,4 do not have remote tripping and monitoring capability for SDS1 or SDSA respectively. Therefore, this has been identified as a PSR2 gap.

- **Gap SF1-11**: Clause 5.2.1.2 of CSA N290.2-11 requires that Emergency Coolant Injection System (ECIS) design requirements be based on the assumption that the least effective of the Shutdown Systems has operated successfully. The Pickering Units 5-8 Safety Report analysis does address this requirement and the requirement is also contained in the Pickering Units 5-8 Design Requirements. However, this requirement cannot be met for Pickering Units 1,4 since there is only one Shutdown System (albeit with tripping capability from separate SDSA and SDSE logic). Therefore, this has been identified as a PSR2 gap.

- **Gap SF1-12**: Clause 5.14.11 of CSA N290.2-11 requires instrumentation to be available to monitor post-accident effectiveness and to determine the extent of plugging of Emergency Coolant Injection System (ECIS) debris interceptors (strainers). While relative health of a strainer can be inferred by a combination of ECIS recovery pump performance and reactor building water level, there is no direct correlation between these conditions and debris loading available. Therefore, this has been identified as a PSR2 gap (which is applicable to both Pickering Units 5-8 and 1,4).

- **Gap SF1-13**: Per CSA N290.3-11, a Containment Energy Management System (EMS) and Radionuclide Management System (RMS) are required to protect containment and minimize radiological releases for Beyond Design Basis Accidents (BDBAs). The Pickering EMS and RMS use the Filtered Air Discharge System (FADS) and Reactor Building Air Cooling Units (ACUs). Enhancements to the AC power supplies to these systems and related loads are being provided by Phase 2 Emergency Mitigating Equipment (EME), which is not yet fully implemented. This PSR2 gap has been identified to track the implementation of Phase 2 EME such that it can be used to support the EMS and RMS.

- **Gap SF1-14**: Clause 4.2 and Clause 5.19 of CSA N290.4-11 require the capability of the Reactor Regulating System (RRS) to be assessed to deal with Anticipated Operational Occurrences (AOOs), by preventing them from escalating into Design Basis Accidents (DBAs) that would require Shutdown System action. In general, the setback function (and stepback in Pickering Units 5-8) addresses this requirement; however, AOOs have not been identified and analyzed in the current Pickering Safety Reports. Therefore, this has been identified as a PSR2 gap. It is being addressed as part of REGDOC-2.4.1 implementation. Note: There are also additional clauses which refer to requirements of RRS during AOOs (Clauses 5.6.2, 5.19, 5.16.1); however, for convenience, all issues related to AOO requirements for RRS in N290.4-11 are captured under this one PSR2 gap.

- **Gap SF1-15**: A gap exists for the Pickering Units 1,4 and 5-8 Instrument Air and Electrical Systems on Clauses 7.1 and 7.4.2 of N290.5-06 (R2011) including Update No. 1 dealing with requirements for Anticipated Operational Occurrences (AOOs). These clauses introduce the requirement for components to be qualified to perform their required functions during normal operation and AOOs. Only the portion of this clause on AOOs is pertinent to nuclear safety. It is likely that AOOs, due to their nature, do not result in a challenge to the qualification of systems, including Instrument Air and Electrical systems. However, AOOs have not been identified and analyzed in the current Pickering

Safety Reports. This issue has therefore been identified as a PSR2 gap. It is being addressed as part of REGDOC-2.4.1 implementation.

- **<u>Gap SF1-16</u>**: The CSA N290.11-13 Clause 5.1.1 to 5.1.5 requirement for back-up heat sinks to mitigate the conditions following an Anticipated Operational Occurrence (AOO) is not specified in governance/procedures. Loss of a division of power, a single component failure, etc., which are likely to be in the set of AOOs, are accounted for in the specification of heat sinks. However, AOOs have not been identified and analyzed in the current Pickering Safety Reports. This issue is therefore a PSR2 gap. It is being addressed as part of REGDOC-2.4.1 implementation.

- **<u>Gap SF1-17</u>**: Clause 5.6.1 of CSA N290.11-13 requires design reliability to be established for outage heat sinks. Although some emergency heat sinks (e.g., Emergency Boiler Water Supply and Emergency Water Supply) have design reliability requirements, design reliability requirements have not been established for all normal and back-up heat sinks used at Pickering. Reliability of all outage heat sinks (including those without explicit targets) is managed under the Risk & Reliability Program (both through unavailability models as well as through Probabilistic Safety Assessment), hence reactor safety impact is assessed and monitored. However, there is a PSR2 gap with respect to establishment of design reliability requirements for Pickering Units 1,4 and 5-8 outage heat sinks.

- **<u>Gap SF1-18</u>**: Clause 5.6.1 of CSA N290.11-13 requires that the designed reliability for process heat sinks be consistent with Anticipated Operational Occurrence (AOO) frequency limits, such that an emergency heat sink does not need to be used for an AOO. AOOs have not been identified and analyzed in the current Pickering Safety Reports. This issue is therefore a PSR2 gap and is being addressed as part of REGDOC-2.4.1 implementation.

- **<u>Gap SF1-19</u>**: Correspondence with the CNSC identifies all of the software application qualifications for software Categories 1, 2 and 3 from January 1, 2007 to the time of the correspondence (June 2016). However, an evaluation of legacy Real-Time Process Computing applications with respect to the requirements of N290.14-15 for Categories 1, 2 and 3 software has not been performed. Therefore, this has been identified as a PSR2 gap.

- **<u>Gap SF1-20</u>**: Clause 6.5.2.2 of CSA N291-15 imposes new requirements for bolted connections in members that are part of the seismic load resisting system. Pickering NGS structures were not explicitly designed to meet these requirements and this is therefore identified as a PSR2 gap.

- **<u>Gap SF1-21</u>**: Clause 9 of CSA N291-15 contains new requirements related to aging management (including design provisions to account for aging) that are not in CSA N291-08 and that may have significance for operation of Pickering beyond 2020. Pickering structures were not explicitly designed to meet these requirements and this is therefore identified as a PSR2 gap.

- **Gap SF1-22**: Clauses 6.1.1(b) and 6.9.2.1.4 of CSA N291-15 state requirements for aspects of the design that are specifically based on the plant service life. Pickering structures were not explicitly designed or assessed in relation to the requirements of these clauses for operation beyond 2020. This is identified as a PSR2 gap.

- **Gap SF1-23**: For OPG Report NK30-REP-71400-10001 R001, "Fire Protection Code Compliance Review Pickering Nuclear Generating Station B", there is an outstanding issue (Deviation # 13301) which relates to NFPA 24 1970 Section 3601: "Yard post indicator valves at PNGS B are not secured in the open position as required by code" (and which applies to Pickering Units 1,4 as well as Units 5-8). Work to resolve this deviation is currently in progress with locks installed on the majority of the affected valves. Based on OPG List P-LIST-71400-00001 R000, there are a number of Structures, Systems and Components (SSCs) in the yard which directly support plant operation and which are defined as being "related to nuclear safety". As a result, fire water supply to these SSCs is a credited safety function. Deviation # 13301 is not yet complete. Therefore, this has been identified as a PSR2 gap.

- **Gap SF1-24**: For Pickering Units 5-8 the baseline for NFPA 24 compliance is the 1970 version of the standard. Pickering Units 1,4 have not been previously assessed against NFPA 24. Although recent changes to the 2013 and 2016 versions of NFPA 24 will be addressed in any firewater system design changes going forward (as a result of Code-over-Code reviews performed for NFPA 24), compliance has not been formally documented for Pickering Units 1,4 or Units 5-8 against the most recent versions of NFPA 24. Furthermore, there have been a large number of significant changes to NFPA 24 since 1970, including the 2002 edition which "represented a complete revision of NFPA 24". Since Pickering NGS has not demonstrated compliance with the 2016 version of NFPA 24, this has been identified as a PSR2 gap. It is noted that OPG is proactively replacing portions of the firewater piping in accordance with NFPA 24, under the Pickering A Firewater Pipe Replacement Project 13-80069.

- **Gap SF1-25**: Containment Leak Tightness for Design Extension Conditions (DECs): Clauses 7.3 and 8.6.12 of REGDOC-2.5.2 require containment to provide a leak tight barrier following DECs with severe core damage for a period sufficient to implement off-site emergency measures. REGDOC-2.5.2 guidance suggests this period be at least 24 hours. Such a requirement does not exist in Beyond Design Basis Accident (BDBA)/Severe Accident (SA) mitigation, so this represents a PSR2 gap.

- **Gap SF1-26**: On-Demand Reliability of Safety Systems: Clause 7.6 of REGDOC-2.5.2 requires all Structures, Systems, and Components (SSCs) important to safety (SIS) to meet an on-demand failure rate of $<10^{-3}$ yrs/yr. This requirement is not met for several systems including Pickering 1,4 Emergency Coolant Injection (ECI) and is therefore identified as a PSR2 gap.

- **Gap SF1-27**: Sharing of Safety Systems and Turbine Hall: Clause 7.6.5 of REGDOC-2.5.2 has a new requirement that sharing of safety systems and the turbine generator building not be permitted. Pickering Units share Emergency Coolant Injection (ECI) and Negative Pressure Containment (NPC), as well as the turbine hall; therefore, this has been identified as a PSR2 gap.

- **Gap SF1-28**: Allowable Times for Crediting On-Site Operator Actions: Clauses 7.10 and 8.10.4 of REGDOC-2.5.2 establish new time limits for crediting operator actions, i.e., 30 minutes for Main Control Room (MCR) actions and 1 hour for field actions. Pickering NGS has not demonstrated that deterministic safety analysis consequences are acceptable if MCR and field action are not credited for these times respectively. Therefore, this has been identified as a PSR2 gap.

- **Gap SF1-29**: Seismic Qualification and Design: Clause 7.13.1 of REGDOC-2.5.2 requires that Beyond Design Basis (BDB) Earthquake seismic margin be a factor of 1.67 beyond that required for the new plant Design Basis Earthquake (DBE). Fragility evaluations were completed for seismic mitigating Structures, Systems and Components (SSCs), however, based on available information it could not be confirmed that the new plant BDB Earthquake margin of 1.67 would be achieved. Therefore, this has been identified as a PSR2 gap.

- **Gap SF1-30**: Human Factors in Design: Clauses 7.21 and 8.10.1 of REGDOC-2.5.2 introduce new requirements for the systematic application of Human Factors Engineering (HFE) principles to plant design. Many years of safe and reliable operating experience indicate that the design and processes for integration of human interactions with the plant were and remain robust. However, Pickering plant design predates the current requirements for incorporating HFE into the design and the existing plant has not been systematically demonstrated to meet the requirements for a new plant. Therefore, this has been identified as a PSR2 gap.

- **Gap SF1-31**: Detection/Isolation of Emergency Coolant Injection (ECI) Heat Exchanger Tube Leak: Clause 8.5 of REGDOC-2.5.2 requires ECI recovery heat exchanger tube leak detection capability. Pickering Units 5-8 ECI recovery heat exchangers do not have leak detection capability on the cooling water side. Therefore, this has been identified as a PSR2 gap.

- **Gap SF1-32**: Safety Parameter Display System Qualification for Design Extension Conditions (DECs): Clause 8.10.1.1 of REGDOC-2.5.2 requires the Main Control Room (MCR) to contain a Safety Parameter Display System (SPDS) that presents sufficient information on safety-critical parameters for the diagnosis and mitigation of Design Basis Accidents (DBAs) and DECs. The SPDSs are to be qualified for DEC and have parameters available in both the MCR and Secondary Control Areas (SCA), per Clause 8.10.2. Pickering SPDSs are not Review Level Condition (RLC) qualified or available in all locations. As part of the Fukushima follow-up, instrumentation to support critical parameters required to function for DECs has been evaluated for survivability. The instrument loops associated with these parameters have been identified for use in Critical Safety Parameter Monitoring (CSPM) and Beyond Design Basis Accident (BDBA) procedures. However, the indications from these loops are not in one central location and, in some cases, require field action (e.g., power) to obtain data. This does not fully satisfy the requirements to have these parameters available from a SPDS in the MCR and SCA. Therefore, this has been identified as a PSR2 gap relating to the new plant requirement to have SPDS that is DEC qualified and with parameters available in the MCR and SCA.

- **Gap SF1-33**: Full provision of Complementary Design Features for containment integrity as required by Clause 4.2.1 of REGDOC-2.3.2 will be addressed with the completion of Phase 2 Emergency Mitigating Equipment. This work is currently scheduled to be fully implemented by the end of 2017. Since this work is still in progress, it has been identified as a PSR2 gap.

- **Gap SF1-34**: Clause 4.7 of CSA N290.8-15 mandates that the technical specification requires the supplier to identify and describe all digital items included in their equipment. In the event that the use of digital items is identified by OPG in advance of issuing a Request for Proposal (RFP) or Request for Quotation (RFQ), existing OPG procedures are adequate for ensuring that requirements related to digital items are documented in the technical specification. However, a requirement for a supplier to self-identify whether their product contains any digital items is not reflected in OPG governing documents. This has therefore been identified as a PSR2 gap.

- **Gap SF1-35**: Darlington Gap IIP-OI 063 was identified based on the requirement to replace single wall fuel oil piping with double wall piping if degraded piping is found. AR# 28175307 was initiated which required revision of N-PROC-MA-0088, "Buried Piping Program Requirements" to use a graded approach for the replacement of single walled piping with double walled material in instances of leakage. AR# 28175307 currently has corrective actions in place and is expected to be completed by Q1 2020. This issue is also applicable to Pickering NGS and is therefore a gap for Pickering PSR2.

- **Gap SF1-36**: Section 6.2 of the Pickering LCH [4] outlines a concession related to exemption from requiring a Canadian Registration Number (CRN) for certain fittings and components associated with fire protection systems. This concession will need to be considered in the context of Pickering PSR2 for operation past 2020 and is therefore a gap for Pickering PSR2.

The review has confirmed, by assessment against the current licensing basis and applicable standards, requirements and practices, that the design of Pickering NGS and its documentation is adequate.

6.0 REFERENCES

[1] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[2] CNSC REGDOC-2.3.3, *Periodic Safety Reviews*, April 2015.

[3] IAEA Safety Guide No. SSG-25, *Periodic Safety Review for Nuclear Power Plants*, 2013.

[4] CNSC Report, LCH-PNGS-R004, *Pickering NGS: Licence Conditions Handbook*, December 23, 2015.

[5] OPG Report, P-REP-03680-00003 R000, *Pickering NGS PSR2: Definition of Safety Factor Review Tasks*, May 30, 2016.

[6] OPG Report, P-REP-03680-00004 R000, *Pickering NGS Periodic Safety Review 2: Code and Standard Reviews for Safety Factors 2 (Actual Condition of SSCs), 3 (Equipment Qualification) and 4 (Aging)*, July 2016.

[7] OPG Report, P-REP-03680-00021 R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 8, 10, 12, 13 and 14*, December 2016.

[8] OPG Report, P-REP-03680-00029 R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 1, 5, 6, and 7*, March 2017.

[9] CSA Impact Statement, *Notification of CSA N286.7 on Quality Assurance of Analytical, Scientific, and Design Computer Programs; Product: New Edition; Product Designation: CSA N286.7; Previous Edition Published: 1999, Reaffirmed 2007 and 2012*, Date not provided.

[10] OPG Correspondence, P-CORR-03680-0607223 R000, *Pickering PSR2 – Change to Review Type for CSA N290.12*, July 25, 2016.

[11] OPG Nuclear Program, N-PROG-MP-0007 R012, *Conduct of Engineering*, October 2012.

[12] OPG Program, N-PROG-MP-0006 R009, *Software*, April 2015.

[13] OPG Program, N-PROG-MP-0005 R005, *Configuration Management*, June 2012.

[14] OPG Program, N-PROG-MP-0009 R011, *Design Management*, January 2015.

[15] OPG Plan, NK30-PLAN-00531-00001 R005, *Pickering 5-8 Continued Operations Plan*, November 2015.

[16] OPG Procedure, N-PROC-RA-0023 R018*, Fleetview Program Health and Performance Reporting*, August 2013.

[17] OPG Report, NK38-REP-03680-10185 R002, *Darlington NGS Integrated Implementation Plan (IIP)*, April 30, 2015.

[18] OPG List, N-LIST-01300-10000 R008, *Bounded Document Set,* November 11, 2014.

[19] OPG Program, OPG-PROG-0001 R009, *Information Management,* September 24, 2015.

[20] OPG Procedure, OPG-PROC-0179 R001, *Nuclear Quality Assurance Records,* March 14, 2016.

[21] OPG Program, N-PROG-MP-0001 R014, *Engineering Change Control,* January 2015.

[22] OPG Procedure, N-PROC-MP-0090 R012, *Modification Process*, April 22, 2015.

[23] OPG Guide, N-GUID-01130-10000 R001, *Modifications for Beyond Design Basis Accidents,* February 6, 2015.

[24] OPG Procedure, N-PROC-MP-0083 R008, *Constructability, Operability, Maintainability and Safety (COMS)*, June 11, 2014.

[25] OPG Form, N-FORM-10480 R009, *COMS Checklist,* April 22, 2016.

[26] OPG List, P-LIST-06937-00001 R000, *Pickering A and B List of Safety Related Systems,* February 18, 2011.

[27] OPG Standard, N-STD-RA-0033 R002, *Reliability Monitoring and Reporting of Systems Important to Safety*, October 24, 2013.

[28] CNSC RD/GD-98, *Reliability Programs for Nuclear Power Plants,* June 2012.

[29] OPG Standard, N-STD-MP-0016 R002, *Safe Operating Envelope*, June 21, 2012.

[30] OPG Report, NA44-REP-03611-00004 R001, *Pickering A Systems Important to Safety*, January 25, 2008.

[31] OPG Instruction, N-INS-03602-10001 R001, *Preparation of Safe Operating Envelope Compliance Tables*, February 9, 2015.

[32] OPG Report, NK30-REP-03611-00024 R00, *Pickering B Systems Important to Safety,* September 29, 2014.

[33] OPG Report, NA44-SR-01320-00001 R015, *Pickering A Safety Report – Part 2,* July 24, 2012.

[34] OPG Report, NK30-SR-01320-00002 R004, *Pickering B Safety Report – Part 2,* October 10, 2012.

[35] CNSC REGDOC-2.4.1, *Deterministic Safety Analysis,* May 2014.

[36] OPG Report, NA44-REP-03611-00031 R000, *Pickering NGS A Level 1 At-Power Risk Assessment (PARA-L1P) – Fukushima Action Item Update*, February 2014.

[37] OPG Report, NK30-REP-03611-00025 R000, *Pickering NGS B Level 1 At-Power Internal Events Risk Assessment – Fukushima Action Item Update*, February 2014.

[38] OPG Report, NA44-REP-03611-00032 R000, *Pickering NGS A Level 2 At-Power Internal Events Risk Assessment (PARA-L2P) - Fukushima Action Item (FAI) Update*, February 2014.

[39] OPG Report, NK30-REP-03611-00010 R000, *Pickering NGS B At-Power Level 2 Probabilistic Risk Assessment (PRA) for Internal Initiating Events,* December 2012.

[40] OPG Technical Basis Document, N-BDB-03600-00002 R000, *OPG Emergency Mitigating Equipment for Beyond Design Basis Accidents: Technical Basis Document,* October 13, 2015.

[41] Province of Ontario Nuclear Emergency Plan, *Provincial Nuclear Emergency Response Plan Master Plan*, 2009.

[42] OPG Program, N-PROG-RA-0001 R014, *Consolidated Nuclear Emergency Plan,* May 8, 2015.

[43] AECL Report, 44RS-06700-HFR-001 R06, *Human Factors Engineering Summary Report – Pickering A Return to Service,* July 2002.

[44] OPG Instruction, N-INS-08920-10002 R006, *Simulator-Based Initial Certification Examinations for Shift Personnel,* November 27, 2013.

[45] OPG Design Guideline, 44RS-06700-HFP-004 R00, *Human Factors Design Guideline: Maintenance, Inspection and Testing,* June 2001.

[46] OPG Design Guideline, 44RS-66000-HFP-001 R00, *Human Factors Minor Change Design Guideline: Pickering 'A' Main Control Room Unit Panels and Field Control Panels, Task C1-28-1,* September 1999.

[47] OPG Design Guideline, 44RS-66000-HFP-002 R01, *Human Factors Minor Change Design Guideline: Pickering 'A' CRT-Based Displays,* December 1999.

[48] OPG Manual, NA44-MAN-60300-00001 R000, *Annunciation Design Guide Pickering A, Units 1 and 4,* November 25, 2005.

[49] OPG Manual, N-MAN-06700-10000 R001, *Human Factors Design Guideline: Maintenance, Inspection and Testing,* June 25, 2003.

[50] OPG Manual, NK30-MAN-06700-00001 R000, *Human Factors Engineering Design Guideline for Main Control Room Unit Panels and Field Control Panels,* June 27, 2003.

[51] OPG Manual, NK30-MAN-06700-00002 R000, *Human Factors Engineering Design Guideline for CRT Displays,* June 27, 2003.

[52] OPG Manual, NK30-MAN-60300-00001 R000, *Annunciation Design Guide Pickering B, Units 5-8*, November 30, 2005.

[53] OPG Form, N-FORM-10959 R016, *Design Scoping Checklist*, June 27, 2016.

[54] OPG Manual, N-MAN-06700-10002 R004, *Guide for OPG Human Factors Engineering Process,* December 18, 2015.

[55] CNSC Regulatory Guide, G-276, *Human Factors Engineering Program Plans,* June 2003.

[56] CNSC Regulatory Guide, G-278, *Human Factors Verification and Validation Plans,* June 2003.

[57] CNSC REGDOC-2.5.2, *Design of Reactor Facilities: Nuclear Power Plants*, May 2014.

[58] U.S. NRC Guideline, NUREG-0711 R03, *Human Factors Engineering Program Review Model,* November 2012.

[59] U.S. NRC, NUREG-0700 R02, *Human-System Interface Design Review Guidelines,* May 2002.

[60] OPG Report, NK30-REP-03680-00008 R002, *Pickering NGS-B Integrated Safety Review – Management,* September 22, 2009.

[61] OPG Report, NA44-REP-41170-00006 R000, *Human Factors Engineering Summary Report – Pickering NGS Units 1,4 Turbine Governor System Upgrade,* July 9, 2014.

[62] OPG Report, NK30-REP-41220-10002 R002, *Human Factors Engineering Summary Report – PB Main Generator Automatic Voltage Regulator (AVR) Replacement Project,* July 25, 2015.

[63] OPG Safety Report, NA44-SR-01320-00002 R004, *Pickering Nuclear 1-4 Safety Report: Part 3 – Accident Analysis,* October 31, 2013.

[64] OPG Safety Report, NK30-SR-01320-00003 R004, *Pickering Nuclear 5-8 Safety Report: Part 3 – Accident Analysis,* October 30, 2014.

[65] OPG Guide, N-GUID-03611-10001 Volume 1 R04, *OPG Probabilistic Risk Assessment (PRA) Guide – Level 1 (At Power),* October 24, 2014.

[66] CNSC Regulatory Guide, G-323, *Ensuring the Presence of Sufficient Qualified Staff at Class I Nuclear Facilities – Minimum Staff Complement,* July 2007.

[67] OPG Instruction, P-INS-09100-00003 R009, *Pickering Minimum Shift Complement*, December 2014.

[68] OPG Instruction, N-INS-03490-10003 R000, *Minimum Shift Complement Resources, Qualifications and Procedures Required for Responding to Resource Limiting Events,* November 12, 2013.

[69] OPG Program, N-PROG-RA-0013 R009, *Radiation Protection,* January 9, 2015.

[70] OPG Standard, N-STD-RA-0018 R007, *Controlling Exposure As Low As Reasonable Achievable,* November 26, 2014.

[71] OPG Procedure, N-PROC-RA-0054 R015, *Control of Space Allocation for Transient Material and Extended Storage of Material within the Site,* April 10, 2015.

[72] OPG Procedure, N-PROC-MA-0060 R005, *Control of Temporary Shielding,* July 7, 2011.

[73] OPG Instruction, P-INS-09071-00002 R002, *Access Control,* February 11, 2016.

[74] OPG Letter, P-CORR-00531-03719, *Application for Renewal of Pickering Nuclear Generating Station Power Reactor Operating Licence,* July 4, 2012.

[75] OPG Procedure, N-PROC-MA-0024 R015, *System Performance Monitoring,* October 28, 2013.

[76] OPG Procedure, N-PROC-RA-0022 R032, *Processing Station Condition Records,* November 13, 2014.

[77] OPG Program, N-PROG-RA-0003 R010, *Corrective Action,* January 9, 2015.

[78] OPG Report, P-REP-03680-00024 R000, *Pickering 5-8 Continued Operations Plan Review in Support of PNGS Periodic Safety Review 2 (PSR2),* January, 2017.

[79] OPG Report, N-REP-69000-10003 R000, *Gap Analysis Between CSA N290.7-14 "Cyber Security Requirements for Nuclear Power Plants and Small Reactor Facilities"*, March 2016.

Appendix A: Nomenclature

ACU	Air Cooling Unit
ALARA	As Low As Reasonably Achievable
AOO	Anticipated Operational Occurrence
AR	Action Request
ASME	American Society of Mechanical Engineers
AVR	Automatic Voltage Regulator
BDB	Beyond Design Basis
BDBA	Beyond Design Basis Accident
CANDU	CANada Deuterium Uranium
CCS	Concrete Containment Structure
CM	Configuration Management
CNSC	Canadian Nuclear Safety Commission
COMS	Constructability, Operability, Maintainability and Safety
COP	Continued Operations Plan
CRN	Canadian Registration Number
CRT	Cathode Ray Tube (display)
CSA	Canadian Standards Association
CSPM	Critical Safety Parameter Monitoring
DBA	Design Basis Accident
DBE	Design Basis Earthquake
DCC	Digital Control Computer
DEC	Design Extension Condition
ECC	Engineering Change Control
ECIS	Emergency Coolant Injection System
EME	Emergency Mitigating Equipment
EMS	Energy Management System
EST	Edwards System Technology
FADS	Filtered Air Discharge System
FAI	Fukushima Action Item
HELBA	High Energy Line Break Assessment
HFE	Human Factors Engineering

IAEA	International Atomic Energy Agency
IIP	Integrated Implementation Plan
ISR	Integrated Safety Review
L/R/C/S	Laws, Regulations, Codes and Standards
LCH	Licence Conditions Handbook
LISS	Liquid Injection Shutdown System
LOCA	Loss of Coolant Accident
MCR	Main Control Room
MSC	Minimum Shift Complement
NFPA	National Fire Protection Association
NGS	Nuclear Generating Station
NPCS	Negative Pressure Containment System
N-PROG	Nuclear Program
OPG	Ontario Power Generation
PACE	Pickering A Control Room Enhancement
PARTS	Pickering A Return to Service
PEVS	Powerhouse Emergency Venting System
PROL	Power Reactor Operating Licence
PSA	Probabilistic Safety Assessment
PSR	Periodic Safety Review
PSR1	Periodic Safety Review 1 (earlier OPG PSR work and other associated assessments)
PSR2	Periodic Safety Review 2 (subsequent PSR per REGDOC-2.3.3)
RFP	Request for Proposal
RFQ	Request for Quotation
RLC	Review Level Condition
RMS	Radionuclide Management System
RRS	Reactor Regulating System
SA	Severe Accident
SCA	Secondary Control Area
SCR	Station Condition Record
SDSA	Shutdown System A
SDSE	Shutdown System Enhancement
SDS1	Shutdown System 1

SDS2	Shutdown System 2
SIS	Systems Important to Safety
SOE	Safe Operating Envelope
SPDS	Safety Parameter Display System
SSC	Structures, Systems and Components
SSS	Special Safety System

Appendix B: OPG Program Effectiveness Review Results

B.1 N-PROG-MP-0007, "Conduct of Engineering"

The Conduct of Engineering Program provides a framework for performing engineering in a consistent manner across Ontario Power Generation (OPG) Nuclear, which includes programs, standards, procedures and instructions. The program establishes the following practices for engineering:

- Ensures plant configuration is maintained in accordance with the design and licensing bases, and operated within its Safe Operating Envelope (SOE).
- Ensures essential plant and nuclear waste management facility equipment performs safely and reliably.
- Complies with relevant legal, statutory, and regulatory requirements.
- Encourages continuous improvement in the conduct of engineering targeted at achieving safe, reliable, and competitive operation of nuclear power generating stations.

The Conduct of Engineering Program is applicable to all organizations performing engineering activities within Pickering. This includes contractors and design agencies performing engineering activities on behalf of OPG Nuclear unless these organizations are performing these activities in accordance with a Quality Program approved by OPG.

The Plant Design department completed a self-assessment in January 2015, P14-001378-SA [B.1.1], to examine various design quality events at Pickering NGS. The self-assessment concluded that performance improvement opportunities applicable to Pickering NGS existed in the areas of event free tools, vendor acceptance practices and staff technical development.

Two ARs were initiated (ARs 28173508 and 28170781) which required corrective actions to be implemented. These ARs have since been completed and the necessary corrective actions were completed to address the underlying issues.

The Operations and Maintenance Support department completed a self-assessment in March of 2013, NO13-000207-SA [B.1.2], in order to assess the health of N-PROG-MP-0007, "Conduct of Engineering" which is applicable for both Darlington and Pickering NGS. This involved a review of the Governance Framework, SCR database, Asset Suite, revision records and previous program assessment reports. No findings/SCRs were initiated as result of this self-assessment.

Nuclear Oversight conducted a performance based audit in October 2015, NO-2015-032 [B.1.3] for Pickering NGS, in order to assess whether the Margin Management requirements defined in OPG Nuclear governance have been met and effectively implemented to support safe and reliable operation (note, N-STD-MP-0020, "Margin Management" standard is a specific element within N-PROG-MP-0007, "Conduct of Engineering"). The audit concluded that performance improvement opportunities applicable to Pickering NGS existed in the areas of Margin Management process requirements (implementation and effectiveness), and fleet alignment in margin management practices.

Three SCRs were initiated to address the above findings (SCRs P-2015-23785, P-2015-23788 and N-2015-23789) which required corrective actions to be completed. The corrective actions from these SCRs have been completed to address the underlying issues. SCR P-2015-23785 has an open action to complete an effectiveness review by Q1 2017, to confirm the effectiveness of the completed actions.

References

[B.1.1] Self-Assessment Report, *Design Engineering – Common Cause Self-Assessment on Design Quality across all Design Engineering,* P14-001378-SA, January 15, 2015.

[B.1.2] Self-Assessment Report, *Program Management Assessment – N-PROG-MP-0007, Conduct of Engineering*, NO13-000207-SA, March 26, 2013.

[B.1.3] Nuclear Oversight Audit, *Conduct of Engineering – Margin Management*, NO-2015-032 (N-REP-01070-0566861), October 16, 2015.

B.2 N-PROG-MP-0006, "Software"

The Software program identifies processes and overall requirements for the classification of software. Software is classified in order to determine the set of applicable standards and procedures for its custom development, maintenance, acquisition, qualification, use and retirement. The Software program complies with Canadian Standards Association (CSA) N286, "Management System Requirements for Nuclear Facilities" and CSA N286.7, "Quality Assurance of Analytical, Scientific, and Design Computer Programs". The program applies to software classified as Real-Time Process Computing (RTPC) and Scientific, Engineering and Safety Analysis (SESA) Software or Software Engineering Tools in Ontario Power Generation (OPG).

Nuclear Oversight conducted an audit, NO-2010-007 [B.2.1], of the SESA software program in March 2010, to assess program compliance and overall program effectiveness for software quality assurance activities. The audit concluded that performance improvement opportunities applicable to Pickering NGS existed in the areas of Program Oversight, execution of SESA software process requirements and SESA software governance.

Five SCRs were initiated during this audit for Pickering NGS (SCRs P-2010-05827, P-2010-05829, P-2010-05832, P-2010-05834 and N-2010-01450) which required corrective actions to be implemented. These SCRs have since been completed and the necessary corrective actions were completed to address the underlying issues.

Nuclear Oversight conducted a performance based audit of the Software Program in June 2015, NO-2015-013 [B.2.2], applicable to both Pickering and Darlington NGSs. The objective of the audit was to determine whether the Software Program requirements for real-time process computing are effectively implemented. The audit determined that the performance of the managed system controls for the Software Program are effective. Improvement opportunities applicable to Pickering NGS were identified in the areas of procurement planning and quality assurance forms, software maintenance plans and software process requirements.

Three SCRs (SCRs N-2015-14731, N-2015-14733 and N-2015-14734) were initiated which required corrective actions to be implemented. The corrective actions from these SCRs have been completed to address the underlying issues.

Design Engineering completed a self-assessment in August of 2015, NO15-001375-SA [B.2.3], for Pickering and Darlington NGS. The purpose of the self-assessment was to perform an annual high level review of the SESA Software Program health. This included a review of previous self-assessments, audits, training statistics and SCRs. No findings/SCRs were initiated as result of this self-assessment.

References

[B.2.1] Nuclear Oversight Audit, *SESA Software Program*, N-01070-T06 (NO-2010-007), March 19, 2010.

[B.2.2] Nuclear Oversight Audit, *Software Program – Real Time Process Computing,* NO-2015-013 (N-REP-01070-0546332), June 26, 2015.

[B.2.3] Self-Assessment Report, *Self-Assessment: Annual High Level Review of SESA Software Program Adherence*, NO15-001375-SA, August 31, 2015.

B.3 N-PROG-MP-0005, “Configuration Management”

The Configuration Management (CM) program is an integrated management process which ensures that:

- Physical and functional characteristics, operation and maintenance conform with the design and licensing basis; and
- Operating, training, modification and maintenance processes are consistent with the design and licensing basis conditions.

The CM program applies to:

- Facility physical configuration, supporting hardware, and software, including: station structures, systems and components (SSCs), waste management facilities, training simulators, engineered tools, nuclear fuel and station process computers.
- Policies, programs and procedures which contain information that could impact the design and licensing basis, physical configuration or any configuration item or information.
- Staff that support operation and preservation of OPG assets (e.g., site staff, Nuclear Engineering, Nuclear Programs and Training, Performance Improvement, Nuclear Operations and contract service providers).

Nuclear Oversight conducted a performance based audit of the CM program in December 2015, NO-2015-033 [B.3.1], for Pickering and Darlington. The purpose of the audit was to determine whether the CM program requirements have been met and effectively implemented to support plant safety and reliability. Performance improvement opportunities were identified in the areas of CM Program governance and implementation.

Two SCRs were initiated (N-2015-29326 and N-2015-29328) which required corrective actions to be implemented. These SCRs have since been completed and the necessary corrective actions were completed to address the underlying issues.

Engineering Mechanics completed a self-assessment in October 2015, N015-000085-SA [B.3.2], in order to assess the health of the Design Management and Configuration Management governance framework, which is applicable for both Darlington and Pickering NGS. This involved a review of SCR databases, operating experience (OPEX) from industry, review of previous design/configuration management self-assessments and code changes versus governance changes. Minor clarifications/insights were proposed as a result of this self-assessment and an Action Request (AR) was generated (AR 28181695). All assignments from this AR have been completed, and the necessary corrective actions were completed to address the underlying issues.

References

[B.3.1] Nuclear Oversight Report, *Configuration Management*, N-REP-01070-0571299 T06 (NO-2015-033), December 2015.

[B.3.2] Self-Assessment Report, *Self-Assessment Related to the Health of Design Management and Configuration Management Governing Documents*, N015-000085-SA, October 2015.

B.4 N-PROG-MP-0009, "Design Management"

The Design Management Program specifies requirements for the following:

- Management of prescribed activities appropriate for execution and control of required design, design support, and documentation for nuclear facilities and organizations owned by Ontario Power Generation (OPG) Nuclear.
- Processes for creating or modifying documentation required for controlling the design basis and design outputs.
- The minimum set of documentation that identifies and describes the design basis, design output, and design process.
- Procurement Engineering processes ensuring implementation and maintenance of the physical nuclear facilities meet the design basis requirements.

The Design Management Program complies with both Canadian Standards Association (CSA) N286, "Management System Requirements for Nuclear Power Plants", as well as CSA N285.0, "General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants" and N-MAN-01913.11-10000, "Pressure Boundary Program Manual", which specifies general requirements for design of pressure retaining systems, components, and their supports, and with complementary standard CSA B51.

Engineering Mechanics completed a self-assessment in October 2015, N015-000085-SA [B.4.1], in order to assess the health of the Design Management and Configuration Management governance framework, both of which are applicable to Darlington and Pickering NGS. This involved a review of SCR databases, operating experience (OPEX) from industry, review of previous Design/Configuration management self-assessments and code changes versus governance changes. The self-assessment concluded that the Design Management Program is in compliance with the applicable codes and licence conditions. AR# 28181695 was initiated to address minor clarifications/insights, which has since been completed.

Nuclear Oversight conducted a performance based audit of the design management program in June 2015, NO-2015-018 [B.4.2], in order to evaluate the level of compliance and effectiveness of the Design Control element of the Pressure Boundary Program Manual and assess the alignment between the Pressure Boundary Program Manual and design control governance. The audit determined that the Design Control elements of the Pressure Boundary Program Manual are effective and in compliance with all aspects of the QA program, and associated codes and standards. No findings were generated as a result of this audit.

Per P-REP-01914-00004, "Pickering Design Quality Assurance Program Review Years 2011-2014" [B.4.3], Pickering NGS carried out an assessment of the effectiveness of the quality management system as it pertains to Pickering Design Engineering, for the period of 2011 through 2014. The assessment also included a review of the effectiveness of corrective actions from the previous program review as well as other self-assessments, reviews and audits performed during the period. The assessment identified the following recommendations:

- Establish a Design Human Performance working group to continuously improve the effectiveness of the design, application and use of Event Free Tools (complete).

- Establish a process to systematically address knowledge transfer. A guide, along with a list of potential training courses for Design Engineers, to be used in conjunction with existing tools for employee development has been prepared and implemented. This information is used with design staff to assist in development planning (complete).

- Improve the clarity of action closure notes. Although this action is still open, the identified gap is administrative in nature and is not significant in the context of PSR2.

References

[B.4.1] Self-Assessment Report, *Self-Assessment Related to the Health of Design Management and Configuration Management Governing Documents*, N015-000085-SA, October 2015.

[B.4.2] Nuclear Oversight Report, *Pressure Boundary Design Control and Procurement Engineering,* NO-2015-018 (N-REP-01070-0547043), June 26, 2015.

[B.4.3] OPG Report, *Pickering Design Quality Assurance Program Review Years 2011-2014,* P-REP-01914-00004 R000, May 7, 2015.

PICKERING NGS PSR2 SAFETY FACTOR 2 REPORT - ACTUAL CONDITION OF STRUCTURES, SYSTEMS, AND COMPONENTS IMPORTANT TO SAFETY

Document No. 705-1680051200-G0012-06

OPG Document No. P-REP-03680-00005 R01

ONTARIO POWER GENERATION

Pickering NGS PSR2 Safety Factor 2 Report - Actual Condition of Structures, Systems, and Components Important to Safety

Document No. P-REP-03680-00005 R01

Document No. 705-1680051200-REP-G0012-06

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Doc No.: P-REP-03680-00005 Rev: 1	
This acceptance does not relieve the contractor from responsibility for errors or omissions or from any obligations or liability under this contract.	

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REVISION HISTORY

REV. NO	ISSUE DATE	PREPARED BY AND DATE	REVIEWED BY AND DATE	APPROVED BY AND DATE	DESCRIPTION OF REVISION
00	May 6, 2016	MM/MD May 6, 2015	VP/DS May 6, 2016	SK May 6, 2016	For OPG Review. Note: Placeholders have been left where there are pending inputs from OPG: 1. Section 4.1.1.1.5 Service Limits Assessment of Nuclear Class 1 Components. 2. Section 4.1.7 for Review Task #7 on Dependence of Dependence on External Essential Services/Supply.
01	June 22, 2016	DS/MD June 22, 2016	VP June 22, 2016	SK June 22, 2016	For OPG Acceptance.
02	July 14, 2016	DS/MD July 14, 2016	VP July 14, 2016	SK July 14, 2016	This revision was issued as R000 of OPG document P-REP-03680-00005
03	February 3, 2017	PW Feb.3, 2017	DS Feb.3, 2017	SK Feb.3, 2017	Draft OPG R01 Issued to OPG. Updated to Reflect Roadmap for OPG SF2 Revision.
04	March 3, 2017	PW/WM March 3, 2017	DS March 3, 2017	SK March 3, 2017	For OPG Acceptance.
05	March 6, 2017	PW/WM March 6, 2017	DS March 6, 2017	SK March 6, 2017	For OPG Acceptance.
06	March 7, 2017	PW/WM March 7, 2017	DS March 7, 2017	SK March 7, 2017	This revision will be issued as R001 of OPG document P-REP-03680-00005.

This revision history is applicable to Tetra Tech Report # 705-1680051200-REP-G0012-06.

Executive Summary

Introduction and Background

Ontario Power Generation (OPG) is performing a Periodic Safety Review (PSR) to support the possibility of continued operation of Pickering Nuclear Generating Station (NGS) beyond 2020. The PSR (referred to as "PSR2") is a subsequent PSR building on the review basis of earlier OPG Integrated Safety Reviews and other associated assessments. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

Part of PSR2 involves the preparation of Safety Factor reports for each of fifteen major topic areas. Safety Factor reports consist of:

- Assessing compliance against "Review Tasks" identified in the PSR2 Basis Document [1]. These Review Tasks are derived from Canadian Nuclear Safety Commission (CNSC) REGDOC-2.3.3, "Periodic Safety Reviews" [2] and International Atomic Energy Agency (IAEA) SSG-25, "Periodic Safety Review for Nuclear Power Plants" [3];

- Documenting assessments against applicable modern Laws, Regulations, Codes and Standards, as defined in Reference [1]; and

- Reviewing the effectiveness of applicable OPG Nuclear Programs that support the above assessments, through review of audit and self-assessment results.

The PSR2 review of Safety Factor 2 (SF2), Actual Condition of Structures, Systems, and Components (SSCs) Important to Safety, is presented in this report. It presents the results of the assessment of the Actual Condition of SSCs Important to Safety at Pickering NGS against Review Tasks as outlined in PSR2 Basis Document [1]. All results of reviews of Laws, Regulations, Codes and Standards and effectiveness reviews of OPG Programs applicable to SF2, are documented in the Safety Factor 4 Report [4].

Scope

There are six Review Tasks associated with the SF2 Review of Actual Condition of SSCs Important to Safety as summarized below (see Section 2.0 for detailed task descriptions):

1. Assess and document the actual condition of station SSCs Important to Safety.
2. Confirm facilities and resources are available for ongoing plant maintenance.
3. After completing the Condition Assessments (CAs), confirm that the design basis assumptions have not been significantly challenged and will remain that way throughout the PSR2 period.

4. Review the condition and operation of the Spent Fuel Storage Facilities.
5. Assess dependence on obsolete equipment for which no direct substitute is available.
6. Assess dependence on essential services and/or supplies external to the plant.

Although not considered a Review Task, the scope also includes a review of Emergency Mitigating Equipment (EME) required to mitigate the consequence of Beyond Design Basis Events (BDBEs).

This review considers continued commercial operation of Pickering Nuclear Generating Station up to 2028. The scope of the review does not address plant states after commercial operation is completed, e.g. final defueling.[1]

Findings and Conclusions

The assessments of the six Review Tasks and additional required reviews have been completed. As documented in this report, these assessments conclude that the majority of the plant SSCs are in good condition and support extended station operation to 2024. This conclusion is supported by the comprehensive and effective set of plant programs in place to ensure the condition of components meet design requirements with margin.

Recommendations for improvement have been made when required, many of which are in progress. For this life extension period, no major concerns have been identified and the SSCs Important to Safety continue to operate as per the design basis requirements. Recommendations to improve Aging Management practices have been made. Additional assessments need to be completed to fully assess life extension to 2028. This required work is documented in the SF2 gaps below.

The condition of Major Components, consisting of Fuel Channels, Feeders, Steam Generators, and Reactor Components and Structures is managed by rigorous Life Cycle Management Plans (LCMPs). There is high confidence that the major components will remain fit for service up to an extended station life to 2024, with limited potential mitigating actions required. However, additional analysis and assessment is required to demonstrate continued Fitness for Service for extended station life to 2028.

The Condition Assessments (CAs) discussed in this report support that for the balance of SSCs Important to Safety, including the Irradiated Fuel Bays, the majority of components (~93%), are rated as Satisfactory or better using the classification criteria described. For the Special Safety Systems there are only two components which are

[1] These post-shutdown phases, i.e. defueling, de-watering and transferring fuel from the IFBs to dry storage, are being considered as part of the Stabilization Activity Plan (SAP). Updated condition assessments are being prepared to cover these phases.

rated Poor. However, as discussed in this report, either their classification has been re-assessed as Satisfactory, or they don't have a safety function in the system. Updated CAs are being completed as part of OPG's ongoing aging management program. The need to complete these CAs is a SF2 gap.

Programs are in place to address required station maintenance and equipment obsolescence. The Irradiated Fuel Bays and supporting equipment are generally in good condition and are able to support the station's spent fuel storage strategy. Also, no vulnerabilities have been identified in obtaining external services and supplies.

The following eighteen gaps need to be addressed further as part of the PSR2 Global Assessment Report and Integrated Implementation Plan process:

- **Gap SF2-1:** Fitness for Service for Fuel Channels has not been demonstrated for station operation to 2028.
- **Gap SF2-2**: OPG does not have approval to operate beyond the current Licence limit of 247,000 Effective Full Power Hours (EFPH) for fuel channels.
- **Gap SF2-3**: The Fuel Channels LCMP has not been formally updated to address extended station operation to 2028.
- **Gap SF2-4**: Fitness for Service for Feeders has not been demonstrated for station operation to 2028.
- **Gap SF2-5**: The Feeders LCMP has not been formally updated to address extended station operation to 2028.
- **Gap SF2-6:** Fitness for Service for Steam Generators has not been demonstrated for station operation to 2028.
- **Gap SF2-7**: The Steam Generators LCMP has not been formally updated to address extended station operation to 2028.
- **Gap SF2-8:** Fitness for Service for Reactor Components and Structures has not been demonstrated for station operation to 2028.
- **Gap SF2-9:** The Reactor Components and Structures LCMP has not been formally updated to address extended station operation to 2028.
- **Gap SF2-10:** Environmental Factors have not been incorporated into the Service Limits Assessment for Class 1 piping.
- **Gap SF2-11:** Condition Assessments for civil structures are not complete for station operation to 2028.
- **Gap SF2-12:** Condition Assessments for in-scope piping systems are not complete for station operation to 2028.
- **Gap SF2-13:** The Cable Surveillance Program risk assessment and condition assessments currently use out of date criticality coding.
- **Gap SF2-14:** The Buried Piping Program risk assessment and condition assessments have not been updated for extended operation to 2028.
- **Gap SF2-15**: Updated Detailed Condition Assessments are not complete for Commodity Groups in the scope of PSR2 for station operation to 2028.
- **Gap SF2-16:** Action plans to correct the leakage in IFB-B are not complete.

- **Gap SF2-17:** The seismic capacity of the current spent fuel basket stacking arrangements in the Pickering IFBs needs to be documented.
- **Gap SF2-18**: The seismic capacity of the Pickering 058 IFB conveyer tunnel needs to be documented.

TABLE OF CONTENTS

1.0 INTRODUCTION

Ontario Power Generation (OPG) is performing a Periodic Safety Review (PSR) to support the possibility of continued operation of Pickering Nuclear Generating Station (NGS) beyond 2020.[2] A comprehensive Integrated Safety Review (ISR) was completed for Pickering Units 5 through 8 in 2009 in support of refurbishment and continued operation. Pickering Units 1 and 4 integrated safety assessments were also performed for Pickering A Return to Service (PARTS) in support of approval to restart these units. In addition to these Pickering-specific studies, the 2013 Darlington ISR performed extensive code and standard reviews that were updated in relation to the versions that were assessed in the 2009 Pickering B ISR. These previous ISRs are considered to constitute the first PSR completed for Pickering (referred to as “PSR1”). The current PSR (referred to as “PSR2”) is a subsequent PSR building on the basis of earlier OPG integrated safety assessments through review of the various studies, assessments and licence renewals performed since PSR1. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

PSR2 will support and complement the licence renewal application for Pickering NGS going forward. Fifteen Safety Factors will be assessed as part of the PSR. The purpose of Safety Factor reviews is to confirm that the design, condition and operation of Pickering Units 1, 4 and 5-8 (as well as common systems) will support continued safe operation for the period of PSR2 by:

- Assessing compliance against “Review Tasks” identified in the PSR2 Basis Document [1], which were derived from Canadian Nuclear Safety Commission (CNSC) REGDOC-2.3.3 and International Atomic Energy Agency (IAEA) SSG-25 [2];

- Documenting assessments against applicable modern Laws, Regulations, Codes and Standards, as defined in Reference [1];

- Reviewing the effectiveness of applicable OPG Nuclear Programs that support the above assessments, through review of audit and self-assessment results.

The PSR2 review of Safety Factor 2 (SF2), Actual Condition of Structures, Systems, and Components (SSCs) Important to Safety, is presented in this SF2 report. It

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2 Pickering Units 5-8 are currently approved to operate to 247,000 Effective Full Power Hours [25]. This operation limit is expected to be reached on some units in 2020. For the purposes of PSR2, OPG assumes operation of Pickering NGS for up to eight additional years, from 2020 until 2028.

represents the results of the assessment of the Actual Condition of SSCs Important to Safety at Pickering NGS against Review Tasks as outlined in PSR2 Basis Document P-REP-03680-00001 R02 [1].

As outlined in IAEA SSG-25 [3], the objective of the review of Safety Performance Safety Factor 2 is to: “determine the actual condition of SSCs important to safety and so to consider whether they are capable and adequate to meet design requirements, throughout the period of PSR2. In addition, the review should verify that the condition of SSCs important to safety is properly documented, as well as reviewing the ongoing maintenance, surveillance and in-service inspection programmes, as applicable.” REGDOC-2.3.3 [2] requires that: “The licensee shall conduct a PSR in accordance with this regulatory document for the period until the next PSR or, if applicable, until the end of commercial operation of the plant.”

Per the Pickering PSR2 Basis Document [1], analysis of gaps and potential safety enhancements for Pickering NGS (including identification of improvements that are reasonable and practicable to implement) is addressed as part of the Global Assessment process. Preparation of a plan for the implementation of safety enhancements is addressed by the PSR2 Integrated Implementation Plan.

2.0 SCOPE OF REVIEW

The scope of the review was based on Review Tasks as documented in the PSR2 Basis Document P-REP-03680-00001 R02 [1]. Generally, the scope of the Safety Factor reports includes reviews against (i) Review Tasks, (ii) Modern Laws, Regulations, Codes and Standards in the PSR2 Assessment Basis, and (iii) effectiveness reviews of OPG Programs applicable to the Safety Factor.

For Safety Factor 2:

- OPG governance, the PNGS design and the actual condition of the plant were assessed to determine the level of compliance with the requirements of each Review Task;
- All results of reviews of Laws, Regulations, Codes and Standards in the PSR2 Assessment Basis applicable to SF2 are documented in the Safety Factor 4 Report [4]; and
- Effectiveness reviews of OPG Programs applicable to Safety Factor 2 are documented in the Safety Factor 4 Report[3] [4].

The Review Task assessments identify Compliances and Gaps as defined below:

- Compliance: Compliance indicates that either the safety requirement or the intent of the Review Task is met.
- Gap: A Gap indicates that the intent of the Review Task is not met.

This review conservatively considers continued operation of Pickering Nuclear Generating Station to 2028. Consistent with the PSR2 Basis Document [5], a freeze-date of January 15, 2016 has been utilized to complete the assessment. All information inputs, e.g. system health reports, used in this assessment are aligned with this date.

It was recognized that, although the period of the PSR extends to 2028, the end of commercial operation is expected to occur prior to that. As such, the groups that are separately planning the life extension and the transition to Safe Storage are collaborating to ensure the plant condition assessment (i.e., aging management plans) adequately address the transition stages.

[3] As documented in Section 4.4, some component aging issues are contained in Safety Factor 4.

OPG's Aging Management program takes into consideration the long term aging management assessments and transition to decommissioning requirements. This is achieved by proactively identifying the plant's Systems, Structures and Components (SSCs) that will also be required to perform for years after the shutdown date. For example, items supporting the irradiated fuel bays will have to operate at least 10 years after shutdown, while some other SSCs may be needed even longer. As a result, the extended operations group has developed a System Transition Boundary Report [96], which documents the required lifespan of the various systems of the plant. This report includes input from the decommissioning/safe storage team, to ensure it reflects the post-shutdown aging management requirements, and they are being documented in the System End State Determination Reports.

2.1 REVIEW TASKS

The PSR2 Basis Document (P-REP-03680-00001 R02) [1] identifies that the Review Tasks for each Safety Factor Report are an interpretation of requirements and guidance set out in IAEA Safety Guide SSG-25 'Periodic Safety Review for Nuclear Power Plants' [3] and CNSC REGDOC-2.3.3 [2]. The Review Tasks were chosen to fulfil the requirements of IAEA SSG-25 [3], and also to retain alignment to the extent practicable with earlier Review Tasks from the Darlington and Pickering B ISRs [11] [12]. These previous ISRs were completed in accordance with CNSC RD-360, Life Extension of Nuclear Power Plants [13] and IAEA NS-G-2.10, Periodic Safety Reviews of Nuclear Power Plants [14], which have since been superseded. By keeping the structure and grouping of similar SSG-25 [3] Review Tasks as close as possible to previously used Review Tasks, the review effort can more effectively focus on safety significant differences including changes in requirements, plant conditions, operating experience and new information. This focus is consistent with the intent of a subsequent PSR as described in REGDOC-2.3.3 [2]. Details of the full alignment of IAEA SSG-25 [3] with the PSR2 Review Tasks are provided in the PSR2 Review Task Definition Report P-REP-03680-00003 [15].

The Objectives and Review Tasks for Safety Factor Report 2 (SF2) are defined in the PSR2 Basis Document R02 [1] as follows:

Objective: The objective of the review of this Safety Factor is to determine the actual condition of SSCs important to safety and to assess whether they are capable and adequate to meet design requirements, throughout the period of PSR2. In addition, the review should verify that the condition of SSCs important to safety is properly documented, as well as reviewing the ongoing maintenance, surveillance and in-service inspection programmes, as applicable.

Review Tasks:

1) Assess and document present conditions of the SSCs important to safety and confirm appropriate measures to address any significant existing or anticipated aging degradation are in place. Any major difference between operating units with respect to aging degradation mechanisms, present condition, or recommended actions shall also be presented.

2) Confirm resources and facilities (on and off site) are available for ongoing plant maintenance.

3) After determining the actual condition of SSCs important to safety, each of these SSCs will be assessed against the current design basis to confirm that design basis assumptions have not been significantly challenged and will remain that way throughout the period of PSR2.

4) Review the condition and operation of spent fuel storage facilities and their effect on the spent fuel storage strategy for Pickering NGS.

5) Assess dependence on obsolescent equipment for which no direct substitute is available.

6) Assess dependence on essential services and/or supplies external to the plant.

2.2 EMERGENCY MITIGATING EQUIPMENT REVIEWS

OPG has established Emergency Mitigating Equipment (EME) at its nuclear power stations to support the station response to Beyond Design Basis Accidents (BDBA). EME is not considered a Safety Related System at OPG, however, OPG has established EME procedures and practices commensurate with the role of the equipment in support of BDBA response. An assessment of the OPG procedures and practices in support of EME and the Pickering NGS EME equipment is included as part of the scope of SF2. This assessment, which was completed by OPG [70], addresses the technical basis, functional requirements, maintenance and testing applicable to EME. A summary of the report’s findings and conclusions is included within this report.

2.3 Additional Reviews

The PSR2 Safety Factor 2 report includes a review of the R04 Pickering Licence Conditions Handbook (LCH) [25] for any impacts of Pickering NGS operation beyond 2020 on the following (related to Safety Factor 2):

- OPG commitments previously made to the CNSC;
- Open CNSC action items; and
- Exemptions granted by the CNSC.

The review of Pickering PSR1 gaps previously identified in the Pickering 5-8 Continued Operations Plan (COP) is provided in a separate PSR2 COP Review Report [88].

Fukushima Action Items (FAIs) were reviewed to identify implications of extending operation beyond 2020 (if any). This review is presented in a separate PSR2 FAI Review Report, P-REP-03680-00022 R00, Fukushima Action Item Review in Support of PNGS Periodic Safety Review 2 (PSR2) [81].

Any PSR2 gaps identified as a result of the Safety Factor 2 review which are relevant to other Safety Factors are discussed in Section 4.4 of this report.

3.0 METHODOLOGY

3.1 Review Task #1 – Actual Condition of SSCs

Review Task #1 is to assess and document the actual conditions of the SSCs important to safety and confirm appropriate measures are in place to address any significant existing or anticipated aging degradation. Any major differences between operating units with respect to aging degradation mechanisms, present condition, or recommended actions are also presented.

The two primary sources of information included in this review are Detailed Condition Assessments (DCAs) and Major Component Life Cycle Management Plans (LCMPs). The LCMPs are prepared and executed to assure the ongoing fitness for service of Fuel Channels, Steam Generators, Feeder Piping, and Reactor Components and Structures. Detailed Condition Assessments are prepared for the balance of Systems, Structures and Components (SSCs) deemed as being AM critical.

The Detailed Condition Assessments and Major Component Life Cycle Management Plans are not documented within this report, however they are Controlled Documents following OPG's well established and robust aging/asset management programs.

OPG's Integrated Aging Management (IAM) Program defines the method of condition assessment for plant SSCs. Documented detailed condition assessments are not required for all SSCs. As per Appendix A in N-PROG-MP-0008 [34], condition assessment is managed by either LCMPs, Condition Assessments (CAs), and/or by System and Component Surveillance. Not all plant SSCs require an assessment of condition to be documented in a DCA. Only those SSCs not-screened out via the aging management process (N-PROC-MP-0060 [7]) require a detailed condition assessment. The condition of screened-out SSCs is addressed by System Surveillance, N-PROC-MA-0024, "System Performance Monitoring" [93] and Component Surveillance, N-PROG-MA-0017, "Components and Equipment Surveillance" [87] and other programs. This is elaborated on further below in the section on the Treatment of SSCs not Requiring a Detailed CA.

An overview of OPG's IAM Program, N-PROG-MP-0008, "Integrated Aging Management" [34] and how it addresses the present condition of SSCs is provided in Appendix C. The Appendix also describes how the IAM program interfaces with and supports the Equipment Reliability Program, N-PROG-MA-0026, "Equipment Reliability" [80].

The following section describes how OPG's Aging Management governance has been applied for the completion of the condition assessments for PSR2.

Detailed Condition Assessments

The Condition Assessment process is defined within N-PROC-MP-0060, “Aging Management Process” [7] as well as P-GUID-01060-10000, “Condition Assessment Preparation Guide” [8]. The process is comprised of three primary steps: (i) Scoping (ii) Screening and (iii) Condition Assessment.

Scoping:

N-PROC-MP-0060 describes the detailed steps followed to derive the scope of OPG’s Aging Management Program. These steps define the boundary of System, Structures and Components (SSCs) that are considered within the program. Appendix C provides a full description of the process. The end result is a scope that includes the majority of station SSCs contained in the station’s Master Equipment List (MEL), encompassing most important station systems (both safety related and production important). Both critical and non-critical SSCs are included, with similar components combined into Commodity Groups (CGs). Component criticality is discussed further in the description of Screening. Both active and passive (e.g. structures, piping) components are included in the scope as required in governance.

From this broad AM scope for the Pickering station, the scope of PSR2 has been defined. In completion of SF2 Review Task #1, this scope is refined down to address SSCs Important to Safety. This is first done at the system level. The scope of the systems within the scope of PSR2 encompasses the Pickering Safety Related Systems, documented in P-LIST-06937-00001 [16] with a focus on Pickering Systems Important to Safety (SIS) (listed in OPG reports NA44-REP-03611-00004 [17] and NK30-REP-03611-00024 [18]), and Safe Operating Envelope (SOE) Systems (listed in N-INS-03602-10001 [19]).

OPG defines Safety Related Systems as those systems and the components and structures thereof, which by virtue of their failure to perform in accordance with the design intent, have the potential to impact on the radiological safety of the public or plant personnel (as defined in P-LIST-06937-00001 [16]). Safety Related Systems are associated with the provision of the following safety related functions: shutdown and regulating (Control), cooling of the reactor core (Cool), and limiting the release of radioactive material and exposure of plant personnel and/or the public during normal operation and accident conditions (Contain). Systems used for critical monitoring functions are also included.

Systems Important to Safety, a sub-set of the Safety Related Systems, are defined following OPG governance standard N-STD-RA-0033 [20], considering the risk importance of system and utilizing expert review panels to select these systems. The identification of SIS is consistent with the requirements of CNSC RD/GD-98 [21].

Safe Operating Envelope (SOE) systems, a sub-set of the Safety Related Systems, are identified per OPG standard N-STD-MP-0016 [22], and it consist of systems and their associated critical components and structures, for which operational safety requirements are specified to conform with the Pickering A and B Safety Reports [23] [24]. OPG has prepared formal Operational Safety Requirements (OSR) documents in support of SOE systems. The SOE systems and associated OSRs are listed in OPG instruction N-INS-03602-10001 [19] and are also listed in the Licence Condition Handbook [25].

The Pickering NGS Units 1,4 SOE/SIS systems included in the PSR2 assessment are listed in Appendix B.1 Table B.1-1 The Pickering NGS Units 5-8 SOE/SIS systems included in the PSR2 assessment are listed in Appendix B.1 Table B.1-2. Per Reference 7, Critical Structures (e.g. Reactor Buildings, Pressure Relief Duct and Vacuum Building) are also addressed in the review.

Within these PSR2 systems, system components having common attributes, e.g. type, criticality, are then arranged into Commodity Groups (CGs). The next step in the CA process is to screen these CGs to determine which are to be subject to detailed condition assessment.

Screening:

The objective of the aging management screening process is to review the large number of SSCs (greater than 500,000 for Pickering) in the scope of the program, and by employing a systematic process using defined criteria to determine which SSCs should be subject to detailed condition assessment. Per OPG Aging Management governance [7], to determine the condition of screened-out SSCs does not require an in-depth condition assessment, but rather, their condition is assessed and managed by other processes in the AM program, e.g. the Equipment Reliability program including system performance monitoring. The treatment of screened-out components (not requiring a Detailed CA) is discussed further below in the section on the “Treatment of SSCs not Requiring a Detailed CA”.

Critical components in the in-scope systems are included in the aging management scope for PSR2. Component Criticality is defined in N-PROC-MA-0077 R006, “Critical Equipment Identification and Categorization” [9]. Using this procedure, components are assigned criticality codes CC1, 2, 3 or 4. Criticality sub-codes are also assigned in the areas of Reactor Safety (RS), Production (P) and Cost, Conventional Safety and Environmental (CCSE). CC1 and 2 components are “critical” components and CC3 and 4 are “non-critical”.

CC1 and 2 components are included in the scope of aging management. For PSR2, CC3/RS3 components are also included. RS3 components are non-critical components having a lower level of importance in reactor safety function, i.e. (i) Components in an OSR system that are also in a non-SIS, whose failure results in a partial loss of redundancy impairment condition, or (ii) is associated with safety component testing. A partial loss of redundancy results when a safety system component is unavailable, however redundancy in the safety function is still present. Other non-critical components can also be added to the scope as requested by the system engineer. Critical structures are not always defined a criticality code and therefore they are included in the scope as per instructions in N-PROC-MP-0060 [7].

The next step of the screening process requires that a preliminary assessment be performed which collects pertinent information needed to conduct further screening. The information collected for each SSC is: (i) Aging Related Degradation Mechanisms (ARDMs), i.e. modes or processes resulting in degradation of the component, e.g. corrosion and (ii) Aging Management Practices (AMPs), i.e. the methods in place to detect and manage component aging.

The objective of the remaining steps in the process is to identify components which require a detailed Condition Assessment. These screening steps are described further in Appendix C. Before describing the condition assessment process used for screened-in components, a description is provided of the methods of managing aging used for screened-out components, i.e. those not requiring a Detailed CA.

Treatment of SSCs not Requiring a Detailed CA:

The condition of screened out components is managed on an ongoing basis via OPG's Equipment Reliability (ER) program and other supporting programs. OPG's ER program is aligned with best industry standards and is comprised of a set of processes whose objective is to ensure that the reliability of systems and components is managed on an ongoing basis, including ensuring that all nuclear safety requirements are met.

Example Supporting Equipment Reliability processes are:

(i) The Corrective Action Program, "Corrective Action", N-PROG-RA-003 [92], executed to identify adverse trends in performance or component failures and put corrective actions in place to prevent re-occurrence of the adverse condition;
(ii) System Performance Monitoring, "System Performance Monitoring", N-PROC-MA-0024 [93] which requires surveillance, tracking, reporting on overall health and preparation of System Health Action Plans to improve system health and component condition; and
(iii) The Component and Equipment Surveillance Program, N-PROG-MA-0017 [87], which addresses a number of different types of components, e.g. Power Operated Valves, Buried Piping, Cables, Heat Exchangers, etc. and;
(iv) The Preventative Maintenance Program, "Conduct of Maintenance", N-PROG-MA-0004 [94] which uses component operating history to optimize component performance and maintenance practices via PM feedback mechanisms and conducts the required maintenance on components. Work reports document the observed condition of equipment subjected to maintenance

Ongoing assessment, monitoring and the documenting information on the condition of station systems and components is conducted per the ER program.

With respect to the documentation of the condition of screened-out components, the objective of screening is not to assign a condition or classification for screened out components. As per N-PROC-MP-0060 [7], this is only required for components for which a detailed condition assessment is performed. However, during the preliminary assessment of component aging used in the screening process, the condition of AM-critical components is reviewed based on operating history, system health reports and other data sources. This information is documented in the System Screening Reports.

In addition, a number of comprehensive programs are in place which document component condition including: System and Component Health Reporting; The Maintenance Program which documents as-found condition of components; Predictive Maintenance, e.g. vibration monitoring; the Corrective Action Program, documenting adverse conditions on equipment in SCRs; Annual Reliability Reports; Design Assessments and many other station processes.

Detailed Condition Assessment:

Aging Management Critical components not screened out are subject to Detailed Condition Assessment (CA). These assessments involve:

a) Identifying and understanding component Age Related Degradation Mechanisms (ARDMs);
b) Collecting data to evaluate the degree of degradation experienced to date, e.g. SHR data, OPEX, SCRs, etc.;
c) Documenting Aging Management Practices (AMPs) in place to mitigate aging.
d) Assessing the adequacy of the AMPs;
e) Evaluating component condition by comparing experienced degradation against established limits; and
f) Establishing actions required to minimize and control Aging Related Degradation Mechanisms (ARDMs) and improving condition.

All of the information above is documented in a Detailed CA. An overall Condition Classification (Very Good, Good, Satisfactory, Poor, and Very Poor) is defined per the criteria in N-PROC-MP-0060 [7] (also provided in Appendix C), which accounts for:

a) The physical condition of the component at the time of assessment, and
b) The adequacy of the practices in place to manage component aging.

Condition Classification is assigned by selecting the limiting of these two criteria. For example if the physical condition of a component is “Good”, but the adequacy of practices is rated as “Satisfactory”, the component condition classification is rated as “Satisfactory”.

Per N-PROC-MP-0060 [7], recommendations for improvement are required for components with a “Poor” or “Very Poor” classification. These are captured as actions in the CAs. In many cases, recommendations are also made for “Satisfactory” or better rated components to maintain or improve this classification. However, these are not essential per governance. AM actions are captured in system health and component health reports. These actions are assessed on an on-going basis and are prioritized and tracked to completion via SHR action plans. OPG plans to implement a new Aging Management database to aid with the tracking and oversight of AM actions.

Review Task #1 also requires the following:

> *“Any major difference between operating units with respect to aging degradation mechanisms, present condition, or recommended actions shall also be presented.”*

Detailed CAs are prepared separately for Pickering Units 1, 4 and 5-8. These separate CAs also address equipment tags for all of the operating units and any differences between units in component condition, OPEX, practices or other factors and resulting recommendations are identified. Components in Units 2 and 3 are in scope if they play a role in supporting operating units, e.g. specific Unit 2 Class III 4.16 kV buses (there are a limited number of Unit 2 and 3 components in scope).

For this review, the contents of existing OPG approved Detailed CAs from screening were reconciled up to the PSR2 Basis Document R00 [5] freeze-date of August 31, 2015. Relevant plant information was reviewed (e.g. SCRs, Health Reporting, Work Orders (WOs), etc.) to establish the condition of each CG as of the freeze date. The Detailed CAs include recommendations (where needed) to reach Plant EOL (2020) and for an additional four years addressing continued operations to 2024.

The primary inputs to Detailed CAs are System and Component Health Reports from the OPG System, Plant, and Program IQ databases. System Health reports are designated by system, year and quarter. Work Orders, Preventative Maintenance Instructions and Station Condition Records were reviewed on as a required basis.

In addition to the above, the Detailed CAs are currently being updated to:

- Use an updated freeze date of January 15, 2016.
- Incorporate updated scoping and screening work. The scoping and screening is being updated to take into account work performed to review and revise component criticality and reflect the extended station operation to 2028
- Include full power operation of the Pickering Units to 2028.
- Address the different phases of permanent station shutdown, e.g. defueling. Although, this aspect of the CAs is not in the scope of PSR2.

This updated condition assessment work is being performed as part of OPG's ongoing IAM Program work, which routinely updates condition assessments as required and is not documented in this SF2 report. The preliminary results of these updated CAs is provided in Section 4.1.1.2.48.

Major Components

OPG maintains Life Cycle Management Plans (LCMPs) for major component groups as per OPG Major Components Program N-PROG-MA-0025, "Major Component" [26].

A review of the Major Components LCMPs (Fuel Channels, Feeder Piping, Steam Generators, and Reactor Components and Structures) Fitness for Service was completed by OPG [27]. OPG reviewed the overall Life Cycle Management Program, in-service inspection data, maintenance, engineering assessments, and research and development findings to determine the viability of these critical SSCs to remain fit for service to 2024. A summary of this review is included within this SF2 report. A review of the LCMPs to assess fitness for service to 2028 is not included in this revision of the report, but will be discussed in section 4.1.1.1.

Additionally, a Class 1 Piping Service Limit Assessment has been completed by OPG [28], the results of which are also summarized in this SF2 report.

3.2 Review Task #2 – Maintenance Facilities and Resources

Review Task #2 is to confirm resources and facilities (on and off site) are available for ongoing plant maintenance. Existing Plant Maintenance facilities used by Pickering NGS (onsite and offsite) were identified and described. Internal audits, self-assessments, and Station Condition Records (SCRs) were reviewed to confirm that the resources and facilities are available for ongoing plant maintenance. The review was based on criteria established within the IAEA Guide NS-G-2.6 'Maintenance, Surveillance and In-Service Inspection in Nuclear Power Plants' [40].

3.3 Review Task #3 – Design Basis Assumptions

Review Task #3 is linked to Review Task #1. As part of the process of determining the actual condition of SSCs important to safety, each of these SSCs is assessed against the current design basis to confirm that design basis assumptions have not been significantly challenged and will remain that way through the period of life extension.

3.4 Review Task #4 – Spent Fuel Storage Facilities

Review Task #4 is to review the condition and operation of spent fuel storage facilities and their effect on the spent fuel storage strategy for Pickering NGS. Inputs include a review of safety, inspection, and analysis reports, audits, self-assessments, and station condition records.

3.5 Review Task #5 – Dependence on Obsolescent Equipment

Review Task #5 is to assess dependence on obsolete equipment for which no direct substitute is available. The results of the condition assessments were reviewed to determine the degree to which station components are potentially obsolete. OPG's process for managing equipment obsolescence was also reviewed for its effectiveness in resolving obsolescence.

3.6 Review Task #6 – Dependence on External Essential Services/Supply

Review Task #6 is to assess dependence on essential services and/or supplies external to the plant. A review of system health and other reports and SCRs for SIS/SOE systems was performed to identify any vulnerabilities. In addition, the processes in place to sustain the required external essential services and supplies are described.

3.7 Additional Reviews

A review of the Pickering Licence Condition Handbook (LCH) [25] was performed to determine if there are any impacts associated with operation of the Pickering Units past 2020 on the following (related to Safety Factor 2):

- Commitments previously made to the CNSC;
- Open CNSC action items; and
- Exemptions granted by the CNSC.

The review of Pickering PSR1 gaps previously identified in the Pickering 5-8 Continued Operations Plan (COP) is provided in a separate PSR2 COP review Report [88]

Fukushima Action Items (FAIs) were reviewed to identify any implications of extending operation beyond 2020. This review is presented in a separate PSR2 FAI Review Report. [81]

Any PSR2 gaps identified as a result of the Safety Factor 2 review which are relevant to other Safety Factor reports are discussed in Section 4.4 of this report.

4.0 REVIEW FINDINGS

4.1 REVIEW TASKS

4.1.1 *REVIEW TASK #1 - ACTUAL CONDITION OF SSCS*

4.1.1.1 *MANAGEMENT OF MAJOR COMPONENTS*

The OPG Major Components Program N-PROG-MA-0025 [26] classifies Fuel Channels, Feeders, Steam Generators, and Reactor Components and Structures as Major Components, and establishes requirements for an integrated set of processes and activities to justify Fitness for Service for these components and to develop long-term Life Cycle Management strategies that support the preservation of these assets. N-PROC-MA-0100 'Major Component Life Cycle Management Plan' [29], provides guidance for the preparation, review, and update of Life Cycle Management Plans (LCMPs) for each of these Major components Fuel Channels [30], Feeders [31], Steam Generators [32], Reactor Components & Structures [33], to facilitate compliance with the Major Components Program N-PROG-MA-0025 [26] and to meet the requirements of Integrated Aging Management (IAM) Program N-PROG-MP-0008 [34].

As described in P-CORR-01060-0632223 [27], major component aging is managed through a comprehensive program of in-service inspections, maintenance, engineering assessment and confirmatory research and development (R&D). These processes provide for the timely detection and mitigation of aging effects in SSCs that impact plant safety, reliability, and economics; thereby providing a decision making process to optimize asset management. The LCMPs addresses legal (Regulatory/Periodic Inspection Program) requirements, Fitness for Service (FFS) and asset preservation activities. The LCMPs are updated on a regular basis to include actions based on inspection results, industry operating experience, and research and development findings.

OPG conducted an assessment of Major Components and have documented the assessment findings and recommendations in P-CORR-01060-0632223 [27]. The key findings and conclusions of this assessment are summarized in the following subsections for each of the Major Components; Fuel Channels, Feeders, Steam Generators, and Reactor Components and Structures.

4.1.1.1.1 *FUEL CHANNELS*

OPG has completed and documented in the Fitness for Service Memorandum [27] a review of Fuel Channels and the results of the activities defined by the Fuel Channel LCMP [30]. This review included assessment of the LCMP principles and methodology, fuel channel inspections, fuel channel Fitness for Service (FFS) requirements, and fuel channel aging mechanisms. A summary of the review in the Fitness for Service Memorandum [27] is included below.

The assessment of fuel channel life cycle management principles and methodology included a listing of strategic goals of the fuel channel LCMP and the means by which those goals were achieved. The LCMP provides projections of the service life using the most up-to-date knowledge of the components condition and updates these projections as new information becomes available. The impact of other components and systems on the performance of fuel changes, and the impacts of fuel channel aging on other systems are also considered. No gaps in the principles and methodology were identified by OPG in their Fitness for Service Memorandum [27].

Fuel channel inspections and fuel channel Fitness for Service are governed by CSA standard N285.4 ‘Periodic Inspection of CANDU Nuclear Power Plant Components’ [35] and compliance to this standard is required by the Power Reactor Operating Licence (PROL). As per the Fitness for Service Memorandum [27], the LCMP for fuel channels includes inspection scope that exceeds the minimum CSA standard requirements of N285.4 [35] to demonstrate FFS. If a flaw is detected by in-service inspection that does not satisfy the acceptance criteria it is necessary to engage in the component disposition process to demonstrate FFS for the next operating interval.

Fuel channel aging mechanisms include pressure tube: axial elongation, sag, wall thinning, and diametral expansion; changes in spacer material properties and spacer mobility; hydrogen ingress and fracture toughness. Predictive models and tools have been developed in support of inspections, FFS, and aging assessments.

The inspection results and conclusions documented in the Fitness for Service Memorandum [27] confirm that “OPG has high confidence that the Pickering FCs will remain fit-for-service for extended operation to 2024. This confidence comes from years of operating experience, extensive research, current assessments and projections of channel condition”, which provides that their Life Cycle Management Strategy will allow timely intervention to predict and correct unacceptable deviations from design.

Additional analyses will need to be performed for the Fuel Channels for extended station operation to 2028. The degradations identified in Reference [27] will need to be projected based on calculated Effective Full Power Hours (EFPH) targets and strategies will need to be put in place for mitigating projected Fitness for Service issues, including increased inspections and replacement strategies. Since Fitness for Service for Fuel Channels has not been demonstrated for station operation to 2028, this results in a PSR2 gap (**Pickering PSR2 Gap SF2-1**).

In addition, OPG does not have CNSC approval to operate beyond the current Licence Condition Handbook [25] limit of 247,000 EFPH for fuel channels, resulting in a PSR2 gap (**Pickering PSR2 Gap SF2-2**).

Lastly, the Fuel Channel LCMP [31] has not been formally updated to address extended station operation to 2028, resulting in a PSR2 gap (**Pickering PSR2 Gap SF2-3**).

4.1.1.1.2 *FEEDERS*

OPG has completed and documented a review in the Fitness for Service Memorandum [27] of Feeders and the results of the activities defined by the Feeder LCMP [31]. This review included assessment of the general program, feeder inspections, and effectiveness of the in-service inspections. A summary of the assessment in the Fitness for Service Memorandum [27] is documented below.

The Feeder LCMP [31] specifies the required Periodic Inspection Program (PIP) and In-Service Inspections (ISI), and maintenance for Feeders. The Feeders LCMP [31] is revised on a regular basis to capture changes that may be required in response to issues identified by inspection, industry experience, and ongoing research activities.

The LCMP incorporates feeder PIP documents, which are Unit specific plans based on the requirements outlined in CSA standard N285.4 [35]. OPG inspects feeders and related components such as supports and instrument lines during planned outages in accordance with the LCMP to confirm that the feeders remain fit for service for the next operating cycle.

As per the Fitness for Service Memorandum [27], the LCMP [31] for feeders defines the periodic and in-service inspection (ISI), and provides a 7-year forward looking plan for all required activities during Pickering Unit planned outages. The PIP documents give specific details for each unit based on CSA Standard N285.4 requirements and represent the base inspection requirements. The ISI scope is based on active and plausible (susceptible) degradation mechanisms for feeder piping, support hardware and other components within the feeder cabinets. The LCMP [31] includes probability of degradation occurring and consequences of degradation (both FFS and economic considerations).

Each Pickering Unit has an inspection outage approximately every 2 years per the Fitness for Service Memorandum [27]. The thickness inspections for future scopes are focused on monitoring lead feeders and dispositioned feeders. Lead feeders are those that are approaching their minimum design-required thickness. Dispositioned feeders are those with specific analysis defining their service limits which were accepted by the CNSC. This population has been determined by previous campaigns where 100% of the feeder inspections were completed. The inspection scope increases as the dispositioned feeder population increases.

As per the Fitness for Service Memorandum [27], the latest inspection results demonstrate that the most recent measured wall thickness remained greater than the minimum allowable wall thickness and will be fit for service for the next operating cycle. As per the LCMP [31], these feeders will continue to be monitored. However, as per the Fitness for Service Memorandum [27], OPG believes there are a small number of feeders (14) that analysis may determine require replacement for reactor operation to 2024. OPG has experience replacing feeders in Pickering (Unit 8 in 2008) and has high confidence that with inspection, analysis, and targeted replacement activities the Pickering feeders will remain fit for service in each unit for extended operation to 2024.

Additional assessments will need to be performed for Feeders for station life extension to 2028. The degradations identified in Reference [27] will need to be projected based on calculated EFPH targets and strategies would need to be put in place for mitigating projected FFS issues, including increased inspections and replacement strategies. Since Fitness for Service for Feeders has not been demonstrated for station operation to 2028, this results in a PSR2 gap (**Pickering PSR2 Gap SF2-4**).

In addition, the Feeder LCMP [31] has not been formally updated to address extended station operation to 2028, resulting in a PSR2 gap (**Pickering PSR2 Gap SF2-5**).

4.1.1.1.3 *Steam Generators*

OPG has completed and documented a review in the Fitness for Service Memorandum [27] of Steam Generators (SG) and the results of the activities defined by the Steam Generator LCMP [32]. This review included assessment of the general program, inspections, and effectiveness of the in-service inspections. A summary of the assessment in the Fitness for Service Memorandum [27] is documented below.

The Steam Generator LCMP [32] specifies the required Periodic Inspection Program and In-Service Inspections, maintenance and modifications for Steam Generators.

The SG LCMP [32] is revised on an annual basis to capture changes that may be required in response to issues identified by inspection, industry experience and ongoing research activities. OPG inspects SG tubes and internals during Unit planned outages in accordance with the SG LCMP [32] to confirm that the SGs remain fit for service until the next planned inspection.

According to the Fitness for Service Memorandum [27], each Pickering Unit has a steam generator inspection approximately every 2 years. The SG LCMP [32] identifies the inspection scope, including Eddy current testing requirements to identify active and plausible tube degradation mechanisms and the extent of the condition in the SGs. Two sample sizes of 30%-50% and 10% are inspected using standard and specialized eddy current probes respectively. Tubes with previous indications are added to the minimum sample size, and the sample size may be further increased based on the particular degradation mechanisms that are active in the Unit. As per the Fitness for Service Memorandum [27], in the Unit 4 planned maintenance outage (P1641), inspection scope was expanded to additional steam generators in Unit 4 due to recent findings. Conditioning monitoring was not acceptable for Pickering Unit 4 during the 2016 outage. Therefore, a disposition period of one effective full power year was requested and accepted by the CNSC. Pickering Unit 4 has been shut down in January 2017 for additional inspections, and targeted ET + UT inspection was performed on 4 SGs to validate previously accepted disposition.

SG tube plugging and tube leakage also provide additional monitoring of SG structural health. No limits on steam generator tube plugging were established in the original design specification or code of construction but as per the Fitness for Service Memorandum [27] OPG has performed station specific assessments of the impact of tube plugging. Reviews of the current number of plugged tubes show that no steam generators in Pickering exceed the current assessment of allowable number of plugged tubes and that margin exists relative to these assessments to support operation to 2024 (per the Fitness for Service Memorandum [27]).

Also, there have been no SG tube leaks in the Pickering Units since mid-2001, although some existing SG tubes have warranted plugging over this period. This conclusion is based on analysis of main steam tritium levels, which in each units are stable and well below levels at which enhanced monitoring is performed to assess suspected steam generator tube leakage.

Based on these results of the SG LCMP [32], OPG has confidence that the Pickering SGs will remain fit-for-service in each of Units 1, 5, 6, 7, 8 for the approved operating interval for extended operation to 2024. Pickering Unit 4 SG inspections, as planned in P1741 outage, have been completed. The SG condition monitoring for P1741 was acceptable and degradations were conservatively bounded by the assessments given in the accepted 2016 disposition.

Additional assessments will need to be performed for Steam Generators for station life extension to 2028. The strategy listed above would need to be projected based on calculated EFPH targets, and solutions would need to be put in place for mitigating projected FFS issues including increased inspections. Since Fitness for Service for Steam Generators has not been demonstrated for station operation to 2028, this results in a PSR2 gap (**Pickering PSR2 Gap SF2-6**).

In addition, the Steam Generator LCMP [32] has not been formally updated to address extended station operation to 2028, resulting in a PSR2 gap (**Pickering PSR2 Gap SF2-7**).

4.1.1.1.4 *REACTOR COMPONENTS AND STRUCTURES*

OPG has completed and documented a review in the Fitness for Service Memorandum [27] of Reactor Components and Structures (RC&S) and the results of the activities defined by the RC&S LCMP [33]. The RC&S LCMP [33] prescribes the work program required to meet legal, Fitness for Service, or asset preservation requirements and embeds results with components dispositions as required to demonstrate Fitness for Service.

Maintenance activities in the RC&S LCMP [33] include CT/LISS gap inspection, Guide Tube gap inspection, and Calandria Tube inspections. According to the Fitness for Service Memorandum [27], OPG believes that for Pickering Units 5-8 there is the potential that the Calandria Tubes (CT) could come into contact with the Shutdown System #2 Liquid Injection Shutdown System (LISS) nozzles due to fuel channel sag. Pickering Units 1 and 4 do not have LISS nozzles.

The current assessments utilize baseline measurements of CT/LISS nozzle gaps in each unit. Additional measurements of CT/LISS nozzle gap will be used to refine the gap closure rate to more accurately predict CT/LISS nozzle contact time. As per the Fitness for Service Memorandum [27], these refinements are expected to show that CT/LISS nozzle contact will not occur prior to the extended operation to 2024 for any of the Pickering 5-8 units. It is expected that operation to end of life will be bridged by repeat inspections as specified in the LCMP [33] to assess gap closure rate and/or mitigation strategies previously employed in the CANDU industry. Additional mitigation measures are also available if required, up to and including single fuel channel replacement, to ensure Fitness for Service for CT/LISS nozzle contact concerns.

Additional assessments will need to be performed for Reactor Components and Structures for station life extension to 2028. The strategy listed above would need to be projected based on calculated EFPH targets, and solutions would need to be put in place for mitigating projected FFS issues including increased inspections and replacement strategies. Since Fitness for Service for Reactor Structures has not been demonstrated for station operation to 2028, this results in a PSR2 gap (**Pickering PSR2 Gap SF2-8**).

In addition, the Reactor Components and Structures LCMP [34] has not been formally updated to address extended station operation to 2028, resulting in a PSR2 gap (**Pickering PSR2 Gap SF2-9**).

4.1.1.1.5 *SERVICE LIMITS ASSESSMENT OF NUCLEAR CLASS 1 COMPONENTS*

OPG has completed a review of the service cycle usage of Nuclear Class 1 components to ensure adequate margin between analyzed cycles and completed cycles for continued operation to 2028 as per memorandum P-CORR-33000-00001 [28].

The methodology employed by OPG included review of the earlier Preliminary Service Limit Assessment [36], and subsequent assessment to incorporate:

a) The analyzed fatigue service limits of Pickering Class 1 systems and components.
b) The Units' operational transient histories.
c) The impact of Flow Accelerated Corrosion (FAC) on PHT piping Fatigue Service Limits.

OPG concluded that the analyzed design basis service limits can cover extended operation to 2028 with adequate safety margin. The number of recorded transient cycles prorated for extended operation would be less than 60% of the analyzed design basis service limit cycles of Pickering NGS 1,4 and 5-8 Units. It was also concluded that FAC induced wall loss is minimal with no impact on the PHT fatigue life calculations for operation up to 2028.

However, Environmental Factors have not been evaluated within the Service Limit Assessments for Class 1 piping, resulting in a PSR2 gap (**Pickering PSR2 Gap SF2-10**).

4.1.1.1.6 *ASSESSMENT FINDINGS FROM MAJOR COMPONENTS*

The Fitness for Service Assessment referenced above documents the current condition of the major components in the Pickering Units. There is high confidence that the major components will remain fit for service up to an extended station life to 2024, with limited potential mitigating actions required. Additional analysis and assessment is required to demonstrate continued Fitness for Service for extended station life to 2028 as per the PSR2 gaps below.

The following gaps were identified from this review:

- **Gap SF2-1:** Fitness for Service for Fuel Channels has not been demonstrated for station operation to 2028.
- **Gap SF2-2**: OPG does not have approval to operate beyond the current Licence limit of 247,000 EFPH for fuel channels.
- **Gap SF2-3**: The Fuel Channels LCMP has not been formally updated to address extended station operation to 2028.
- **Gap SF2-4**: Fitness for Service for Feeders has not been demonstrated for station operation to 2028.
- **Gap SF2-5**: The Feeders LCMP has not been formally updated to address extended station operation to 2028.
- **Gap SF2-6:** Fitness for Service for Steam Generators has not been demonstrated for station operation to 2028.
- **Gap SF2-7**: The Steam Generators LCMP has not been formally updated to address extended station operation to 2028.
- **Gap SF2-8:** Fitness for Service for Reactor Components and Structures has not been demonstrated for station operation to 2028.

- **Gap SF2-9:** The Reactor Components and Structures LCMP has not been formally updated to address extended station operation to 2028.
- **Gap SF2-10:** Environmental Factors have not been incorporated into the Service Limits Assessment for Class 1 piping.

4.1.1.2 CONDITION ASSESSMENTS FOR SSCS IMPORTANT TO SAFETY

This section provides results of the Condition Assessments for the balance of SSCs in two parts. In the first part System Summaries are presented. The System Summaries document the results of the Detailed CAs prepared prior to 2016. These results reflect a freeze date of August 31, 2015 and were performed in accordance to the methodology in Section 3.1. In the second part (Section 4.1.1.2.49), an overview is presented of the preliminary results of Detailed CAs currently being prepared to address an assessment for a station life of 2028, with a January 15, 2016 freeze date.

System Summaries:

The results of the DCAs are presented in this report as System Summaries. The System Summaries document the key DCA findings including component condition and any improvement actions required.

Prior to the PSR2 effort, the scope of Aging Management (AM) Systems were merged between Pickering 1,4 and 5-8. This resulted in 70 combined AM Systems, encompassing all Aging Management Program related systems in Pickering. 47 of these 70 AM Systems were identified as being in-scope for PSR2 (per the methodology described in section 3.1) and the DCA results for these systems are presented in this report. The 47 SF2 systems encompass all SOE/SIS systems as described in Section 3.1.

Screening reports encompassing all SOE/SIS systems were then prepared to identify those AM components that required a DCA to be prepared. These screening reports are issued as controlled documents. The results of the screening identified that 1202 unique Commodity Groups (CGs) required a DCA be prepared. The remaining CGs were screened-out based on the criteria in N-PROC-MP-0060 [7].

The System Summaries are documented in sections 4.1.1.2.1 to 4.1.1.2.47 for each of the 47 in-scope systems[4]. The following information is provided for each system:

- The relevant System Health Report(s).

[4] The results are also documented in Appendix B.2 in tabular form for each system and commodity group and discussed in section 4.1.1.2.48.

- Current Initiatives (not including periodic maintenance practices) credited within each Detailed CA for improvement and/or to sustain condition. This includes currently planned work, e.g. planned inspection work orders.
- Incremental Recommendations to support continued operation to 2020 (i.e. not currently planned).
- Incremental Recommendations to support extended operation to 2024.[5]

Where specific equipment tags/populations in a CG are referenced, recommendations apply to only that equipment. In cases where the recommendations do not identify specific equipment populations, the recommendations apply to the entire CG. CGs are not contained in these sections if no recommendations are required to improve their condition, indicating that their condition and aging management practices are acceptable. The entire list of CGs is addressed in Appendix B.2.

Selected Civil structures are included in the System Summaries and are in system numbers: 406, 425, 436, 441, 452, 455 and 466, and 0469. All structures that are part of Containment are included, e.g. the Reactor Buildings, Pressure Relief Duct. The remaining non-containment in-scope civil structures are in the process of being addressed as part of the current condition assessment updates. Since these condition assessments are not complete this results in a PSR2 gap (**Pickering PSR2 Gap SF2-11**).

The System Summaries include piping for many critical systems, e.g. the heat transport and moderator systems. The remaining in-scope system piping, e.g. ECIS, is in the process of being screened and CAs completed as required. Since these assessments are not complete this results in a PSR2 gap (**Pickering PSR2 Gap SF2-12**).

Pipe supports have been screened out from further assessment. There is a pipe support program N-GUID-04980-10002, "Guideline for Critical Pipe Support Inspection and Results Processing" [90] for critical systems and in addition pipe supports are included in system engineer surveillance practices. Selected critical hangers in Nuclear Class systems are also inspected via the Periodic Inspection Program (PIP) [82]. Continued execution of these programs would identify any pipe support deficiencies. Any deficiencies are then assessed for fitness for service and corrected via the work management process.

Lastly, compared to OPG Revision 0 of this report, recommendations have been removed for CGs now screened out based on the latest screening discussed in Section 4.1.1.2.49. These removed items correspond to the shaded out rows in the Appendix B-2 Tables which are discussed in section 4.1.1.2.48.

All of the recommendations documented in following sections will be assessed, dispositioned, prioritized and tracked via station work programs.

[5] Detailed CAs are being prepared to consider life extension to 2028. This is addressed in SF2 Gap 17. The preliminary results of these CAs is presented in section 4.1.1.2.49.

4.1.1.2.1 REVIEW OF ANNULUS GAS SYSTEM

System Number: 0400

System Health Report Name: Annulus Gas
Review Period for System Health Report: Q4-2015
Overall System Health Rating: Green

Unit 1	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8
Green	Green	Green	Green	Green	Green

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

- CG 011268, Unit(s) 5,6,7,8: 34980 Annulus Gas System-Valves - NV-Non-Return CAT 1&2 - Replace 6/7/8-34980-NV102. This requires implementation of CAT ID 709329 (which is currently going through the design processes to address obsolescence issues with CAT ID 118136).

Incremental recommendations for Plant EOL (2020):

- CG 008001, Unit(s) 1,4: Analyzer, Cat 1/2 - Implement a new PM to perform periodic calibration and function checking.

Incremental recommendations for CO EOL (2024):

- CG 008010, Unit(s) 1,4: FAN, Cat 1/2 - Implement a new PM to periodically check the fan and motor bearings and add lubrication if necessary.

4.1.1.2.2 REVIEW OF BOILER BLOW-OFF SYSTEM

System Number: 0401

System Health Report Name: Main Steam and Blowdown
Review Period for System Health Report: Q4-2015
Overall System Health Rating: White

Unit 1	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8
Yellow	Yellow	Green	Green	White	White

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

- CG 008060, Unit(s) 1,4: Valve Pneumatic / Pneumatic AC, Cat 1/2, large Boiler Blowdown Valves - Complete WOs 2731629, 2731630, 2284458, 2243671 to overhaul and/or replace valve internals and actuator.

Incremental recommendations for Plant EOL (2020):

- CG 011297, Unit(s) 5,6,7,8: 36410 Boiler Blow-off System – AOVs - Initiate one-time inspection of valve internals, and replace/repair if degraded. Procure additional spare valves to ensure corrective action can be expedited if required.

Incremental recommendations for CO EOL (2024):

- CG 008058, Unit(s) 1,4: Valve Pneumatic / Pneumatic AC, Cat 1/2 - Perform a one-time inspection of valve internals against ARDMs (elastomer embrittlement, material loss on bushing, spindle, seat, disc, bellows failure, valve body corrosion), weld inspections, and repair/replace valve internal parts if degraded.

- CG 011297, Unit(s) 5,6,7,8: 36410 Boiler Blow-off System – AOVs – Initiate new PMs for external inspections of valves every two years.

- CG 011298, Unit(s) 5,6,7,8: 36410 Boiler Blowoff System - Mechanical - Spare ball joint components to be procured to facilitate corrective actions if/when required.

4.1.1.2.3 *Review of Boiler Emergency Cooling System*

System Number: 0402

System Health Report Name: Main Steam and Blowdown
Review Period for System Health Report: Q4-2015
Overall System Health Rating: White

Unit 1	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8
Yellow	Yellow	Green	Green	White	White

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

- CG 011477, Unit(s) 5,6,7,8: 36710 Boiler Emergency Cooling System – Valves – NV-Non-Return, Cat 1/2 - Perform internal inspection per WO 1618394. Also, resolve spare parts and obsolescence issues.

Incremental recommendations for Plant EOL (2020):

- CG 008081, Unit(s) 1,4: VALVE, CHECK/NONRETURN/BACK FL, Cat 1/2, BECS-02 - Perform one-time inspection of 4-36710-NV3, NV13, 1-36710-NV3, and NV13.

- CG 008086, Unit(s) 1,4: VALVE, PRESSURE REGULATING, Cat 1/2 - Complete a one-time calibration/set point check for 1/4-36710-PRV3006 to reach Plant EOL (2020).

Incremental recommendations for CO EOL (2024):

- CG 008086, Unit(s) 1,4: VALVE, PRESSURE REGULATING, Cat 1/2 - Complete a one-time calibration/set point check for 1/4-36710-PRV3006 to reach CO EOL (2024).

- CG 008089, Unit(s) 1,4: Solenoid / Solenoid Operated Val, Cat 1/2 - Perform one-time overhaul of Actuator and Solenoid Valve.

4.1.1.2.4 REVIEW OF THE BOILER FEED SYSTEM

System Number: 0403

System Health Report Name: Boiler Feed and Main Condensate System
Review Period for System Health Report: Q4-2015
Overall System Health Rating: White

Unit 1	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8
Yellow	Red	White	White	White	White

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

- CG 008144, Unit(s) 1,4: Valve, Control, Cat 1/2 - Complete work orders, and resolution to the following issues:
    1. 4-43230-CV213 leaking WO#4703681
    2. Pressure controller failure 1-43230-CV206 WO#4762318
    3. Repacking of 1-43230-CV220 WO#4809923
    4. Check welds for cracks 1-43230-CV202 WO#3060177.

- CG 008146, Unit(s) 1,4: Valve, Manual / Hand Operated, Cat 1/2 - Complete outstanding CM/DM work orders to correct leaking valves.

- CG 011156, Unit(s) 5,6,7,8: 43230 Boiler Feed Pump Recirculation Control Valves:

1. Complete outstanding WOs to replace limit switches with a more robust model.
2. Complete outstanding AR 28175792 assignments to resolve issues regarding valve stem/plug separation.

- CG 011157, Unit(s) 5,6,7,8: 43230 Boiler Inlet Feedwater Isolation and Auxiliary BFP Discharge MOVs - Complete outstanding work requests, and resolve the issue of solenoid valve failures.
- CG 011185, Unit(s) 5,6,7,8: 43230 Main Boiler Feed Pump Discharge Motorized Valves – Complete work orders to overhaul actuators and repair leaks/packing issues.

Incremental recommendations for Plant EOL (2020):

- CG 008127, Unit(s) 4: Relay, Cat 1/2 - Implement recurring inspections to assess degradation of relay.
- CG 008148, Unit(s) 1,4: Valve, Motorized / Motor Operate, Cat 1/2 - Perform diagnostics including a review of current data to reach Plant EOL (2020).
- CG 008153, Unit(s) 1,4: VALVE, CHECK/NONRETURN/BACK FL, Cat 1/2, FW-02 - Perform a one-time internal inspection of at least one “Sample” valve to use as a reference of condition for the other valves.

Incremental recommendations for CO EOL (2024):

- CG 008100, Unit(s) 1,4: ELEMENT, Cat 1/2 - Perform one-time inspection, and if degraded replace.
- CG 008153, Unit(s) 1,4: VALVE, CHECK/NONRETURN/BACK FL, Cat 1/2, FW-02 – Initiate PM’s for non-intrusive testing of all valves in the CG every 104 weeks.

4.1.1.2.5 *Review of Boiler Steam and Water Systems*

System Number: 0404

System Health Report Name: Main Steam and Blowdown
Review Period for System Health Report: Q4-2015
Overall System Health Rating: White

Unit 1	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8
Yellow	Yellow	Green	Green	White	White

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

- CG 008197, Unit(s) 1,4: VALVE, MOTORIZED/MOTOR OPERATE, Cat 1/2 - Complete outstanding WO 04849728 for 1-36110-MV5B, and WO 04845032 for 1-36110-MV6A.

- CG 008202, Unit(s) 1,4: VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2 –
    1. Complete WO 4850648 for 1-36110-MV38 overhaul and WO 4727738 for 1-36110-MV40 overhaul.
    2. Complete WO 4823379 for 1-36110-MV36 packing replacement and WO 4727739 for 1-36110-MV41 packing adjustment.
    3. Complete WO 04813836 (4-36110-MV38) and WO 04908520 (4-36110-MV40) for air leaks.
    4. Complete WO 02704201 (4-36110-MV38) and WO 02704199 (4-36110-MV39) to replace rubber hoses.

Incremental recommendations for Plant EOL (2020):

- CG 008164, Unit(s) 1,4: Controller, Hand, Cat1/2 - Implement new PM for inspection/overhaul and calibration to be scheduled every 104 weeks.

- CG 008197, Unit(s) 1,4: VALVE, MOTORIZED/MOTOR OPERATE, Cat 1/2 - Complete a one-time actuator overhaul and diagnostic testing for all MVs in this CG.

- CG 008199, Unit(s) 1,4: VALVE, CHECK/NONRETURN/BACK FL, Cat 1/2, MS-01 - Perform one-time inspections of valves 1/4-36110-NV7/NV8 to assess valve condition.

- CG 011221, Unit(s) 5,6,7,8: 63615 Boiler Steam and Water Systems-CONTROLLER -HAND-CAT 1&2 - Obtain a replacement for 5-63615-P1-HC1 and WX5 and perform calibration every 104 weeks.

- CG 011349, Unit(s) 5,6,7,8: 36110 Boiler Steam and Water Systems-Valves - NV-Non-Return -CAT1&2 - Inspect 7-36110-NV8 as part of a sampling strategy to determine required maintenance on 5/6/7/8-36110-NV7/NV8. Also, inspect 7-36110-NV63 to establish a baseline condition.

- CG 011350, Unit(s) 5,6,7,8: Boiler Steam and Water Systems-Valves - MOV-Standard-CAT1&2 –
    1. Inspect one Group 1 valve (MV1/MV2) to determine if future maintenance is needed and overhaul actuators.
    2. Replace Group 2 valves 5-36110-MV60, 6-36110-MV60 and 7-36110-MV62 with new Cat ID 609880.
    3. Procure spares for Group 1 valves and four valves along with Rotork actuator for Group 2 valves.

Incremental recommendations for CO EOL (2024):

- CG 008197, Unit(s) 1,4: VALVE, MOTORIZED/MOTOR OPERATE, Cat 1/2 –
    1. Implement new PMs for 1/4-36110-MV1, MV2, MV3A, MV4A, MV5A, and MV6A for actuator inspection, diagnostics and functional test on a 4 year frequency.
    2. Implement PMs for 1/4-36110-MV3B, MV4B, MV5B, and MV6B for actuator lubrication on a 3 year frequency.

4.1.1.2.6 *Review of Calandria Vault / Vault Structure Cooling / Shield Tank*

System Number: 0406

The Calandria Vault structure does not require a System Performance Monitoring Plan, and hence does not have a System Health Report.

The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Incremental recommendations for Plant EOL (2020):

- CG 011249, Unit(s) 5,6,7,8: Calandria Vault, Vault Structure Cooling, Shield Tank - Structural Concrete - Concrete Walls and Slabs - Conduct one-time visual inspection of accessible areas of Calandria Vault structural concrete for Unit(s) 5-8 to inspect for potential damage and perform any required mitigating/remedial actions.

- CG 011250, Unit(s) 5,6,7,8: 21300 Calandria Vault/Vault Structure Cooling/Shield Tank-Liners - Steel Liners - Conduct one-time inspection for Unit 8, and for Unit(s) 5 to 7 only if evidence of component failure or degradation is discovered. For Unit 8 the condition of epoxy patch as a temporary solution to repair a weld defect is unknown and it is recommended to initiate a permanent repair (i.e. welding).

- CG 011251, Unit(s) 5,6,7,8: 21300 Calandria Vault/Vault Structure Cooling/Shield Tank-Penetrations - Steel Sleeves Surrounding Concrete Vault Openings - Perform one-time visual inspection of accessible areas and local leakage testing to confirm components' suitability.

- CG 011252, Unit(s) 5,6,7,8: 21300 Calandria Vault/Vault Structure Cooling/Shield Tank - Embedded Parts and Supports - Conduct one-time visual inspection of accessible areas to confirm the suitability of the components.

- CG 011253, Unit(s) 5,6,7,8: 21300 Calandria Vault/Vault Structure Cooling/Shield Tank - Seals & Sealants –
    1. Complete one-time inspection of seals.
    2. Replace the elastomeric seals if required.
    3. Replace seals if they show advanced degradation.

Incremental recommendations for CO EOL (2024):

- CG 011253, Unit(s) 5,6,7,8: 21300 Calandria Vault/Vault Structure Cooling/Shield Tank - Seals & Sealants –
    1. Replace the elastomeric seals if required.
    2. Replace seals if they show advanced degradation.

4.1.1.2.7 *Review of Class 1 & 2 Electrical and battery room HVAC Systems*

System Number: 0408

System Health Report Name: Class I and II Electrical Equipment and Battery Rooms HVAC Systems
Review Period for System Health Report: Q4-2015
Overall System Health Rating: White

Unit 1	Unit 2	Unit 3	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8
White	Green	Green	Green	White	White	White	White

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

- CG 008265, Unit(s) 4: DAMPER, Cat 1/2 - Complete MDP actuator replacements via master Non-Identical Component Replacement (NICR) 104778.

Incremental recommendations for Plant EOL (2020):

- CG 008265, Unit(s) 4: DAMPER, Cat 1/2 - Revise PMs 92533, 98078, 98079 and 98080 to add tasks for lubrication and inspection every 1 year on 4-73230-MDP2030/2031/2032/2033 (4 components).

- CG 008279, Unit(s) 4: SWITCH, Cat 1/2 - Calibrations to be scheduled at a frequency adequate for each pressure switch.

Incremental recommendations for CO EOL (2024):

- No additional practices are recommended to reach CO EOL (2024).

4.1.1.2.8 *Review of Common Water Supply*

System Number: 0410

System Health Report Name: Screenhouse
Review Period for System Health Report: Q3-2015

Overall System Health Rating: White

Unit 012	Unit 034	Unit 056	Unit 078
Yellow	Yellow	Yellow	White

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here.
The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

- CG 008326, Unit(s) 012,034: Screen, Cat 3/4 - Repair failed Bar Screens 012-71110-SC11 and 034-71110-SC12 (WO 3249081).

- CG 011247, Unit(s) 056,078: 71100, 71120 Common Water Supply - Screens and Conveyors - Overhaul the trash conveyors, 056, 078-71120 TC1 (WO 2809130 and WO 2795171).

- CG 011299, Unit(s) 056,078: 54130, 53300 Common Water Supply-Motor Control Centre (MCC) 600V - Complete structural inspection/maintenance activities on remaining CC2 MCCs (056-54130-MCC541, 078-54130-MCC741). Also, accelerate the implementation of all MCC cells replacement as per NK30-ESI-50000-00006.

Incremental recommendations for Plant EOL (2020):

- CG 011247, Unit(s) 056,078: 71100, 71120 Common Water Supply - Screens and Conveyors –
    1. Complete a one-time cleaning of the screens to remove debris and zebra mussels.
    2. Resolve spare parts issues.

Incremental recommendations for CO EOL (2024):

- CG 011247, Unit(s) 056,078: 71100, 71120 Common Water Supply - Screens and Conveyors –
    1. Complete one-time inspection of Screenhouse concrete structures and equipment supports.
    2. Initiate a new PM for zebra mussel cleaning for the bar screens.

- CG 011303, Unit(s) 056,078: 71100 Common Water Backwash Strainers - Purchase one additional strainer.

4.1.1.2.9 *REVIEW OF CONTAINMENT*

System Number: 0412

System Health Report Name: Airlocks
Review Period for System Health Report: Q4-2015

Overall System Health Rating: Green

Unit 1	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8	Unit 018
Green	Green	Green	Green	Green	Green	Green

System Health Report Name: Filtered Air Discharge
Review Period for System Health Report: Q4-2015
Overall System Health Rating: White

Unit 018
White

System Health Report Name: Negative Pressure Containment and H2 Ignition
Review Period for System Health Report: Q3-2015
Overall System Health Rating: White

Unit 1	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8
White	White	Green	White	Green	Green

System Health Report Name: Reactor Building Cooling
Review Period for System Health Report: Q3-2015
Overall System Health Rating: Green

Unit 1	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8
White	Green	White	Green	Green	Green

System Health Report Name: Vacuum Building (NPC/ESW) System
Review Period for System Health Report: Q4-2015
Overall System Health Rating: Green

Unit 0	Unit 018
Green	Green

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

- CG 008418, Unit(s) 1,4: Igniter, Cat 1/2 - Complete proactive replacement of hydrogen igniters which failed recently in Units 1 and 4 (ref. EC 112707).

- CG 011182, Unit(s) 5,6,7,8: 73110 RB Cooling-Fuelling Machine Vault ACUs - Follow-up OEM and parallel company investigation of premature coil failures.

- CG 011242, Unit(s) 018: 34230 Filtered Air Discharge System (FADS) MOVs - Complete the actions listed below:
    1. Replace parts of 018-34220-MV12/MV13/M14 (Vacuum Pump Suction Isolation Valves) and 018-34220-MV201/MV203 per EC120350 (WOs 1887012/ 2886975/ 2887049/ 2887432/ 2887431).
    2. Replace 018-34230-MV119/MV122 (Vent to U1 Stack Isolation Valves) per EC108218 and EC108257.
    3. Replace 018-34230-MV124/MV125 (Stack Discharge Isolation Valves) per WR 1663427 and 1663428.
    4. Replace 018-34230-MV112 (F102 Discharge Isolation Valve) if required based on results from WO4939713.
    5. Complete investigation of MV112 failure, and initiate actions as required to address any AM related findings for the other affected valves.
    6. Procure spares/parts to support the above activities, as well as potential valve replacements resulting from inspection PMs.
- CG 011262, Unit(s) 018: 21100 Containment-ALARM-ALARM UNIT-CAT 1&2 - Procure a suitable replacement for the FADS pressure alarm Unit(s) per EC 129463.
- CG 011335, Unit(s) 5,6,7,8,018: 21103, 21130, 25230 Containment-Valves - NV-Non-Return -CAT1&2 - Complete the following work orders:
    1. WO 2096010 to remove and replace 018-25230-AL3-NV203.
    2. WO 619970 by installing new locking tabs on PRD AL3.

Incremental recommendations for Plant EOL (2020):

- CG 011181, Unit(s) 5,6,7,8: 73110 RB Cooling - Boiler Room ACUs - Confirm the integrity of the condensate drain lines (leak tight and not plugged) for all ACUs.

- CG 011182, Unit(s) 5,6,7,8: 73110 RB Cooling-Fuelling Machine Vault ACUs - Perform one-time replacement of coils which have not been replaced since 2009.

- CG 011277, Unit(s) 5,6,7,8: 73110 RB Cooling Dampers –
    1. Perform a one-time replacement of all dampers.
    2. Address obsolescence issues with CatID#151434.
    3. Perform a review to determine the availability of spares and stock for at least one complete overhaul of all dampers.
- CG 011340, Unit(s) 058,018: 34230, 71330 Containment-Valves - Manual-Criticality Category 1 (RS2) - Ensure that adequate spares are available to reach Plant EOL (2020).

- CG 011356, Unit(s) 018: 34200 - Containment - Manual Valves – FADS - Ensure that adequate spares are available to reach Plant EOL (2020).

Incremental recommendations for CO EOL (2024):

- CG 008430, Unit(s) 1,4,018: Regulator, Cat 1/2 - Perform one-time inspection, and if degraded repair/replace.

- CG 008436, Unit(s) 1,4,018: Switch, Cat 1/2 - Implement a new PM for the position switches of the FM shielding doors to reach CO EOL (2024).
- CG 008453, Unit(s) 1,4: VALVE, PRESSURE REGULATING, Cat 1/2 - Perform one-time inspection, and if degraded repair/replace.
- CG 011333, Unit(s) 5,6,7,8,018: 21000, 25000, 67000, 73000 Containment-Valves - SV-Solenoid Valves-CAT1&2 - Perform one-time inspection for all the Solenoid Valves that currently have no PMs (e.g. 5-21060-D3–SV1), replace/repair if needed.
- CG 011340, Unit(s) 058,018: 34230, 71330 Containment-Valves - Manual-Criticality Category 1 (RS2) - Ensure that adequate spares are available to reach CO EOL (2024).
- CG 011356, Unit(s) 018: 34200 - Containment - Manual Valves – FADS - Ensure that adequate spares are available to reach CO EOL (2024).

4.1.1.2.10 *REVIEW OF DEAERATOR AND STORAGE TANK*

System Number: 0417

System Health Report Name: Boiler Feed and Main Condensate System
Review Period for System Health Report: Q4-2015
Overall System Health Rating: White

Unit 1	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8
Yellow	Red	White	White	White	White

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

- CG 008628, Unit(s) 1,4: Valve, Motorized / Motor Operate, Cat 1/2 - Complete WO 04793232 which requires actuator removal, a valve repack and stem/ drive nut replacement for 1-43210-MV17.
- CG 008629, Unit(s) 1,4: Valve, Check/Nonreturn/Back Fl - Complete WO 3005442 & 3005443 to inspect and overhaul U1 NV18 & NV19. Based on inspection results, overhaul U4 NVs.
- CG 011211, Unit(s) 5,6,7,8: 43120 Deaerator & Storage Tank - Complete modifications to the Unit(s) 5-8 Deaerator Storage Tanks (DST) supports to increase the seismic capacity beyond the Pickering B Review Level Earthquake (RLE) for applicable deaerators per Master EC124589.

Incremental recommendations for Plant EOL (2020):

- CG 008601, Unit(s) 1,4: Expansion Joint, Cat 1/2 –
    1. Perform a one-time proactive external inspection of 1/4-43210EJ4 & 4-43210-EJ3001.
    2. Resolve MEL/Bill of Materials (BOM) issues. 4-43210-EJ3001 & 4-43210-EJ4 are linked to incorrect CIDs, identify if CID 606527 (1-43210-EJ4) is appropriate.
- CG 008626, Unit(s) 1, 4: VALVE, MANUAL/HAND OPERATED, Cat 1/2 –
    1. Resolve outstanding MEL/BOM (Cat ID should be in approved status) issues on valves 1, 4-43210-V12, 1, 4-43210-V301 & 1, 4-43210-V69.
    2. Complete a one-time elastomer replacement and stem lubrication of all valves in this CG.
- CG 008628, Unit(s) 1,4: Valve, Motorized / Motor Operate, Cat 1/2 - Complete a one-time overhaul & diagnostics of all MV actuators as per instructions in INACTIVE PMs (e.g. 6969).
- CG 011211, Unit(s) 5,6,7,8: 43120 Deaerator & Storage Tank - Evaluate the cost/benefits of completing weld inspections (of the two welds on each side of the HX4 of the water box plate to the 43120-HX4 shell) on U2 or U3 and implement additional AM practices on U5-8 if necessary based on inspection results.

Incremental recommendations for CO EOL (2024):

- CG 008601, Unit(s) 1,4: Expansion Joint, Cat 1/2 - Conduct an assessment for additional AM practices for CC1/SPV EJ4 and EJ3001 based on expected life after one-time inspection.
- CG 008626, Unit(s) 1,4: VALVE, MANUAL/HAND OPERATED, Cat 1/2 - Review the need to initiate a PM for periodic stem lubrication for all valves in this CG, based on condition after one-time Plant EOL (2020) work.
- CG 008629, Unit(s) 1,4: Valve, Check/Nonreturn/Back Fl - Initiate a PM for in-service testing of 1/4-43210-NV18 & 1/4-43210-NV19 at frequency of 4 years.

4.1.1.2.11 *Review of Digital Control Computers*

The Digital Control Computers (DCCs) have been assessed by OPG under memorandum P-CORR-66400-0632085 [37], the following content is extracted from there.

System Number: 0418

P1, 4 Digital Control Computers:
The P1,4 DCC systems are expected to support the plant to end of Extended Operating Life (EOL 2024/8 + 3 years for DCC mission). The existing strategies for P1,4 DCC system operation to 2020 were adequate.

To maintain current operations, a number of projects are under way or were completed to deal with the obsolescence of the DCC equipment such as:

- Replacement of DCC power supplies;

- PACE computer refurbishment;
- Class II auto transfer panel PL170/171 replacements;
- DCC Digital Output card replacements;
- Display monitors and ANO desktop keyboards/monitors.

For operation during life extension, several recommendations are made in the Detailed CA (see Appendix B.2).

P5-8 Digital Control Computers:
The P5-8 DCC systems are expected to support the plant to end of Extended Operating Life (EOL 2024/2028 + 3 years for DCC mission).
The existing strategies for P5-8 DCC system operation to 2020 were adequate.

To maintain current operations, a number of projects are under way or were completed to address aging and obsolescence of the DCC hardware such as:
- DCC core memory boards;
- DCC power supplies and Ramtek display system.

For operation during life extension, it is recommended as indicated in the Detailed CA recommendations (see Appendix B.2) that selected DCC peripherals and associated power supplies be replaced due to obsolete computer technology.

4.1.1.2.12 *Review of Electrical Systems*

System Number: 0420

System Health Report Name: Class I
Review Period for System Health Report: Q4-2015
Overall System Health Rating: Green

Unit 1	Unit 2	Unit 3	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8
Green	Green	Green	Green	Green	Green	Green	Green

System Health Report Name: Class II
Review Period for System Health Report: Q4-2015
Overall System Health Rating: Green

Unit 1	Unit 2	Unit 3	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8
Green	Green	Green	Green	White	Green	Green	Green

System Health Report Name: Class III / Class III Transfer
Review Period for System Health Report: Q4-2015
Overall System Health Rating: Green

Unit 1	Unit 2	Unit 3	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8
Green	Green	Green	Green	White	Green	Green	Green

System Health Report Name: Class IV / Class IV Transfer
Review Period for System Health Report: Q4-2015
Overall System Health Rating: Green

Unit 0	Unit 1	Unit 2	Unit 3	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8
Green	Green	Green	Green	Green	Green	Green	Green	Green

System Health Report Name: Main Power Output
Review Period for System Health Report: Q4-2015
Overall System Health Rating: Green

Unit 0	Unit 1	Unit 2	Unit 3	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8
Green	White	Green	Green	White	Green	Green	Green	Green

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

- CG 008719, Unit(s) 1,3,4: Heat Tracing, Cat 1/2 –
    1. Ensure WOs 3014186, 2332936 are completed for replacement of 1-51200-HX1/2.
    2. Complete the proactive replacement of the System Service Transformer (SST) Heat Exchangers (1-52100-HX1, 4-52100-HX1, 1-52100-HX2 and 4-52100-HX2) and the GST Heat Exchangers (1-52200-HX2, 4-52200-HX2, 1-52200-HX1 and 4-52200-HX2 (example WOs are 04939705, 04939724).

- CG 008751, Unit(s) 1,2,3,4: Rectifier, Cat 1/2 – Complete WOs to replace capacitors and address other deficiencies.

- CG 011284, Unit(s) 018,0,056,078,058,068,5,6,7,8: 53000, 54000, 55000, 65000, 79000 Electrical Systems - Cables - 4.16 kV, 600V, 125V, 250V -
    1. Complete inspection of the MOT/SST/GST and S/Y underground cables per P-2015-03783.
    2. Complete inspection and/or replacement of SES 4kV cable between the P/H and HPECI building per WO# 1681231.

- CG 011287, Unit(s) 5,6,7,8: 57100 Electrical Systems - Control Cable - Complete inspection of the MOT/SST/GST and S/Y underground cables per P-2015-03783.

- CG 011331, Unit(s) 5,6,7,8,058,018,0: 54200 Class II Electrical Systems - UPS / Batteries - Replace 5-54230-UPSB under WO# 4917436.

- CG 011332, Unit(s) 5,6,7,8,056,078,058,018,0: Electrical Systems Relays-Control & Auxiliary - Procure spare CC1 & CC2 relays (under Relay Obsolescence Project 41042).

- CG 011476, Unit(s) 5,6,7,8,056,078,058,018,0: 50000 - Protective Relays-250 / 125 / 48 V D.C. - Replace protective relays on U5-8 MOT, SST & GST (under Project 13-40691) and address protective relay obsolescence issues (under I&C Obsolescence Project 41042).

Incremental recommendations for Plant EOL (2020):

- CG 008686, Unit(s) 1,2,4: Battery, Cat 1/2 –
  1. Replace CH 'J' cells of 1/ 4-54400-BY2.
  2. U1/U4 Class I battery banks 1/ 4-55100-BY1/BY2 to be replaced.

- CG 008698, Unit(s) 1,2,3,4,014: Contactor, Cat 1/2 - Proactive replacement of U1-4 CL II contactors to be completed.

- CG 008719, Unit(s) 1,3,4: Heat Tracing, Cat 1/2 – Procure spare heat exchangers.

- CG008753, Unit(s) 1,2,3,4: Regulator, Cat 1/2 - Replacement of regulators to be completed.
- CG 008754, Unit(s) 1,2,3,4,012,034,014,0: Relay (Protective), Cat 1/2 –
  1. Proactive replacement of ETS relays to be completed.
  2. Calibrate U2/U3 spares so they are available if needed.

- CG 011283, Unit(s) 5,6,7,8: 71310 Electrical Systems AOV's - Procure three MV844/845 and three CV826 for replacement, and actuator overhaul kits in case of valve failure.

- CG 011285, Unit(s) 5,6,7,8,058,018,0: 54100, 54200, 54300, 53300 Electrical System - Motor Control Centres - 600 V / 208 V - Replace all the CC1 and CC2 MCC cells in P058.

- CG 011331, Unit(s) 5,6,7,8,058,018,0: 54200 Class II Electrical Systems - UPS / Batteries -
  1. Replace Units 5,6,7 and 8 UPS-Cs.
  2. Replace DC and AC capacitors of 54230-UPSA and - UPSB at a frequency of 5 years.
  3. Procure one spare for UPSA/UPSB for future maintenance.
  4. Procure replacement for GUI for 5/6/7/8-54230-UPSA/UPSB.
  5. Replace the Class II batteries within the next 2-3 years.

- CG 011524, Unit(s) 5,6,7,8: 54240 Class II Main Transformer - T1/T2 – Procure a spare transformer.

Incremental recommendations for CO EOL (2024):

- CG 008694, Unit(s) 1,2,3,4,012,034,014,0: Circuit Breaker, Cat 1/2 – Perform a one-time overhaul of all the circuit breakers with no active PM.

- CG 008698, Unit(s) 1,2,3,4,014: Contactor, Cat 1/2 - Class IV MCC contactors to be maintained along with the recommendation for one-time maintenance for CL III/ IV MCCs to reach CO EOL (2024).

- CG 008731, Unit(s) 1,2,3,4,014,5,018,0: Motor Control Centre, Cat 1/2 - Perform one-time maintenance for CL III/ IV MCCs to reach CO EOL (2024).

- CG 008742, Unit(s) 1,2,3,4,0: Motor Starter, Cat 1/2 –
  1. Perform one-time inspection of Class II CC1 motor starter cells.
  2. Class III/IV MCC motor starters to be maintained along with the recommendation for one-time maintenance for CL III/ IV MCCs to reach CO EOL (2024).

- CG 008746, Unit(s) 1,2,3,4: Power Supply, Cat 1/2 - Perform proactive replacement of the 48VDC power supply (PS) modules for transfer contactors 1/4 - 54130-CN101/CN102-B2-PS1, and 4-55200-RF50B.

- CG 008751, Unit(s) 1,2,3,4: RECTIFIER, Cat 1/2 - Resolve obsolescence issues with 55200-RF20A/B/C/D/E/F/G/H/T.

- CG 008785, Unit(s) 1,2,3,4,014: Transformers, Cat 1/2 - Implement new PMs for transformers in the group not having a PM, similar to the PMs for the transformers that have PMs.

- CG 011286, Unit(s) 5,6,7,8,056,078,058: 53200,54120,54320,54600 Electrical Systems-Breakers-4.16 kV - Overhaul nine 4kV breakers not covered by PM program.

- CG 011289, Unit(s) 5,6,7,8,058,018,0: 53300, 54130, 54240, 562XX Electrical Systems - Dry-type Power Transformers; Regulating, 9kVA, 4.16kV/600V - Perform a one-time inspection of transformers used for lighting/receptacles (56210/56220-T10/T14) and if found degraded, repair/replace.

- CG 011325, Unit(s) 5,6,7,8: 65425 - 40V DC Distr.System Power Supplies - Perform one-time inspection, and if degraded replace.

4.1.1.2.13 *Review of Emergency Coolant Injection System*

System Number: 0421

System Health Report Name: Emergency Coolant Injection
Review Period for System Health Report: Q4-2015
Overall System Health Rating: White

Unit 1	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8	Unit 056	Unit 058	Unit 078
Green	Green	Green	Green	Green	Green	White	White	White

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

- CG 008835, Unit(s) 1,4: SOLENOID/SOLENOID OPERATED VAL, Cat 1/2 - Complete Work Orders for a one-time replacement of all test SVs.

- CG 011324, Unit(s) 056,078,058: 33350 ECIS - Injection Pumps - Ensure examination of wearing rings at higher magnification is included in existing maintenance program (as recommended in NK30-33350-LOF dated 11-Jun-2010).

- CG 011529, Unit(s) 056,078: 33350 ECI Recovery Pump Motors-4kV - Procure one (1) spare motor for the ECI Recovery Pump Motor strategy documented in NK30-ESI-05600-0000.

- CG 011535, Unit(s) 5,6,7,8: 63330 - ECI - Controller - Hand - CAT 1&2 – Complete strategy to proactively replace hand controllers.

Incremental recommendations for Plant EOL (2020):

- CG 008817, Unit(s) 1,4,014: SWITCH, Cat 1/2 –
    1. Obtain spares for the valve limit switches of 1/4-33350-MV571.
    2. Replace the door switches of 1-63335-NS203 & NS204.
    3. Replace the temperature switch of 014-33350-HTR202-TS1.
- CG 011255, Unit(s) 5,6,7,8,056,078,058: 33350 Emergency Coolant Injection System - Copes-Vulcan AOVs - Complete the procurement of replacements for obsolete Copes-Vulcan ECI valves pneumatic positioners (Cat ID: 653079, 627875, and 606006).

- CG 011259, Unit(s) 5,6,7,8,056,078,058: 33350, 63335 Emergency Coolant Injection System - Check Valves - Perform one-time inspection and overhaul NV3 as necessary, and complete the replacement of 058-33350-NV3020.

- CG 011355, Unit(s) 5,6,7,8,058: 63335 Emergency Coolant Injection System-Hoke & Valcor AOVs - Investigate a more robust replacement to mitigate passing Hoke valves.

Incremental recommendations for CO EOL (2024):

- CG 008817, Unit(s) 1,4,014: SWITCH, Cat 1/2 - Implement new PMs for the position switch of ECI recovery injection isolation valves (33350-V476/V477) and Moderator Room door limit switches (63335- S201/NS203/NS204).

- CG 008835, Unit(s) 1,4,014: SOLENOID/SOLENOID OPERATED VAL, Cat 1/2 – Perform one-time replacement of all SVs to ensure all components in this CG reach CO EOL (2024).

4.1.1.2.14 REVIEW OF EMERGENCY POWER SUPPLY

System Number: 0422

System Health Report Name: Emergency Power Distribution System
Review Period for System Health Report: Q4-2015
Overall System Health Rating: Green

Unit 5	Unit 6	Unit 7	Unit 8	Unit 058
Green	Green	Green	Green	Green

System Health Report Name: Emergency Power Generators
Review Period for System Health Report: Q3-2015
Overall System Health Rating: White

Unit 058
White

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

- CG 011376, Unit(s) 058: 54800 - Emergency Power Supply-Lube and fuel oil Pumps - Complete WO 3199660 / 3199647 for Proactive replacement of EPG1/2-P5.

Incremental recommendations for Plant EOL (2020):

- CG 011368, Unit(s) 5,6,7,8: 54300 - Emergency Power Supply-Rectifiers-48 Volt D.C - Replace major components (e.g., control cards, blocking diodes and SCRs) to support continued operation to Plant EOL (2020).

- CG 011376, Unit(s) 058: 54800 - Emergency Power Supply-Lube and fuel oil Pumps - Add inspection of P1, P2, and P5 to PM 18249.

- CG 011378, Unit(s) 058: 54800 - Emergency Power Supply - Emergency Power Turbine-driven Generator - Perform inspection and overhaul of EPG1 and EPG2 generators. Also, replace the exhaust stacks for EPG1. Note that all remaining equipment tags are subject to inspection and overhaul.

Incremental recommendations for CO EOL (2024):

- CG 011376, Unit(s) 058: 54800 - Emergency Power Supply-Lube and fuel oil Pumps - Procure 3 spare pumps to have one spare for each type of the pumps (058-54800-EPG1/G2-P1, 058-54800-EPG1/G2-P2, 058-54800-EPG1/G2-P5).

4.1.1.2.15 REVIEW OF EMERGENCY WATER SUPPLY (EWS) SYSTEM

System Number: 0423

System Health Report Name: Emergency Water Supply
Review Period for System Health Report: Q3-2015
Overall System Health Rating: Green

Unit 5	Unit 6	Unit 7	Unit 8	Unit 056	Unit 058	Unit 078
Green	Green	Green	Green	Green	Green	Green

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

- CG 008880, Unit(s) 1,4: Valve, Motorized / Motor Operate, Cat 1/2 - Complete repair of 1-71330-MV501.

- CG 011177, Unit(s) 056,078: 71380 EWS Recovery Pump Motors - Procure spare motor as approved in MR #2858124.

- CG 011445, Unit(s) 058: 71380 - Emergency Water Supply (EWS) System - Travelling Screens –
    1. Resolve issues to enable overhaul of screens (WO 1665303 - SC1, WO 1665305 - SC2).
    2. Obtain critical spare parts for traveling screens to be used if required, this will ensure the screens can be replaced quickly if failure occurs prior to end of life.

Incremental recommendations for Plant EOL (2020):

- CG 008836, Unit(s) 1,4: Actuator, Cat 1/2 - Implement a diagnostics program for actuator testing.

- CG 008845, Unit(s) 3,4,0: Expansion Joint, Cat 1/2 - Perform baseline inspection and initiate new PM for routine inspection of Expansion Joints.

- CG 008865, Unit(s) 1,4,0: Strainer, Cat 1/2 - Initiate a PM to complete regular inspection and cleaning of Unit 0 Strainers. Note: All of the remaining equipment tags are subject to cleaning and inspection.

- CG 008878, Unit(s) 018,0: Valve, Manual / Hand Operated, Cat 1/2 - Initiate a PM to inspect condition of valves 018-71330-V2007 and V2008.

- CG 008880, Unit(s) 1,4: Valve, Motorized / Motor Operate, Cat 1/2 - Implement diagnostic testing.

- CG 011440, Unit(s) 5,6,7,8,056,078,058: 71380 Emergency Water Supply (EWS) System - Valves - Manual - Criticality Category 1 (RS2) –
    1. Procure spare parts for maintenance and overhaul. Removed valves to be overhauled and used for future replacements.
    2. Resolve obsolescence issues.
    3. Initiate a one-time replacement of the shear pins on 056-71380-V3 and 078-71380-V3 to prevent reoccurrence of shear pin failure.
- CG 011441, Unit(s) 5,6,7,8,056,078,058: 71380 Emergency Water Supply (EWS) System-Valves - Manual-Criticality Categories 1 (P1, P2) And 2 (RS3) - Address obsolescence issues and source replacement valves.

- CG 011442, Unit(s) 5,6,7,8,058: 71380 Emergency Water Supply (EWS) System-Valves - MOV-Standard-CAT1&2 - Resolve spare parts and obsolescence issues.

- CG 011471, Unit(s) 058: 54320 Emergency Water Supply (EWS) System-Stand Alone Motor Starters-4.16 kV - Perform a one-time refurbishment of motor starters.

Incremental recommendations for CO EOL (2024):

- CG 008850, Unit(s) 0: Heater (Generic), Cat 1/2 - Complete a one-time sample inspection of an HX to ensure condition for CO EOL (2024).

- CG 011177, Unit(s) 056,078: 71380 EWS Recovery Pump Motors - Procure spare bearings.

- CG 011443, Unit(s) 5,6,7,8,056,078,058: 71380 Emergency Water Supply (EWS) System-Non Return Valves -CAT1&2 - Initiate PMs for NVs in accordance with check valve strategy manual P-MAN-04946 00001 that have no PM practices.

- CG 011470, Unit(s) 5,6,7,8: 67138 Emergency Water Supply (EWS) System-Power Transformers - 600V to 120/208V - Implement a one-time transformer inspection program, repair/replace if required.

4.1.1.2.16 *Review of EWS Intake Structure*

System Number: 0425

System Health Report Name: Emergency Water Supply
Review Period for System Health Report: Q3-2015
Overall System Health Rating: Green

Unit 5	Unit 6	Unit 7	Unit 8	Unit 056	Unit 058	Unit 078
Green	Green	Green	Green	Green	Green	Green

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Incremental recommendations for Plant EOL (2020):

- No additional practices are recommended to reach Plant EOL (2020).

Incremental recommendations for CO EOL (2024):

- CG 011463, Unit(s) 058: 23910 - 23910 - EWS Intake Structure-Structural Concrete - Concrete Walls and Slabs of Water Passages - Perform a one-time inspection of the concrete walls and slabs of water passages for EWS intake structure.

4.1.1.2.17 *REVIEW OF FEEDWATER AND CONDENSATE SYSTEM*

System Number: 0426

System Health Report Name: Boiler Feed and Main Condensate System
Review Period for System Health Report: Q4-2015
Overall System Health Rating: White

Unit 1	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8
Yellow	Red	White	White	White	White

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

- CG 008901, Unit(s) 1,4: Controller (LEVEL), Cat 1/2 - Complete WO's 1709429 to 1709433 to replace controllers.

Incremental recommendations for Plant EOL (2020):

- CG 008907, Unit(s) 1,4: Expansion Joint, Cat 1/2 –
    1. Perform one-time inspection of 1-43110-EJ5/6/8/10, and pending inspection findings repair/replace as required.

2. Consider acquisition of spares to address potential replacement depending on inspection result.

- CG 008956, Unit(s) 1,4: Valve, Manual / Hand Operated, Cat 1/2 - Perform one-time inspection for valves 1/4-45310-V22 (inspection to include valve internals), and if degraded repair/replace.

- CG 011414, Unit(s) 5,6,7,8: 43210 / 43110 Feedwater And Condensate System-Piping-Expansion Joints - Procure one spare for each unique CAT ID with priority based on component criticality (i.e. CC1 and CC2).

Incremental recommendations for CO EOL (2024):

- CG 008935, Unit(s) 1,4: Relay, Cat 1/2 - Procure sufficient spares.

- CG 008938, Unit 1,4: Switch, Cat 1/2 - Perform one-time inspection/calibration checks on equipment in this CG not covered under existing PM program.

- CG 011414, Unit(s) 5,6,7,8: 43210 / 43110 Feedwater And Condensate System-Piping-Expansion Joints - Implement a one-time inspection with priority based on component criticality (i.e. CC1 and CC2).

4.1.1.2.18 *Review of Fuel Transfer System*

System Number: 0428

System Health Report Name: Fuel Transfer System
Review Period for System Health Report: Q3-2015
Overall System Health Rating: White

Unit 1	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8
Yellow	Yellow	White	White	White	White

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

- CG 009056, Unit(s) 012,034: Conveyor Unloader –
  1. Complete the implementation of EC 94748 to modify the IFB-A unloader gate cylinder linkage for 034 Conveyor Unloader.
  2. Complete the following work orders: WO’s 3249406 and 3249409 for proactive replacement of the unloader pulley bearings. WO 1832501 to install “basket in position” switch for 034 irradiated fuel receiving bay.

- CG 009057, Unit(s) 1,4,012,034: Cylinder Cat 1/2 - Complete WOs: 4805164 for replacement of parker fitting/tubing on 4-35200-NLR-M1E and 2456198 to inspect/rebuild/replace conveyor stop cylinder.

- CG 009069, Unit(s) 1,4,012,034: Hose, Cat 1/2 - Complete WO#3148365, 3149052, 3149053, 3149054, 3149055 and 3017012 to replace hoses.

- CG 009080, Unit(s) 1,4: Mechanism Cat 1/2 –
    1. Complete EC (DCR) # 124483 for drawing changes to enable timely ram overhauls.
    2. Complete ECR# 19333 to perform the TM U/L Ram position indication change.
    3. Complete FH reliability plan work orders not yet done; WOs 2811775 (P1551), 2811812(P1671), 2814749(P1671), 2811816(P1681), 2806566 (P1681), 2808315 (P1681), 2808335 (P1681), 2814794 (P1681).

- CG 011505, Unit(s) 5,6,7,8: 63524 DC Motor Controllers and Torque Controllers - Procure spares to facilitate prompt replacement if required.

Incremental recommendations for Plant EOL (2020):

- CG 009057, Unit(s) 1,4,012,034: Cylinder Cat 1/2 - Initiate proactive replacement of cylinders on (7 components) 012/034-35200-NGS-M1, 012-35200-NIS/NMS-M1, 4-35200-NVR-M3E, 1/4-35200-NVR-M3W.

- CG 009069, Unit(s) 1,4,012,034: Hose, Cat 1/2 - Complete a one-time inspection of hoses which have not been replaced in the last 2 years.

Incremental recommendations for CO EOL (2024):

- CG 009056, Unit(s) 012,034: Conveyor Unloader - Complete one-time replacement of unloader bearings based on condition of bearings replaced on WO’s 3249406 and 3249409.
- CG 009057, Unit(s) 1,4,012,034: Cylinder Cat 1/2 - Complete a one-time replacement of (19 components) 012/034-35200-NGS/NIS/NMS/NRS/NSS/NTS/NUS/NWS-M1, 4-35200-NVR-M3E, 1/4-35200-NVR-M3W.

- CG 009069, Unit(s) 1,4,012,034: Hose, Cat 1/2 –
    1. Complete a one-time replacement of the D2O flex hoses and all other hoses in the CG.
    2. Complete a one-time hose inspection 2 years after replacement.

- CG 011505, Unit(s) 5,6,7,8: 63524 DC Motor Controllers and Torque Controllers - Perform one-time inspection, and if degraded, repair/replace.

4.1.1.2.19 Review of Fuelling Machine Ancillaries

System Number: 0430

System Health Report Name: Fuelling Machines
Review Period for System Health Report: Q4-2015
Overall System Health Rating: Yellow

Unit 1	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8
White	Yellow	White	White	White	White

System Health Report Name: Fuelling Machine Auxiliaries
Review Period for System Health Report: Q4-2015
Overall System Health Rating: White

Unit 1	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8
White	White	White	Green	White	White

System Health Report Name: Fuel Transfer System
Review Period for System Health Report: Q3-2015
Overall System Health Rating: White

Unit 1	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8
Yellow	Yellow	White	White	White	White

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

There are no incremental recommendations per the Detailed CAs.

4.1.1.2.20 Review of Fuelling Machine Auxiliary Systems

System Number: 0431

System Health Report Name: Fuelling Machine Auxiliaries
Review Period for System Health Report: Q4-2015
Overall System Health Rating: White

Unit 1	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8
White	White	White	Green	White	White

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

- CG 009244, Unit(s) 1,4: Motor, Tank, Cat 1/2 - Complete proactive replacement of PM3 under WO #3007230 (Unit 1) and #3007229 (Unit 4).
- CG 009254, Unit(s) 1,4: Pump, Cat 1/2 - Ensure existing WO's (1458455, 2811100) for overhaul/replacement are completed. Ensure spare pump parts and spare pump assembly procured and ready for replacement.

Incremental recommendations for Plant EOL (2020):

- CG 009244, Unit(s) 1,4: Motor, Tank, Cat 1/2 - Procure new assembly (CID 674769).
- CG 009263, Unit(s) 1,4: Switch, Pressure, Cat 1/2 - Complete a one-time proactive replacement of pressure switches (Unit 1 -WR 854456 & 854463 for PS1 & PS2 respectively) by December 2017.
- CG 011503, Unit(s) 5,6,7,8: 35390, 63530, 63536 D2O Valves - Ensure spare parts for equipment in this CG are procured as part of FH 100 parts initiative to support run to failure maintenance strategy.
- CG 011519, Unit(s) 5,6,7,8: 63530, 63535 FM Oil Hydraulic System Valves – Implement PMs for all tags without PMs in the CG.

Incremental recommendations for CO EOL (2024):

- CG 009263, Unit(s) 1,4: Switch, Pressure, Cat 1/2 - Add a task in existing PMs for control maintenance to calibrate PS's during pump overhaul.

4.1.1.2.21 *Review of Fuelling Machine Carriages and Bridges*

System Number: 0432

System Health Report Name: Fuelling Machine Bridges and Rams
Review Period for System Health Report: Q4-2015
Overall System Health Rating: Yellow

Unit 1	**Unit 4**	**Unit 5**	**Unit 6**	**Unit 7**	**Unit 8**
White	White	Yellow	Yellow	Yellow	Yellow

System Health Report Name: Fuelling Machines
Review Period for System Health Report: Q4-2015
Overall System Health Rating: Yellow

Unit 1	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8
White	Yellow	White	White	White	White

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

- No current initiatives required to be completed for Plant EOL (2020).

Incremental recommendations for Plant EOL (2020):

- CG 009287, Unit(s) 1,4: Bridge, Cat 1/2 - Ensure a set of spare ball nuts are available.

Incremental recommendations for CO EOL (2024):

- No additional practices are recommended to reach CO EOL (2024).

4.1.1.2.22 *Review of Fuelling Machine Head System*

System Number: 0433

System Health Report Name: Fuelling Machines
Review Period for System Health Report: Q4-2015
Overall System Health Rating: Yellow

Unit 1	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8
White	Yellow	White	White	White	White

System Health Report Name: Fuelling Machine Bridges and Rams
Review Period for System Health Report: Q4-2015
Overall System Health Rating: Yellow

Unit 1	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8
White	White	Yellow	Yellow	Yellow	Yellow

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

- CG 009293, Unit(s) 1,4: Magazine, Cat 1/2 - Complete replacement of FM Magazine Ferguson drive bearings and FM Magazine rear shaft bearings per outstanding Work Orders (WO # #2453453, 2453452, 2416636, 2416634).

- CG 011509, Unit(s) 5,6,7,8: 35310, 35313 FM Magazine DR Gearbox & FM Magazine Assembly –
    1. Complete WO 04826430 for 7-35313-MAGAZINE W, to repair / replace a large Graylok seal leak.
    2. Complete WO 02625577 for 7-35313-MAGAZINE W magazine seal replacement.
- CG 011511, Unit(s) 5,6,7,8: 35314, 35317 FM Main Ram Assembly and Ram Adaptor - Complete AR#28153602-06 for new RAM ball screw seals.

Incremental recommendations for Plant EOL (2020):

- CG 009293, Unit(s) 1,4: Magazine, Cat 1/2 - Initiate proactive one-time replacement of the Ferguson drive assemblies at the end of expected design life of Ferguson drive assemblies.

- CG 011509, Unit(s) 5,6,7,8: 35310, 35313 FM Magazine DR Gearbox & FM Magazine Assembly – Resolve obsolescence and issues.

- CG 011511, Unit(s) 5,6,7,8: 35314, 35317 FM Main Ram Assembly and Ram Adaptor - Resolve obsolescence/spares issues.

- CG 011512, Unit(s) 5,6,7,8: 35319 FM Cradle Assembly - Resolve spare parts issues.

Incremental recommendations for CO EOL (2024):

- CG 011509, Unit(s) 5,6,7,8: 35310, 35313 FM Magazine DR Gearbox & FM Magazine Assembly – Revise existing analysis (Reports 30-35310-SR-001 Rev. 01, 30-35310-ASD-001 Rev. 0) to demonstrate adequacy to CO EOL (2024).

4.1.1.2.23 *Review of HPECI Storage Tank*

System Number: 0436

System Health Report Name: Emergency Coolant Injection
Review Period for System Health Report: Q4-2015
Overall System Health Rating: White

Unit 1	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8	Unit 056	Unit 058	Unit 078
Green	Green	Green	Green	Green	Green	White	White	White

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. Detailed CAs have evaluated the work recommended for maintaining current life or extended operating life. Refer to Appendix B.2 for more information.

There are no incremental recommendations per the Detailed CAs.

4.1.1.2.24 REVIEW OF HPECI SUPPLY AND RECIRCULATION

System Number: 0437

System Health Report Name: Emergency Coolant Injection
Review Period for System Health Report: Q4-2015
Overall System Health Rating: White

Unit 1	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8	Unit 056	Unit 058	Unit 078
Green	Green	Green	Green	Green	Green	White	White	White

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

- CG 011207, Unit(s) 058: 71260 HPECI Supply & Recirculation-Valves - NV-Non-Return - CAT1&2 - Repair 058-71310-NV2140.

Incremental recommendations for Plant EOL (2020):

- CG 011207, Unit(s) 058: 71260 HPECI Supply & Recirculation-Valves - NV-Non-Return - CAT1&2 - Perform one-time internal inspection of 058-71260-NV11.

Incremental recommendations for CO EOL (2024):

- No additional practices are recommended to reach CO EOL (2024).

4.1.1.2.25 REVIEW OF INSTRUMENT AIR (B)

System Number: 0439

System Health Report Name: Instrument Air
Review Period for System Health Report: Q4-2015
Overall System Health Rating: White

Unit 1	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8	Unit 014	Unit 058
White	Green	White	Yellow	White	Yellow	Green	White

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here.
The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

- CG 009446, Unit(s) 034: Element, Cat 1/2 - Complete replacement of 034-67513-ME2142 as per WO#1975449.

- CG 011227, Unit(s) 5,6,7,8,056,078,058,018: 75120 Instrument Air Check Valves - Excluding Airlocks - Complete work orders 02906169, 02906171, 02964221, 02908772, 02964232, 02964234, and 02723248, to inspect and replace specified check valves.

- CG 11202, Unit(s) 5,6,7,8: 75120 Instrument Air Manual Valves Inside Containment - Complete current diaphragm valve replacement campaign.

Incremental recommendations for Plant EOL (2020):

- CG 009443, Unit(s) 018: Disc, Cat 1/2- Complete a one-time replacement of the rupture disc.

- CG 009455, Unit(s) 018: Hose, Cat 1/2 – Perform a one-time inspection of the hoses to reach Plant EOL (2020).

- CG 009460, Unit(s) 034: Motor, Cat 1/2 - Implement a new PM to perform inspection and oil change for 034-75130-DRM2047/DRM2048/DRM2049 every year.

- CG 009467, Unit(s) 1,4: Regulator, Cat 1/2- Perform a one-time overhaul of these pressure regulating valves.

- CG 009475, Unit(s) 1,4: Station, Cat 1/2- Implement a TOV replacement program.

- CG 009477, Unit(s) 1,4: Strainer, Cat 1/2 - Update maintenance procedures to inspect the strainers when SRV maintenance is being performed.

- CG 009488, Unit(s) 1,012: Valve, Control, Cat 1/2 - Implement a one-time inspection and/or overhaul program for these control valves.

- CG 009490, Unit(s) 1,034: Valve, Isolation, Cat 1/2 - Implement one-time inspections/replacements of these isolation valves.

- CG 011198, Unit(s) 5,6,7,8,058: 75120 Instrument Air System Dryers - Obtain sufficient spares for Instrument Air CV’s.

- CG 011226, Unit(s) 058: 75120 Instrument Air Pressure Regulating Valves - Resolve obsolescence by procuring an adequate replacement PRV.

Incremental recommendations for CO EOL (2024):

- CG 009446, Unit(s) 034: Element, Cat 1/2 - Implement new PM for routine calibrations.

- CG 009451, Unit(s) 4,018: Gauge, Cat 1/2 - Implement new PM for routine calibrations.

- CG 009460, Unit(s) 034: Motor, Cat 1/2 - Resolve spares issues for 034-75130-CPM2044/ CPM2045/ CPM2046.

- CG 009469, Unit(s) 012,034: Relay, Cat 1/2 –
    1. Implement new PM for routine calibrations/tests.
    2. Resolve obsolescence and spares issues.

- CG 009482, Unit(s) 1,4,018: Tank, Cat 1/2 - Implement new PMs for internal/external inspection of 018-67513-TK2000, 1-75130-TK2001, 4-75130-TK2001, and 1-75130-TK2002.

- CG 011226, Unit(s) 058: 75120 Instrument Air Pressure Regulating Valves - Complete a one-time replacement of all PRVs.

4.1.1.2.26 *Review of Irradiated Fuel Bay Auxiliaries*

System Number: 0440

System Health Report Name: IFB-A, IFB-B, AIFB and Interbay Transfer Systems
Review Period for System Health Report: Q4-2015
Overall System Health Rating: White

Unit 0	Unit 014	Unit 058
White	White	Yellow

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

There are no incremental recommendations per the Detailed CAs.

4.1.1.2.27 REVIEW OF IRRADIATED FUEL BAYS

System Number: 0441

System Health Report Name: IFB-A, IFB-B, AIFB and Interbay Transfer Systems
Review Period for System Health Report: Q4-2015
Overall System Health Rating: White

Unit 0	Unit 014	Unit 058
White	White	Yellow

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

- CG 011318, Unit(s) 058: 21500 – IFB- (Irradiated Fuel Bay) - Complete repairs to liner cracks under Project 13-40703.

There are no incremental recommendations per the Detailed CAs

4.1.1.2.28 REVIEW OF THE LIQUID ZONE SYSTEM

System Number: 0444

System Health Report Name: Liquid Zone Control
Review Period for System Health Report: Q4-2015
Overall System Health Rating: Green

Unit 1	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8
Green	White	Green	Green	White	White

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

- CG 011129, Unit(s) 5,6,7,8: 34810 Liquid Zone Control Compressor Motors - Complete CR2012-01045 to re-instate PM for motor replacement every 4Yr.

- CG 011131, Unit(s) 5,6,7,8: 34810 Liquid Zone Control H2O Supply Valves - Continue replacing positioners with smart positioners as per ECR 15052.

- CG 011374, Unit(s) 5,6,7,8: 34810 Liquid Zone System-Helium Bubbler & Balance Header Pressure And Storage Tank Level Control Valves - Complete outstanding work orders to proactively replace the Bubbler Header Control Valves and the Back-up Balance Header Control Valves.

- CG 011453, Unit(s) 5,6,7,8: 34810- Liquid Zone System- Non-Return Valves - CAT 2 - Continue to execute recommended actions from previous CA NK30-REP-34810-00012 R002 i.e. procure spares for NV225, one-time internal inspection of one of 5-8-34810-NV225 (see AR 1646265, WO 4814904), and inspect one of NV16, NV55, NV58 from U7 due to these valves being in service the longest.

Incremental recommendations for Plant EOL (2020):

- CG 009613, Unit(s) 1,4: Compressor, Cat 1/2 - Address compressor seal flow conditions through MV31/33. This is a known cause for compressor degradation.

- CG 011129, Unit(s) 5,6,7,8: 34810 Liquid Zone Control Compressor Motors - Implement proactive replacement of compressor motors to address concerns regarding premature winding failures (ref. SCR P-2012-16220).

- CG 011131, Unit(s) 5,6,7,8: 34810 Liquid Zone Control H2O Supply Valves - Address vendor quality issues for Unit(s) 1&4 valves leaking through the O-rings.

- CG 011132, Unit(s) 5,6,7,8: 34810 Liquid Zone Control Return Header Isolating Valves - Follow-up on BOM for MVs and ensure spare parts procured.

- CG 011453, Unit(s) 5,6,7,8: 34810- Liquid Zone System- Non-Return Valves - CAT 2 - Complete one-time inspection of 7-34810-NV37.

- CG 011465, Unit(s) 5,6,7,8: 34810 - Liquid Zone System-Tanks - Implement periodic tank inspections.

Incremental recommendations for CO EOL (2024):

- CG 011131, Unit(s) 5,6,7,8: 34810 Liquid Zone Control H2O Supply Valves - Perform a one-time overhaul of the U6 actuators to reach CO EOL (2024).

- CG 011132, Unit(s) 5,6,7,8: 34810 Liquid Zone Control Return Header Isolating Valves - Re-initiate PMs for diagnostics every 4 years e.g. PM 97834-01.

- CG 011374, Unit(s) 5,6,7,8: 34810 Liquid Zone System-Helium Bubbler & Balance Header Pressure And Storage Tank Level Control Valves - Implement one-time internal inspections of valves.

- CG 011465, Unit(s) 5,6,7,8: 34810 - Liquid Zone System-Tanks - Depending on inspected condition, establish procurement route to purchase a new tank.

4.1.1.2.29 REVIEW OF THE MODERATOR SYSTEM

System Number: 0445

System Health Report Name: Moderator and Auxiliaries
Review Period for System Health Report: Q4-2015
Overall System Health Rating: White

Unit 1	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8	Unit 058
White	White	White	Green	White	Green	Green

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

- CG 009671, Unit(s) 1,4: Actuator, Cat 1/2 –
    1. Complete work orders to perform one-time valve and actuator replacement/overhaul: WO 04976469, 04976480, 04976489, 4967469, 04974812, and 1677304.
    2. Complete outstanding work order 03175399 “4-32110-CV27 Replace Calandria Outlet Actuator / Valve”.
- CG 009677, Unit(s) 1,4: Alarm, Cat 1/2 - Complete proactive replacement per WOs 1549266, 1549267, 1549251, and 1549269.

- CG 009680, Unit(s) 1,4: Arrestor, Flame/Fire, Cat 1/2 - Complete WO#3005445 to complete a visual inspection of 1-32310-FA2 to determine the integrity of the flame arrestor. The inspection should include a review of internal screen material to reach Plant EOL (2020).

- CG 009691, Unit(s) 1,4: Controller, HC1/PIC1/TIC1 - Complete proactive replacement of 1/4-63210-L8-HC1 and 1/4-63230-P2-PIC1 per WOs 2073755, 2073756, 3003138, and 3001713.

- CG 009699, Unit(s) 1,4: Disc, Rupture, Cat 1/2 - Complete the outstanding WOs to replace 1-32310-Y6 and 4-32310-Y8/Y9.

- CG 009743, Unit(s) 1,4: Station, Current Output, Cat 1/2 - Complete outstanding WO# 3004764 to replace 1-3231-CV104-HC1.

- CG 009788, Unit(s) 1,4: Transmitters (LT), Cat 1 - Complete one-time replacement for 1/4-63210-L8-LT2 and 1-63210-L1-LT1/LT2 (per WO#02073776 and WO#02073779).

- CG 009792, Unit(s) 1,4: Transmitter (Temp), Cat 1/2 - Complete life cycle replacement for 1/4-63210-T11-TT1 per WO#02953863/4.

- CG 009801, Unit(s) 1,4: EXPANSION JOINT, Cat 1/2 - Complete the inspection's scheduled in WO#03005464 and WO#3003872 for Unit(s) 1 and 4 EJ3.

- CG 009831, Unit(s) 1,4: VALVE, CONTROL, Cat 1/2, Calandria inlet/outlet, DT outlet:
    1. Perform a one-time replacement/overhaul of Calandria outlet valves (1/4-32110-CV2/4/26/27) before EOL per WOs 4976443, 4976469, 4976480, 4976489, 4967469, 4974812, 4974813, 3175399.
    2. Obtain required spare parts for the Calandria outlet valves 1/4-32110-CV2/4/26/27.

- CG 009835, Unit(s) 1,4: VALVE, CHECK/NONRETURN/BACK FL, Cat 1/2 - Complete WOs 03180892, 02791486, 02791484, 02791483 for overhaul and internal inspections.

- CG 011100, Unit(s) 5,6,7,8: 32110 Moderator Head Tank –
    1. Complete outstanding work order WO#3056287 (internal inspection of moisture separator 5-32310-TK1).
    2. Complete work order WO#3005445 to inspect Pickering A flame arrestor 1-32310-FA2. Based on findings, determine if inspection of Pickering B flame arrestors is warranted, and establish work orders for inspection/replacement.

- CG 011109, Unit(s) 5,6,7,8: 32110 Moderator to EWS Tie Non-Return Valves - Complete WO 1580499 to replace 5-32110-NV37.

Incremental recommendations for Plant EOL (2020):

- CG 009671, Unit(s) 1,4: Actuator, Cat 1/2 –
    1. Ensure that there are adequate spare parts for the moderator outlet valve actuators.
    2. Review EQA N-EQA-04944-00036 for the CV actuators and complete a one-time replacement of components which have not been replaced for 30 years.
    3. Review EQA NA44-EQA-04944-00009 for MV9 and complete a one-time replacement of components which have not been replaced for 40 years.

- CG 009691, Unit(s) 1,4: Controller, HC1/PIC1/TIC1 - Resolve obsolescence and spares issues.

- CG 009699, Unit(s) 1,4: Disc, Rupture, Cat 1/2 - Perform a one-time replacement of 1/4-32310-Y7 rupture discs. Note that all the remaining components are subject to replacement.

- CG 009708, Unit(s) 1,4: Element, Cat 2– Resolve obsolescence and spares issues.

- CG 009710, Unit(s) 1,4: Element (FE), Cat 1– Resolve obsolescence and spares issues.

- CG 009717, Unit(s) 1,4: Ion Exchanger (Column), Cat 1/2 - Perform a one-time inspection of Ion exchange columns to determine their condition (pressure boundary and strainer integrity).

- CG 009730, Unit(s) 1,4: Recombination Unit, Cat 1/2 - Implement a one-time replacement of the catalyst and perform an internal and external inspection.

- CG 009756, Unit(s) 1,4: Valve, Control, Cat 1/2, Small Moderator Level CV - Obtain spares either by purchasing or from harvesting Unit 2/3.

- CG 009815, Unit(s) 1,4: Pump, Cat 1/2 - Complete a one-time inspection of one pump internals to assess degradation.

- CG 009818, Unit(s) 1,4: Regulator, Cat 1/2 for Calandria inlet/outlet - Replace all PRV's that have not been replaced since 2015.

- CG 011100, Unit(s) 5,6,7,8: 32110 Moderator Head Tank–
    1. Address spare parts supply issues with 32210-FR1/FR2.
    2. Complete a one-time inspection of selected components from all component groups.

- CG 011101, Unit(s) 5,6,7,8: 32110 Moderator Heat Exchangers - Complete shell side and LPSW inlet/ outlet nozzles inspections. These inspections should be added to the current PM instructions.

- CG 011108, Unit(s) 5,6,7,8: 32110 Moderator Pump Motors-4.16kV - Resolve obsolescence and spares issues.

- CG 011112, Unit(s) 5,6,7,8: 32710 Moderator Liquid Poison Addition Pumps - Purchase one spare pump and perform BOM verification of subcomponent CIDs to ensure spare parts are available.

- CG 011165, Unit(s) 5,6,7,8: 63210 Moderator Wide Range Level Transmitters – Rosemount 1152DP5N - Complete proactive replacement.

- CG 011172, Unit(s) 5,6,7,8: 71310 Moderator HX Temperature Control Valves - Purchase spares for valves: 5/8-32210-MV39/MV40, 5/8-32310-CV1/CV2, & 5/8-32710-MV7/MV8.

- CG 011359, Unit(s) 5,6,7,8: 32310 Moderator Cover Gas Compressors - Complete replacement of compressors under Master EC# 129405 prior to 2018.

- CG 011360, Unit(s) 5,6,7,8: 32110 Moderator System-Valves - HX Manual Isolators - Perform a one-time test/inspection of 5/6/7/8-32110-V20 to reach Plant EOL (2020).

- CG 011363, Unit(s) 5,6,7,8: 32000 Moderator System-Valves - Manual Isolators - Perform a one-time inspection of the CC2 components in this CG and repair / replace as required to reach Plant EOL (2020) (excluding V15 & V21, which have associated PMs in place to replace diaphragm).

Incremental recommendations for CO EOL (2024):

- CG 009677, Unit(s) 1,4: Alarm, Cat 1/2 - Perform a one-time calibrations.

- CG 009680, Unit(s) 1,4: Arrestor, Flame/Fire, Cat 1/2 - Perform a one-time inspection of 1-32310-FA2 to determine the integrity of the flame arrestor if necessary based on inspection results of WO#3005445 to reach CO EOL (2024).

- CG 009699, Unit(s) 1,4: Disc, Rupture, Cat 1/2 - Perform a one-time replacement of 1/4-32310-Y7 rupture discs. Note that all the remaining components are subject to replacement.

- CG 009701, Unit(s) 1,4: Element (temp), Cat 2 - Perform a one-time calibrations on the temperature loops associated with 1-63230-T3/T4-TE1.

- CG 009707, Unit(s) 1,4: Element, Cat 1/2 - Perform one-time calibration on the temperature loops associated with 4-63230-T3/T4-TE1. Note that all the remaining components are subject to calibration.

- CG 009708, Unit(s) 1,4: Element, Cat 1/2 - Perform calibrations at a frequency consistent with the approved IQ Review Maintenance Template.

- CG 009710, Unit(s) 1,4: Element (FE), Cat 1 - Inspect and clean the flow elements and verify their operation every 6 years.

- CG 009717, Unit(s) 1,4: Ion Exchanger (Column), Cat 1/2 - Perform a one-time inspection of Ion exchange column (vessel) to address effects of aging if necessary based on results of the last inspection.

- CG 009730, Unit(s) 1,4: Recombination Unit, Cat 1/2 - Perform a one-time inspection to verify the internal and external condition of the recombination unit if necessary based on the results of the last inspection.

- CG 009738, Unit(s) 1,4: Recorder, Cat 1/2 –
    1. Calibrate the moderator purification conductivity CRs as part of the loop calibration.

2. Perform one-time replacement of the capacitors in all CRs.

- CG 009743, Unit(s) 1,4: Station, Current Output, Cat 1/2 –
  1. Perform calibrations at a frequency consistent with the approved IQ Review Maintenance Template.
  2. Complete replacement of 4-32310-CV104-HC1.

- CG 009770, Unit(s) 1,4: Valve, Pneumatic/ Pneumatic Ac, Cat 1/2, Dump Tank MVs - Implement a one-time seat replacement.

- CG 009783, Unit(s) 1,4: Solenoid/Solenoid Operated Val, Cat 1/2, Dump MVs - Refurbish the SVs as part of the MV PMs.

- CG 009784, Unit(s) 1,4: Solenoid/Solenoid Operated Val, Cat 1/2, Valcor V70900-98-07 - Implement new PMs for the solenoid valves of 32210-MV15 and 32510-MV43.

- CG 009787, Unit(s) 1,4: Transmitters (FT), Cat 1/2 - Complete one-time proactive replacement.

- CG 009801, Unit(s) 1,4: EXPANSION JOINT, Cat 1/2 - Perform a one-time inspection of all expansion joints based on inspection results from WO# 03005464 and WO# 3003872.

- CG 009805, Unit(s) 1,4: Heater (Generic), Cat 1/2 - Perform periodic heater inspection with a contingency task to replace as required.

- CG 009815, Unit(s) 1,4: PUMP, Cat 1/2 - Perform further pump inspections/overhauls as required based on previous sample inspection.

- CG 009818, Unit(s) 1,4: Regulator, Cat 1/2 for Calandria inlet/outlet - Replace all PRV's that have not been replaced by 2019.

- CG 009834, Unit(s) 1,4: Valve, Motorized/Motor Operate, Cat 1/2 - Revise EQ assessment NA44-EQA-04940-00020 to allow extending the life of the existing valves seats to reach CO EOL (2024), or perform replacement.
- CG 011100, Unit(s) 5,6,7,8: 32110 Moderator Head Tank - Perform a one-time inspection of selected components from each of the component groups if required based on results from previous inspections.

- CG 011109, Unit(s) 5,6,7,8: 32110 Moderator to EWS Tie Non-Return Valves - Initiate WOs to inspect 50% of this valve group.

- CG 011357, Unit(s) 5,6,7,8: 32310 Moderator Cover Gas Check Valves –
  1. Initiate a sampling strategy to inspect one valve per valve group and replace based on inspection results.
  2. Procure spare NV's.

- CG 011360, Unit(s) 5,6,7,8: 32110 Moderator System-Valves - HX Manual Isolators - Perform a one-time test/inspection of 5/6/7/8-32110-V20 to reach CO EOL (2024), if required based on previous inspection/test.

- CG 011362, Unit(s) 5,6,7,8: 32110 Moderator System-Valves – Pump Suction Manual Isolators - Initiate a new PM to stroke test CC1/CC2 valves every 104 weeks (or at minimum during unit outages).

- CG 011363, Unit(s) 5,6,7,8: 32000 Moderator System-Valves - Manual Isolators - Perform a one-time inspection/replacement for the CC2 valves in this CG to reach CO EOL (2024) (excluding V15 & V21, which have associated PMs in place to replace diaphragm).

- CG 011365, Unit(s) 5,6,7,8: 32310 - Moderator Cover Gas Heater - 120 Volt - Periodically conduct heater inspection with a contingency task to replace as required.

4.1.1.2.30 *Review of the Powerhouse Emergency Venting System*

System Number: 0447

System Health Report Name: Powerhouse Emergency Venting
Review Period for System Health Report: Q4-2015
Overall System Health Rating: White

Unit 1	Unit 2	Unit 3	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8	Unit 014
Green	Green	Green	Green	Green	Green	White	White	Green

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Incremental recommendations for Plant EOL (2020):

- CG 011166, Unit(s) 5,6,7,8: 67324 Powerhouse Emergency Venting Panels - Replace rusted latching plates on all panels.

- CG 011424, Unit(s) 058: 67322 - Powerhouse Emergency Venting System-SIGNAL CONDITIONER-CAT 1&2 - Resolve obsolescence and spares issues.

Incremental recommendations for CO EOL (2024):

- No additional practices are recommended to reach CO EOL (2024).

4.1.1.2.31 REVIEW OF THE PRIMARY HEAT TRANSPORT SYSTEM

System Number: 0449

System Health Report Name: Primary Heat Transport and Aux
Review Period for System Health Report: Q4-2015
Overall System Health Rating: White

Unit 1	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8
White	White	White	Green	White	White

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

- CG 009924, Unit(s) 1,4: Heat Exchanger (Collection), Cat 1/2 - Complete WO 1418348 &1645320 to replace HXs.

- CG 009984, Unit(s) 1,2,4: Transmitter , Cat 1/2/3 - Complete life cycle replacements and procure spares.

- CG 009991, Unit(s) 1,4: Valve, Manual/Hand Operated, Cat 1/2 - Perform scheduled replacements of elastomeric gaskets/packing and/or valves with materials vulnerable to aging (e.g. diaphragm valves).

- CG 009998, Unit(s) 1,4: Valve, Check/Nonreturn/Back Fl, Cat 1/2, Swing HTS-05, 11 - Inspect/repair/replace 4-33310-NV14.

- CG 011117, Unit(s) 5,6,7,8: 33320 Heat Transport Bleed Condensers - Inspection of Unit 7 tube bundle is scheduled under WO 4819201 during P1671. Further inspections for P058 side tube bundles to be scheduled based on the outcome of this inspection.

- CG 011243, Unit(s) 5,6,7,8: 33120 HT Pump Discharge & Boiler Inlet/Outlet Isolation Valves - Complete planned work orders for gasket/packing replacement and actuator overhaul.

- CG 011258, Unit(s) 5,6,7,8: 33000 Primary Heat Transport System - Check Valves - Swing Type - Inspect valve internals of selected check valves:

  - 5-33340-NV312 (WO# 2719342) Inspect NV,
  - 8-33340-NV312 (WO# 2878274) Inspect NV,
  - 5-33710-NV8 (WO# 2722790) to inspect and overhaul NV.

- CG 011270, Unit(s) 5,6,7,8: 33000 Primary Heat Transport System-Valves - MOV-Standard-CAT1&2 - Complete WOs for actuator overhauls.

- CG 011352, Unit(s) 5,6,7,8: 33000 Primary Heat Transport System - Check Valves - Ball & Piston Type - Complete planned Work Orders to inspect valve internals of selected check valves (i.e. 5-33330-NV216, 8-33330-NV216, 7-33310-NV2 and 8-33310-NV2).

Incremental recommendations for Plant EOL (2020):

- CG 009921, Unit(s) 1,4: Heat Exchanger(Bleed Cooler), Cat 1/2 - Perform one-time inspection of heat exchangers, and if degraded repair / replace.

- CG 009923, Unit(s) 1,4: Heat Exchanger (Gland), Cat 1/2 –
  1. Remove an HX from U2 or U3, inspect to confirm good condition, then use it to replace an HX from Unit 4. The HX from U4 to be used to determine the current condition of the active gland cooler heat exchangers.
  2. Based on findings, create WOs for replacement of HXs.
  3. Purchase spare HXs as needed based on findings.

- CG 009945, Unit(s) 1,4: MOTOR (PRESS PUMP), Cat 2- Include vibration monitoring in Super Route PM, to reach Plant EOL (2020).

- CG 009946, Unit(s) 1,4: Motor (Recovery), Cat 2 - Complete proactive replacement of motors.

- CG 009947, Unit(s) 1,4: Motor (Transfer), Cat 1 - Implement new PM to perform electrical testing every 2 years and vibration analysis to be performed every year.

- CG 009959, Unit(s) 1,4: Pump, Cat 1/2 - Perform a one-time inspection of pump internals (impeller, casing, shaft etc.). Inspect one primary heat transport circulation pump, pressurizing pump and D2O transfer pump. Based on inspection results create supplemental WOs to inspect/overhaul other pumps within this CG.

- CG 009976, Unit(s) 1,4: Tank (33810), Cat 1/2 - Perform a one-time inspection, and if degraded repair/replace (for all tanks).

- CG 009978, Unit(s) 1,4: Tank (BUIA 33610), Cat 1/2 - Perform a one-time inspection, and if degraded repair/replace (for all tanks).

- CG 009991, Unit(s) 1,4: Valve, Manual/Hand Operated, Cat 1/2 –
  1. Resolve obsolescence issues with 1/4-33320-V110/V119, 1/4-33540-V21/V2.
  2. Update SPMP to include all manual valves in CG.

- CG 009993, Unit(s) 1,4: Valve, Motorized/Motor Operate, Cat 1/2 –
  1. Proactively refurbish a contingency of actuators for fast swapping with malfunctioning actuators during outages (for Boiler Isolating Valves and Pump Discharge Valves.
  2. Perform one-time inspection, and if degraded repair/replace.
  3. Replace worn out stem nuts (for all valves).

- CG 009999, Unit(s) 1,4: VALVE, CHECK/NONRETURN/BACK FL, Cat 1/2, Swing HTS-06 - Perform a one-time inspection of valve internals and if degraded, repair/replace valve.

- CG 010001, Unit(s) 1,4: VALVE, CHECK/NONRETURN/BACK FL, Cat 1/2, Ball & Piston, HTS-01, 01A - Perform one-time inspection, and if degraded repair/replace valves.

- CG 010003, Unit(s) 1,4: VALVE, CHECK/NONRETURN/BACK FL, Cat 1/2, Ball & Piston, HTS-14- Perform one-time inspection, and if degraded repair/replace valves.

- CG 010005, Unit(s) 1,4: VALVE, CHECK/NONRETURN/BACK FL, Cat 1/2, Swing HTS-02 - Perform a one-time inspection of a valve from this CG. Based on the findings generate WOs for inspection/overhaul/replacement.

- CG 010007, Unit(s) 1,4: VALVE, CHECK/NONRETURN/BACK FL, Cat 1/2, Ball&Piston, HTS-23 - Perform a one-time sample inspection of 1-33610-NV3001. Determine if further inspections are required from the results of inspection.

- CG 010008, Unit(s) 1,4: VALVE, CHECK/NONRETURN/BACK FL, Cat 1/2, Ball&Piston, HTS-18 - Perform one-time inspection for one representative valve in each unit. From inspection results, determine if any further inspections and/or replacements are required.

- CG 010009, Unit(s) 1,4: VALVE, CHECK/NONRETURN/BACK FL, Cat 1/2, Ball&Piston, HTS-16 - Inspect 1/4-33120-NV49, 1/4-33120-NV54, 1/4-33120-NV58 & 1/4-33120-NV64 and determine if further inspections or replacements are required based on inspection results.

- CG 010020, Unit(s) 1,4: SOLENOID/SOLENOID OPERATED VAL, Cat 1/2 - Obtain sufficient spares for 1/4-33310-MV5-SV1, 1/4-33330-MV217-SV1 & 1/4-34960-MV3008-SV1.

- CG 011116, Unit(s) 5,6,7,8: 33320 Bleed Condenser Inlet Isolating Valve –
  1. Add a task to the current PM actuator overhaul to replace the limit switch.
  2. Create separate PMs for 5-8-33320-MV103 stroke tests.

- CG 011117, Unit(s) 5,6,7,8: 33320 Heat Transport Bleed Condensers - Procure spares as required to support inspection/tube bundle replacement.

- CG 011244, Unit(s) 5,6,7,8: 33710 HT D2O Inter-Unit Transfer Pump - Acquire one spare pump to enable replacement in the event of component failure.

- CG 011258, Unit(s) 5,6,7,8: 33000 Primary Heat Transport System - Check Valves - Swing Type - Resolve obsolescence issues and obtain necessary spare parts.

- CG 011270, Unit(s) 5,6,7,8: 33000 Primary Heat Transport System-Valves - MOV-Standard-CAT1&2 - Resolve valve obsolescence issues.

- CG 011274, Unit(s) 5,6,7,8: 33120 Primary Heat Transport Pumps –
  1. Complete a one-time proactive replacement of the HTS pump seals.
  2. Spares/Parts to be procured to support repair/replacement of the seals. (In addition, safe storage of 3 assembled seal cartridges is recommended).
  3. Spares/Parts to be procured to support overhaul of the pump.

- CG 011292, Unit(s) 5,6,7,8: 33000 Primary Heat Transport System - Manual Valves - D2O Inside Containment –
  1. One-time replacement/repair of manual valves with known negative OPEX.
  2. Establish a performance monitoring program to trend of failure rates of Boiler Drain Valves to determine if a future maintenance strategy is required.
  3. Resolve outstanding spare parts issues.

- 011352, Unit(s) 5,6,7,8: 33000 Primary Heat Transport System - Check Valves - Ball & Piston Type - Obsolescence issues to be resolved. Procure replacement valves and spare parts to support valve inspection and potential replacement/overhaul activities.

Incremental recommendations for CO EOL (2024):

- CG 009903, Unit(s) 1,3,4: ELEMENT (TEMP.), Cat 1/2/3 - Initiate new PMs for Boiler Outlet, Pump Motor & Pump Gland Circuit TEs to perform calibrations at a frequency adequate for each temperature loop.

- CG 009924, Unit(s) 1,3,4: Heat Exchanger (Collection), Cat 1/2 - Based on the condition of the replaced HXs, implement PMs to inspect/clean HX at frequency of 4 years, if required.

- CG 009970, Unit(s) 1,4: Strainer, Cat 1/2 - Complete a one-time strainer removal, inspection and replace or clean as necessary. Based on condition of strainers, supplemental WOs to be generated.

- CG 009984, Unit(s) 1,2,4: Transmitter, Cat 1/2/3: Perform one-time replacement of all transmitters without replacement PM.

- CG 009986, Unit(s) 1,4: VALVE, CONTROL, Cat 1/2, 657/667 w/ A or DBQ - Perform a one-time internal inspection of a representative control valve sample from each functional group. Raise supplemental WO's for overhaul or replacement as a result of internal inspection findings.

- CG 009993, Unit(s) 1,4: VALVE, MOTORIZED/MOTOR OPERATE, Cat 1/2 - Perform recurring inspection/lubrication at a frequency of 2 years for all valves/actuators.

- CG 010020, Unit(s) 1,4: SOLENOID/SOLENOID OPERATED VAL, Cat 1/2 –
    1. Perform one-time replacement of associated SV's for 1/4-33910-MV2002 & 1/4-34960-MV3008.
    2. Perform one-time replacement of associated SVs for 1/4-33610-CV1-SV1/2, 1/4-33610-CV2-SV1/2, 1/4-33610-CV3-SV1/2, and 1/4-33610-CV4-SV1/2.

- CG 011119, Unit(s) 5,6,7,8: 33340 Heat Transport Pump Gland Seal Coolers - Perform sample inspections (similar to U2 planned inspections) to ensure condition is acceptable to reach CO EOL (2024). Based on inspection results, as well as U2 inspection results, determine any further AM activities to reach CO EOL (2024).

- CG 011214, Unit(s) 5,6,7,8: 33000 Primary Heat Transport System-TRANSMITTER-ANALYTICAL-CAT 1&2 - Resolve obsolescence and spares issues for transmitters.
- CG 011352, Unit(s) 5,6,7,8: 33000 Primary Heat Transport System - Check Valves - Ball & Piston Type - Perform additional inspections if currently scheduled inspections show degraded condition to ensure condition is maintained to CO EOL (2024).

- CG 011276, Unit(s) 5,6,7,8: 33810 Primary Heat Transport System-Vessels-PV's, Tanks, Drums, etc.-CAT1&2 - Complete a one-time inspection, and if degraded repair/replace (for all tanks).

- CG 011291, Unit(s) 5,6,7,8: 33810 PHT Leakage Collection - Heat Exchanger -
    1. Perform a one-time inspection and if degraded, repair/replace.
    2. Proactively procure replacement heat exchangers for all units.

4.1.1.2.32 *Review of Reactor Building*

System Number: 0452

System Health Report Name: Negative Pressure Containment and H2 Ignition
Review Period for System Health Report: Q3-2015
Overall System Health Rating: White

Unit 1	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8
White	White	Green	White	Green	Green

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

There are no incremental recommendations per Detailed CAs.

4.1.1.2.33 *Review of Reactor Regulating System*

System Number: 0453

System Health Report Name: Reactor Regulating
Review Period for System Health Report: Q3-2015
Overall System Health Rating: Green

Unit 1	**Unit 4**	**Unit 5**	**Unit 6**	**Unit 7**	**Unit 8**
Green	Green	Green	Green	Green	Green

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

- CG 010147, Unit(s) 1,4: POWER SUPPLY, Cat 1/2 - Complete replacement of AC Transfer Panels 4-63715-R1A/R1C-PS1 (ref. EC#30104, WO's 1377367 & 1377370), ensure spares are available for 1/4-63715-R1A/B/C-PS1.

- CG 011455, Unit(s) 5,6,7,8: 63171/63177 - Reactor Regulating System-SIGNAL CONDITIONER-CAT 1&2 - Complete replacement program for AA & CA ESPM modules.

Incremental recommendations for Plant EOL (2020):

- No additional practices are recommended to reach Plant EOL (2020).

Incremental recommendations for CO EOL (2024):

- CG 010135, Unit(s) 1.4: DETECTOR, Cat 1/2 - Replace all detectors that were last replaced before 2004.

- CG 010141, Unit(s) 1,4: ION CHAMBER, Cat 1/2 - Replace all detectors that were last replaced before 2004.

- CG 010147, Unit(s) 1,4: POWER SUPPLY, Cat 1/2 - Complete the rest of the lifecycle replacements of power supplies to reach CO EOL (2024).

4.1.1.2.34 *Review of Recirc Cooling Water (RCW)*

System Number: 0454

System Health Report Name: Recirculating Cooling Water
Review Period for System Health Report: Q4-2015
Overall System Health Rating: Green

Unit 1	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8
Green	Green	Green	Green	Green	Green

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Current initiatives that need to be completed to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

- CG 010158, Unit(s) 1,4: ELEMENT, Cat 1/2 - Perform a one-time inspection of 1/4-67132-T501-TE1 and if degraded replace.

- CG 010163, Unit(s) 1,4: HEAT EXCHANGER, Cat 1/2 - Maintain established PM frequency (cessation of repetitive PM deferral history), and complete outstanding WO#2164403 to obtain a spare floating end cap from Unit 2 and WO#03088812 to replace cover gasket.

- CG 010189, Unit(s) 1,4: VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2 - Complete NICRs to replace Unit 4 MV531/MV560.

- CG 011174, Unit(s) 5,6,7,8: 71320 RCW Heat Exchangers - Maintain established PM frequency (to address deferred PMs). Monitor wall losses and leakages, and raise Work Orders as necessary to address aging management issues.

- CG 011402, Unit(s) 5,6,7,8: 71320 Recirc Cooling Water (RCW)-Expansion Joints:
    1. Investigate an alternate style/type/material of expansion joint that can withstand the piping misalignment, or a way of aligning the piping to prevent the expansion joint from leaking. Once a solution is found, complete replacement of discharge expansion joints EJ544/EJ545/EJ546 under Master NICR 124073.
    2. Complete open WOs to replace both suction and discharge expansion joints and initiate WOs to replace the remaining suction side expansion joints, if necessary, based on the results of visual inspection during pump maintenance.
    3. Complete outstanding corrective WOs to address leakage problems.

- CG 011403, Unit(s) 5,6,7,8: 71320 Recirc Cooling Water (RCW)-Strainers - Complete removal of the filter element from 6-71320-STR501.

Incremental recommendations for Plant EOL (2020):

- CG 011174, Unit(s) 5,6,7,8: 71320 RCW Heat Exchangers –
    1. Procure sufficient spare parts to support potential gasket replacements, as needed.
    2. Proactively establish procurement requirements for new HX, tube bundle and channel cover to reduce procurement time, should a replacement be required.

Incremental recommendations for CO EOL (2024):

- CG 010158, Unit(s) 1,4: ELEMENT, Cat 1/2 - Implement a PM program to test and inspect these TEs.

4.1.1.2.35 *Review of Relief Ducts*

System Number: 0455

System Health Report Name: Negative Pressure Containment and H2 Ignition
Review Period for System Health Report: Q3-2015
Overall System Health Rating: White

Unit 1	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8
White	White	Green	White	Green	Green

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. Detailed CAs have evaluated the work recommended for maintaining current life or extended operating life. Refer to Appendix B.2 for more information. There are no incremental recommendations per the Detailed CAs.

4.1.1.2.36 *REVIEW OF SERVICE WATER SYSTEMS*

System Number: 0456

System Health Report Name: Low Pressure Service Water
Review Period for System Health Report: Q3-2015
Overall System Health Rating: White

Unit 0	Unit 1	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8	Unit 058
Green	Yellow	Yellow	Green	White	White	White	Green

System Health Report Name: High Pressure Service Water
Review Period for System Health Report: Q3-2015
Overall System Health Rating: Green

Unit 0	Unit 1	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8	Unit 058
Green	White	Yellow	Green	Green	Green	Green	Green

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

- CG 010212, Unit(s) 1,2,3,4: CONTROL, Cat 1/2 - Implement late/deferred Predefine WO# 2432900 to overhaul/diagnose 4-71310-CV825.

- CG 010241, Unit(s) 1,4: MOTOR, Cat 1/2, CLIV Pumps - Execute PMs that have not been performed in the last 4 years (e.g. PM #126075).

- CG 010251, Unit(s) 1,4: PUMP, Cat 1, LPSW Pumps -
    1. Complete outstanding weld repair of pump top case and suction bell (WO# 2933902).
    2. Significant negative OPEX (SCR's) suggests that deferred PM activities have resulted in the downgrade of the condition. The PM activities need to be completed as per their approved frequency to reach Plant EOL (2020).

- CG 010252, Unit(s) 1,4: PUMP, Cat 2, HPSW Pumps - Complete the following WOs: WO 2850568 to overhaul pump 1-71340-P3, WO 1667952 to overhaul pump 4-71340-P3, WO 2826402 to replace 1-71340-P4, WO 02854187 to perform daily IR & vibration monitoring 4-71340-P4.

- CG 010290, Unit(s) 2,0: VALVE, CONTROL, Cat 1/2, Diaphragm Actuator, Rising Stem Cooling CVs, Fisher –

1. Complete WO for replacement of 2-71310-CV697 and 2-71310-CV698 (WO# 2480901, 2480903).
2. Complete WO 04861286 for one-time actuator overhaul of 0-71310-CV810.

- CG 010303, Unit(s) 1,4: VALVE, CHECK/NONRETURN/BACK FL, Cat 1/2, LPSW-05, 05A – Implement a one-time inspection/overhaul/repair/replacement of 1-71310-NV20.

- CG 010311, Unit(s) 1,4: VALVE, CHECK/NONRETURN/BACK FL, Cat 1/2, HPSW-08, 09 - Complete outstanding work orders WO 2786118, WO 2968105, WO 2967008 and WO 2967009 to inspect/repair the respective valves.

- CG 010315, Unit(s) 1,2,3,4,0: VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2, Piston, Butterfly Keystone MVs - Complete a one-time overhaul/refurbishment of valve, actuator and instruments for valves 1,4-71310-MV186, MV187.

- CG 010317, Unit(s) 1: VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2, Piston, Butterfly Jamesbury MV- Complete outstanding WO to refurbish valve 1-71310-MV176 and instruments (WO# 02254655).

- CG 010318, Unit(s) 1,4: VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2, Piston, Centerline Butterfly MV –
    1. Complete replacement of valve assembly 1-71310-MV502 (ref. NICR 105583; WO# 2878189).
    2. Overhaul valve and actuator of 1-71310-MV505 (WO 939721).
    3. Replace valve assembly of 71340-MV660 (WO #02417971).

- CG 010320, Unit(s) 4: VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2, Piston, Butterfly Keystone MVs Associated UPP Supply MV and High Pressure pump discharge MV - Complete WOs for valve replacement (1561412, 1561410).

- CG 011170, Unit(s) 5,6,7,8: 71310 Emergency LPSW Pumps – PM deferral will result in further degradation of the pumps. PM's need to be completed at their scheduled frequency to reach Plant EOL (2020).

- CG 011175, Unit(s) 5,6,7,8: 71340 HPSW Pumps - PM deferral will result in further degradation of the pumps. PM's need to be completed at their scheduled frequency to reach Plant EOL (2020).

- CG 011371, Unit(s) 5,6,7,8: 71340 Emergency CLIII HPSW Pump Motors-4kV - Accelerate motor refurbishment program that was started in 2009, to support continued operation.

- CG 011383, Unit(s) 5,6,7,8: 71310, 71340 Service Water Systems-Valves – Manual Diaphragm – Cat 1&2 - Complete Work Orders; 2928072, 3108125, 2928086, 2928089, 2928091, 2928092 and 2940512.

- CG 011387, Unit(s) 056,078,058: 71310 Service Water Systems-Valves – MOV – Butterfly – CAT1&2 – Complete inspections of 056-71310-MV402/ MV403 and 078-71310-MV404/405.

- CG 011388, Unit(s) 5,6,7,8,056,078: 71310 / 71340 Service Water Systems-Valves - AOV-Standard-CAT1&2 - Complete outstanding WOs: WO for 8-71310-CV585 replacement (WO 4907606), WO for 5-71310-CV583 monitoring temp changes (WO 4907649), WO for 5-71310-CV585 replacement (WO 2508842), WO for 7-71310-CV585 troubleshooting (WO 3067647). Generate WR for 6-71310-CV585 replacement (no WO).

- CG 011391, Unit(s) 5,6,7,8,056,078: 71310 / 71340 Service Water Systems-Non Return Valves - CAT1&2 –
    1. Complete a one-time practice per the check valve strategy manual P-MAN-04946-00001 for all equipment tags that are not covered by an active PM to reach Plant EOL (2020).
    2. Complete all Work Orders and Change Requests in the Action List of the CA.
    3. Complete WO’s: 2853616, 04781418 02736496, 02733802, 02626810, 03060255, 01703138, 02626810, 02736588 and 2735252.

- CG 011399, Unit(s) 5,6,7,8: 71310, 71340 Service Water Systems-Valves - Manual Butterfly - Criticality 1 & 2 - Complete WO 02945929, WO 02945931, WO 02657191.

Incremental recommendations for Plant EOL (2020):

- CG 010217, Unit(s) 1,4: EXPANSION JOINT, Cat 1/2, Moderator - Perform a one-time replacement of all expansion joints.

- CG 010219, Unit(s) 1,4: EXPANSION JOINT, Cat 1/2 - Complete a one-time inspection of the EJs in this CG to reach Plant EOL (2020).

- CG 010235, Unit(s) 1,4: LUBRICATOR, Cat 1/2 - Complete a one-time inspection of the lubricators to reach Plant EOL (2020).

- CG 010252, Unit(s) 1,4: PUMP, Cat 2, HPSW Pumps - Complete a one-time task for pump monitoring, inspection and lubrication to reach Plant EOL (2020).

- CG 010262, Unit(s) 1,4: STRAINER, Cat 1/2 -
    1. Complete a one-time internal inspection of 1-71310-STR3310, 1-71310-STR3311, and 1-71310-STR3312 to reach Plan EOL (2020).

2. Increase the frequency of existing PM (e.g. 10938) execution from every 3 years to every 2 years. If performance does not improve, increase the frequency further.

- CG 010290, Unit(s) 2,0: VALVE, CONTROL Cat 1/2,Diaphragm Actuator, Rising Stem Cooling CV's Fisher – Complete a one-time diagnostic/calibration of 0-71310-CV813.

- CG 010300, Unit(s) 1,3,4,0: VALVE, MOTORIZED/MOTOR OPERATE, Cat 1/2-
    1. Complete a one-time diagnostic testing of 0-71310-MV401 and 3-71310-MV2012.
    2. Complete a one-time inspection and lubrication of 1/4-71310-MV2070, 1/4-71340-MV2191, MV2332, 0-71310-MV459, MV401 and 3-71310-MV2012 to reach Plant EOL (2020).

- CG 011383, Unit(s) 5,6,7,8: 71310, 71340 Service Water Systems-Valves – Manual Diaphragm – Cat 1&2 – Replace valves as they reach their 10 year life expectancy.

- CG 011388, Unit(s) 5,6,7,8,056,078: 71310 / 71340 Service Water Systems-Valves - AOV-Standard-CAT1&2 – Complete a sample inspection of one CV, based on results repair/replace as necessary and implement further PM activities for remaining CV's.

- CG 011394, Unit(s) 5,6,7,8,056,078: 71310 Service Water Systems-Valves - Cast Iron Manual Butterfly - CAT1&2 - Replace 5, 7, 8-71310-V370. Replace or refurbish six other valves in this CG based on OPEX.

- CG 011399, Unit(s) 5,6,7,8: 71310, 71340 Service Water Systems-Valves - Manual Butterfly - Criticality 1 & 2 - Perform a one-time replacement of valves as they approach their 10 year service life.

- CG 011400, Unit(s) 5,6,7,8: 71310, 71340 Service Water Systems - Valves - Manual Gate & Globe - Cat 1 & 2 –
    1. Perform a one-time replacement of valves as they approach their 10 year service life.
    2. Procure spare CAT ID 123565.

Incremental recommendations for CO EOL (2024):

- CG 010219, Unit(s) 1,4: EXPANSION JOINT, Cat 1/2 - Complete a one-time inspection of the EJs in this CG to reach CO EOL (2024), depending on outcome of last inspection.

- CG 010235, Unit(s) 1,4: LUBRICATOR, Cat 1/2- Complete a one-time inspection of the lubricators to reach CO EOL (2024), depending on previous inspection condition.

- CG 010240, Unit(s) 4: MOTOR, Cat 1/2, MOVs - Change PM frequency for associated MV diagnostic testing from every 8 years to every 4 years, to reach CO EOL (2024).

- CG 010252, Unit(s) 1,4: PUMP, Cat 2, HPSW Pumps - Complete a one-time task for pump monitoring, inspection and lubrication to reach CO EOL (2024), depending on condition from last completion.

- CG 010256, Unit(s) 1,4: RELAY, Cat 1/2 - Validate status of spare relays to ensure between 5 &10% spares are available to reach CO EOL (2024).

- CG 010262, Unit(s) 1,4: STRAINER, Cat 1/2- Complete a further one-time internal inspection of 1-71310-STR3310, 1-71310-STR3311, and 1-71310-STR3312 to reach CO EOL (2024).

- CG 010269, Unit(s) 1,4: SWITCH, Cat 1/2, Flow Switch - Initiate PM to calibrate and function check CC2 items 67131-FS195, FS196 & FS197 (ref. PM 117096).

- CG 010290, Unit(s) 2,0: VALVE, CONTROL, Cat 1/2, Diaphragm Actuator, Rising Stem Cooling CVs, Fisher –
    1. Complete a one-time overhaul of 0-71310-CV813.
    2. Complete a one-time diagnostics/calibration 2 years after CV813 overhaul.

- CG 010300, Unit(s) 1,3,4,0: VALVE, MOTORIZED/MOTOR OPERATE, Cat 1/2 - Complete a further one-time inspection and lubrication of 1/4-71310-MV2070, 1/4-71340-MV2191, MV2332, 0-71310-MV459, MV401 and 3-71310-MV2012 to reach CO EOL (2024).

- CG 010320, Unit(s) 4: VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2, Piston, Butterfly Keystone MVs Associated UPP Supply MV and High Pressure pump discharge MV - Complete a one-time actuator overhaul for 4-71340-MV28/29.

- CG 010329, Unit(s) 4: SOLENOID/SOLENOID OPERATED VAL, Cat 1/2, Associated UPP Supply MV and High Pressure pump discharge MV- Implement PM to overhaul/replace SVs during MV overhaul PM which is currently set at 208 week interval.

- CG 010337, Unit(s) 1,4: TRANSFORMER, Cat 1/2- Perform a one-time inspection of transformers and if degraded repair/replace to ensure equipment reaches CO EOL (2024).

- CG 011170, Unit(s) 5,6,7,8: 71310 Emergency LPSW Pumps - Complete a one-time detailed inspection of the concrete supporting pads.

- CG 011370, Unit(s) 5,6,7,8: 71310 Emergency LPSW Pump Motors-4kV - Refurbish all ELPSW pump motors before 2020 to restore condition of all ELPSW motors to reach CO EOL (2024).

- CG 011383, Unit(s) 5,6,7,8: 71310, 71340 Service Water Systems-Valves – Manual Diaphragm – Cat 1&2 - Replace the valves that have not been replaced for 10 years.

- CG 011387, Unit(s) 056,078,058: 71310 Service Water Systems-Valves – MOV – Butterfly – CAT1&2 - Complete a one-time actuator inspection for 056-71310-MV402/403 and 078-71310-MV404/405 similar to PM#18360.

- CG 011399, Unit(s) 5,6,7,8: 71310, 71340 Service Water Systems-Valves - Manual Butterfly - Criticality 1 & 2 - Perform a one-time replacement of valves that have not been replaced for 10 years.

- CG 011400, Unit(s) 5,6,7,8: 71310, 71340 Service Water Systems - Valves - Manual Gate & Globe - Cat 1 & 2- Perform a one-time replacement of valves that have not been replaced for 10 years.

4.1.1.2.37 *Review of Shield Cooling System*

System Number: 0458

System Health Report Name: End Shield Cooling
Review Period for System Health Report: Q4-2015
Overall System Health Rating: Green

Unit 1	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8
Green	Green	Green	Green	Green	Green

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

- CG 010363, Unit(s) 1,4: VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2 - Ensure 1-34310-MV1/MV5 limit switches are replaced per EC 67465 and EC 67464.

- CG 010373, Unit(s) 1,4: ALARM, Cat 1/2 - Complete WO 2796608 to correct deficiencies with 1-63431-L1-L8060IA1 and L4-LIA1.

- CG 011126, Unit(s) 5,6,7,8: 34110 End Shield Cooling Heat Exchangers -
  1. Complete outstanding WOs 02868142, 02868136, 02868103 to replace rear channels.
  2. Complete WOs 02320095, 02839923 for inspection and cleaning activities.

- CG 011479, Unit(s) 5,6,7,8: 34110 End Shield Cooling System-Non Return Valves -CAT1&2 - Complete WOs 02719289, 02733800, 02737230, 02734283, 02525539, 02734357 to overhaul/replace NVs.

Incremental recommendations for Plant EOL (2020):

- CG 010353, Unit(s) 1,4: EXPANSION JOINT - Add a task to routine RB operator rounds to perform a visual inspection of the expansion joints. Look for signs of mechanical/thermal degradation of the joints, as well as material deterioration resulting in cracking and failure of the pressure boundary.

- CG 010357, Unit(s) 1,4: RELAY, Cat 1/2 –
  1. Implement a PM to periodically replace the relays.
  2. Implement a PM to perform periodic inspection and testing.
  3. Populate model information and Cat ID in Asset Suite for 1/4-34310-MV5-R61.
  4. Contact vendor to confirm spare part availability for 1/4-34310-MV5-R61 following identification of the model(s).

- CG 011479, Unit(s) 5,6,7,8: 34110 End Shield Cooling System-Non Return Valves -CAT1&2 - Resolve spare parts issue associated with Cat ID 118501.

Incremental recommendations for CO EOL (2024):

- CG 010356, Unit(s) 1,4: VALVE, CONTROL, Cat 1/2-
  1. Re-activate PMs to calibrate the actuator & components every 728 days.
  2. Complete a one-time actuator overhaul / replace sub-components for all control valves.

- CG 010372, Unit(s) 4: VALVE, CHECK/NONRETURN/BACK FL, Cat 1/2 - Implement PMs for NV replacements.

- CG 010373, Unit(s) 1,4: ALARM, Cat 1/2 - Perform proactive replacement of Alarm Unit(s).

- CG 011479, Unit(s) 5,6,7,8: 34110 End Shield Cooling System-Non Return Valves -CAT1&2 - Perform one-time NIT for valve groups consistent with P-MAN-04946-00001.

4.1.1.2.38 REVIEW OF SHUTDOWN COOLING SYSTEM

System Number: 0459

System Health Report Name: Primary Heat Transport and Aux
Review Period for System Health Report: Q4-2015
Overall System Health Rating: White

Unit 1	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8
White	White	White	Green	White	White

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

- CG 010380, Unit(s) 1,4: HEAT EXCHANGER, Cat 1/2 - Complete the implementation of EC's 107659 and 108155 to install new leak off valves for the isolators on 1, 4-33410-HX-1, 2, 3, 4.

Incremental recommendations for Plant EOL (2020):

- CG 010380, Unit(s) 1,4: HEAT EXCHANGER, Cat 1/2 –
    1. Perform a one-time inspection of 1, 4-33410-HX1, 2, 3, 4 to include inspection of tubes for cleanliness and eddy current inspection.
    2. Complete internal and external inspections of the shell for Microbiologically Induced Corrosion (MIC) on 1, 4-33410-HX1, 2, 3 and 4 every outage.
    3. Initiate a one-time inspection and tube cleaning of 1, 4-33410-HX5, 6, 7, 8.

- CG 011294, Unit(s) 5,6,7,8: 33410 Shutdown Cooling System - AOV's - Perform one-time overhauls of the outstanding depressurization and warm-up valve actuators.

Incremental recommendations for CO EOL (2024):

- CG 011347, Unit(s) 5,6,7,8: 33410 Shutdown Cooling System - Solenoid Valves - Perform one-time proactive replacement.

- CG 010380, Unit(s) 1,4: HEAT EXCHANGER, Cat 1/2 –
    1. Initiate new PMs for tube cleaning and internal inspections of 1, 4-33410-HX5, 6, 7, 8 every 4 years.
    2. Complete shell replacements on 1-33410-HX1, 2.

- CG 010385, Unit(s) 1,4: MOTOR, Cat 1/2 - Reinstate Baker Analysis and Megger Check for pump motors (e.g. PM #6709-1/3).

4.1.1.2.39 Review of Shutdown System (SDS1)

System Number: 0460

System Health Report Name: Shutdown System 1 (SDS1)
Review Period for System Health Report: Q3-2015
Overall System Health Rating: White

Unit 5	Unit 6	Unit 7	Unit 8
White	White	White	White

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Incremental recommendations for Plant EOL (2020):

- CG 011434, Unit(s) 5,6,7,8: 31730 Shutdown System 1 (SDS1) - Major Equipment - Shut-off Unit(s) - Replace SA Rod Ready Reed Switch assemblies on the SA Mechanisms in Unit(s) 5, 6, 7 and 8.

- CG 011468, Unit(s) 5,6,7,8: 63721 Shutdown System 1 (SDS1)-Independent PCB-S/R Perform proactive replacements on Unit 7 & 8.

Incremental recommendations for CO EOL (2024):

- No additional practices are recommended to reach CO EOL (2024).

4.1.1.2.40 Review of Shutdown System (SDS2)

System Number: 0461

System Health Report Name: LISS Shutdown System 2
Review Period for System Health Report: Q4-2015
Overall System Health Rating: Green

Unit 5	Unit 6	Unit 7	Unit 8
Green	Green	Green	Green

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

- CG 011256, Unit(s) 5,6,7,8: 63730 SDS2 Check Valves - Complete all active work orders to replace and/or inspect valve internals.

Incremental recommendations for Plant EOL (2020):

- CG 011128, Unit(s) 5,6,7,8: 34710 LISS Poison Addition & Sampling Pumps - Acquire a spare pump for 34710-P1.

Incremental recommendations for CO EOL (2024):

- CG 011256, Unit(s) 5,6,7,8: 63730 SDS2 Check Valves - Complete one-time replacement of 6-34710-NV217 and 8-34710-NV15.

- CG 011281, Unit(s) 5,6,7,8: 31760 Shutdown System 2 (SDS2)-Piping-Piping Components-CAT1&2- Complete mechanical design evaluation of the LISS Piping for continued operation and implementation of any findings.

- CG 011282, Unit(s) 5,6,7,8: 34710 Liquid Injection Shutdown System (LISS) Tanks:
    1. Perform internal inspections of one of the LISS Tanks in each of the Units (5/6/7/8-34710-TK1, TK2, TK3, TK4, TK5, and TK6).
    2. Perform an internal inspection of the mixing tanks in each of the units (5/6/7/8-34710-TK11).
    3. Perform an internal inspection of the Helium Supply Tank (5/6/7/8-34710-TK10) in each of the units.

4.1.1.2.41 Review of Shutdown System A (SDSA)

System Number: 0462

System Health Report Name: Shutdown System (SDSA)
Review Period for System Health Report: Q3-2015
Overall System Health Rating: Green

Unit 1	Unit 4
Green	Green

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

- CG 010429, Unit(s) 1,4: MOTOR, Cat 1/2 - Proactive replacement to be completed: Backlogged Work Orders to replace all Unit 1, 4 aged Shut Off Rod (SOR) motor to be completed.

- CG 010445, Unit(s) 1,4: SWITCH, Cat 1/2 - Replace all hand switches/pushbuttons when performing SDS tests.

- CG 010451, Unit(s) 1,4: TRANSMITTER, Cat 1/2 - Implement outstanding WOs for the pro-active replacement of Boiler Room High Pressure (BRHP) Pressure Transmitters.

Incremental recommendations for Plant EOL (2020):

- CG 010412, Unit(s) 4: CAPACITOR, Cat 1/2 - Implement comprehensive capacitor monitoring (thermography) PMs.

- CG 010420, Unit(s) 1,4: DETECTOR, Cat 1/2 - Inspect all detectors, if needed repair/replace.

- CG 010425, Unit(s) 1,4: INDICATOR, Cat 1/2 - Complete one-time replacement of meters and position indicators.

- CG 010434, Unit(s) 1,4: POWER SUPPLY, Cat 1/2- Implement a PM for proactive life cycle replacement.

- CG 010456, Unit(s) 1,4: VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2- Replace all valves with USI 63101 and USI 63744 that have not been replaced since 2012.

Incremental recommendations for CO EOL (2024):

- CG 010407, Unit(s) 1,4: AMPLIFIER, Cat 1/2 - Initiate EQ PM to replace the amplifiers, as the amplifiers will be reaching their qualified life.
- CG 010409, Unit(s) 1,4: ANALYZER Cat 1/2 – Perform one-time replacement.

- CG 010412, Unit(s) 4: CAPACITOR, Cat 1/2- Implement proactive replacement PMs.

- CG 010420, Unit(s) 1,4: DETECTOR, Cat 1/2 - Replace all ICFD.

- CG 010456, Unit(s) 1,4: VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2 - Perform one-time replacements of all valves which have not been replaced since 2016.

- CG 010458, Unit(s) 1,4: SOLENOID/SOLENOID OPERATED VAL, Cat 1/2 - Implement one-time replacement.

- CG 010462, Unit(s) 1,4: POWER SUPPLY, Cat 1/2 –
    1. Procure sufficient spares.
    2. Perform one proactive replacement.
    3. Initiate PM to Monitor AC Ripple and DC Output Voltage before start-up.

4.1.1.2.42 *Review of Shutdown System E (SDSE)*

System Number: 0463

System Health Report Name: Shutdown Sys Enhanced (SDSE)
Review Period for System Health Report: Q4-2015
Overall System Health Rating: Green

Unit 1	Unit 4
Green	Green

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

- CG 010468, Unit(s) 1,4: AMPLIFIER, Cat 1/2, Fission Chamber - Complete installation of modified amplifiers per Design ECs 102583 and 102590.

- CG 010496, Unit(s) 1,4: POWER SUPPLY, Cat 1/2 - Proactive replacements to be completed for the power supplies (e.g. WO# 3003011).

- CG 010500, Unit(s) 1,4: RELAY, Cat 1/2 - Replace the relays 1-63731-R93G/R94G/R96G per WO# 3261833.

Incremental recommendations for Plant EOL (2020):

- CG 010468, Unit(s) 1,4: AMPLIFIER, Cat 1/2, Fission Chamber - Resolve spare part issues.

- CG 010469, Unit(s) 1,4: ARRESTOR (Dump Arrest Unit), Cat 1/2- Procure sufficient spares.

- CG 010478, Unit(s) 1,4: DETECTOR, Cat 1/2 - Inspect all detectors, if needed repair/replace.

- CG 010479, Unit(s) 1,4: DETECTOR, Fission Chamber, SPV –
    1. Perform Fission Chamber detector Plateau Checks every 2 years.
    2. Based on the results of the Plateau Checks, replace the detectors if required.
- CG 010499, Unit(s) 1,4: Regulator, Cat 1/2 – Initiate a one-time replacement of all valves in this commodity group.

Incremental recommendations for CO EOL (2024):

- CG 010469, Unit(s) 1,4: ARRESTOR (Dump Arrest Unit), Cat 1/2 - Implement new PMs for calibration.

- CG 010478, Unit(s) 1,4: DETECTOR, Cat 1/2 –
    1. Perform prompt fraction checks as required prior to CO EOL (2024).
    2. Replace all ICFD.

- CG 010496, Unit(s) 1,4: POWER SUPPLY, Cat 1/2 - Implement a new PM for monitoring AC ripple and output voltage.
- CG 010518, Unit(s) 1,4: SOLENOID/SOLENOID OPERATED VAL, Cat 1/2 - Implement new PMs for the solenoid valves for use in SDSE HTS High/Low Pressure Trips.

4.1.1.2.43 *Review of Site Electrical System*

System Number: 0464

System Health Report Name: Class IV / Class IV Transfer
Review Period for System Health Report: Q4-2015
Overall System Health Rating: Green

Unit 0	Unit 1	Unit 2	Unit 3	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8
Green	Green	Green	Green	Green	Green	Green	Green	Green

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. Detailed CAs have evaluated the work recommended for maintaining current life or extended operating life. Refer to Appendix B.2 for more information.

There are no incremental recommendations per the Detailed CAs.

4.1.1.2.44 *Review of Standby Generators*

System Number: 0465

System Health Report Name: Standby Generators
Review Period for System Health Report: Q4-2015
Overall System Health Rating: Yellow

Unit 0	Unit 012	Unit 034	Unit 056	Unit 058	Unit 078
Green	Yellow	Yellow	White	Green	White

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

- CG 010564, Unit(s) 012,034: ACTUATOR, Cat 1/2 –
    1. Complete implementation of Master EC 114279 and WO's 02620689, 02620691 for governor valve replacement.
    2. Complete WO 04802166 to ensure that the louver failure was not a result of actuator failure.

- CG 010572, Unit(s) 012,034: CIRCUIT BREAKER, Cat 1/2 - EC#118495 and WOs# 4857973, 3272375, 44675553, 44675558, 44675563, 44675567, 44675571 to be completed.

- CG 010594, Unit(s) 012,034: FAN, Cat 1/2 - Ensure spare parts/spare fans are available to facilitate repair/replacement/corrective action if fans fail regular function test.

- CG 010603, Unit(s) 012,034: GENERATOR, Cat 1/2 - Complete WOs 4701963, 2834659, 3141105 to restore condition.

- CG 010608, Unit(s) 012,034: HEAT EXCHANGER, Cat 1/2 - Ensure adequate spares are available to facilitate replacement if required.

- CG 010664, Unit(s) 012,034: SWITCH, Cat 1/2 - Complete EC (e.g. 29774) that removes CO2 fire protection system from SGs.

- CG 010693, Unit(s) 012,034: SOLENOID/SOLENOID OPERATED VAL, Cat 1/2 - Complete NICR 125627 for 034 SG3 to replace SV3007.

- CG 011260, Unit(s) 056,078: 54600 Standby Generators - Lube Oil Cooler Dampers/Generator air cooling Dampers –
    1. Perform one-time replacement of Generator air cooling system Dampers (056/078-54600-SG1/2/3-MDP1/2) and spare parts through NICRs.

2. Perform one-time replacement of lube oil cooler dampers and spare parts through NICRs.
3. Perform one-time replacement of actuator motor for the PT lube oil cooler.
4. Procurement engineering to BOM and CAT ID components of new dampers (replaced via NICRs from above recommendations).

- CG 011305, Unit(s) 056,078: 54600 Standby Generators-Standby Generator - Complete WO's for one-time inspections of power turbines (i.e. WO's 02696119, 02696120, 02696121, 02696116, 02696117 and 02696118).

Incremental recommendations for Plant EOL (2020):

- CG 010564, Unit(s) 012,034: ACTUATOR, Cat 1/2 – Complete a one-time inspection of governor actuator.

- CG 010572, Unit(s) 012,034: CIRCUIT BREAKER, Cat 1/2 –
  1. Implement PMs on the CBs that are not subject to already under a PM.
  2. Resolve obsolescence issues on applicable breakers.

- CG 010594, Unit(s) 012,034: FAN, Cat 1/2 - Perform a one-time lubrication of fan/motor bearings, and inspection of fan for wear and proper alignment to reach Plant EOL (2020).

- CG 010602, Unit(s) 012,034: GEARBOX, Cat 1/2 - Complete cost/benefit for obtaining one spare.

- CG 010603, Unit(s) 012,034: GENERATOR, Cat 1/2 - Perform major generator inspection (electrical and mechanical testing) on 012-SGs and 034-SGs.

- CG 010608, Unit(s) 012,034: HEAT EXCHANGER, Cat 1/2 - Complete a one-time cleaning, internal inspection, NDE inspection and leak testing.

- CG 010678, Unit(s) 012,034: VALVE, CONTROL, Cat 1/2 - Perform a one-time inspection of all valves (including valve internals) that are not covered under PM's for periodic replacement (i.e. only CV3029 is regularly replaced). If inspection indicates aging degradation, then repair or replace to reach Plant EOL (2020).

- CG 010680, Unit(s) 012: VALVE, CONTROL, Cat 1/2 - Perform a one-time inspection of valve internals, and repair / replace if degraded to reach Plant EOL (2020).

- CG 010688, Unit(s) 012, 034: VALVE, PRESSURE REGULATING, Cat 1/2 – Complete a one-time inspection, and repair/replace if degraded.

- CG 010645, Unit(s) 012,034,0: PUMP, Cat 1/2 –
  1. Ensure PdM for thermography is conducted routinely (at 12 week frequency) for fuel forwarding pumps only (i.e. 012/0-54660-P101/102/201/202/301/302).
  2. Complete a one-time alignment inspection on all pumps except fuel forwarding pumps (i.e. 012/034-54660-P101/102/201/202/301/302).

- CG 011305, Unit(s) 056,078: 54600 Standby Generators-Standby Generator - Complete a one-time inspection of generator on 056 bank, similar to inspection performed on 078 bank (078-SG2, WO 2146890). Results of inspection to determine if corrective actions or further inspections on other SGs are required.

Incremental recommendations for CO EOL (2024):

- CG 010594, Unit(s) 012,034: FAN, Cat 1/2 - Perform a further one-time lubrication of fan/motor bearings, and inspection of fan for wear and proper alignment to reach CO EOL (2024).

- CG 010608, Unit(s) 012,034: HEAT EXCHANGER, Cat 1/2 – Complete a further one-time cleaning, internal inspection, NDE inspection and leak testing.

- CG 010645, Unit(s) 012,034,0: PUMP, Cat 1/2:
  1. Ensure PdM for thermography is conducted routinely (at 12 week frequency for fuel forwarding pumps only (i.e. 012/034-54660-P101/102/201/202/301/302).
  2. Complete a one-time alignment inspection on all pumps except fuel forwarding pumps (i.e. 012/034-54660-P101/102/201/202/301/302).

- CG 010664, Unit(s) 012,034: SWITCH, Cat 1/2 - Complete proactive replacement of 056/078-54600 SG louvres 11/12 and the actuator motor.

- CG 010670, Unit(s) 012,034: TRANSMITTER, Cat 1/2 - Perform one-time inspection/maintenance checks of all transformers in this CG, and if degraded repair/replace to ensure components reach CO EOL (2024).

- CG 010678, Unit(s) 012,034: VALVE, CONTROL, Cat 1/2 - Perform a further one-time inspection for all valves (including valve internals) that are not covered under PM's for periodic replacement (i.e. only CV3029 is regularly replaced). If inspection indicates aging degradation, then repair or replace to reach CO EOL (2024).

- CG 010680, Unit(s) 012: VALVE, CONTROL, Cat 1/2 - Perform a further one-time inspection of valve internals, and repair/replace if degraded to reach CO EOL (2024).

4.1.1.2.45 REVIEW OF STRUCTURES

System Number: 0466

System Health Report Name: Screenhouse
Review Period for System Health Report: Q3-2015
Overall System Health Rating: White

Unit 012	Unit 034	Unit 056	Unit 078
Yellow	Yellow	Yellow	White

System Health Report Name: Negative Pressure Containment and H2 Ignition
Review Period for System Health Report: Q3-2015
Overall System Health Rating: White

Unit 1	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8
White	White	Green	White	Green	Green

System Health Report Name: Vacuum Building (NPC/ESW) System
Review Period for System Health Report: Q4-2015
Overall System Health Rating: Green

Unit 0	Unit 018
Green	Green

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information. Note, the Calandria Vault structure does not require a System Performance Monitoring Plan, and hence does not have a System Health Report. The CGs in this system were all screened out.

4.1.1.2.46 REVIEW OF VB / EMERGENCY WATER TANK AND SPRAY SYSTEM

System Number: 0469

System Health Report Name: Vacuum Building (NPC/ESW) System
Review Period for System Health Report: Q4-2015
Overall System Health Rating: Green

Unit 0	Unit 018
Green	Green

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

- CG 010748, Unit(s) 0: PUMP, Cat 1 –
  Complete the following:
    1. Complete WOs 2810540 & 2956861 to do a post-mortem inspection of 0-34220-P3 (Sent to OEM, Waiting for Report).
    2. Implement the applicable ACE recommendations written for SCR P-2014-30189, and implement the recommended changes resulting from the OEM inspection.
    3. Complete WOs 2898045 and 2898043 to inspect the strainer/ separator on 0-34220-P1 and 0-34220-P3 respectively.
    4. Complete WO 02956864 to overhaul 0-34220-P1.
    5. Complete WO 04930103. 0-34220-P3 could not shutdown at target pressure. Repair / replace as necessary.
    6. Complete WO 02953941 to rebuild the spare main volume pump and return to stores.
    7. Complete WO 04808854 to investigate an oily mist observed at the pump inboard bearing.
    8. Complete WO 02956863 0-34220-P2. Investigate and fix oil loss issue with P2.
    9. Complete WO 04773326 and install In-pro Seals on P3.
    10. Progress ECR 18400 to install permanent vibration transducers in pump bearing cartridges for improved transmission of high frequency signal.
    11. Complete EC 88952 and install proper oil sampling ports on the pumps.
    12. Complete EC 74633 to change the seal water distribution to each of the main volume vacuum pumps.
    13. Complete EC 83997 to modify the breathers on the main volume vacuum pumps (0-34220-P1, P2, P3) to prevent pump oil leakage.
    14. Complete EC 107060 for 0-34220-P1 and 0-34220-P2 to update the OEM flow diagram to as built conditions.
    15. Complete EC 108025 for P2 0-71620-NICR Demin. Water piping change to support 0-34220-P2 Seal water float valve replacement.

- CG 010753, Unit(s) 018, 0: Relay (U0, 018), Cat 1/2 - Complete a proactive replacement of relays in the system that are experiencing higher failure rates.

- CG 010781, Unit(s) 1,4,018,0: SOLENOID/SOLENOID OPERATED VAL, Cat 1/2 - Execute PMs that have not been performed in the last 10 years (e.g. PM #88954).

Incremental recommendations for Plant EOL (2020):

- CG 010745, Unit(s) 0: MOTOR, Cat 1/2 - Replace Main Vacuum Pump Motors.

- CG 010748, Unit(s) 0: PUMP, Cat 1 – Implement PdM activities to perform vibration testing, infrared readings and oil analysis.

Incremental recommendations for CO EOL (2024):

- CG 010748, Unit(s) 0: PUMP, Cat 1 – Review OEM inspection / recommendation from WOs 2810540, 2956861 and SCR P-2014-30189, and apply to assure condition of components to CO EOL (2024).

4.1.1.2.47 REVIEW OF FUELLING MACHINE

System Number: 0472

System Health Report Name: Fuelling Machines
Review Period for System Health Report: Q4-2015
Overall System Health Rating: Yellow

Unit 1	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8
White	Yellow	White	White	White	White

System Health Report Name: Fuelling Machine Rams and Bridges
Review Period for System Health Report: Q4-2015
Overall System Health Rating: Yellow

Unit 1	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8
White	White	Yellow	Yellow	Yellow	Yellow

On-going Station activities geared at maintaining and/or improving the System Health Rating are included in the System Health Report and are not repeated here. The incremental work identified in the Detailed CAs as being recommended for maintaining current life or extended operating life is listed below. Refer to Appendix B.2 for more information.

Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

- CG 009129, Unit(s) 1,4: Catenary Assembly, Cat 1/2 - Correct configuration management errors in current PMs.

- CG 009134, Unit(s) 1,4: Control, Cat 1/2 - Complete one-time replacement per WOs 2636180, 2636179, 2731627, and 2636257.

- CG 009138, Unit(s) 1,4: Cylinder, Cat 1/2 - Complete one-time replacements of 1/4-63535-NZM-A1E & W and 1/4-63535-NZM-A2E & W via WO#3163574, 3163568, 3163587 and 3163654 (scheduled for 2019) to reach Plant EOL (2020).

- CG 009143, Unit(s) 1,4: Fueling Machine, Cat 1/2 - Resolve outstanding spare parts issues for Fuelling Machines, sufficient spares should be on hand to complete outstanding WOs for replacement and support maintenance.

- CG 009155, Unit(s) 1,4: Meter, Cat 1/2 - Complete one-time replacement of the x-drive train (WO # 2416645, 4940414, 4940416, 2676831) to restore condition to reach Plant EOL (2020).

- CG 009161, Unit(s) 1,4: Motor, Hydraulic, Cat 1 - Complete a one-time replacement of the x-drive motors, including WOs 4940414, 4940416, 2416645 and 2676831.

- CG 009195, Unit(s) 1,4: Valve, Control, Cat 1/2 –
  1. Complete one-time replacements of Latch Ram and Z-Drive valves not completed via WO 2260381, 2260387, 2260389, 2392149, and 2392150.
  2. Complete ECs #106679 to 106686 to replace the 90-degree rotation control valves.

- CG 009197, Unit(s) 1,4: Valve, Mag Pressure Control, Crit 1- Complete WOs 04803924, 4948548 and 4832604 for 1/4-63536-PMA-CV1E/W, REPLACE/OVERHAUL VALVE.

- CG 009200, Unit(s) 1,4: Valve, Hydraulic, Cat 1/2 - Complete WOs for snout valve replacement (QLD-MVs): WO 3006097 (U1E) ACTIVE WO 2917819 (U1W) ACTIVE WO 4822459 (U4E) ACTIVE WO 4731348 (U4W) ACTIVE.

- CG 009206, Unit(s) 1,4: VALVE, CHECK/NONRETURN/BACK FL, Cat 1/2 - Complete WOs: 2106863, 1972285, 2362488, 2362478 to replace Magazine return check valves (FMA-NV2).

- CG 009208, Unit(s) 1,4: Valve, Pneumatic/ Pneumatic Ac, Cat 1/2 - Replace 1/4-63536-QLD-MV3E/MV3W via NICR 131568 and associated WOs (e.g. WOs 2451453, 2451451) and monitor performance to determine replacement frequency based on vendor's expected design life and operating duty.

- CG 009215, Unit(s) 1,4: SOLENOID/SOLENOID OPERATED VAL, Cat 1/2 - Complete currently in-progress NICR (e.g. EC #116372) to address availability of spare parts.

Incremental recommendations for Plant EOL (2020):

- CG 009134, Unit(s) 1,4: Control, Cat 1/2 - Resolve obsolescence and spare part issues.

- CG 009161, Unit(s) 1,4: Motor, Hydraulic, Cat 1 - Complete a one-time replacement of Magazine motors (NMA) upon resolution of vendor quality issue with subject bent axis motors.

- CG 009176, Unit(s) 1,4: Regulator, Cat 1/2 - Complete a one-time replacement of all PRVs in this CG to reach Plant EOL (2020).

- CG 009195, Unit(s) 1,4: Valve, Control, Cat 1/2 –
  1. Perform a one-time replacement of the Latch Ram Speed valves (SLA).
  2. Complete a one-time replacement of the Z-Drive speed valves (SZM).

3. Complete a one-time replacement of the Y-Correction Speed valves (SYC).
4. Complete a one-time replacement of the FM 90 degree rotation valves (SRO).

- CG 009197, Unit(s) 1,4: Valve, Mag Pressure Control, Crit 1- Complete one-time replacement for all valves (except 1/4-63536-PMA-CV1E/W) to reach Plant EOL (2020).

- CG 009209, Unit(s) 1,4: Valve, Pressure Regulating, Cat 1/2 - Complete a one-time replacement of 1/4-63535-PHS-PRV4E/W, 1/4-63536-PSE-PRVE/W and 1/4-63535-PZM-PRVE/W and monitor performance to determine replacement frequency based on vendor's expected design life and operating duty.

- CG 009211, Unit(s) 1,2,4: Valve, Pressure Relief, Cat 1/2 - Complete a one-time replacement of 1/4-63536-PCR-RV2/3/E/W to reach Plant EOL (2020).

- CG 009213, Unit(s) 1,4: Valve, Speed Control, Cat 1/2 - Complete a one-time replacement of all valves in this CG to reach Plant EOL (2020).

Incremental recommendations for CO EOL (2024):

- CG 009118, Unit(s) 1,4: Actuator, Cat 1/2 - Reactivate PMs (#121195-01,12297-01, 121198-01, and 121199-01) to replace the catenary isolation valve and FM D2O valves every 5 years to each CO EOL (2024).

- CG 009136, Unit(s) 1,4: Converter, Cat 1/2 - Perform calibrations at a frequency consistent with approved IQ Review Maintenance Template.

- CG 009138, Unit(s) 1,4: Cylinder, Cat 1/2 - Complete one-time replacements of 1/4-63535-NZM-A1E & W and 1/4-63535-NZM-A2E & W to reach CO EOL (2024).

- CG 009143, Unit(s) 1,4: Fueling Machine, Cat 1/2 - Update recent fatigue analysis for Fuelling Machine Pressure Boundary (i.e. NA44-CALC-35310-00002 R001), to account for life extension to CO EOL (2024).

- CG 009161, Unit(s) 1,4: Motor, Hydraulic, Cat 1 - Complete a one-time replacement of the X-Drive Motors (NXM). Complete a one-time replacement of the Magazine motors (NMA).

- CG 009176, Unit(s) 1,4: Regulator, Cat 1/2 - Complete a one-time replacement of all PRVs in this CG to reach CO EOL (2024).

- CG 009195, Unit(s) 1,4: Valve, Control, Cat 1/2-
    1. Complete a one-time replacement of the Z-Drive speed valves (SZM).
    2. Complete a one-time replacement of the Y-Correction Speed valves (SYC).
    3. Complete a one-time replacement of the FM 90 valves (SRO).

- CG 009197, Unit(s) 1,4: Valve, Mag Pressure Control, Crit 1- Complete a one-time replacement of all valves to reach CO EOL (2024).

- CG 009211, Unit(s) 1,2,4: Valve, Pressure Relief, Cat 1/2 - Complete a one-time replacement of 1/4-63536-PCR-RV2/3/E/W to reach CO EOL (2024).

- CG 009213, Unit(s) 1,4: Valve, Speed Control, Cat 1/2 - Complete a one-time replacement of all valves in this CG to reach CO EOL (2024).

- CG 009215, Unit(s) 1,4: Solenoid/Solenoid Operated Val, Cat 1/2 - Reactivate PMs to replace NMA-SV1, NSC-SV, NYC-SV2 and NZM-SV1 every 5 years to reach CO EOL (2024).

4.1.1.2.48 *Assessment Findings from System Summaries*

Detailed Condition Assessments have been completed for 1202 commodity groups, encompassing all SSCs in Systems Important to Safety and Safe Operating Envelope systems. Detailed recommendations were provided in the previous sections for each of the 47 PSR2 systems. The System Summaries are provided in Appendix B-2 in tabular form and document the CG classifications and related improvement initiatives. Appendix B-2 contains a number of CG entries that are in shaded rows. These shaded rows represent CGs that have now been screened-out as not requiring a DCA based on the 2016 updated screening work[6]. This was done to focus on the CGs that are relevant to PSR2 moving forward. As a result of the updated screening, 546 of the original 1202 CGs remain in SF2 scope compared to Revision 0 of this report (shown in non-shaded rows).

As required in Review Task #1, the condition of SSCs Important to Safety is documented in the System Summaries per OPG's Aging Management Program. For those components not requiring a Detailed CA, their condition is documented via a significant number of robust and comprehensive programs including: System and Component Health Reporting; the Maintenance Program which documents as-found condition of components; Predictive Maintenance, e.g. vibration monitoring; the Corrective Action Program which documents adverse conditions on equipment in SCRs; Annual Reliability Reports; Design Assessments and many other station products/processes. These processes provide confidence that station components are maintained such that their condition is acceptable, or improvement actions are put in place. Further information on this was provided in section 3.1.

As shown in Table B.2, a large number of CGs are classified as "Good" or "Very Good" (58%). Thirty-five percent are assigned a "Satisfactory" classification. Only 6% are assigned a classification of "Poor" and 0.4% are "Very Poor. For the Units 1, 4 and Units

[6] These updated screening reports are approved OPG Controlled Documents.

5-8 Special Safety Systems, there are only two CGs rated “Poor” that have a primary safety function.[7]

Another assessment observation is a large fraction of the CGs rated less than “Satisfactory” are not associated with components in a SOE/SIS having nuclear safety functions, e.g. main boiler feed pump recirculation valves and the boiler steam isolation valves. Another contributing factor for classifications being less than “Satisfactory” is there are a number of cases where the current condition of the CG is “Good”, but future recommendations are required to address obsolescence and/or spare parts procurement.

633 recommendations were identified within the CAs, largely focused on incremental actions to improve condition and/or aging management practices. Approximately thirty percent of the recommendations are current station initiatives (i.e. already identified, planned and in some cases completed after the freeze date, or are in progress). The majority of the recommendations are already in place for CGs classified as “Satisfactory” or lower.

A breakdown on the predominant recommendation types is as follows:

- 32% recommend a one-time replacement (e.g. instrumentation)
- 26% recommend a one-time inspection (e.g. tanks)
- 9% are to procure spare parts

Approximately half of the recommendations pertain to activities that are only required to be executed once before the end of extended station life to 2024. This was identified as an observation in OPG Revision 0 of this report. In many cases, these one-time actions are to perform an infrequent Pre-defined Maintenance (PM) activity, e.g. calibration that was managed by a PM, but given the previously planned station life to 2020, was suspended or cancelled. Other examples falling into this category are life cycle replacements, recommended to maintain component reliability to extended station life. These recommendations will also apply for extended life to 2028.

Other one-time recommendations are associated with performing a one-time inspection on a component for ARDMs that require long periods of time to become apparent, e.g. inspection of motorized butterfly valve or pump internal degradation. This was also identified as an observation in OPG Revision 0 of this report. Valve program activities are primarily associated with actuator capability and set-up and therefore degradation of some

[7] Of the two CGs rated as Poor for the Special Safety Systems, the ECI system has one CG for a hand controller used for testing purposes only and therefore does not have a primary safety function, i.e. does not have a role in the successful actuation of the SSS. There is one CG rated Poor for the Pickering B SDS1. The CG contains switches used for shutoff rod position indication only, i.e. they are not active components required to initiate shutdown the reactor (there are also redundant indications for this function). Also for information, in the updated CAs these components have been classified as Satisfactory. This is based on the good performance, but there is an obsolescence issue that needs to be addressed.

valve internal components, e.g. seats, bodies is not necessarily identified. In general, it is not expected that these one-time inspections will result in unacceptable condition of the component being observed. Other periodic activities, e.g. safety related system tests are in place for critical safety components to confirm safety requirements are met. These one-time inspections are proactive to identify potential degradation prior to component failure.

A number of recommendations pertain to passive components, e.g. piping, cabling, structures, tanks and expansion joints. Some passive components are addressed in OPG Component Programs, N-PROG-MA-0017, “Component and Equipment Surveillance” [87] and other activities. Example programs are:

- Periodic Inspection Program (PIP) for structures, piping and supports in place to fulfill the requirements of CSA Standards and to address emerging OPEX [82];
- Pressure Vessel Certification Program for tanks and other pressure vessels [83];
- Buried Piping Program for Underground Piping [39];
- Cable Surveillance Program, N-PROC-MA-0099 [38];
- Pipe Support Program [90].

Some CGs are screened out based on these effective aging management programs being in place. For components not having a program in place, condition assessments are performed as required for SSCs in the Aging Management program. An example recommendation from these passive component condition assessments are inspection of back-up air tanks for safety related components. The CA process is the vehicle for assessing the condition of this group of passive components and therefore their presence is not unexpected and does not represent a PSR2 gap. Recommendations for internal inspections will be addressed by current station work programs.

However, there are two issues with risk assessments for the Cable and Buried Piping Programs. For the Cable Surveillance Program, the current risk assessments are based on component criticality coding that has not been updated to the latest available. In addition, the condition assessments need to be prepared to reflect this new coding. Since these assessments are not yet complete this results in a PSR2 gap (**Pickering PSR2 Gap SF2-13**).

The Buried Piping Program Risk Assessment identifies activities required to be performed in the program. This risk assessment and buried piping condition assessments have not been updated for extended operation to 2028, resulting in a PSR2 gap (**Pickering PSR2 Gap SF2-14**).

Another observation made in OPG Revision 0 of the report was that many CGs had recommendations to change PMs. A PM is a work management type of activity for completing pre-defined recurring maintenance tasks. Approximately 10% of the recommendations re-instate or create a new PM. These modifications to PMs are one of the types of recommendations that will be assessed, prioritized and tracked in System Health Reports.

A number of improvement initiatives were identified during the completion of the CAs which will or have improved component condition and equipment reliability for life extension, including the:

- Fuel Handling Improvement Initiatives
- DCC Improvements
- Obsolescence Management (see section 4.1.5 for further information)
- Critical Spares Initiative

In addition, OPG has an ongoing “Equipment Reliability Cornerstone”, which is a mechanism used to drive specific initiatives and station behaviors to improve equipment reliability. ER Initiatives from 2016 included:

- Fuel Handling Reliability
- Chemistry Performance Improvement
- Vendor Parts Quality
- Work Management Initiatives to reduce backlogs and increase work completion rates, e.g. backlog blitzes.

All of these initiatives play an important role in maintaining and improving the condition of critical plant equipment. The following additional information is provided for the Fuel Handling Reliability Initiatives as a sample of the work being completed via initiatives.

Fuel Handling Reliability Initiatives

The focus of the Fuel Handling (FH) Reliability Initiative for 2016 was to:

- Identify Critical Spare Parts which support fueling reliability and operational readiness
- Increase the quantity of these critical spares on hand
- Improve the quality of FH project work plans in order for needed projects to be executed safely, per schedule and with the required quality

To support these initiatives a comprehensive review of FH criticality was performed and changes made where required. FH Top 100 Parts Weekly Meetings were put in place to drive improvements in parts availability. Staffing changes were made in Supply Chain and Engineering to put in place higher priority support. Also, governance changes were made to support critical activities.

By mid-2016 the percent of available critical inventory increased to 86%. Also, a FH Equipment Reliability Index (ERI) is in place to track the critical indicators in a number of areas which measure availability and reliability of FH systems. This indicator has improved from 53 in July 2015 to 66 in January 2016.

Work is being completed to update the CAs and preliminary results are discussed in the next section. This work will result in changes to the recommendations contained in the System Summaries and Appendix B-2 Tables.

The following gaps were identified resulting from the System Summaries:

- **Gap SF2-11:** Condition Assessments for civil structures are not complete for station operation to 2028.
- **Gap SF2-12:** Condition Assessments for in-scope piping systems are not complete for station operation to 2028.
- **Gap SF2-13:** The Cable Surveillance Program risk assessment and condition assessments currently use out of date criticality coding.
- **Gap SF2-14:** The Buried Piping Program risk assessment and condition assessments have not been updated for extended operation to 2028.

4.1.1.2.49 *Results of the Updated CAs For Operation to 2028*

In addition to the results from the System Summaries described above, updated Detailed CAs are currently being completed as part of the ongoing Aging Management Program to:

- Use an updated freeze date of January 15, 2016;
- Reflect updated scoping and screening work;
- Include full power operation of the Pickering Units to 2028;
- Address the different phases following station shutdown, e.g. defueling. (This phase of operation is outside the scope of PSR2).

A preliminary overview of the results of these updated CAs follows.

Updated Scoping:

No changes have been made to the scope of systems addressed in the aging management scope for PSR2. The scope still encompasses all Systems Important to Safety and Safe Operating Envelope systems.

Updated Screening:

Updated screening is in process of being completed as part of the current CA revisions for all of the PSR2 systems. The previous screening results that were the basis for CGs in OPG Revision 0 of this report were prepared in 2010/2012. Updated screening was primarily required to address extended operation to 2028 and changes made to criticality coding. As described in the previous section, of the original 1202 CGs in the SF2 report OPG Revision 0, 656 were screened out with the primary basis being:

- Component criticality of the CG has been re-classified to non-critical, i.e. to CC3/4 (and non-RS1/2/3).
- A re-assessment of aging management practices for certain groups of components, e.g. handswitches, push-buttons and other simple devices, have resulted in them being classified as adequate.

A Detailed CA is no longer required for these CGs and their condition is managed via other programs as described in Section 3.1. To maintain traceability from OPG Revision 0 of the SF2 report, these CGs have not been removed from the Appendix B.2 Tables, but have been shaded to show they are no longer applicable to SF2.

Also, based on this updated screening, additional DCAs were determined to be required for SF2 (not included in Appendix B.2), resulting in a total of 716 CGs[8] requiring a Detailed CA be prepared. The primary reasons for the additional number of CAs are:

- Additional components with a reactor safety ranking of RS3 were not screened out.
- Aging management practices were re-assessed, and those that were presently deemed adequate and screened out, now require a DCA to address life extension beyond 2020, e.g. practices with long duration PMs were retired since they would not be required again before 2020.
- One-time internal inspections were conservatively deemed required for life extension for susceptible pumps/valves. A subsequent condition assessment will determine the need for these inspections based on recorded history.

Updated CA Results:

To date, the preliminary results of the Detailed CAs are largely consistent with the System Summaries reported on in Section 4.1.1.2.48 which considered operation to 2024. No states of degradation have been observed that compromise the current design basis or design basis assumptions for operation to 2028.

In general, the condition of the assessed components is not expected to be significantly different from the conclusions of the System Summaries. This is expected since the difference in the freeze dates used in the assessments (changed from August 2015 to January 2016), has not resulted in significant changes to the operating history used in the assessment.

[8] This represents a delta of 170 DCAs from what is in this report, i.e. (716-546 per section 4.1.1.2.48).

Preliminary results from the updated CAs have identified a larger number of recommendations to improve component classifications. For the updated CAs a more in-depth assessment has been made in the areas of obsolescence and spare parts requirements. Approximately half of the updated recommendations made fall into these two categories. Consistent with the criteria in N-PROC-MP-0060 [7], if aging management practices are not effective in only one area, e.g. in obsolescence, the classification rating can be no greater than “Satisfactory”. Therefore, given the findings in the obsolescence and spare parts areas, the classification of the affected CGs can be no greater than “Satisfactory”. The potential impact of this finding is discussed further in Section 4.1.5 in Review Task #5 on Obsolescence.

The majority of the recommendations shown in the System Summaries in sections 4.1.1.2.1 through 4.1.1.2.47 and Appendix B.2 Tables, are still applicable and those required for life extension to 2024 are also required for life extension to 2028. The types of recommendations are the same in the updated CAs, the predominant ones being for:

- Resolving obsolescence
- Procuring spare parts
- One-time inspections
- One-time calibrations
- Component replacements

As described above these results are still preliminary. The set of recommendations will still be assessed and prioritized as per normal practice associated with Aging Management assessments. These recommendations will be managed under OPG’s Equipment Reliability and System and Component Surveillance programs, with required actions being tracked to completion in the System Health Reports.

No new PSR2 gaps have been identified based on the preliminary work completed to date. However, updated Detailed Condition Assessments are not complete for Commodity Groups in the scope of PSR2 for station operation to 2028, resulting in a PSR2 gap (**Pickering PSR2 Gap SF2-15**).

4.1.1.2.50 *Summary Of Review Task #1 Findings – Condition Assessments*

Detailed Condition Assessments have been performed and documented as per the requirements of OPG’s Aging Management Program. The majority of SSCs Important to Safety have been found to have a condition of Satisfactory or better. Recommendations for improvement have been documented where required to improve component condition. These recommendations will be tracked in System and Component Health Reports per normal station practices. As described above, the condition of all SSCs Important to Safety is documented by a comprehensive set of programs. The following gaps have been identified in completion of this Review Task:

- **Gap SF2-11:** Condition Assessments for civil structures are not complete for station operation to 2028.
- **Gap SF2-12:** Condition Assessments for in-scope piping systems are not complete for station operation to 2028.
- **Gap SF2-13:** The Cable Surveillance Program risk assessment and condition assessments currently use out of date criticality coding.
- **Gap SF2-14:** The Buried Piping Program risk assessment and condition assessments have not been updated for extended operation to 2028.
- **Gap SF2-15**: Updated Detailed Condition Assessments are not complete for Commodity Groups in the scope of PSR2 for station operation to 2028.

4.1.2 *Review Task #2 - Maintenance Facilities and Resources*

The purpose of Review Task #2 is to confirm resources and facilities (on and off-site) availability for ongoing plant maintenance. This assessment was performed against the IAEA guidelines ‘Maintenance, Surveillance and In-service Inspection in Nuclear Power Plants’ issued as NS-G-2.6 [40]. These guidelines were selected as they are in general prescriptive and therefore allow a more direct comparison with existing PNGS maintenance facilities than the CNSC Regulatory Standard S-210 ‘Maintenance Programs for Nuclear Power Plants’ [41], which has a similar intent of the IAEA guideline with respect to maintenance facilities and resources.

Additionally, they are referenced in CNSC Regulatory Standard S-210 [41] with regards to maintenance program requirements that Nuclear Power Plant (NPP) licensees must implement.

NS-G-2.6 [40], against which the earlier PSR1 review was also performed, has not changed following the issue of PSR1 NK30-REP-03680-00002 [42]. As such, the following assessment provides an update to the maintenance facilities information in PSR1 NK30-REP-03680-00002 [42]. Additionally, a review was performed on the acceptability of resource allocation to Maintenance for safe operation. The latter follows from page 18 of the PSR2 Basis Document R02 [1] which requires “focus attention on requirements that are new or that have changed in relation to the requirements that were used as the basis for PSR1 ISRs, so that their impact on Pickering NGS can be assessed”.

4.1.2.1 *Maintenance Facilities Description and Operation*

The support facilities available to Pickering NGS were described in PSR1 NK30-REP-03680-00002 [42] and assessed against the recommendations of Sections 8.6 through 8.20 of IAEA Safety Guide NS-G-2.6 [40], Maintenance, Surveillance and In-service Inspection in Nuclear Power Plants.

The IAEA recommendations focus on the following areas for maintenance facilities:

1. Workshop Facilities (IAEA NS-G-2.6 [40] Sections 8.6 to 8.8).
2. Facilities for Maintenance on Radioactive Items (IAEA NS-G-2.6 [40] Sections 8.9 to 8.11).
3. Decontamination Facilities (IAEA NS-G-2.6 [40] Sections 8.12 to 8.14).
4. Other Facilities, Tools and Equipment (IAEA NS-G-2.6 [40] Sections 8.15 to 8.20).

4.1.2.1.1 WORKSHOP FACILITIES

As a summary, IAEA NS-G-2.6 [40] Sections 8.6 to 8.8 makes the following recommendations for workshop facilities:

- Facilities should include a workshop for mechanical, electrical and instrumentation equipment.
- Facilities should be equipped with facilities for processing and storage of records and procedures.
- Facilities should be equipped with necessary work areas for assembly and disassembly.
- Facilities should be equipped with tools and testing equipment needed for maintenance.

Pickering NGS workshop facilities consist of number of function-specific workshops that together meet the intent of recommendations listed above. These facilities consist of Facilities inside the Operating Island and Facilities outside the Operating Island, and are described separately below.

4.1.2.1.1.1 FACILITIES INSIDE THE OPERATING ISLAND

The workshop facilities inside the operating island at Pickering NGS consist of the following:

- The mechanical maintenance shop is located in the service wing extension. The major maintenance activities performed in the shop are repairing mechanical equipment, prefabrication of pipe and supports and pressure boundary repairs. A fitting and overhaul area with suitable work benches for disassembly, repair and reassembly of components is also available.
- The machine shop is located in the service wing on elevation 254'. Machine tools such as lathes, milling machines, pedestal drills, grinders and presses are available. Space for sheet metal and plate fabrication and handling of heavy equipment and materials is also available.
- The control maintenance shops are located on the second floor of the service wing, the service wing extension, and the East Annex. Test benches with appropriate power supplies and pneumatic and hydraulic supplies and test equipment are provided. Calibration and testing facilities for instruments, relays

and portable calibration equipment are provided. Both electrical and instrumentation and control maintenance activities are carried out in the shops.

- The valve testing shop is located on elevation 254' in the service wing. Test facilities are available for maintenance and testing of pneumatic and motorized valve actuators and relief valves. The relief valve testing program for both nuclear and non-nuclear relief valves is under the Relief Valve Certificate of Authorization issued to PNGS by the Technical Standards and Safety Authority (TSSA).
- Test facilities are also available for maintenance and testing of pneumatic and motorized valve actuators and relief valves.
- The welding shop is located in the service wing on elevation 254'. Welding and brazing of pressure boundary materials is carried out in the shop.
- The fuelling machine shop is located on elevation 254’ in the service wing. The repair of fuelling machines, overhaul of contaminated equipment, and decontamination of large parts is carried out in the shop.
- The control maintenance electronic shop is located on elevation 274' in the service wing. Calibration and repair of electronic hardware is carried out in the shop.
- There are two tool cribs located on elevation 254' and 274' in the service wing. All necessary tooling for carrying out activities in the field are stored in the tool cribs.
- A calibration laboratory calibrates measurement and test equipment used at Pickering to site standards that are periodically calibrated to national standards. The climate of the laboratory is controlled. On occasion during the summer, the environment cannot be controlled within specifications. At those times, calibration of measurement and test equipment is prohibited.
- Two annex buildings were constructed for the Large Scale Fuel Channel Replacement Program (LSFCRP). The west annex is beside the west end of the Pickering A reactor auxiliary and turbine auxiliary bays. The east annex is beside the east end of the Pickering B reactor auxiliary and turbine auxiliary bays. Both annex buildings are connected to the powerhouse, and are accessible from the reactor auxiliary bays via truck and personnel doors on the 254’ and 274’ elevations. Transfer of tooling from one annex to the other is along the 254’ elevation reactor auxiliary bay corridor. The two annexes contain:

    - West Annex: Space and facilities are provided for the decontamination and repair of LSFCRP equipment and tooling at the 254’ elevation. A maintenance welding and fabrication shop is located on the 274’ elevation. Also on the upper elevation are offices for station personnel.
    - East Annex: Storage facilities for decontaminated LSFCRP tooling are provided at the 254’ and 274’ elevations. The 254’ elevation at the south end also has a steel-framed and skinned enclosure. This is a controlled environment for pressure tube sampling, or manual decontamination of large flasks and shielding cabinets which cannot be accommodated in the west annex.

Workshop facilities also include provisions for the processing and storage of both paper-based and electronic records.

As recently re-approved by the TSSA on April 15, 2014, the Pickering site holds a Certificate of Authorization (C of A) for carrying out the following work. The C of A has a three year term, next due on April 15, 2017, which covers:

- Fabrication of welded and non-welded Category A, B, D and H type fittings in accordance with CSA Standard B51, Boiler & Pressure Vessel and Pressure Piping Code.
- Repair and alterations of boiler and pressure vessel fittings and piping in accordance with CSA Standard B51, Boiler & Pressure Vessel and Pressure Piping Code.
- Fabrication of process piping in accordance with CSA Standard B51, Boiler & Pressure Vessel and Pressure Piping code and ASME 31.3 Process Piping.
- Fabrication of Class 1, 2 & 3 welded and non-welded Category A, B, D & H type fittings in accordance with CSA N285.0, General Requirements for Pressure Retaining Systems and Components in CANDU Nuclear Power Plants.
- Construction of Class 1, 2, 3 & 4 Piping Systems and Class 1, 2, 3 Shop Assembly, as a Material Organization supplying ferrous and non-ferrous material with material supply from the Central Warehouse.
- Fabrication of Class 1, 2, 3 & 4 welded and non-welded supports with or without design responsibility in accordance with CSA N285.0 General Requirements for Pressure Retaining Systems and Components in CANDU Nuclear Power Plants.

Based on the description above, the Pickering NGS workshop facilities inside the operating island are properly equipped, are capable of maintaining the required equipment, and can address contaminated equipment as needed; which satisfies the recommendations of paragraphs 8.6, 8.7, 8.8, and 8.9 of IAEA Safety Guide NS-G-2.6 [40].

4.1.2.1.1.2 Facilities Outside the Operating Island

The major workshop facilities outside the operating island at Pickering NGS are as follows:

- The east site shops are located just east of Pickering NGS Unit 8. The east site shops include: a pipe fabrication shop, a paint shop, a carpentry shop, a machine shop, a sandblasting shop, and a vehicle maintenance shop. The facility holds a Certificate of Authorization for carrying out pressure boundary work on Class 1, 2, 3 and 6 systems. The facility is primarily used for carrying out modifications for Pickering NGS.

- The Inspection & Maintenance Services (IMS) Division has facilities at the Pickering Training and Mock-up Building (TMB). IMS supports Pickering NGS with steam generator, feeder and fuel channel inspections and CSA N285.4 and CSA N285.5 periodic inspections. IMS carries out specialized inspections for cracks in pipes. IMS also develops specialized inspection tooling.
- Babcock & Wilcox (B&W) has facilities in Cambridge, Ontario. B&W supports Pickering NGS with water lancing of steam generators and development of specialized tooling.
- SNC-Lavalin (previously Atomic Energy of Canada Limited (AECL)) has facilities in Sheridan Park, and CNL (previously AECL) in Chalk River. These facilities support OPG including Pickering NGS with analysis of pressure tube scrape samples, development of specialized tooling and fuel handling support.
- Kinectrics has facilities at the Kipling Research Facility. Kinectrics supports OPG including Pickering NGS with analysis of failed components such as steam generator tubes and with CSA N287.7 inspections.
- Siemens supports Pickering NGS with chemical cleaning of steam generators and turbine-generator maintenance.
- Multiple approved vendors support Pickering NGS in various maintenance activities such as rewinding of electrical motors.

Based on the description above, the major workshop facilities outside the operating island provide the required support and required qualifications to adequately support PNGS maintenance and therefore satisfy the recommendations of sections 8.15 and 8.16 of IAEA Safety Guide NS-G-2.6 [40]. These facilities also assist in meeting the requirements of sections 8.6, 8.7, 8.8, and 8.9 of the Safety Guide.

4.1.2.1.2 *Facilities for Maintenance on Radioactive Items*

As a summary, IAEA Guide NS-G-2.6 [40] Sections 8.9 to 8.11 makes the following recommendations for Facilities for maintenance on radioactive items:

- Maintenance of contaminated plant items.
- Temporary arrangements for work on radioactive plant items.
- Specific considerations such as access control and change rooms, ventilation filters, radiation monitoring and radiation protection facilities, shielding and decontamination facilities.

Workshop facilities are physically located in Zone 2 (in the service wing). This allows plant items to be maintained within permitted levels of fixed contamination (after decontamination). The Tool Cribs also allow segregated storage of tools with permitted levels of fixed contamination. All radioactive work is controlled via station Radiological Processes and Procedures (RP&Ps).

Temporary arrangements for work on radioactive plant items are also addressed in station RP&Ps. As an example, items or areas may be tented to allow contamination control.

Specific considerations include PNGS facilities layout and procedures for access control, change rooms, ventilation discharge filters, radiation monitoring and radiation protection equipment, shielding and decontamination facilities to control spread of contamination and prevent worker exposure. Details are addressed through station RP&Ps and satisfy the recommendations described in IAEA Guide Section 8.11.

Based on the description above, the maintenance facilities for radioactive items satisfy the requirement to contain and minimize radioactive contaminations within acceptable limits control and worker exposure, therefore meeting the recommendations of sections 8.9, 8.10 and 8.11 of IAEA Safety Guide NS-G-2.6 [40].

4.1.2.1.3 *Decontamination Facilities*

As a summary, IAEA Guide NS-G-2.6 [40] Sections 8.12 to 8.14 makes the following recommendations for Decontamination Facilities:

- Facilities to remove radioactive contamination from plant items, tools and equipment prior to their maintenance or any other disposition.
- Adequate worker facilities such as change rooms during maintenance work during peak use.

The decontamination shops are located in the service wing and in the West Annex. On elevation 254' on the north side of the corridor of the service wing, there is an area for decontaminating small parts. On the south side of the corridor, there is an area for decontaminating large parts. On elevation 274' in the service wing there is an area for the decontamination of plastic and rubber goods. The facilities are equipped with access control and changing rooms, ventilation with discharge filters, handling, storage and disposal of liquid and solid radioactive waste, decontamination tanks and equipment capable of decontaminating the largest plant items. Equipment is also provided for local decontamination at worksites.

Change rooms for male employees are located on the 254' and 274' elevations of the service wing. Change rooms for female employees are located on the 274' elevation of the service wing. Change rooms for contractors are located outside the powerhouse near Unit 4 and the 254’ and 274’ change rooms are also available for contractor use. The change rooms have sufficient capacity for peak use during outages.

Based on the above, the decontamination shops at Pickering NGS and the worker facilities such as change rooms adequately support the requirement for safe handling and decontamination of radioactive materials and therefore satisfy the recommendations of sections 8.12, 8.1 3 and 8.14 of IAEA Safety Guide NS-G-2.6 [40]. It should be noted, however, that reactor defueling may present additional challenges such as in equipment utilization rate and OPG may need to evaluate the adequacy of existing decontamination facilities to address post operation demands. However, this is not considered a gap as defuelling is outside the scope of the PSR2 per the PSR2 Basis Document R02 [1].

4.1.2.1.4 OTHER FACILITIES, TOOLS, AND EQUIPMENT

OPG has a Tool Control System (TCS) to control and track tools used for maintenance across OPGN [101]. It also provides a general history of the tools available in the system. In this system, the inspection scheduling and labels for tools requiring periodic inspection can be identified [100]. This will ensure that certain tools, such as gamma metres, are calibrated and inspected to minimize factors that can cause delays and inaccurate findings during maintenance.

In addition, IAEA NS-G-2.6 [40] Sections 8.15 to 8.20 are considered below.

4.1.2.1.4.1 MOCK-UPS

Mock-ups have been used for rehearsing work that is to be carried out in high radiation areas. In addition, mock-ups are used in the development and improvement of tools, equipment and training of personnel. Examples where mock-ups have been used include:

- Development of the Universal Delivery Machine (UDM) for remote inspection of fuel channels. A mock-up was built to develop the tool prior to implementing the tool in the field.
- Development of the Feeder Cutting Tool for carrying out remote cutting of feeder pipe. A mock-up facility was built for tool development and training of personnel.
- Pressure Tube Replacement.
- Cobalt Adjuster Rod Rehearsal Facility and Dynamic Learning Activity mock-ups.

4.1.2.1.4.2 SPECIAL EQUIPMENT

To reduce exposure special equipment is utilized. Some examples of the special equipment used at Pickering NGS are:

- Remote handling manipulators for steam generator tube inspections.
- UDM for pressure tube inspections.
- Remote viewing equipment such as fiberscopes for steam generator secondary side inspections.
- Communication equipment for use inside the reactor building.
- HVAC mobile tent/glove box with dedicated venting.
- Robotic maintenance for high dose work where possible.
- Thermography.

- Remote video recording is used at Pickering NGS to establish baseline and degradation data. Subsequent inspections use video recording and photographs.
- Cobalt Adjuster Element Processing System (CAEPs)

4.1.2.1.4.3 MAJOR LIFTING AND TRANSPORT FACILITIES

Adequate fixed and mobile lifting and transport facilities are provided at Pickering NGS. Examples of the major lifting and transport facilities are as follows:

- The boiler room cranes (31.7 Mg) support lifting in the reactor building. Monorails are provided for moving equipment to the main crane.
- In the reactor auxiliary bay there are hatches for the transfer of reactor building equipment from elevation 274' to grade via a 27.2 Mg monorail.
- The irradiated fuel can be loaded into a shipping flask by the fuel-handling gantry and the flask can be lifted from the bay by a 113 Mg hoist.
- The turbine hall is equipped with two cranes each with a capacity of 130 Mg. The cranes can be operated in tandem for lifting heavy items such as low-pressure turbine spindles.

4.1.2.1.4.4 SUMMARY OF OTHER FACILITIES, TOOLS, AND EQUIPMENT

The review did not identify any shortcomings with regard to the recommendations of paragraphs 8.6 through 8.20 of IAEA Safety Guide NS-G-2.6, Maintenance, Surveillance and In-service Inspection in Nuclear Power Plants [40].

Note: Station SSCs that identify any system or equipment findings which might result from gaps in the maintenance facilities or resource allocation are outside the scope of this assessment, however the findings from the CAs are within Review Task #1 in Section 4.1.1.2 where applicable.

4.1.2.2 MAINTENANCE RESOURCES

PNGS conducts numerous internal audits and self-assessments in all areas including maintenance. Internal processes at OPG require that recommendations from internal audits and self-assessments generate SCRs, when gaps in performance are identified. A review was performed of the PNGS SCR database for the period 2011-2015 for performance gaps in maintenance facilities or maintenance resource allocation. The SCR review did not identify any significant findings.

This assessment also reviewed the findings of industry audits of OPG. The results of this review did not identify any significant findings related to maintenance facilities or maintenance resource allocation. The audits did identify a maintenance backlog finding, but these are not necessarily attributable to maintenance resources.

In 2003, the CNSC raised Action Item 2003-8-01 to track the Pickering B maintenance backlogs. In 2009, CNSC staff acknowledged [43] that the corrective maintenance backlogs at Pickering B had met its target. In 2012, "CNSC staff agreed that the Deficient (previously called Elective) Maintenance backlog at Pickering B is now consistent with the industry benchmark target and that the Corrective Maintenance backlog at Pickering B is also now consistent with other OPG facilities. Hence, Action Item 2003-8-01 is hereby closed." CNSC also indicated they expected OPG to continue to monitor backlogs to ensure continuous improvement (per CNSC correspondence [44]). All backlogs are tracked and documented in the OPG Self-Assessment database as per the requirements in Section 1.1 of N-PROC-MA-0008 [45]. Gaps are documented and tracked via SCRs.

On-Line maintenance work scheduling is performed as per the requirements in OPG procedure N-PROC-MA-0022 [46]. It is intended to ensure "maintenance work is organized, well-coordinated station collaboration by which fully planned work, system and component tests, Corrective Maintenance (CM), Deficient Maintenance (DM), Preventive maintenance (PM), Other Maintenance (OM) and Projects are systematically identified, scoped, scheduled, executed, monitored and reported". Key parameters such as the corrective maintenance backlog target by Unit are also updated weekly and tracked against the Target backlog. The results are available to all OPG personnel to review on the intranet.

It should be noted that there are station systems with "Yellow" or "Red" categorized maintenance backlogs. However, these are not attributable to facilities deficiencies or lack of maintenance resources.

Maintenance performance is assessed on a quarterly basis that use the methodology described in INPO-05-005: Guidelines for Performance Improvement at Nuclear Power Stations [47], and newly revised N-INS-01966.1-10000 R05, "Trending Analysis Instruction and Performance Improvement Reporting" [48]. Any findings, which includes gaps in the adequacy of maintenance facilities and/or maintenance resources allocation that could impact Station safety performance is monitored and tracked to completion. The data from the most recent assessment are documented in report P-REP-01966-0575908 [49].

A quarterly report is provided to the CNSC on safety performance indicators including indicators such as maintenance backlogs (see above), Forced Loss Rate performance metrics pursuant to the Regulatory Document REGDOC-3.1.1 Reporting Requirements for Nuclear Power Plants [98], Section 3.1, 25 Safety Performance Indicators.
Any impacts from facilities or resource gaps would be reflected in the performance indicators. The data from the most recent assessment are documented in correspondence P-CORR-00531-04625 [50].

The Pickering NGS Licence Renewal Application also documented improvements in Equipment Reliability, reflecting acceptability of many factors including maintenance resource allocation. “The industry recognized indicator for Forced Loss Rate (FLR) has improved by approximately 40%.” as documented in reference [51]. The FLR data from 2011 onwards is provided in letters between OPG and the CNSC in References [50], [52], [53] and [54]. The FLR for the six Pickering Units was 2.9% and 4.1% in 2016 [95], which demonstrates significant improvement in station performance (a 66% reduction since 2011). Refer to Appendix D: Forced Loss Rate (FLR) Data for visual representation of the FLR trend for the past six years.

Training and development of staff at OPG, including Maintenance staff, follow the requirements in N-PROG-TR-0005, R016 ‘Training’ [55]. This Training Program provides the structure, processes, and tools for defining, developing, implementing, documenting, assessing, and improving the training required to ensure Nuclear staff have the appropriate knowledge, skill, and attitudes for safe and efficient plant operation. The Program applies to trained performance areas, which includes maintenance, as identified in N-LIST-08920-10001, ‘Trained Performance Areas’ [56]. The Performance Areas in this List identify Training and Qualification Descriptions (TQDs), the organization responsible for the performance area training, and those TQDs which are part of the Major Trained Performance Areas. For example, TQDs to qualify Mechanical Maintenance, Electrical and Control Maintenance, Civil Maintenance and non-BTU Contractor personnel to work independently are defined in N-TQD-301-00001, Rev 018A ‘Nuclear Maintenance Training And Qualification Description’ [57]. OPG audits and self-assessments (programmatically driven) provide mechanisms for assurance that OPG remains in compliance with its training and development requirements.

The minimum maintenance group complement for PNGS is identified in OPG Instruction P-INS-09100-00003 [58]. A review of the PNGS SCR databases for the period 2011-2015 did not identify any SCRs where the minimum maintenance group complement was not met.

Also, N-PROG-AS-0005, ‘Nuclear Business Planning Program’ [86] is in place to ensure OPG’s performance, including in one of its four cornerstones on Equipment Reliability, meets objectives and is aligned with best industry performance. The objective of the business planning process is to ensure all station resources, material, etc., including maintenance resources, are in place to achieve the station’s reliability objectives.

4.1.2.3 *Maintenance Facilities and Resources Assessment Summary*

No gaps with respect to maintenance facilities and resources against the guidance from IAEA NS-G-2.6 [40] were identified that could challenge continued operations for PNGS to 2028. Therefore Review Task #2 is assessed as compliant for PSR2.

4.1.3 Review Task #3 - Design Basis Assumptions

This Review Task is to assess the SSCs against the current design basis to confirm that design basis assumptions have not been significantly challenged and will remain that way through the period of PSR2. This Review Task is an integral part in the completion of Review Task #1 (see Section 4.1.1).

In the context of this SF2 Report Design Basis Assumptions are the set of requirements identified in the Safety Report, OSR documents and other relevant Design Basis Documentation. These requirements are used to assess the SSCs in the condition assessments and therefore the CA process demonstrates Design Basis Assumptions are met or required actions are recommended to maintain this.

Technical Operability Evaluations (TOEs) are prepared to document cases when the operability of a safety related component is in question. TOEs are addressed in the CA process as one of the inputs where they exist, since they are documented in the Health Reports and Station Condition Record database.

The CAs have identified where recommendations are required to improve or maintain condition, or to improve aging management practices. However, none of these recommendations are required to ensure Design Basis Assumptions are currently met. They all represent improvement actions. Therefore, Review Task #3 is assessed as compliant for PSR2.

4.1.4 Review Task #4 - Spent Fuel Storage Facilities

This Review Task is to review the condition and operation of spent fuel storage facilities and their effect on the spent fuel storage strategy for Pickering NGS.

Facility and System Descriptions in the following subsections have been extracted from relevant Safety Reports [23] [24] and Design Manuals [60], [61].

4.1.4.1 Storage Facilities Description and Operation

Pickering reactors are fuelled on-power. Each reactor is serviced by two remotely controlled fuelling machines, one at each reactor face, which, operating at opposite ends of the same fuel channel, match Primary Heat Transfer system pressure, open the channel and insert fresh fuel or remove irradiated fuel, without interrupting reactor operation.

The Pickering Nuclear Generating Station (PNGS) contains three Irradiated Fuel Bays (IFBs): the IFB-A, IFB-B, and the Auxiliary Irradiated Fuel Bay (AIFB). The IFB-A and AIFB support P014 Units and the IFB-B supports P058 Unit’s requirements. Each bay has a cooling and purification system to remove used fuel decay heat, contain radioactive by-products, and shield the public and personnel from harmful radioactive by-products. Fuel is stored and cooled for a minimum of ten (10) years in the IFBs prior to being moved to a Dry Storage Container (DSC) and sent to the Pickering Waste Management Facility (PWMF) for storage.

The irradiated fuel storage bays are located in the Reactor Auxiliary Bay (RAB) at the ground floor level between reactor buildings 2 and 3 for P014 Units, and between reactor buildings 6 and 7 for P058 Units. The AIFB is located southwest of Unit 4 RAB and connected via a corridor to the IFB-A.

In Units 1 and 4 irradiated fuel is transferred by an elevator and conveyor to the IFB-A where it remains for a minimum of four years before transfer to the AIFB. The discharge of fuel into the transfer system, the transfer of fuel to the receiving bay, and its placement into storage baskets may be done remotely. The removal of filled baskets to the storage area and their placement into stacking frames is done manually with powered tools operated from the bay gantry.

After a minimum four year cooling period, the fuel baskets can be removed from the stacking frames and loaded under water into the on-site flask for transfer to the AIFB. The on-site flask is sized to hold eight baskets of fuel (a total of 256 fuel bundles) and provides a total containment barrier for the fuel during the transfer. In the AIFB, fuel is transferred to modules that are stored in seismically qualified stacking frames in the storage area. The empty baskets are returned, in the on-site flask, to the primary storage bay. The modules hold 96 fuel bundles and provide a higher storage density than baskets. They have also been qualified as transportation devices for possible future shipments to off-site long-term storage depots.

In P5-8, two receiving bays serve a single storage bay. Baskets are used in the receiving bays to accept fuel, as at P1,4, but the fuel is then transferred to modules holding 48 pairs of bundles for stacking in the main storage bay. Similar to P1,4 modules provide better storage density than baskets, and are qualified transportation devices for shipment to long-term storage depots. Space is also allocated in the storage bay for loading irradiated fuel into shipping flasks.

The discharge of fuel into the transfer system, the transfer of fuel to the receiving bay, and the placement of fuel into storage baskets may be done remotely and automatically. The removal of fuel baskets is done manually with powered tools.

The irradiated fuel transfer and storage facilities are sized for the handling and accumulation of irradiated fuel at the average fuelling rate of the reactors.

4.1.4.2 COOLING AND PURIFICATION DESCRIPTION AND OPERATION

The system consists of a cooling circuit and a purification circuit, which return water to a common header at the north end of the storage bay.

Three parallel pumps draw water from dual intakes at one end of the bay and feed it through three parallel heat exchangers. Permanent skimmers are the intakes for the purification system and maintain the water surfaces free of foreign matter. Since emergency shutdown of the purification system is acceptable, a single pump is used ahead of the ion exchange columns and filters. There is also a separate, portable underwater vacuum cleaner. The two receiving bays, the main storage bay, and its south extension may each be emptied for inspection or repair.

The irradiated fuel bay cooling and purification system is designed (Section 10.1.4.1 of Pickering A Safety Report [23], Section 10.2.4.4 of Pickering B Safety Report [24], and Design Manual [60]) to meet the following:

1. Maintain the temperature of the storage bay water between 28ºC - 32°C (82ºF - 90°F); based on an accumulation of irradiated fuel over 5.5 years operation at a 90% capacity factor.
2. Maintain optical clarity of the water to permit observation of irradiated fuel handling operations.
3. Remove suspended and dissolved radionuclides to allow access to the work areas.
4. Ensure a minimum water level is maintained in the bays for shielding during all phases of fuel handling and storage.
5. Provide a cooling water supply to the irradiated fuel elevators and conveyors.

For the IFB-A, the system piping, valves and equipment which contact the storage bay water are made of stainless steel or other corrosion resistant material. All components conform to Section VIII of the ASME Boiler and Pressure Vessel Code, 1964 edition.

For IFB-B, the piping, valves, and equipment of this system are made of stainless steel or other corrosion resistant material. All components conform to Section III Subsection ND (Class 3) of the ASME Boiler and Pressure Vessel Code.

The AIFB cooling and purification system is designed to meet the following (per the Design Manual [61]:

1. Maintain the temperature of the storage bay water at less than 29 ºC (84ºF); from a full bay load of irradiated fuel that has been previously cooled at least 4 years.
2. Maintain optical clarity of the water to permit observation of irradiated fuel handling operations.
3. Remove suspended and dissolved radionuclides to allow access to the work areas.
4. Ensure a minimum water level is maintained in the bays for shielding during all phases of fuel handling and storage.

The AIFB cooling and purification system is designed to ASME Section III, Subsection ND (currently classified as Class 6), except for piping and fittings between CV1 and the Irradiated Fuel Bay Structure designed to ANSI B31.1 and the shell side of the heat exchangers designed to ASME Section VIII, Division 1.

4.1.4.3 FACILITIES CONDITION

The bays structures (IFB and AIFB), cooling systems and purification are the SSCs that contribute to Fuel Storage functionality. The Irradiated Fuel Systems provide the following critical functions:

1. Remove the heat transferred to the bay water by irradiated fuel bundles.
2. Reduce the radioactivity level and turbidity of the bay water by removing suspended and dissolved solids; maintain clarity of the bay water to permit observation of irradiated fuel handling operations.
3. Provide shielding of the irradiated fuel to reduce radiation fields around the bays to acceptable levels.
4. Ensure that contaminants do not escape to the environment.
5. Supply water to the Elevator Sprays.

The overall health rating of the Irradiated Fuel Bay - A, Irradiated Fuel Bay – B and Auxiliary Irradiated Fuel Bay and Interbay Transfer System as of Q4 2015 is WHITE. One of the major reasons for the downgrade from GREEN, and common across all bays and systems, is the Corrective Maintenance backlog (discussed below) related to cooling and purification systems, and addressing the spares for pumping systems in IFB-B and AIFB.

There is one recommendation in the System Summaries for IFB and Auxiliary systems operation until 2024 (refer to Section 4.1.1.2.27). It is to correct the problem with IFB-B leakage and is discussed below. In addition, updated DCAs will be completed for IFB and Auxiliary systems as part of the current DCA update. Completion of these DCAs is addressed by gap SF2-15 identified earlier covering all of the DCA updates.

A CNSC Type II inspection (P-CORR-00531-04477) [62] of the IFBs resulted in two action notices. One action (PRPD-2015-010-AN1) requires OPG to develop and implement a corrective action plan to correct equipment deficiencies in the IFB-A, IFB-B and AIFB. This action is a result of multiple Work Requests (WRs) located in the field that have not been addressed for a prolonged period of time. The second CNSC action (PRPD-2015-010-AN2) was related to lighting deficiencies in IFB-A which have since been addressed.

The issue related to maintenance backlogs (CNSC Action - PRPD-2015-010-AN1) is being tracked by OPG. OPG has already submitted a detailed action plan for addressing this issue (P-CORR-00531-04624 [64]. This work is being managed through the station work program and as per the latest available update [97], one third of the work has been completed. A review of the remaining work has shown that it primarily consists of maintenance on non-critical components (outside the scope of SF2) and other work that is relatively minor in nature. This is consistent with none of the work being addressed within the Condition Assessments for these systems. Given these factors, existing station work management processes and Management Actions updating progress with the CNSC, will effectively track this work to completion and a PSR2 gap addressing the issue is not required.

Another issue with the fuel storage facilities is the chronic leakage from IFB-B to the collection sumps that has been increasing since 2007. This issue was raised in CNSC Action – 2014-4-5386. Equipment improvements and revised procedures are in place to maintain the water levels in the collection sumps below groundwater level so that any leakage is inward and not outward. Monitoring for tritium in groundwater at the IFB is conducted regularly to confirm the adequacy of the current fixes. OPG has taken initiatives to make repair of the IFB a high priority and has committed to the CNSC (P-CORR-00531-04624) [63] of its intention to mitigate leaks from the P058 IFB, and its collection sumps, to minimize the leak rate and to reduce the potential for environmental risk. Since these action plans are not complete, this results in a PSR2 gap **(Pickering PSR2 SF2-16)**.

In addition, per SCRs P-2013-05015, P-2015-11143 and Action Request AR 28182003, the seismic capacity of the current spent fuel basket stacking arrangements in the Pickering IFBs has not been adequately documented, resulting in a PSR2 gap **(Pickering PSR2 SF2-17)**.

These same SCRs and Action Request identify the seismic capacity of the Pickering 058 IFB fuel conveyor has not been adequately documented, resulting in a PSR2 gap **(Pickering PSR2 SF2-18)**.

4.1.4.4 Spent Fuel Storage Strategy

Assessments (P-REP-34400-00001) [65] have been performed and have confirmed that both structural loading and heat removal capacity are sufficient to accommodate the full design storage capacity. Therefore, heat removal and structural loading are not limiting factors for the long term storage facilities.

The irradiated fuel bays are designed to have a storage capacity for all the irradiated fuel accumulating over approximately 12 station-years. Since the fuel is stored and cooled for about ten (10) years in the bays prior to being moved to a DSC and sent to PWMF, there is sufficient bay space available provided movement to dry storage is performed in a timely manner.

However, recent field walkdowns have identified unusable space in each of the bays. Unusable bay space is defined as basket/module spaces in each bay that are inaccessible, damaged, filled with non-fuel material, filled with scrap fuel and/or non-fuel matter, and any space that cannot be occupied by used intact irradiated fuel. According to an assessment performed following the walkdowns (P-REP-34400-00002) [66], the number of bundles that cannot be optimally stored represent the amount of fuel stored in approximately one reactor in each of IFB-B and AIFB, and approximately three reactors for the IFB-A.

As per the Bay Storage Assessment at End of Life (P-REP-34400-00003) [67], given the unavailable space in the bays, and DSC and ITB transfer rates, there are challenges to meeting the Bay Storage requirements for EOL core defueling. However, this is a production issue and can be effectively managed as part of the station work programs.

4.1.4.5 SPENT FUEL HANDLING

Pickering Auxiliary Irradiated Fuel Bay (AIFB) and Irradiated Fuel Bay Pickering B (IFB-B) each employ a module handling mechanism (Supertool) attached to existing gantry to securely latch, lift, transport and deposit Irradiated Fuel baskets/modules.

The existing AIFB module handling device has reached its end of life and the IFB-B module requires a major overhaul to fix issues encountered on a regular basis. It is more cost effective to replace the entire mechanism rather than conducting a major replacement of its existing potentially obsolete parts/components.

The modification per Master EC 122773 will provide two new module handling devices (Supertool) which will ensure ergonomic concerns raised are eliminated for AIFB & IFB-B. The replacements for the AIFB and IFB-B are designed to be ergonomically superior with a longer operational life (300,000 fuel cycles) to support defueling and regular operation of the units. These new modules are planned to be installed by 2018 or earlier. Given this is being tracked by an existing modification EC, this will be tracked by system health reporting and therefore is not a PSR2 gap.

Spent fuel older than 4 years is transferred from IFB-A to AIFB (as described above). The lifting equipment in IFB-A to support this activity was replaced in 2011 and there are no issues with this transfer operation. The fuel transport is performed by a purpose-built transport truck called the ‘King-Kong”. There are age related maintenance issues associated with this equipment. The equipment meets its current functional requirements, but outstanding maintenance issues will be addressed via a planned refurbishment program under EC 126141. Since the equipment continues to meet its functional requirements, completion of this work is not considered a PSR2 gap.

4.1.4.6 SPENT FUEL STORAGE FACILITIES ASSESSMENT SUMMARY

A review of the condition and operation of spent fuel storage facilities and their effect on the spent fuel storage strategy has been completed. The condition of the storage facilities is generally good. One recommendation was identified in the Detailed CAs on IFB-B leakage and there is a committed action plan addressing the issue. Updates to the Detailed CAs for spent fuel storage facilities for operation to 2028 is addressed by gap SF2-15. The assessment has confirmed that all of the IFBs are designed to accommodate the de-fueling rates from all of the reactors with margin.

The following gaps were identified from this Review:

- **Gap SF2-16:** Action plans to correct the leakage in IFB-B are not complete.
- **Gap SF2-17:** The seismic capacity of the current spent fuel basket stacking arrangements in the Pickering IFBs needs to be documented.
- **Gap SF2-18**: The seismic capacity of the Pickering 058 IFB conveyer tunnel needs to be documented.

4.1.5 *Review Task #5 - Dependence on Obsolescent Equipment*

During the preparation of condition assessments a review is performed to address the availability of any parts for critical components required for replacement or periodic maintenance. Having the necessary supply of parts is essential to maintain the condition of critical components. Unavailability of parts may be due to them becoming obsolete. Equipment obsolescence is an issue that becomes more apparent during the latter phases of nuclear power plant operation. This section describes the OPG processes in place to address obsolescence and the results from the condition assessments pertaining to obsolescence.

As identified in the system reviews, Pickering relies on some equipment that is no longer available for purchase from original equipment manufacturers to the existing technical and quality requirements. To improve upon the processes in place to deal with these situations, in 2015 OPG set up a section within the Design Division and issued N-STD-MA-0024, "Obsolescence Management" [91] to, "*define and implement a sustaining program to manage the proactive and reactive obsolescence issues associated with critical equipment and components. The program activities interface with equipment reliability and life-cycle management strategies designed to sustain continued safe and reliable plant operation*."

The Nuclear Utility Obsolescence Group[9] was used to ensure the standard aligns with industry best practice. It includes the following key elements:

1. Identification of obsolescence.
2. Prioritization of obsolescence issues.
3. Determining a solution.

[9] The Nuclear Utility Obsolescence Group is a North American organization formed to support the nuclear industry to identify obsolete items and potential replacements to reduce station risk and vulnerabilities associated with Equipment Reliability.

4. Implementation of obsolescence solutions according to prioritization.

The standard documents how OPG uses industry tools to proactively assess the material management system to see what catalogue items are no longer available. OPG staff prioritize obsolescence issues to support critical spares, emergent on-line work and outage schedule demands. Standard industry software is also used to gain access to previously solved obsolescence issues. The OPG staff may use these solutions to determine the resolution path to be taken and develop the appropriate change paper to address obsolescence proactively. N-STD-MA-0024 [91] provides all the details of how this process is performed and overseen.

Obsolescence issues associated with safety related SSCs at Pickering NGS may also be identified through N-PROG-MP-0008, “Integrated Aging Management” [34] and N-PROG-MA-0026, “Equipment Reliability” [80] programs. The Integrated Aging Management program requires obsolescence of a component to be identified in condition assessments prior to it having a significant station impact. In particular, N-PROC-MP-0060, “Aging Management Process” [7] requires obsolescence to be considered as an aging related degradation mechanism, when conducting a condition assessment of an SSC. In determining whether obsolescence is an issue of concern, consideration is given to the quantity of spares available and their predicted usage. Once an equipment obsolescence issue has been identified, actions to resolve the issue are identified in the condition assessments and tracked in system health report action plans.

In addition, the Equipment Reliability program requires critical spares be identified for safety related SSCs and the status of these spares monitored for any issues including obsolescence. N-INS-00680-10000 “Identification of Critical Spares” [88] outlines the process of identifying critical spares together with roles and accountabilities. System Engineers are responsible for identifying critical spares and summarizing their status in system health reports by identifying issues with the procurement of these parts. Once a component or part has been identified as critical, the responsible System Engineer is required to provide the information to Nuclear Supply Chain. Nuclear Supply Chain flags the component CAT ID in Asset Suite as a critical spare and then develops an appropriate stocking strategy which takes into consideration any other instances where the specific CAT ID is used.

System Engineers periodically review obsolescence and inventory levels of critical spares as part of system health monitoring. Critical spares not in stock are identified in system action plans for prioritization. In addition, Nuclear Supply Chain also reports on the health of Critical Spares through Supply Chain Performance Reporting.

Through all of these processes, OPG manages obsolescence tied to safety related SSCs at Pickering NGS. The program is sustained in the five year business plan, and fully staffed, so no further improvement actions are required.

The System Summaries contain 25 recommendations to address equipment parts that are presently assessed as being obsolete. That represents 4% of the recommendations in the condition assessments. Some of these items are already recognized and are being addressed in System Health Report action plans (which address the most critical parts). The majority of the obsolete items are in components having a Satisfactory or better condition and none are in Special Safety Systems. Therefore, these obsolescence issues do not introduce a nuclear safety risk. Recommendations to address obsolescence will be tracked in System Health Reports along with the other CA recommendations.

Based upon the above, OPG has effective processes in place aligned with industry standards. Obsolete equipment is identified in condition assessments and are tracked in OPG processes and system health reports. Therefore, Review Task #5 is assessed as compliant for PSR2.

4.1.6 *Review Task #6 – Dependence on External Essential Services/Supply*

To complete this task on the dependence on external essential services and supplies, the following information sources were reviewed:

- System Health Reports
- Station Condition Records (SCRs)
- Applicable OPG Governance
- Potential Issues affecting the Global Nuclear Industry

System Health Reports:

A review of SHRs was conducted to identify any issues in this area. No relevant problems were observed identifying potential risks associated with the supply of external services or supplies. In addition, available Chemistry Reports and Self-Assessments were reviewed and again no vulnerabilities were identified.

Review of Station Condition Records:

A review of SCRs was also conducted and no significant vulnerabilities were identified within the scope of this Review Task.

Only one SCR was found using the search criteria “Possible Supplier Interruption” noted below in the governance review. Station Condition Record P-2014-03700 was issued to document that a supplier of calibration services was unwilling to be subject to an OPG audit. The issue was resolved by using another available supplier and therefore this issue does not introduce a vulnerability to the supply of external services or supplies.

Governance Reviews:

Obsolescence of services and supplies for any reason (business discontinuation/interruption, failure of a qualification audit, low volume interactions) is addressed by OPG-PROC-0058 “Procurement Activities” [84], Section 1.19.5, which states:

“Any potential disruptions to the supply of items and/or services that may adversely impact OPG shall be addressed in a consistent and expedient manner.
(a) For nuclear and/or OPG wide suppliers:
(1) When OPG becomes aware of a potential supplier interruption a Station Condition Record (SCR) titled “Possible Supplier Interruption: [vendor name]” shall be initiated.”

Specific actions to address a supplier interruption of any type is then addressed by N-PROC-MM-0010 “Establishing and Maintaining Ontario Power Generation Approved Suppliers List” [85]. Among other things, this governance mandates an Approved Supplier List Oversight Committee meeting (ASLOC), which reviews impacts on internal stakeholders and initiates corrective actions as may be required. This governance ensures the risk posed by potential discontinuation of any external supplies or services is effectively mitigated.

Potential Vulnerabilities Affecting the Nuclear Industry:

Various information sources (Industry and Regulatory web-sites and periodicals)[10] were reviewed to identify any vulnerabilities affecting the nuclear industry or other related industries. No significant issues were identified. The sole potential issue identified was the supply of Lithium. Lithium hydroxide is used for protection against corrosion. There has been a significant increase in the demand for lithium for energy storage technologies. However, this is not considered to be a risk given potential other materials that can be used for corrosion protection and there is sufficient supply beyond 2028.

In addition to the reviews performed in the above areas, there are a number of established engineering service providers serving OPG that are capable of providing continued specialized engineering tasks.

Lastly, many of the same external services and supplies required for Pickering are also required for the Darlington station. Darlington is being refurbished to operate far beyond the proposed Pickering life extension date of 2028. OPG’s processes will provide assurance of supply of external services and supplies for the extended life of Darlington and Pickering stations.

[10] NRC, Nuclear Energy Institute, Power Engineering and other web-sites were searched for shortages/vulnerabilities in the power industry.

Given the above, there are no existing vulnerabilities associated with this Review Task and OPG processes are in place to mitigate future risks. Therefore, Review Task #6 is assessed as compliant for PSR2.

4.2 Emergency Mitigation Equipment Review

REGDOC 2.10.1 'Nuclear Emergency Preparedness and Response' [68] includes requirements to identify essential emergency response equipment and how its operation and effectiveness are assured during an emergency. These regulations are generic to all classes of accidents. Subsequent to the failure of the Fukushima Daiichi Nuclear Power Plant to maintain critical safety functions following a BDBA, the CNSC established a "Task Force" to manage the CNSC Fukushima review. Based on recommendations from the "Task Force", Fukushima Action Items (FAI) were raised pertaining to Emergency Mitigating Equipment (EME), which is the term adopted by Canadian Nuclear Power Plants to define the equipment that provides additional lines of defense against BDBA. All FAIs assigned to OPG for EME are now closed (per final status report N-REP-03600-10003 [69]).

As part of the Fukushima response, OPG has established EME at its nuclear power stations to support station response to BDBA. As per OPG EME memorandum [70], Nuclear Safety has programmatic accountability for EME, the Emergency Response Team is responsible for EME maintenance and testing, and Station Performance Engineering provides engineering support for the equipment.

Although EME is not considered a Safety Related System, OPG has established EME procedures and practices commensurate with the role of the equipment in support of BDBA response.

The technical basis pertaining to EME is documented in N-BDB-03600-00002 'OPG Emergency Mitigating Equipment for Beyond Design Basis Accidents – Technical Basis Document' [71]. This document summarizes the analyses that have been done for each step of event progression, and modifications undertaken by OPG to improve the depth and capability to arrest BDBA progression.

The analysis documented in the Technical Basis document has established technical requirements that have led OPG to ensure that portable diesel powered pumps and generators, and required support equipment, are in service and available to be deployed following an event to limit accident progression and mitigate the impact of a BDBA.

A detailed listing of the EME is provided in N-BDB-03600-00001 'Emergency Mitigating Equipment Inventory' [72]. As documented in the OPG memorandum [70], EME from other Canadian utilities is also available for use under the establish Mutual Aid Agreement with details provided in N-CORR-0542905, 'Beyond Design Basis Event Response –

Emergency Mitigating Equipment (EME) – Equipment Available from other Canadian Nuclear Facilities' [73].

EME has also been established within the OPG governance framework as per the following documents.

N-STD-MP-0019 'Beyond Design Basis Accident Management' defines the reactor safety program governing Beyond Design Basis Accident Management [74]. The overall approach for management of BDBAs includes measures to prevent BDBAs from progressing to Severe Accidents (SAs) as well as measures to mitigate the consequences should a BDBA progress to a SA. This document includes both Emergency Mitigating Equipment Guidelines (EMEG) and Severe Accident Management Guidelines (SAMG) where the intent of EMEG is to prevent a BDBA sequence from progressing to a SA. Defined within the Beyond Design Basis Accident Management Standard [74] are BDB Functional Safety Requirements, BDB Challenge Evaluation Process, and Roles and Accountabilities. Also included in the implementation requirements of the BDBA Standard [74] is a requirement for EME testing to support sustainable operation, sufficient to demonstrate that the required functional performance for BDBAs is achieved.

As per the BDBA Accident Management Standard [74], BDB Functional Safety Requirements have been identified and documented for each station. NA44-GUID-03600-00001 'Pickering 1-4 Beyond Design Basis Functional Safety Requirements' [75], and NK30-GUID-03600-00001 [76] 'Pickering 5-8 Beyond Design Basis Function Safety Requirements' include BDBA credited equipment requirements (similar to safety limits) and degradation conditions (similar to impairments). These documents summarize functional requirements for SSCs credited to manage and/or mitigate BDBAs, or to prevent progression to SAs and include specific functional safety requirements for EME.

Testing and maintenance of EME is addressed in N-INS-03600-10002 'Beyond Design Basis Emergency Mitigating Equipment Testing and Maintenance Process' [77]. This document establishes instructions for testing and maintenance of EME and associated connection points to station systems credited to support BDBA mitigation, which is aligned with INPO governance AP-913 [10]. This instruction document ensures fleet wide consistency and rigour of the process of testing, maintenance, monitoring, and reporting in support of BDBA management. This document also identifies all portable EME as Equipment Important to Emergency Response (EITER) as per N-PROC-RA-0133 [78]. Management of EITER includes maintenance, testing, and response requirements to unavailability of equipment.

As per the EME memorandum [70] and OPG Instruction N-INS-03600-10002 [77], as part of the testing and maintenance pre-defined Preventative maintenance and equipment checks are performed and major EME equipment availability is reported in the daily Pickering Alignment Meeting and in the Chief Nuclear Officer report. Also as per the

Memorandum [70], Emergency Preparedness drills and exercises are conducted using the equipment to confirm integrated response capability.

A review of testing and maintenance practices is documented in N-CORR-01130-0495435, 'Review of Testing and Maintenance Practices for Emergency Mitigation Equipment at Darlington and Pickering' [79] indicating that the overall performance has been good, EME is being well maintained, there has been few equipment problems, and that there is confidence that any issues are being identified and corrected in a timely manner.

According to the EME memorandum [70], OPG is also planning to add additional EME to further enhance response capability, and as part of the OPG change control processes the new equipment will have appropriate maintenance and testing requirements and procedures in place.

As per the discussion above, the Fukushima action items associated with EME have been completed (per Status Report N-REP-03600-10003 [69]) and have led to having EME management incorporated into station governance, the establishment of EME requirements, and the implementation of EME including ongoing maintenance and testing.

As no issues are identified in the review of EME at OPG, EME is assessed as compliant for PSR2.

4.3 Additional Reviews

As discussed in Section 3.7, the SF2 assessment also included a review of commitments previously made to the CNSC, open CNSC action items, and exemptions granted by the CNSC, as identified in the R04 Pickering Licence Conditions Handbook LCH [25], to determine if there are any impacts associated with operation of the Pickering Units past 2020. The only item in the LCH related to SF2 is contained in section 16.3 of the LCH [25], which documents a hold point for not exceeding 247,000 EFPHs on any unit without prior CNSC approval. This has already been identified as PSR2 gap. This assessment did not find any additional PSR2 gaps for Safety Factor 2.

OPG has also conducted a review of the Pickering 5-8 Continued Operations Plan in support of PSR2 [88]. Fitness for service aspects from that review (e.g., items related to life cycle management plans and condition assessments) parallel the assessment conducted in this Safety Factor 2 Report. Implications were identified if closed COP actions needed to be re-assessed given the potential to operate Pickering to 2028 rather than 2020, which had been factored into the closure criteria for some COP actions. PSR2 implications were also identified if they were applicable to Pickering Units 1,4. Where there were implications for extended operation, a PSR2 gap has been identified in the COP report. These gaps will be considered in the Global Assessment Report.

In addition, Fukushima Action Items (FAIs) were reviewed to identify any implications of extending operation beyond 2020. This review is presented in a separate PSR2 FAI Review Report [81]. There are no SF2 gaps associated with this review.

4.4 SF2 Gaps that Impact Other Safety Factors

Safety Factor Report 4, "Pickering NGS PSR2, Safety Factor 4 Report – Aging" [4] reviews the aging of spent fuel facilities, as does Section 4.1.4 in this report. This reference contains a gap to complete the condition assessments for these facilities. This gap is addressed by gap SF2-15 in this report. In addition, some of the gaps raised in this report on spent fuel storage facilities, i.e. SF2-16 and SF2-17 are discussed in Reference [4].

No other SF2 gaps have been identified that are related to other PSR2 Safety Factors. A number of gaps in SF2 relate to work ongoing related to aging management (updating LCMPs, fitness for service of components, completion of condition assessments, etc.). These do not directly impact other Safety Factors, but the completion of this work supports the continued safe operation of the plant and to maintain operation within the credited design basis.

5.0 RESULTS AND CONCLUSIONS

5.1 RESULTS

5.1.1 MAJOR COMPONENTS

Major components (Fuel Channels, Feeders, Steam Generators, and Reactor Components and Structures) are managed through a comprehensive program of in-service inspections, maintenance, engineering assessment and confirmatory research and development documented in life cycle management plans. There is high confidence that the major components will remain fit for service up to an extended station life to 2024, with limited potential mitigating actions required. Additional analysis and assessment is required to demonstrate continued Fitness for Service for extended station life to 2028. The review identified the following gaps:

- **Gap SF2-1:** Fitness for Service for Fuel Channels has not been demonstrated for station operation to 2028.
- **Gap SF2-2**: OPG does not have approval to operate beyond the current Licence limit of 247,000 Effective Full Power Hours (EFPH) for fuel channels.
- **Gap SF2-3**: The Fuel Channels LCMP has not been formally updated to address extended station operation to 2028.
- **Gap SF2-4**: Fitness for Service for Feeders has not been demonstrated for station operation to 2028.
- **Gap SF2-5**: The Feeders LCMP has not been formally updated to address extended station operation to 2028.
- **Gap SF2-6:** Fitness for Service for Steam Generators has not been demonstrated for station operation to 2028.
- **Gap SF2-7**: The Steam Generators LCMP has not been formally updated to address extended station operation to 2028.
- **Gap SF2-8:** Fitness for Service for Reactor Components and Structures has not been demonstrated for station operation to 2028.
- **Gap SF2-9:** The Reactor Components and Structures LCMP has not been formally updated to address extended station operation to 2028.
- **Gap SF2-10:** Environmental Factors have not been incorporated into the Service Limits Assessment for Class 1 piping.

5.1.2 CA Summary Tables

Detailed condition assessments have been performed for 1202 Commodity Groups (CGs), including all SSCs in Systems Important to Safety and Safe Operating Envelope systems. No states of degradation were observed that compromise the current design basis or design basis assumptions. 58% of the CGs are classified as “Good” or “Very Good”. 35% are assigned a “Satisfactory” classification. Only 6% are assigned a classification of “Poor” and 0.4% are “Very Poor.

For the Special Safety Systems for both P1,4 and P5-8, there are no CGs credited in the Safety Factor 2 Report which are rated Poor or Very Poor. For the Units 1,4 and Units 5-8 Special Safety Systems, there are only two CGs rated “Poor” that have a primary safety function. However, as described above these have minimal safety impact.

Approximately 500 recommendations were identified within the CAs, largely focused on incremental actions to improve condition and/or aging management practices. Approximately thirty percent of the recommendations are current station initiatives (i.e. already identified and planned). The majority of the recommendations are in place for CGs rated Satisfactory or lower. The following gaps were identified during completion of the condition assessments:

- **Gap SF2-11:** Condition Assessments for civil structures are not complete for station operation to 2028.
- **Gap SF2-12:** Condition Assessments for in-scope piping systems are not complete for station operation to 2028.
- **Gap SF2-13:** The Cable Surveillance Program risk assessment and condition assessments currently use out of date criticality coding.
- **Gap SF2-14:** The Buried Piping Program risk assessment and condition assessments have not been updated for extended operation to 2028.

Work is currently in progress to complete updated CAs to address extension of station life to 2028. This outstanding condition assessment work introduces the following PSR2 gap:

- **Gap SF2-15**: Updated Detailed Condition Assessments are not complete for Commodity Groups in the scope of PSR2 for station operation to 2028.

5.1.3 Maintenance Facilities and Resources

The review of Maintenance Facilities and Resources did not identify any issues when reviewed against the relevant sections of IAEA Safety Guide NG-G-2.6 ‘Maintenance, Surveillance and In-Service Inspection in Nuclear Power Plants’ [40]. This review included assessment of SCRs and the findings of industry audits, and did not identify any issues of significance. Also, maintenance resources are assigned via station business planning to ensure that equipment reliability goals are achieved.

5.1.4 CONDITION AND OPERATION OF SPENT FUEL STORAGE FACILITIES

An assessment of the condition and operation of the Irradiated Fuel Bays (IFB-A, IFB-B), and the Auxiliary Irradiated Fuel Bay (AIFB), including their cooling and purification systems, was completed. Overall, the condition of the facilities is good. Three gaps have been identified as outlined below:

- **Gap SF2-16:** Action plans to correct the leakage in IFB-B are not complete.
- **Gap SF2-17:** The seismic capacity of the current spent fuel basket stacking arrangements in the Pickering IFBs needs to be documented.
- **Gap SF2-18**: The seismic capacity of the Pickering 058 IFB conveyer tunnel needs to be documented.

5.1.5 EMERGENCY MITIGATING EQUIPMENT

The EME program within OPG has been reviewed including its governance and programs, technical basis and maintenance practices. No issues were identified in the review of the EME program at Pickering.

5.2 OVERALL ASSESSMENT OF THE SAFETY FACTOR

An assessment of the six SF2 Review Tasks has been completed and the objectives of each have been satisfied as documented in this report.

As part of this review, a comprehensive assessment of the Systems, Structures and Components (SSCs) Important to Safety at Pickering NGS has found they are generally in good condition. This conclusion is supported by the presence of comprehensive and effective programs in place to ensure the condition of components meet design requirements with margin.

This assessment has not identified any major concerns that the SSCs will continue to operate as per the design basis requirements during life extension. As part of SF2, eighteen PSR2 gaps have been identified that will need to be addressed further as part of the PSR2 Global Assessment Report as detailed in the PSR2 Basis Document [1].

6.0 REFERENCES

[1] P-REP-03680-00001 R002, OPG Report ‘Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document’, July 2016.
[2] REGDOC-2.3.3, CNSC Regulatory Document ‘Periodic Safety Reviews’, April 2015.
[3] Safety Guide No. SSG-25, IAEA Safety Guide ‘Periodic Safety Review for Nuclear Power Plants’, 2013.
[4] P-REP-03680-00007 R00, OPG Report ‘Pickering NGS Periodic Safety Review 2 (PSR2) Safety Factor 4 Report: Aging’, July 2016.
[5] P-REP-03680-00001 R002, OPG Report ‘Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document’, June 2016.
[6] P-CORR-01060-0595301, OPG Memorandum ‘Pickering NGS: Changes in Criticality Coding’, June 2016.
[7] N-PROC-MP-0060 R005B, OPG Procedure ‘Aging Management Process’, September 2015.
[8] P-GUID-01060-10000 R001, OPG Guide ‘Condition Assessment Preparation Guide’, February 2016.
[9] N-PROC-MA-0077 R007, OPG Procedure ‘Critical Equipment Identification and Categorization’, December 2016.
[10] INPO AP-913 Revision 4, INPO Process Description ‘Equipment Reliability Process Description’, October 2013.
[11] NK30-REP-03680-00002 R000, OPG Report ‘Pickering NGS-B Integrated Safety Review – Actual Condition of Systems, Structures, and Components Safety Factor Report’, May 2008.
[12] NK38-REP-03680-10078 R001, OPG Report ‘Darlington NGS Integrated Safety Review – Aging and Actual Condition of Systems, Structures And Components Safety Factor Report’, October 2011.
[13] REGDOC RD-360, CNSC Regulatory Document ‘Life Extension of Nuclear Power Plants’, February 2008.
[14] Safety Guide No. NS-G-2.10, IAEA Safety Guide ‘Periodic Safety Review of Nuclear Power Plants’, 2003 (Note: superseded by SSG-25 [3]).
[15] P-REP-03680-00003 R000, OPG Report ‘Pickering NGS Periodic Safety Review 2 (PSR2): Definition of Safety Factor Review Tasks’, dated May 2016.
[16] P-LIST-06937-00001 R000, OPG List ‘Pickering A and B List of Safety Related Systems’, April 2011.
[17] NA44-REP-03611-00004 R001, OPG Report ‘Pickering A Systems Important to Safety’, January 2008.
[18] NK30-REP-03611-00024 R000, OPG Report ‘Pickering B Systems Important to Safety’, September 2014.
[19] N-INS-03602-10001 R001, OPG Instruction ‘Preparation of Safe Operating Envelope Compliance Tables’, February 2015.
[20] N-STD-RA-0033 R003 OPG Standard ‘Reliability Monitoring and Reporting of Systems Important to Safety’, August 2016.
[21] RD/GD-98, CNSC Regulatory Document ‘Reliability Programs for Nuclear Power Plants’, June 2012

[22] N-STD-MP-0016 R002, OPG Standard ‘Safe Operating Envelope’, June 2012.
[23] NA44-SR-01320-00001 R015, OPG Report ‘Pickering A Safety Report’, July 2012.
[24] NK30-SR-01320-00002 R004, OPG Report ‘Pickering B Safety Report Part 2’, October 2012.
[25] LCH-PNGS R005, Licence Conditions Handbook, November 2016.
[26] N-PROG-MA-0025 R002, OPG Program ‘Major Components’, March 2015.
[27] P-CORR-01060-0632223, OPG Correspondence ‘Fitness for Service of Major Components’, February 2017.
[28] P-CORR-33000-00001, OPG Correspondence ‘Service Limits Assessment of Nuclear Class 1 Components for Potential of Life Extension’, June 2016.
[29]N-PROC-MA-0100 R002, OPG Procedure ‘Major Components Life Cycle Management Plan’, March 2013.
[30] N-PLAN-01060-10002 R017, OPG Plan ‘Fuel Channel Life Cycle Management Plan’, October 2016.
[31] N-PLAN-01060-10001 R018, OPG Plan ‘Feeders Life Cycle Management Plan’, October 2016.
[32] N-PLAN-33110-10009 R007, OPG Plan ‘Steam Generators Life Cycle Management Plan’, October 2016.
[33] N-PLAN-01060-10003 R014, OPG Plan ‘Reactor Components & Structures Life Cycle Management Plan’, October 2016.
[34] N-PROG-MP-0008 Rev 006B, OPG Program, ‘Integrated Aging Management’, April 2016.
[35] CSA N285.4, CSA Standard ‘Periodic Inspection of CANDU Nuclear Power Plant Components’, 2009.
[36] NK30-REP-33110-00046 R001, OPG Report ‘Pickering B Preliminary Service Limits Assessment Of Class 1 Components For Potential Plant Life Extension’, August 2012.
[37] P-CORR-66400-0632085, OPG Memorandum ‘Pickering DCC System input to SF2’, January 2017.
[38] N-PROC-MA-0099 R001, OPG Procedure ‘Cable Surveillance’, August 2014.
[39] N-PLAN-04916-10002 R002, OPG Plan ‘Buried Piping Program Asset Management Plan’, June 2015.
[40] Safety Guide No. NS-G-2.6, IAEA Safety Standard ‘Maintenance, Surveillance and In-Service Inspection in Nuclear Power Plants’, 2002.
[41] S-210, CNSC Regulatory Document ‘Maintenance Programs for Nuclear Power Plants’, July 2007.
[42] NK30-REP-03680-00002 R000, OPG Report ‘Pickering NGS-B Integrated Safety Review - Actual Condition of Systems, Structures and Components Safety Factor Report’, May 2008.
[43] NK30-CORR-00531-05295, OPG Letter from T.E. Schaubel to P. Pasquet, ‘Pickering Nuclear Generating Station B (NGS-B) – Corrective and Elective Maintenance Backlogs - Request for Closure of Action Item 2003-8-01’, August 2009.
[44] NK30-CORR-00531-06252, OPG Correspondence ‘Pickering NGS-B – 2011 Fourth Quarter Update of Corrective Maintenance and Deficient Maintenance Backlogs, CNSC Action Item 2003-8-01 (RIB # 2505),’ March 2012.
[45] N-PROC-MA-0008 R022, OPG Procedure ‘Work Initiation, Approval and Prioritization’, December 2016.

[46] N-PROC-MA-0022 R022, OPG Procedure 'Integrated On-Line Work Schedule', February 2016.
[47] INPO-05-005, INPO Guideline 'Guidelines for Performance Improvement at Nuclear Power Stations', August 2005.
[48] N-INS-01966.1-10000 R006, OPG Instruction 'Trending Analysis Instruction and Performance Improvement Reporting', April 2014.
[49] P-REP-01966-0575908, OPG Report 'Pickering Maintenance Department Performance Improvement Report for Q4-2015', February 2016.
[50] P-CORR-00531-04625 R000, OPG Correspondence 'Pickering Quarterly Report on Safety Performance Indicators - Third Quarter 2015', December 2015.
[51] P-CORR-00531-03860 R000, OPG Correspondence 'Notice of Participation Pursuant to Rule 18 of CNSC Rules of Procedure – Pickering NGS Licence Renewal Application Hearing – February 20, 2013', January 2013.
[52] P-CORR-00531-04622 R000, OPG Correspondence 'Pickering NGS: Response to CNSC Staff's Request for Trending Data for the 2015 NPP report (for FLR and ISAR)', December 2015.
[53] P-CORR-00531-04490 R000, OPG Correspondence 'Pickering Quarterly Report on Safety Performance Indicators - First Quarter 2015', June 2015.
[54] P-CORR-00531-04553 R000, OPG Correspondence 'Pickering Quarterly Report on Safety Performance Indicators - Second Quarter 2015', September 2015.
[55] N-PROG-TR-0005 R016, OPG Program 'Training', January 2016.
[56] N-LIST-08920-10001 R008, OPG List 'Trained Performance Areas', March 2016.
[57] N-TQD-301-00001 R020, OPG Training Qualification Document 'Nuclear Maintenance Training and Qualification Description', January 2017.
[58] P-INS-09100-00003 R009, OPG Instruction 'Pickering Minimum Shift Complement', August 2015.
[59] P-REP-09320-0576780, OPG Report 'Pickering Outage Interval Extension to 30 Months: Preliminary Regulatory and Implementation Risk Assessment – R02', January 2016.
[60] NK30-DM-34400-00001 R001, OPG Design Manual 'Irradiated Fuel Storage Bay Cooling and Purification System', June 2013.
[61] NK30-DM-35710-00001 R001, OPG Design Manual 'Auxiliary Irradiated Fuel Bay Facility Cooling and Purification System', July 2015.
[62] P-CORR-00531-04477 R000, OPG Correspondence 'Pickering NGS: CNSC Type II Compliance Inspection Report: PRPD-2015-010 System Inspection Irradiated Fuel Bays, Action Item 2015-48-6500', May 2015.
[63] P-CORR-00531-04624 R000, OPG Correspondence 'Response to CNSC Action Item 2014-48-5386 Status Update: CNSC Review of 2013 Groundwater Monitoring Results Report – Pickering B IFB Leak Mitigation', February 2016.
[64] P-CORR-00531-04511 R000, OPG Correspondence 'Pickering NGS: CNSC Type II Compliance Inspection, Irradiated Fuel Bays, Report #PRPD-2015-010 – CNSC Action Item 2015-48-6500', July 2015.
[65] P-REP-34400-00001 R000, OPG Report 'Review of Fuel Bundle Storage Capability in IFB-A and IFB-B', October 2005.

[66] P-REP-34400-00002 R000, OPG Report ‘PNGS Irradiated Fuel Bay Space Unavailability Assessment’, February 2013.
[67] P-REP-34400-00003 R001, OPG Report ‘Pickering Bay Space Assessment for End of Life Defuelling’, December 2016.
[68] REGDOC-2.10.1 Version 2, CNSC Regulatory Document ‘Nuclear Emergency Preparedness and Response’, February 2016.
[69] N-REP-03600-10003 R007, OPG Report ‘Fukushima Action Item Status Report’, November 2015.
[70] P-CORR-76801-0586468, OPG Correspondence ‘Pickering NGS Extended Operations – Emergency Mitigating Equipment Support’, March 2016.
[71] N-BDB-03600-00002 R000, OPG Technical Basis Document ‘OPG Emergency Mitigating Equipment for Beyond Design Basis Accidents – Technical Basis Document’, October 2015.
[72] N-BDB-03600-00001 R000, OPG Document ‘Emergency Mitigating Equipment Inventory’, November 2015.
[73] N-CORR-09013-0542905, OPG Memorandum ‘Beyond Design Basis Event Response – Emergency Mitigating Equipment (EME) – Equipment Available from other Canadian Nuclear Facilities’, June 2015.
[74] N-STD-MP-0019 R002, OPG Standard ‘Beyond Design Basis Accident Management’, August 2016.
[75] NA44-GUID-03600-00001 R000, OPG Guideline ‘Pickering 1-4 Beyond Design Basis Functional Safety Requirements’, October 2014.
[76] NK30-GUID-03600-00001 R000, OPG Guideline ‘Pickering 5-8 Beyond Design Basis Functional Safety Requirements’, October 2014.
[77] N-INS-03600-10002 R000, OPG Instruction ‘Beyond Design Basis Emergency Mitigating Equipment Testing and Maintenance Process’, March 2015.
[78] N-PROC-RA-0133 R000, OPG Procedure ‘Management of Equipment Important to Emergency Response’, December 2014.
[79] N-CORR-01130-0495435, OPG Correspondence ‘Review of Testing and Maintenance Practices for Emergency Mitigating Equipment at Darlington and Pickering’, April 2014.
[80] N-PROG-MA-0026 R002, OPG Program ‘Equipment Reliability’, May 2015.
[81] P-REP-03680-00022 R00, OPG Report ‘Fukushima Action Item Review in Support of PNGS Periodic Safety Review 2 (PSR2)’, February 2017.
[82] N-PROC-MA-0065 R006, OPG Procedure ‘Administrative Requirements For the Periodic Inspection of Nuclear Power Plants’, October 2016.
[83] N-PROC-MA-0073 R008, OPG Procedure ‘Pressure Vessel Certification Management’, August 2015.
[84] OPG-PROC-0058 R010, OPG Procedure ‘Procurement Activities’, July 2016.
[85] N-PROC-MM-0010 R020, OPG Procedure ‘Establishing and Maintaining Ontario Power Generation Approved Suppliers List’ May 2016.
[86] N-PROG-AS-0005 R006, OPG Program ‘Nuclear Business Planning Program’, June 2016.
[87] N-PROG-MA-0017 R008, OPG Program ‘Components and Equipment Surveillance’, June 2015.
[88] P-REP-03680-00024 R000, OPG Report ‘Pickering 5-8 Continued Operations Plan Review in Support of PNGS Periodic Safety Review 2 (PSR2)’, January 2017.

[89] N-INS-00680-10000 R002, OPG Instruction ‘Identification of Critical Spares’, March 2016.
[90] N-GUID-04980-10002 R03, OPG Guide ‘Guideline for Critical Pipe Support Inspection and Results Processing’, August 2015.
[91] N-STD-MA-0024 R000, OPG Standard ‘Obsolescence Management’, August 2015.
[92]N-PROG-RA-0003 R010, OPG Program ‘Corrective Action’, January 2015.
[93]N-PROC-MA-0024 R016, OPG Procedure ‘System Performance Monitoring’, December 2016.
[94]N-PROG-MA-0004 R011, OPG Program ‘Conduct of Maintenance’, May 2015.
[95]OPG Report, NUGEN Report, Forced Loss Rate Data, Jan.1, 2016 to Dec. 31, 2016.
[96]OPG Report, P-REP-01060-00002 R000, Pickering Component Assessment Project System Transition Boundary Report, November 2016.
[97]P-CORR-00531-04713 R000, OPG Correspondence ‘Pickering NGS: CNSC Type II Compliance Inspection, Irradiated Fuel Bays, Report #PRPD-2015-010 – CNSC Action Item 2015-48-6500’, April 2016.
[98] REGDOC-3.1.1, CNSC Regulatory Document “Reporting Requirements for Nuclear Power Plants”, April 2016.
[99] IAEA Safety Standards, Ageing Management for Nuclear Power Plants, Safety Guide No. NS-G-2.12, 2009.
[100] N-PROC-MA-0015 R001, OPG Procedure ‘Tool Control’, July 2012.
[101] N-INS-01516-10006 R000, OPG Instruction ‘Tool Control System Instruction’, November 2004.

APPENDIX A. NOMENCLATURE

ACU	Air Conditioning Unit
AIFB	Auxiliary Irradiated Fuel Bay
AM	Aging Management
ASLOC	Approved Supplier List Oversight Committee
ASME	American Society of Mechanical Engineers
BDB	Beyond Design Basis
BDBA	Beyond Design Basis Accident
BDBE	Beyond Design Basis Event
BOM	Bill of Materials
CA	Condition Assessment
CAT ID	Catalogue Identifier
CG	Commodity Group
CID	Catalogue Identifier
CL	Class
CNSC	Canadian Nuclear Safety Commission
CO	Continued Operations
COP	Continued Operations Plan
CSA	Canadian Standards Association
CT	Calandria Tubes
DCC	Digital Control Computers
ECR	Engineering Change Request
EFPH	Effective Full Power Hours
EJ	Expansion Joint
EITER	Equipment Important to Emergency Response
EME	Emergency Mitigating Equipment
EMEG	Emergency Mitigating Equipment Guidelines
EOL	End of Life
EQ	Environmental Qualification
ER	Equipment Reliability

FAC	Flow Accelerated Corrosion
FADS	Filtered Air Discharge System
FC	Fuel Channels
FFS	Fitness for Service
FLR	Forced Loss Rate
FM	Fuelling Machine
GST	Generator Service Transformer
GUI	Graphical User Interface
HX	Heat Exchanger
IAEA	International Atomic Energy Agency
IAM	Integrated Aging Management
IFB	Irradiated Fuel Bay
IIP	Integrated Improvement Plan
IMS	Inspection and Maintenance Services
IR	Infrared
ISR	Integrated Safety Review
LCMP	Life Cycle Management Plan
LISS	Liquid Injection Shutdown System
LSFCRP	Large Scale Fuel Channel Replacement Program
MCC	Motor Control Centre
MIC	Microbiologically Induced Corrosion
MOT	Main Output Transformer
MV	Motorized Valve
NGS	Nuclear Generating Station
NICR	Non-Identical Component Replacement
NPC	Negative Pressure Containment
NV	Non-Return Valve
OPG	Ontario Power Generation
OSR	Operational Safety Requirements
PdM	Predictive Maintenance
PM	Preventative Maintenance

PMID	Preventative Maintenance Identifier
PNGS	Pickering Nuclear Generating Station
PROL	Power Reactor Operating Licence
PSR	Periodic Safety Review
PWMF	Pickering Waste Management Facility
RB	Reactor Building
RC&S	Reactor Components and Structures
RP&Ps	Radiological Processes and Procedures
SCR	Station Condition Record
SG	Steam Generator
SIS	Systems Important to Safety
SOE	Safe Operating Envelope
SOR	Shut off Rod
SSC	Structures, Systems, and Components
SST	System Service Transformer
SV	Solenoid Valve
S/Y	Switchyard
TCS	Tool Control System
TE	Temperature Element
TMB	Training and Mock-up Building
TSSA	Technical Standards and Safety Authority
UDM	Universal Delivery Machine
UPS	Uninterruptable Power Supply
UT	Ultrasonic Testing
WO	Work Order

APPENDIX B. SOE/SIS SYSTEMS INCLUDED IN SF2

Appendix B.1

The systems included in this review are focused on the Pickering NSG 1, 4 SOE/SIS systems as per Table B.1-1 and Pickering NGS 5-8 SOE/SIS systems as per Table B.1-2 in this Appendix. In preparation of the DCAs, system numbers are assigned to the in-scope systems. These system numbers are shown in Tables 1 and 2 for each SOE/SIS system. Note: Some SOE/SIS systems cover more than one system/sub-systems, therefore multiple system numbers can be assigned to a single SOE/SIS system.

APPENDIX B.1 SIS/SOE SYSTEMS

Table B.1-1: Pickering NGS 1, 4 SOE/SIS Systems

#	Pickering NGS 1,4 Systems	SOE[1]	SIS[2]
1.	Emergency Coolant Injection System System Numbers: 0421, 0436, 0437	√	√ (Also includes associated recovery system (with Moderator Pumps and Moderator Room Active Drainage Sump Pumps))
2.	Shutdown System A System Number: 0462	√	√
3.	Shutdown System E System Number: 0463	√	√
4.	Negative Pressure Containment Systems System Numbers: 0412, 0423, 0452, 0455, 0466, 0469	√	√
5.	Powerhouse Emergency Venting System System Number: 0447	√	√
6.	Reactor Regulating System System Numbers: 0444, 0453	√	
7.	Service Water Systems System Numbers: 0410, 0454, 0456	√	√ (Limited to Emergency Low and High Pressure Service Water)
8.	Moderator System System Number: 0445	√	
9.	Electrical Power System System Numbers: 0408, 0420, 0464	√	√ (Limited to Standby Class III power and Class III 600V Interstation Transfer Bus, Emergency Transfer Scheme and Class III 600V Motor Control Centres 54130-MCC-18 and MCC-19)
10.	Emergency Boiler Water Supply System Numbers: 0423, 0456	√	
11.	Heat Transport System System Number: 0449	√	√
12.	Shutdown Cooling System System Number: 0459	√	
13.	Boiler Emergency Cooling System System Number: 0402	√	
14.	Feedwater System System Numbers: 0401, 0403, 0417, 0426	√	
15.	Main Steam Supply System System Number: 0404	√	
16.	Fuel and Reactor Physics System Number: N/A[3]	√	
17.	Annulus Gas System System Number: 0400	√	

#	Pickering NGS 1,4 Systems	SOE[1]	SIS[2]
18.	Critical Safety Parameter Monitoring Instrumentation System Numbers: 404, 421, 437, 458, 462, 463	√	
19.	Fuel Handling System & Irradiated Fuel Bays System Numbers: 0428, 0430, 0431, 0432, 0433, 0440, 0441, 0472	√	
20.	Shield Cooling System System Number: 0458	√	
21.	Interstation Transfer Bus System Number: 0420	√	
22.	Powerhouse Environmental Protection System System Number: 0448[4]	√	
23.	Critical Structures (e.g., Reactor Buildings, Pressure Relief Duct and Vacuum Building) [5] System Numbers: 0406, 0425, 0436, 0441, 0452, 0455, 0466, 0469		

1. Specific USIs are provided in associated system OSRs listed in References [16] and [19]. Also, it includes elements of specified support systems (e.g. Instrument Air) where it is required for credited design basis functions (This is generally reflected in the criticality coding).
2. Specific USIs and required functional elements are detailed in [16] and [17].
3. There are no specific equipment credits in the Fuel & Reactor Physics (F&RP) OSR. Any implicitly credited instrumentation for measurement purposes are included in other systems, e.g. SDSA. Therefore there is no system number or Appendix B-2 Table entries for F&RP.
4. System 448 is not in Tables in Appendix B-2 since all components were screened out and CAs were not required.
5. Critical Structures supporting SOE/SIS operation are included in the review.

Table B.1-2: Pickering NGS 5-8 SOE/SIS Systems

#	Pickering NGS 5-8 Systems	SOE[1]	SIS[2]
1.	Emergency Coolant Injection System System Numbers: 0421, 0436, 0437	√	√
2.	Shutdown System One System Number: 0460	√	√
3.	Shutdown System Two System Number: 461	√	√
4.	Negative Pressure Containment Systems System Numbers: 0412, 0423, 0452, 0455, 0466, 0469	√	√
5.	Powerhouse Emergency Venting System System Number: 0447	√	√
6.	Reactor Regulating System System Numbers: 0444, 0453	√	
7.	Service Water Systems System Numbers: 0410, 0454, 0456	√	√ (Limited to Class III Service Water (Low and High Pressure))
8.	Moderator System System Number: 0445	√	
9.	Group 1 Electrical Power System System Numbers: 0408, 0420, 0464, 0465	√	√ (Limited to Standby Class III power / Class II Power and also includes Class II UPS Room Ventilation)
10.	Emergency Water Supply System System Number: 0423, 0425	√	√ (Limited to Emergency Water Supply to Boilers, Heat Transport, and Moderator)
11.	Heat Transport System System Number: 0449	√	
12.	Shutdown Cooling System System Number: 0459	√	√
13.	Boiler Emergency Cooling System System Number: 0402	√	
14.	Feedwater System System Numbers: 0401, 0403, 0417, 0426	√	√ (Limited to Auxiliary Boiler Feedwater and Auxiliary Condensate Systems)
15.	Main Steam Supply System System Number: 0404	√	
16.	Fuel and Reactor Physics System Number: N/A[3]	√	
17.	Emergency Power Supply System Number: 0422	√	√
18.	Fuel Handling System & Irradiated Fuel Bays System Numbers: 0428, 0430, 0431, 0432, 0433, 0440, 0441, 0472	√	

#	Pickering NGS 5-8 Systems	SOE[1]	SIS[2]
19.	HPECI Power Supplies System Numbers: 403, 404, 417, 420, 453, 464	√	
20.	Annulus Gas System System Number: 0400	√	
21.	Critical Safety Parameter Monitoring Instrumentation System Numbers: 404, 421, 423, 445, 449, 451, 453, 460, 461	√	
22.	Shield Cooling System System Number: 0458	√	
23.	Critical Structures (e.g., Reactor Buildings, Pressure Relief Duct and Vacuum Building)[4] System Numbers: 0406, 0425, 0436, 0441, 0452, 0455, 0466, 0469		

1. Specific USIs are provided in associated system OSRs listed in [16] and [19]. Also includes elements of specified support systems (e.g., Instrument Air) where required for credited design basis functions (This is generally reflected in the criticality coding).
2. Specific USIs and required functional elements are detailed in [16] and [17].
3. There are no specific hardware credits in the Fuel & Reactor Physics (F&RP) OSR. Any implicitly credited instrumentation for measurement purposes are included in other systems, e.g. SDS1. Therefore there is no system number or table entries for F&RP.
4. Note: Critical Structures supporting SOE/SIS operation are included in the review.

Appendix B.2

The Tables in Appendix B.2 contain the key findings from the System Summaries documented in section 4.1.1.2. The Tables are shown on a system basis listing the system number and name, e.g. System 0400 Annulus Gas. For each of the systems, the DCA results are shown for each Commodity Group (CG) which are reactor safety or production critical. The following columns are provided:

Column	Description
CG	Commodity Group Number
Units	Applicable Units
USI#	All USIs/sub-USIs covered in the CG
CA Description	Component Type
Function	Description of Reactor Safety or Production Role
Classification	Aging Management Classification (more information provided below)
Potential Aging Related Degradation Mechanisms (ARDMs)	ARDMs applicable to the component
Current Practices	Summary of Aging Management practices currently in place (more information provided below)
Incremental Work Recommended for Current and Extended Operating Life	Aging Management Recommendations (more information provided below)

Aging Management Classification

An overall Condition Classification (Very Good, Good, Satisfactory, Poor, or Very Poor) is established in N-PROC-MP-0060 [7], which accounts for:

a) The physical condition of the component at time of assessment, and
b) The adequacy of the practices in place to manage component aging.

Condition Classification is assigned by selecting the lesser of these two criteria. For example if the physical condition of a component is “Good”, but the adequacy of practices is “Satisfactory”, the component condition classification is “Satisfactory”. Aging Management recommendations are required for CGs having a Poor or Very poor classification. The criteria used for assigning classification is provided in Table C.2.

Of note, the condition classification of “Satisfactory” can be a result of:

- The component still meets all it functional design requirements, but operating margins are significantly eroded. This can be attributed to evidence of significant aging degradation, or,

- The *aging management* practices are ineffective in only one area and should be reviewed and/or changed.

Given these definitions of Satisfactory, improvement recommendations would nominally be expected. Of the approximately 350 CGs with this classification, the majority have a number of associated recommendations. Only about 90 CGs (~25%) currently do not contain recommendations. This is acceptable for the following reasons:

- The physical condition is rated as “Good”, however, only one practice is ineffective and therefore classification is “Satisfactory”. In some cases practices may not be practical or justified to improve practices for a component in a physically “Good” condition.
- The physical condition is rated as “Satisfactory”. The classification for “Good” is indicative of a component having only a slight reduction in operating margins. There is a significant difference between these two condition definitions, i.e. a “slight reduction” vs. “significantly eroded”. Usually, the classification will be assessed conservatively, resulting in Satisfactory being selected.
- The criticality of the component is low, i.e. CC3 or CC4.
- Per OPG governance, recommendations are only required for “Poor” and “Very Poor” classifications.

Current Practices

This column lists the aging management practices in place for the components across the CG. It includes high level activities. These practices were extracted from the updated DCAs. Additional details, e.g. PM #s, etc. are documented in the DCAs. In some cases the practice on the parent component is listed, e.g. limit switches are addressed during the diagnostics test performed on a valve.

There can also be cases where a practice is listed for a CG, e.g. calibration, yet there is a recommendation to perform calibrations on a specific component(s) in that CG. This is acceptable, as in this case, calibration is performed on the majority of components in the CG, but it is not performed in the components that are listed in the recommendations. There are standard practices that are in place generically which are not listed as they are applicable to all relevant CGs. These include: Operator Rounds, System Walkdowns, and Trending of Safety System Component Failure Rates in accordance with REGDOC-3.1.1 [98]. Operator Rounds and System Engineer surveillance are listed for components where these are the primary practices.

Incremental Work Recommended for Current (2020) and Extended Operating Life (2024)

This column identifies recommendations resulting from the Condition Assessments for each CG. Although recommendations are only required for Poor or Very poor classifications, recommendations can also be made for Satisfactory or better classifications to maintain that rating or to make improvements.

Three different types of recommendations can be made:

a) Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices.

   These are recommendations to complete currently planned work for plant end of life in 2020. For example, complete currently planned inspection work orders (WOs). They are termed “incremental” as they are not currently complete, and are not in the set of periodic aging management practices.

b) Incremental recommendations for Plant EOL (2020).

   These are recommendations to complete work that is not currently planned (i.e. incremental) for current plant end of life in 2020.

c) Incremental recommendations for CO EOL (2024)

   These are recommendations to complete work that is necessary for life extension. For the System Summaries a life extension date of 2024 was considered. The DCAs currently in progress address life extension to 2028.

Information for selected CGs has been shaded in this revised version of the report. These shaded rows represent cases where the information is no longer applicable, as the latest scoping and screening has determined that these CGs can be screened out and no longer require a Detailed CA to be produced. The condition of these components and recommendations made are no longer relevant to this SF2 assessment and have not been updated. They are included to illustrate which CGs have been removed compared to OPG Revision 0 of this report.

As an example, for the Annulus Gas System, system 0400, Commodity Group 11475, on CO2 monitors has been shaded, i.e. screened out. The justification for this is the criticality code was changed from cat 1/2 to cat 3/4 and therefore a Detailed CA is no longer required. This change in criticality code occurred as a result of the criticality code review project performed by OPG. The recommendation provided in the Table to complete EC 111431 is also removed in section 4.1.1.2.1. However, this action is still in the SHR and will be tracked, prioritized and executed via the work management system.

Also, the content of the tables has not been updated to reflect the DCA work currently in progress. The tables reflect a snapshot of the CA results representative of a freeze date of Aug 31, 2015. The results of the updated CAs with a freeze date of January 15, 2016 and addressing extended operation to 2028 is a PSR2 gap. The CA recommendations from these updated CAs will be captured in system health reports, and prioritized and tracked to completion via the System Health process. OPG plans to implement a new Aging Management (AM) database to aid with the tracking and oversight of AM actions.

System 0400 Annulus Gas

CG:	Units	USI #	CA Description	Function	Classification	Potential Aging Related Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008000	1,4	34980 63498	ALARM, Cat 1/2	CO2 concentration in strategic areas surrounding the Annulus Gas System is monitored continuously for each unit. Level instrumentation is provided for drains tank, alarms in MCR.	Good	Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008001	1,4	34980 63498	ANALYZER, Cat 1/2	Oxygen is added to the AGS to maintain 02 concentration between 0.5% and 5% by volume at all times. Oxygen analyzer continuously monitors concentration on-line during normal operation.	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging	Surveillance	Incremental recommendations for Plant EOL (2020): Implement a new PM to perform periodic calibration and function checking. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
008003	1,4	34980 34980	COIL, Cat 1/2	The finned cooling coil is used with two cooling fans and a separator for condensing and collecting sufficient liquid to alarm moisture beetles.	Good	Corrosion / General Corrosion	Surveillance	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008004	1,4	34980 34980	COMPRESSOR, Cat 1/2	There are stainless steel, bellows type compressors which provide air flow for recirculation. Each compressor has a nominal flow rating of 170 NLlmin (6 Ncfm).	Satisfactory	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Lubricant Degradation Corrosion / Oxidation		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete EC 119541 to replace the applicable compressors with CID 670078. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI #	CA Description	Function	Classification	Potential Aging Related Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008007	1,4	34980 34980	DISC, Cat 3/4	The overpressure protection for the compressors is provided by rupture discs 3498-RD1, RD2 and RD3. The rupture discs have been set to burst at a maximum differential pressure of 227.5 kPa(d).	Satisfactory	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008008	1,4	34980 63498	ELEMENT, Cat 1/2	Moisture element (beetle) measures dewpoint of CO2 in order to detect pressure tube, calandria tube leaks or leakage from the End Shield Cooling.	Good	Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008010	1,4	34980 34980	FAN, Cat 1/2	The cooling fans are used with a finned cooling coil and a separator for condensing and collecting sufficient liquid to alarm moisture beetles.	Satisfactory	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Functional Test Logic Test Lubrication Predictive Maintenance - Thermography Predictive Maintenance - Vibration Monitoring	Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Implement a new PM to periodically check the fan and motor bearings and add lubrication if necessary.
008013	1,4	34980 63498	FUSE, Cat 1/2	40VDC supply fuse for Moisture Collection Tank level instrumentation loop.	Good	Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008014	1,4	34980 63498	GAUGE, Cat 1/2	Collection Tank level gauge.	Good	Fatigue / Mechanical Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008018	1,4	34980 63498	INDICATOR, Cat 1/2	Flow meter to monitor purge flow	Good	Fatigue / Mechanical Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008019	1,4	34980 63498	INDICATOR, Cat 3/4	1-63498-F4-FI515 is a flowmeter for measuring beetle blast flow, currently not in use and valved out. 1-	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI #	CA Description	Function	Classification	Potential Aging Related Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				63498-F5-FI527 is a flow meter used for monitoring gas flow during cold finger sampling, normally valved out. PI instruments are system local pressure gauges.		Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		
008021	1	34980 63498	MONITOR, Cat 1/2	Fixed CO2 monitors & detectors	Good	Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008022	4	34980 63498	MONITOR, Cat 3/4	Fixed CO2 monitors & detectors	Good	Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008023	1,4	34980 34980	MOTOR, Cat 1/2	Compressor motor & motor cooling fan.	Satisfactory	Environmental degradation, General corrosion, Mechanical fatigue, Thermal fatigue / 81		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008026	1,4	34980 63498	PROCESSOR, Cat 1/2	Collection Tank rate calculation microprocessor.	Good	Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008027	1,4	34980 63498	RECORDER, Cat 3/4	Chart recorders for AG system	Good	Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008028	1,4	34980 34980	RELAY, Cat 1/2	Relay in cooling fan motor start logic.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI #	CA Description	Function	Classification	Potential Aging Related Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008029	1,4	34980 34980	RESISTOR, Cat 1/2	Resistor in cooling fan start logic.	Good	Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008038	1,4	34980 63498	TRANSMITTE R, Cat 1/2	Collection tank level transmitter	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging Fatigue / Mechanical Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011263	5,6,7,8	34980 63498	63498 Annulus Gas System-Oxygen Analyzer	Analyser provides local oxygen concentration indication.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Perform probe and digital display replacement for O2 Analyser per EC 122171. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011265	5,6,7,8	34980 63498	63498 Annulus Gas System-Moisture Hygrometer Transmitter	Transmits measurements of CO2 dewpoint to MCR, detects pressure tube or calandria tube leaks or leakage from the End Shield Cooling.	Good	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Electronic Aging Radiation Induced Degradation / Radiation Embrittlement		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI #	CA Description	Function	Classification	Potential Aging Related Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011267	5,6,7,8	34980	34980 Annulus Gas System - Power Cables - 600V, 125V, 250V, 120V	To supply power & control to Annulus Gas system equipment.	Good	Mechanical and Thermal Degradation / Radiation Embrittlement Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue	Inspection - Condition Monitoring Program	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI #	CA Description	Function	Classification	Potential Aging Related Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011268	5,6,7,8	34980 34980	34980 Annulus Gas System-Valves - NV-Non-Return - CAT1&2	34980-NV102-Prevents backflow from bottle. 34980-NV103-Prevents backflow from panel. 34980-NV33-CO2 Bottle Stn Supply NV-Prevents Annulus Gas from back flowing to the Carbon Dioxide Bottle supply. 34980-NV34-CO2 Bulk CO2 Supply NV-Prevents Annulus Gas from back flowing to the Bulk Carbon Dioxide supply. 34980-NV50-Annulus Gas PRV41 Non Return-Prevents Annulus Gas from back flowing to the Bulk Carbon Dioxide supply. 34980-NV54-Annulus Gas CP1 discharge NV-Prevents Annulus Gas from back flowing through CP1. 34980-NV57-Annulus Gas CP2 discharge NV-Prevents Annulus Gas from back flowing through CP2. 34980-TK1 D2O drain tanks receive liquid drainage from the beetle (M1) and monitors the collection rate. 34980-NV60-Annulus Gas CP3 discharge NV-Prevents Annulus Gas from back flowing through CP3	Satisfactory	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Corrosion / General Corrosion Obsolescence / Immediate Obsolescence Concern Mechanical and Thermal Degradation / Creep and Stress Relaxation	Component Replacement SRST - Functional Test	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Replace 6/7/8-34980-NV102. This requires implementation of CAT ID 709329 (which is currently going through the design processes to address obsolescence issues with CAT ID 118136). <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI #	CA Description	Function	Classification	Potential Aging Related Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011475	5,6,7,8	34980 63498	63498 - Annulus Gas System - CO2 Monitors - CAT 1&2	CO2 concentration in strategic areas surrounding the Annulus Gas System is monitored continuously for each unit.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Electronic Aging		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Proactive replacement to be completed (EC 114431). Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

System 0401 Boiler Blow-Off System

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008058	1,4	36400 36410	VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2	This CA includes the local boiler blowdown isolating valves. They are normally closed, and opened intermittently to assist in chemical control in the boilers.	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue	Surveillance SRST - Functional Test	Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Perform a one-time inspection of valve internals against ARDMs (elastomer embrittlement, material loss on bushing, spindle, seat, disc, bellows failure, valve body corrosion), weld inspections, and repair/replace valve internal parts if degraded.
008060	1,4	36400 36410	VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2, large boiler blowdown valves	This CA includes the main boiler blow-off isolating valves. These valves are operated intermittently, and operate in conjunction with local boiler blow-off isolating valves to maintain reasonable solids accumulation levels and assist in chemical control in the boiler.	Satisfactory	Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue	Overhaul/Refurbishment SRST - Functional Test Valve Diagnostics	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete WOs 2731629, 2731630, 2284458, 2243671 to overhaul and/or replace valve internals and actuator. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
008061	1,4	36400 36410	SOLENOID/SOLENOID OPERATED VAL, Cat 1/2	The Blow off system has individual boiler blow off valves 36410-MV2 through MV24. The pneumatic valves are operated by solenoid valves, 36410-MV2-SV1 through MV24-SV1.	Good	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Electronic Aging Radiation Induced Degradation / Radiation Depletion of Material Properties Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		
008063	1,4	36400 36410	SOLENOID/SOLENOID OPERATED VAL, Cat 1/2 for large boiler blowdown valves	The Blow-off system has two large pneumatic valves, 3641-MV37 and 3641-MV38, mounted on header lines for the two groups of boilers. The pneumatic valves are operated by two solenoid valves, 3641-MV37-SV1 and 3641-MV38-SV1.	Very Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear Radiation Induced Degradation / Radiation Depletion of Material Properties Mechanical and Thermal Degradation / Thermal Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011278	5,6,7,8	36400 36410	36410 Boiler Blow-off System NVs	These check valves protect against a backflow of water/steam from the intermittent blowdown and steam when the steam reject valves 36110-MV36 and MV 42 operate	Satisfactory	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear		<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Complete a one-time replacement of all check valves in this CG.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Wear Mechanisms - Erosion		
011297	5,6,7,8	36400 36410	36410 Boiler Blow-off System - AOVs	This CA includes individual blowdown isolating valves and main blowdown isolating valves. The main isolating valves are normally open and provide a common isolation point for the blowdown system from the east or west bank of boilers. They are operated in a coordinated sequence with the individual boiler blowdown valves (normally closed) allowing an operator to select a boiler for blowdown, to provide chemical control and sludge removal from the boiler water.	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Thermal Aging Fatigue / Mechanical Fatigue	Component Replacement Inspection - Visual SRST - Functional Test	<u>Incremental recommendations for Plant EOL (2020):</u> Initiate one-time inspection of valve internals, and replace/repair if degraded. Procure additional spare valves to ensure corrective action can be expedited if required. <u>Incremental recommendations for CO EOL (2024):</u> Initiate new PMs for external inspections of valves every two years.
011298	056, 078	36400	36410 Boiler Blowoff System- Mechanical	This CA includes the pipe, tube, fittings, flanges, ball joints, pipe supports, snubbers and anchors that form the boiler blowdown system flow path from the boilers to the screen house intake channel.	Good	Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Corrosion / General Corrosion Fatigue / Thermal Fatigue Fatigue / Mechanical Fatigue	Inspection - Pipe Wall Thinning Program	<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendation for CO EOL (2024):</u> Spare ball joint components to be procured to facilitate corrective actions if/when required.

TETRA TECH

System 0402 Boiler Emergency Cooling System

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008064	1,4	36710 63670	ALARM, Cat 1/2	Level and temperature alarms for the Boiler Emergency Cooling System.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Calibration	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008065	1,4	36710 63670	GAUGE, Cat 1/2	These components are BECS water storage tanks level gauges.	Good	Corrosion / General Corrosion Fatigue / Mechanical Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008066	1,4	36710 63670	INDICATOR, Cat 1/2	Pressure indicator (on CR PL3C) and associated hand-switch allow operator to view pressure from 63670-P1-PT1 or 63670-P2-PT1 which belong to the Boiler Emergency Cooling System.	Good	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Electronic Aging		
008070	1,4	36710 63670	SWITCH, Cat 1/2	Pressure alarms for the Boiler Emergency Cooling system.	Good	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008071	1,4	36710 36710	SWITCH, Cat 1/2	Limit switches on MVs indicating Valve position for the Boiler Emergency Cooling Valves.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion Fatigue / Mechanical Fatigue Radiation Induced Degradation / Radiation Depletion of Material Properties	Overhaul/Refurbishment Valve Diagnostics	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008073	1,4	36710 36710	TANK, Cat 1/2	36710-TK1 and TK2 are water storage tanks to provide injection water. 36710-TK3, TK4, TK5 are pressurized air tanks to provide pressure – assistance for water injection.	Good	Corrosion / General Corrosion Corrosion / Pitting Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008074	1,4	36710 63670	TRANSMITTER, Cat 1/2	Pressure Transmitter converts pneumatic input signal to output current required for providing alarms.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging Radiation Induced Degradation / Radiation Depletion of Material Properties		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008081	1,4	36710 36710	VALVE, CHECK/NONRETURN/BACKFL, Cat 1/2, BECS-02	BECS injection non-return valves to prevent reverse flow from boilers to the BECS water tanks.	Good	Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms – Wear	Inspection - Internal Inspection - Non - Intrusive	<u>Incremental recommendations for Plant EOL (2020):</u> Perform one-time inspection of 4-36710-NV3, NV13, 1-36710-NV3, and NV13. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
008082	1,4	36710 36710	VALVE, CHECK/NONRETURN/BACK	BECS air non-return valves, prevent backflow of air from the high pressure instrument air station.	Good	Corrosion / Flow Assisted Corrosion (Erosion-Corrosion)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
			FL, Cat 1/2, BECS-01			Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear		
008083	1,4	36710 36710	VALVE, CHECK/NONRETURN/BACKFL, Cat 1/2, BECS-04, 05	BECS air supply line non-return valves.	Good	Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008084	1,4	36710 36710	VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2	MV2 and MV14 are normally closed isolators which permit discharge of BECS tanks TK1 & TK2 to boilers.	Very Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008085	1,4	36710 36710	VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2, Worcestor Rack & Pinion, Ball Valve	MV15 is a motor operated valve which permits the BECS tanks to be pressurized via instrument air.	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear	SRST - Stroke Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008086	1,4	36710 36710	VALVE, PRESSURE REGULATING, Cat 1/2	PRV1 regulates instrument air pressure to controls of end device MV15. PRV3006 regulates instrument air pressure to BECS tanks.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Creep and Stress Relaxation	Calibration SRST - Functional Test	Incremental recommendations for Plant EOL (2020): Complete a one-time calibration/set point check for 1/4-36710-PRV3006 to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Complete a one-time calibration/set point check for 1/4-36710-PRV3006 to reach CO EOL (2024).
008087	1	36710 36710	VALVE, PRESSURE REGULATING, Cat 1/2, related to POV program valve	The PRVs serve to regulate instrument air pressure to the air controls of their end devices MV2 and MV14.	Good	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Creep and Stress Relaxation	Overhaul/Refurbishment SRST - Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008088	1,4	36710 36710	VALVE, PRESSURE RELIEF, Cat 1/2	The RVs serve as overpressure protection devices for BECS 1, 4-36710-TK1, 2, 3, 4, 5 and associated piping.	Very Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Creep and Stress Relaxation Corrosion / Fouling (accumulation of deposits) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008089	1,4	36710 36710	SOLENOID/SOLENOID OPERATED VAL, Cat 1/2	Solenoid valves directs air to operate MVs.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging Radiation Induced Degradation / Radiation Depletion of Material Properties	SRST - Functional Test	Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Perform one-time overhaul of Actuator and Solenoid Valve.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008090	1,4	36710 36710	SOLENOID/SOLENOID OPERATED VAL, Cat 1/2, related to POV program valve, EF8320	Solenoid valves provide instrument air to operate MV14 and MV2.	Very Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging Radiation Induced Degradation / Radiation Depletion of Material Properties		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011366	5,6,7,8	36710 36710	36710 Boiler Emergency Cooling System-Tanks	5/6/7/8-36710-TK1/TK2 - Boiler Emergency Storage Water Tank 5/6/7/8-36710-TK3/TK4/TK5 - Boiler Emergency Cooling Pressurizing Air Tank	Good	Corrosion / General Corrosion Corrosion / Pitting Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Inspection - Visual SRST - Air Holding Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

TETRA TECH

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011477	5,6,7,8	36710 36710	36710 Boiler Emergency Cooling System-Valves - NV-Non-Return - CAT1&2	5/6/7/8-36710-NV3, NV34 NV3 and NV34 are swing check valves located in the injection lines. They prevent hot, high pressure water from entering the BECS tanks in event that injection valve CV2 or CV35 open spuriously. 5/6/7/8-36710-NV4 NV4 is a piston type check valve in the BECS drain/overflow line. 5/6/7/8-36710-NV22 NV22 is a piston type check valve in the instrument air supply line to prevent backflow from the BECS to the Instrument Air System. 5/6/7/8-36710-NV26, NV27 Instrument air supply non return valve to boiler Emergency Cooling tanks 1 & 2.	Satisfactory	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Obsolescence / Immediate Obsolescence Concern	Component Replacement Inspection - Non - Intrusive SRST - Functional Test	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Perform internal inspection per WO 1618394. Also, resolve spare parts and obsolescence issues. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011484	5,6,7,8	36710	36710 Boiler Emergency Cooling System-Piping - Cat1&2	The system is designed as a safety support system to provide water injection into the boilers upon loss of feedwater.	Good	Corrosion / General Corrosion Corrosion / Microbiological Influenced Corrosion	Inspection - Pipe Wall Thinning Program SRST - Leak Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

System 0403 Boiler Feed System

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008091	1,4	43230 64323	ALARM, Cat 1/2	Components in this CA control Auxiliary Boiler Feed pump #5 (ABFP#5) operation and provide alarms and trips for very high boiler levels. AS212 to AS215 control ABFP#5 operation. L51A-LIA1 provide very high boiler level alarm and L51A-LA1 sends trip signals to turbine governor valves.	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008093	1,4	43230 43230	BOOSTER, Cat 1/2	AF1's (boosters) provide a high volume air signal to improve response of associated control valves (CV). CVs control flow to boilers to maintain boiler level at set point.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008096	1,4	43230 64323	CONTROL, Cat 1/2	PC206 controls Auxiliary Boiler Feed Pump #5 outlet pressure. LIC1's control boiler level control valves position to maintain boiler level (Boiler Level Control).	Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008098	1,4	43230 64323	CONVERTER, Cat 1/2	The convertors compute rational signals from Steam Flow, Feedwater flow and Boiler level to derive an output signal that can be used to accurately control boiler level.	Good	Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008100	1,4	43230 64323	ELEMENT, Cat 1/2	The flow elements produce differential pressure changes that are related to changes in flow rate.	Good	Corrosion / Flow induced wear of the leading edge	Calibration	Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Perform one-time inspection, and if degraded replace.
008102	1,4	43230 43230	FILTER, Cat 1/2	These air filters are used to clean air for the reliable operation of CVs.	Very Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008106	1,4	43230 43230	HEAT TRACING, Cat 3/4	HX3001/3003/3005: Removes heat from circulated lubrication oil for Boiler Feed Pump (BFP) 6/7/8. HX3002/3004/3006: Removes heat from circulated lubrication oil for Boiler Feed Pump Motor (BFPM) 6/7/8.	Good	Corrosion / Fouling (accumulation of deposits) Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008107	1,4	43230 43230	HEATER (GENERIC), Cat 1/2	Pump motor heater is used to heat motor windings and reduce moisture buildup when pump is off line.	Very Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008109	1,4	43230 43230 64323	INDICATOR-CAT 1/2	Indicate Feedwater flow or valve position.	Good	Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008114	1,4	43230 43230	MOTOR, Cat 1/2	Pump motor provides motive power to drive Auxiliary Boiler Feed pump #5.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring,	Overhaul/Refurbishment Predictive Maintenance - Vibration Monitoring	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Predictive Maintenance - Lubrication Analysis Predictive Maintenance - Thermography SRST - Logic Test SRST - Functional Test	
008115	1,4	43230 43230	MOTOR, Cat 3/4	Pump motors provide motive power to operate pumps.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008118	1,4	43230 43230	POSITIONER, Cat 1/2	Valve positioners (NC1) accurately set valve stem position based on a control signal.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008120	1,4	43230 64323	PROCESSOR, Cat 1/2	These processors compute input signals to generate an output control signal based on the function of the LM or FTX.	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008121	1,4	43230 43230	PUMP, Cat 1/2	The auxiliary boiler feed pump is fed from the Class III power supply and is used for supplying boiler feedwater.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Predictive Maintenance - Vibration Monitoring Predictive Maintenance - Thermography SRST - Logic Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		
008122	1,4, 014	43230 43230	PUMP, Cat 3/4	These Boiler Feed pumps include the main boiler feed pumps that delivers feedwater to the boilers, recirculation pumps that pumps feedwater back to the deaerator storage tank as well as oil pumps that provide lubrication to the boiler feedwater pumps and associated motors.	Satisfactory	Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete outstanding CM/DM work orders for repairing oil leaks, pump refurbishment, vibrational fixes, water leaks and pump tundish leak. <u>Incremental recommendations for Plant EOL (2020):</u> Implement a vibration monitoring program. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
008123	1,4	43230 64323	RECEIVER, Cat 1/2	The receivers provide local back-up instrument air for control valves valve 43230-CV200/CV206	Good	Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008127	4	43230 43230	RELAY, Cat 1/2	83-R12 relay automatically transfers P5 logic control power (48 VDC) from one failed supply to a backup supply.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Predictive Maintenance - Thermography	<u>Incremental recommendations for Plant EOL (2020):</u> Implement recurring inspections to assess degradation of relay. <u>Incremental recommendations for CO EOL (2024):</u>

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Thermal Aging		No additional practices are recommended to reach CO EOL (2024).
008131	1,4	43230 43230	STRAINER, Cat 1/2	This strainer filters out debris, preventing it from affecting the performance of a main boiler feedwater pump.	Good	Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008134	1,4	43230 64323	SWITCH, Cat 1/2	43230-CV213-NS1 is control valve (CV213)'s position switch used in valve operational logic. 64323-F51-PS1 supplies DC power to the feedwater flow measuring loop. 64323-P11-PS239 senses 43230-P5 suction strainer differential pressure. 64323-F5-NS204 senses isolation valve V125, position. 64323-SS200 detects 43230-P5 speed and pump direction.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008136	1,4	43230 43230 64323	SWITCH, Cat 1/2	Hand Switches (HS) used for changing the operating state of equipment (ON/OFF or Auto).	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008139	1,4	43230 43230 64323	TRANSMITTER, Cat 1/2	1-43230-MV191-NT1 senses and transmits MV position to Indicator in the MCR. 1-64323-L51A-LT1 senses Boiler Level for use in the 3 element Boiler Level Controller. 1-64323-L51-AX1 converts electrical boiler level control	Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				signals to pneumatic signals for level Control Valves. 1-64323-F12-FT207 measures boiler feed pump discharge flow rate. 1-64323-F51-FT1 measures feed water flow rate for use in Boiler Level Controller.				
008140	1,4	43230 64323	I/P CONVERTER, Cat 1/2	AX's convert electrical control signals to pneumatic signals for operation of pneumatic control valves.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008144	1,4	43230 43230	VALVE, CONTROL, Cat 1/2	These Control Valves are used to recirculate boiler feedwater for boiler level control purposes.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Overhaul/Refurbishment Valve Diagnostics Inspection - Visual SRST - Stroke Test SRST - Functional Test	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete work orders, and resolution to the following issues: 1) 4-43230-CV213 leaking WO#4703681 2) Pressure controller failure 1-43230-CV206 WO#4762318 3) Repacking of 1-43230-CV220 WO#4809923 4) Check welds for cracks 1-43230-CV202 WO#3060177. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
008146	1,4	43230 43230	VALVE, MANUAL/HAN	These manual valves in the Boiler Feed Water circuits	Good	Corrosion / General Corrosion	SRST - Functional Test	<u>Current initiatives that need to be completed for Plant EOL (2020) that are</u>

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
			D OPERATED, Cat 1/2	used primarily for component isolation.		Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear		incremental to current periodic maintenance practices: Complete outstanding CM/DM work orders to correct leaking valves. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
008148	1,4	43230 43230	VALVE, MOTORIZED/ MOTOR OPERATE, Cat 1/2	3A / 114A: HP Heater bank inlet valve MV113B / 114B: HP Heater bank inlet bypass valve MV115A / 116A: HP Heater bank outlet valve MV115B / 116B: HP Heater bank outlet bypass valve MV163A / 165A / 167 / 169 / 171A / 173A / 175A / 177A / 179 / 181 / 183A / 185A: Isolation valve for Boiler Level MV190 / 191 / 192 / 193 / 194 / 195 / 196 / 197 / 198 / 199 / 200 / 201: Motorized trim valve for boiler	Satisfactory	Corrosion / General Corrosion Mechanical and Thermal Degradation / Lubricant Degradation Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Inspection - Internal Inspection - Visual Lubrication Valve Diagnostics	Incremental recommendations for Plant EOL (2020): Perform diagnostics including a review of current data to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
008150	1,4	43230 43230	VALVE, CHECK/NONR ETURN/BACK FL, Cat 1/2	These valves prevent the backflow of feed water from the boilers.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Fatigue / Mechanical Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008153	1,4	43230 43230	VALVE, CHECK/NONR ETURN/BACK	These valves prevent backflow to the main boiler feed water pumps.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Inspection - Non - Intrusive SRST - Functional Test	Incremental recommendations for Plant EOL (2020): Perform a one-time internal inspection of at least one “Sample” valve to use as a

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
			FL, Cat 1/2, FW-02			Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Fatigue / Mechanical Fatigue		reference of Classification for the other valves. Incremental recommendations for CO EOL (2024): Initiate PM's for non-intrusive testing of all valves in the CG every 104 weeks.
008154	1,4	43230 43230	VALVE, CHECK/NONRETURN/BACKFL, Cat 1/2, FW-01	The valves act as non-return valves for the auxiliary boiler feed pump.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008155	1,4	43230 64323	VALVE, CHECK/NONRETURN/BACKFL, Cat 1/2, CA-45	The non-return valves prevent backflow of instrument air from the boiler feed control system.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Corrosion / General Corrosion Fatigue / Mechanical Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008156	1,4	43230 64323	VALVE, PRESSURE RELIEF, Cat 1/2	This relief valve provides overpressure protection in the boiler feedwater system	Good	Fatigue / Mechanical Fatigue Corrosion / General Corrosion Mechanical and Thermal Degradation		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						/ Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		
008158	1,4	43230 64323	SOLENOID/SOLENOID OPERATED VAL, Cat 1/2	Solenoid valves used in these applications operate Boiler Feed pump recirculation control valves.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011154	5,6,7,8	43230 43230	43230 Auxiliary Boiler Feed Pump	The Auxiliary Boiler Feed Pumps operate under Class III power and pump boiler feed water from the de-aerator storage tank to the boilers.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Lubricant Degradation	Lubrication Overhaul/Refurbishment Predictive Maintenance - Vibration Monitoring Predictive Maintenance - Thermography SRST - Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011155	5,6,7,8	43230 43230	43230 Auxiliary Boiler Feed Pump Motor-4kV	The Auxiliary Boiler Feed Pump is fed from the class III power supply and is used for supplying Boiler feedwater upon Class IV power failure conditions and for cooldown periods.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011156	5,6,7,8	43230 43230	43230 Boiler Feed Pump Recirculation Control Valves	These valves are used for recirculation from each main boiler feed pump back to the deaerator storage tank to ensure safe minimum flow for the MBFP.	Poor	Fatigue / Mechanical Fatigue Corrosion / General Corrosion		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u>

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		1. Complete outstanding WOs to replace limit switches with a more robust model. 2. Complete outstanding AR 28175792 assignments to resolve issues regarding valve stem/plug separation. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011157	5,6,7,8	43230 43230	43230 Boiler Inlet Feedwater Isolation and Auxiliary BFP Discharge MOVs	The function of 5/6/7/8-43230-MV112 is to isolate the ABFP when reverse rotation is detected at the pump. The function of 5/6/7/8-43230-MV112B is used to equalize the pressure across MV112 when it is being opened. The function of 5/6/7/8-43230-MV190/MV191/MV192/MV193//MV194/MV195/MV196/MV197/MV198/MV199/MV200/MV201 is to regulate the flow of water entering the boilers.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Fatigue / Mechanical Fatigue	Lubrication Inspection - Visual SRST - Functional Test SRST - Stroke Test	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete outstanding work requests, and resolve the issue of solenoid valve failures. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011158	5,6,7,8	43230 43230	43230 Main Boiler Feed Pumps	The Main Boiler Feed Pumps (5/6/7/8-43230-P6/P7/P8) take suction from the deaerator feedwater heater storage tank and discharge into a header eventually entering the reactor boilers. The MBF Auxiliary lube Oil pump (43230-P3001, P3005, P3009) supplies lubricating oil during start-up. The Shaft-Driven Lubrication Pump (43230-P3002, P3006, P3010) provides oil to maintain the system.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Lubricant Contamination		Incremental recommendations for Plant EOL (2020): Ensure adequate spares are available for the replacement of the MBFP auxiliary oil pumps, shaft-driven lubrication pumps. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011185	5,6,7,8	43230 43230	43230 Main Boiler Feed Pump Discharge Motorized Valves	The main BFP Discharge Motorized Valves are designed to be opened when the MBFP is running or on stand-by.	Satisfactory	Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Lubrication Inspection - Internal	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete work orders to overhaul actuators and repair leaks/packing issues. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011186	5,6,7,8	43230 43230	43230 Boiler Feed HP Heater Isolation MV's	The HP Feed Heater Isolators are used to isolate a Feed Heater bank for maintenance or in the event of an HP Feed Heater tube leak.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Thermal Aging Corrosion / Flow Assisted Corrosion (Erosion-Corrosion)	Lubrication Inspection - Internal SRST - Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Corrosion / Fouling (accumulation of deposits)		
011187	5,6,7,8	43230 43230	43230 Boiler Level Control CV Isolating MV's (large)	The Large Boiler Level Control Valve (BLCV) isolators are used to isolate the BLCVs during BLCV on power changeover (to allow functional testing) or for isolation of the non-operating BLCV following change over.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Thermal Aging Corrosion / General Corrosion	Calibration Inspection - Internal Inspection - Visual Lubrication SRST - Functional Test SRST - Stroke Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011188	5,6,7,8	43230 43230	43230 Boiler Level Control CV Isolating MV's (small)	These motorized valves are downstream of the Boiler Level Control Valve (BLCV) and provide isolation for a given boiler quadrant.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / Flow Assisted Corrosion (Erosion-Corrosion)	Lubrication Inspection - Internal Overhaul/Refurbishment Valve Diagnostics	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011189	5,6,7,8	43230 43230	43230 Auxiliary Boiler Feed Pump Minimum Flow CV200	The Minimum Flow Control Valve, 5/6/7/8-43230-CV200 function is to provide protection to the Auxiliary Boiler Feed Pumps to ensure the design flow rate (or higher) is available.	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Based on CG history, perform a one-time replacement of the packing for all valves in the CG.
011190	5,6,7,8	43230 43230	43230 ABFP Discharge CVs	CV 206 protects the ABFP by providing an artificial back pressure during the times when the back pressure does not meet the pump requirements.	Good	Corrosion / Galvanic Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Overhaul/Refurbishment SRST - Stroke Test Valve Diagnostics	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011191	5,6,7,8	43230	43230 Boiler LCVs (large)	These Control Valves are used to control flow of feedwater for a given boiler quadrant.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete outstanding work orders to address maintenance issues (including repacking valves to resolve leaks, resolving sticky valves, overhaul of valves, etc.). Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024):

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Fatigue / Mechanical Fatigue		No additional practices are recommended to reach CO EOL (2024).
011192	5,6,7,8	43230	43230 Boiler Level Control Valves (small)	These CVs are used for feed water level control for a given quadrant of boilers.	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete outstanding work orders to address leaking and sticky valves, and ensure there are adequate spare parts to support Plant EOL (2020). Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011193	5,6,7,8	43230	43230 Main Boiler Feed Pump Discharge Non-Return Valves	These non-return valves prevent backflow in the discharge lines when the main Boiler Feed Pumps are not in service.	Satisfactory	Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011194	5,6,7,8	43230 43230	43230 Auxiliary Boiler Feed Pump Discharge Non-Return Valves	The function of the NV is to prevent backflow through the Auxiliary Boiler Feed Pumps when the pump is not in service.	Good	Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Overhaul/Refurbishment Inspection - Internal SRST - Stroke Test Valve Diagnostics	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011195	5,6,7,8	43230	43230 Boiler Feed Boiler Quadrant Inlet Non-Return Valves	In case of a boiler feed line rupture outside of the reactor building, these non-return valves function to maintain boiler inventory.	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011196	5,6,7,8	43230	43230 Boiler Feed MBFP Oil Pump Suction Non-Return Valves	These non-return valves ensure the suction line to shaft driven oil pump remains filled with oil when the boiler feed pump is not in service	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011197	5,6,7,8	43230 64323	64323 Pressure Controller PC206	Control Auxiliary Boiler Feed Pump (ABFP) outlet pressure.	Good	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011201	5,6,7,8	43230	43230 Boiler Feed Pump Concrete Pads	The primary function of these concrete pads are to support the Main Boiler Feed pumps and Auxiliary Boiler feed pump along with their associated motors.	Good	Mechanical and Thermal Degradation / Cracking Due to Cyclic Loading Mechanical and Thermal Degradation / Cracking Due to Vibration		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete initiated WO's for visual inspection of all Main and Auxiliary Boiler Feed Pump Concrete foundations. Carry out mitigating/remedial measures required for the component to reach Plant EOL (2020). <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> A routine condition monitoring of the boiler feed pump/motor concrete pads (above 254' level) including the sole plate, grout and anchor bolts needs to be performed

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
								each time the pump/motor is removed for maintenance.
011238	5,6,7,8	43230	43230 Boiler Feed System-Cable - 4.16 kV, 600V, 125V, 250V - Power Cables	Cables feed power from source to load.	Good	Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Inspection - Condition Monitoring Program	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011538	5,6,7,8	43230 43230	43230 Main Boiler Feed Pump Motors-4kV	Main Boiler Feed Pump (MBFP) motors drive boiler feed pumps	Satisfactory	Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Corrosion / Microbiological Influenced Corrosion Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete the refurbishment of all the MBFP motors. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).

System 0404 Boiler Steam and Water Systems

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008159	1,4	36000 63615	ALARM, PRESSURE , Cat 1/2	Annunciations of Boiler Emergency Cooling abnormal conditions (Boiler secondary side pressure / 36710 MV2 fail open)	Good	Mechanical and Thermal Degradation / Electronic Aging	Calibration SRST - Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008161	1,4	36000 63615	AMPLIFIER, SERVO FOR SPEEDER GEAR , Cat 1/2	63615-P1-AF1 controls turbine speed when Unit synchronized.	Good	Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008162	1,4	36000 36110	BOOSTER, Cat 1/2	The volume boosters are used to increase the stroking speed of the valve.	Very Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008164	1,4	36000 63615	CONTROLLER , HAND, Cat 1/2	Manual control of steam reject valves (SRV)	Good	Mechanical and Thermal Degradation / Electronic Aging	Calibration SRST - Functional Test	Incremental recommendations for Plant EOL (2020): Implement new PM for inspection/overhaul to be scheduled every 104 weeks. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
008165	1,4	36000 63615	CONTROLLER , SPEED, Cat 1/2	Control Turbine Speed when unit synchronized.	Good	Mechanical and Thermal Degradation / Electronic Aging	Calibration SRST - Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008168	1,4	36000 36110	EXPANSION JOINT, Cat 1/2	The expansion joints accommodate for thermal expansion within the Boiler Steam and Water System.	Good	Fatigue / Thermal Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008170	1,4	36000 36110	FILTER, Cat 1/2	Provide clean instrument air to Steam Reject Valves (SRV) and control components.	Very Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008172	1,4	36000 36110	GAUGE, Cat 1/2	The component pressure gauge used for the Service Air to MVXX-AX1 (Transducer I/P Converter).	Good	Corrosion / General Corrosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Incremental recommendations for Plant EOL (2020): Perform a one-time replacement of the pressure gauges. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
008174	1,4	36000 63615	INDICATOR, PRESSURE, Cat 1/2	Provide indication of steam pressure at Line 3611L6 west	Good	Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008179	1,4	36000 36110	POSITIONER, Cat 1/2	NC's (valve positioners) accurately position SRV travel based on controlled signal.	Very Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008180	1,4	36000 63611	POWER SUPPLY, Cat 1/2	Provide loop power for boiler steam flow measurement loop (1-63611-F1-FT1 and 1-63611-F1-FM1)	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008181	1,4	36000 63611	PROCESSOR, Cat 1/2	1-63611-F1-FM1 provide a square root steam flow signal for boiler level control.	Very Good	Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008183	1,4	36000 36110	REGULATOR, Cat 1/2	The design intent of PRV1 is to provide a constant control pressure to Transducer I/P Converter AX1. The design intent of PRV2 is to provide a constant control pressure to signal booster B4. The design intent of PRV3 is to provide a control pressure to the pilot valve actuator, to keep the pilot valve closed.	Very Good	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008185	1,4	36000 36110	SWITCH, Cat 1/2	Limit switches, NS's, provide indication of SRV position.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008188	1,4	36000 36110	TRANSMITTER, Cat 1/2	AX's convert electrical signals from electronic controllers to pneumatic signals for SRV control instrumentation.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		<u>Incremental recommendations for Plant EOL (2020):</u> Replace AXs with high temperature model. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
008190	1,4	36000 63615	TRANSMITTER, PRESSURE, Cat 1/2	Measure and transmit channelized boiler steam pressure.	Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008191	1,4	36000 63611	TRANSMITTER, FLOW Cat 1/2	Flow transmitters measure and transmit steam flow rate for boiler control and indication	Good	Mechanical and Thermal Degradation / Electronic Aging		<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Proactively replace transmitters.
008193	1,4	36000 36110	VALVE, AIR RELEASE, Cat 1/2	The quick release valve provides capacity for the quick exhausting of actuators to reduce cycling times.	Very Good	Mechanical and Thermal Degradation - Deterioration of Material / 62		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue		
008194	1,4	36000 36110	VALVE, CONTROL, Cat 1/2	The valve is a three way switching valve used to assist in the proper operation of the Stream Reject Valve.	Very Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008197	1,4	36000 36110	VALVE, MOTORIZED/ MOTOR OPERATE, Cat 1/2	These motorized valves are used to control/stop flow between the main steam header and the steam inlets	Satisfactory	Corrosion / General Corrosion Mechanical and Thermal Degradation / Lubricant Degradation Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue	Inspection - Internal Inspection - Visual	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete outstanding WO 04849728 for 1-36110-MV5B, and WO 04845032 for 1-36110-MV6A <u>Incremental recommendations for Plant EOL (2020):</u> Complete a one-time actuator overhaul and diagnostic testing for all MVs in this CG. <u>Incremental recommendations for CO EOL (2024):</u> 1. Implement new PMs for 1/4-36110-MV1, MV2, MV3A, MV4A, MV5A, and MV6A for actuator inspection, diagnostics and functional test on a 4 year frequency. 2. Implement PMs for 1/4-36110-MV3B, MV4B, MV5B, and MV6B for actuator lubrication on a 3 year frequency.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008199	1,4	36000 36110	VALVE, CHECK/NONRETURN/BACKFL, Cat 1/2, MS-01	These are non-return valves in the main steam supply lines to the deaerator.	Good	Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Corrosion / General Corrosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Surveillance	Incremental recommendations for Plant EOL (2020): Perform one-time inspections of valves 1/4-36110-NV7/NV8 to assess valve condition. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
008201	1,4	36000 36110	VALVE, CHECK/NONRETURN/BACKFL, Cat 1/2, MS-02	These components are non-return valves in the steam supply bypass lines to the deaerator.	Good	Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008202	1,4	36000 36110	VALVE, PNEUMATIC/PNEUMATIC AC, Cat 1/2	These are pneumatically operated valves that provide isolation for main stream supply to screenhouse equipment and steam release to atmosphere.	Satisfactory	Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / Flow Assisted Corrosion (Erosion-Corrosion)	Calibration Inspection - Visual Overhaul/Refurbishment SRST - Functional Test SRST - Stroke Test Valve Diagnostics	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: 1. Complete WO 4850648 for 1-36110-MV38 overhaul and WO 4727738 for 1-36110-MV40 overhaul. 2. Complete WO 4823379 for 1-36110-MV36 packing replacement and WO 4727739 for 1-36110-MV41 packing adjustment. 3. Complete WO 04813836 (4-36110-MV38) and WO 04908520 (4-36110-MV40) for air leaks. 4. Complete WO 02704201 (4-36110-MV38) and WO 02704199 (4-36110-MV39) to replace rubber hoses. Incremental recommendations for Plant EOL (2020):

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
								No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
008203	1,4	36000 36110	VALVE, PRESSURE REGULATING, Cat 1/2	PRVs provide regulated instrument air pressure to SRV and its control devices.	Very Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008204	1,4	36000 36140	VALVE, PRESSURE RELIEF, Cat 1/2	These RVs provide over pressure protection to the Boiler Steam and Water Systems	Very Good	Fatigue / Mechanical Fatigue Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008205	1,4	36000 36110	SOLENOID/SOLENOID OPERATED VAL, Cat 1/2	SV's provide a signal path for operation of SRVs during normal operation and testing.	Very Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008206	1,4	36000 36110	SOLENOID/SOLENOID OPERATED VAL, Cat 1/2	SV1 is used to ensure SRV pilot valve is locked open when main SRV valve plug lifts off its seat.	Very Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		
008207	1,4	36000 63615	RELAY, Cat 1/2	Detects turbine runback due to reactor trip and illuminates "Turbine Runback Interlock" lamp at SDSE MCR panel.	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008208	4	36000 63610	Fuse	63615-FU1 provides protection for turbine speed control electronic circuits.	Satisfactory	Fatigue / Thermal Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008209	4	36000 63615	BREAKER, CIRCUIT	63615-EF1-CB1, when closed, applies power to speeder gear motor.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear		<u>Incremental recommendations for Plant EOL (2020):</u> Add inspection of CB1 in PM for speeder gear. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
008210	1,4	36000 63615	SPEEDER GEAR SERVOMOTO R	Accurately adjust governor valve position to regulate turbine speed.	Very Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Component Replacement Inspection - Internal	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008213	4	36000 63615	CONTACTOR	Relay 4-63615-P1-CN1 is energised when operated via 63615-P1-HS4 to apply power to Speeder Gear Amplifier and Motor.	Satisfactory	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Inspection - Visual Predictive Maintenance - Thermography	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008214	1,4	36000 63615	SWITCH, HAND FOR REACTOR	HS5 is used in Turbine Run-Up and Control Logic circuits.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
			SETBACK CANCEL					
008215	4	36000 63615	SWITCH, PRESSURE	P1-PS1 supplies DC power to turbine speed controller	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Incremental recommendations for Plant EOL (2020): Develop new PM to check DC voltage output ripple and to replace power supply at periodic intervals. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
008216	4	36000 63615	TRANSFORMER, ISOLATION	Transformer P1-T1 isolates field 120 Vac from amplifier circuits to reduce noise in the Amplifier.	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Incremental recommendations for Plant EOL (2020): Add calibration of 4-63615-P1-PM1 to PM 89235. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011133	5,6,7,8	36000 36110	36110 Large Boiler Steam Reject Valves	The Large Steam Reject Valves (5/6/7/8-36110-MV37/MV38/MV39/MV40/MV41/MV43/MV44/MV45/MV46/MV47) are used to provide crash cooling of the Heat Transport System following a LOCA.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue Corrosion / General Corrosion	Overhaul/Refurbishment SRST - Stroke Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011221	5,6,7,8	36000 63615	63615 Boiler Steam and Water Systems-CONTROLLER-HAND-CAT 1&2	5-63615-P1-WX5 provides DC power to HC5. HC5 is used to manually control operation of SRV's during testing or loss of auto control.	Good	Thermal Aging / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear	SRST - Functional Test SRST - Stroke Test	Incremental recommendations for Plant EOL (2020): Obtain a replacement for 5-63615-P1-HC1 and WX5 and perform calibration every 104 weeks. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Fatigue / Environmentally-Assisted Fatigue Mechanical and Thermal Degradation / Electronic Aging		
011222	5,6,7,8	36000	36110, 45220, 63611 Boiler Steam and Water Systems - Cables 4.16 kV, 600V, 125V, 250V, 120V	Cables transmit electrical power from power source to load.	Good	Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Radiation Embrittlement Fatigue / Mechanical Fatigue	Inspection - Condition Monitoring Program	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011223	5,6,7,8	36000 36110	36110 Boiler Stop and Bypass Valves (MOVs)	5/6/7/8-36110- MV3A, MV4A, MV5A & MV6A are the main Boiler Stop Valves are used to isolate the turbine from main steam when the turbine is not operating. 5/6/7/8-36110-MV3B/MV4B/MV5B/MV6B are bypass valves to balance the pressure between the main steam piping and the turbine steam chest when opening the main stop valves	Good	Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Lubricant Degradation	Inspection - Internal Inspection - Visual Lubrication	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011224	5,6,7,8	36000 36110	36110 Boiler Steam and Water Systems- Valves - NV- Non-Return - CAT1&2	The check valve is required to prevent reverse flow of instrument air to the instrument air station and not leak externally in the event of a loss of instrument air.	Satisfactory	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / Flow Assisted Corrosion (Erosion-Corrosion)	Overhaul/Refurbishment Inspection - Visual Inspection - Internal SRST - Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		
011225	5,6,7,8	36000 36110	36110 Boiler Steam and Water Systems-Piping-Expansion Joints-CAT3&4	The expansion joints accommodate for thermal expansion of the boiler steam lines.	Good	Fatigue / Mechanical Fatigue Corrosion / Flow Assisted Corrosion (Erosion-Corrosion)		<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Perform a one-time inspection of one EJ to confirm the CG will reach CO EOL (2024).
011349	5,6,7,8	36000 36110	36110 Boiler Steam and Water Systems-Valves - NV-Non-Return - CAT1&2	The check valves protect against a back flow of water into the main steam headers under conditions where the deaerator becomes flooded.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Surveillance	<u>Incremental recommendations for Plant EOL (2020):</u> Inspect 7-36110-NV8 as part of a sampling strategy to determine required maintenance on 5/6/7/8-36110-NV7/NV8. Also, inspect 7-36110-NV63 to establish a baseline condition. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011350	5,6,7,8	36000 36110	Boiler Steam and Water Systems-Valves - MOV-Standard-CAT1&2	The motorized valves provide isolation of main stream supply to the condensate system deaerator.	Good	Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Obsolescence / Immediate	Lubrication Inspection - Visual Inspection - Internal SRST - Stroke Test	<u>Incremental recommendations for Plant EOL (2020):</u> 1. Inspect one Group 1 valve (MV1/MV2) to determine if future maintenance is needed and overhaul actuators. 2. Replace Group 2 valves 5-36110-MV60, 6-36110-MV60 and 7-36110-MV62 with new Cat ID 609880. 3. Procure spares for Group 1 valves and four valves along with Rotork actuator for Group 2 valves. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Obsolescence Concern		

System 0406 Calandria Vault, Vault Structure Cooling Shield Tank

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011249	5,6,7,8	31200	Calandria Vault, Vault Structure Cooling, Shield Tank - Structural Concrete - Concrete Walls and Slabs	The main purpose of the vault structure is to provide shielding against the radiation from an operating unit and to provide structural support to Calandria in its operating and non-operating modes. The Calandria Vault and its demineralized light water provides operational and shutdown shielding for the immediate surrounding areas. The water also provides cooling for the Calandria assembly and the vault concrete.	Good	Corrosion / Corrosion of Embedded Steel Radiation Induced Degradation / Radiation Embrittlement Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Leaching Calcium Hydroxide	Surveillance	<u>Incremental recommendations for Plant EOL (2020):</u> Conduct one-time visual inspection of accessible areas of Calandria Vault structural concrete for Unit(s) 5-8 to inspect for potential damage and perform any required mitigating/remedial actions. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011250	5,6,7,8	31200	21300 Calandria Vault/Vault Structure Cooling/Shield Tank-Liners - Steel Liners	Steel liner is a crucial part of the calandria shielding system and provides a leak-tight seal for containment of the vault demineralised light water.	Good	Corrosion / General Corrosion	Surveillance	<u>Incremental recommendations for Plant EOL (2020):</u> Conduct one-time inspection for Unit 8, and for Units 5 to 7 only if evidence of component failure or degradation is discovered. For Unit 8 the condition of epoxy patch as a temporary solution to repair a weld defect is unknown and it is recommended to initiate a permanent repair (i.e. welding). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011251	5,6,7,8	31200	21300 Calandria Vault/Vault	With the exception of the large diameter opening in each end shield wall and the	Good	Corrosion / General Corrosion	Surveillance	<u>Incremental recommendations for Plant EOL (2020):</u>

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
			Structure Cooling/Shield Tank-Penetrations - Steel Sleeves Surrounding Concrete Vault Openings	rectangular opening in the roof slab, the only other openings and penetrations in the vault wall are to provide for moderator and other systems piping. These penetrations and their associated embedded parts for piping and other systems are designed to contain the shield water within the confines of the Calandria Vault.		Radiation Induced Degradation / Radiation Embrittlement		Perform one-time visual inspection of accessible areas and local leakage testing to confirm components' suitability. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011252	5,6,7,8	31200	21300 Calandria Vault/Vault Structure Cooling/Shield Tank - Embedded Parts and Supports	Reactivity Mechanism Deck Supports and Connections: The Reactivity Mechanism Deck is part of the reactor vault assembly. It closes the top of the Calandria Vault, thus providing a boundary between the vault and the boiler room atmospheres. The deck is supported by the Calandria Vault and it seals the vault atmosphere by seal plates welded to both the lower plate and the vault liner. End Shield Manhole: A manhole is located at the top of each End Shield, passing through both the support shell and the End Shield shell. It was used to provide access during fabrication and ball filing and was permanently sealed by welded cover plates.	Good	Corrosion / Corrosion of Embedded Steel Radiation Induced Degradation / Radiation Embrittlement	Surveillance	<u>Incremental recommendations for Plant EOL (2020):</u> Conduct one-time visual inspection of accessible areas to confirm the suitability of the components. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011253	5,6,7,8	31200	21300 Calandria Vault/Vault Structure Cooling/Shield Tank - Seals & Sealants	Atmospheric SS Seals & Rubber Seals: The Atmospheric seals together with the silicone rubber seals at elevation 317'-6" (extending to elevation 324'-0"), separate the east and west F/M room atmospheres. The Elastomer Silicone rubber seals at elevation 274'-0" prevent liquid spills into the 25mm (1inch) and 76mm (3inch) gaps around the Calandria Vault. Integrity of the barrier/seal has a direct effect on environment inside the Calandria Vault. Inconel 600 Manhole Seal: The Inconel 600 Manhole expansion joint seal is necessary to allow free movement of the seal plate due to a temperature differential. Shear Key Joints: The Horizontal keys (elev. 312'-0") and vertical keys (elev. 312'-0") in north and south cross-walls are provided to resist forces due to earthquake and unbalanced header failure pressure on the east or west face of the vault.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion Mechanical and Thermal Degradation / Cracking Due to Expansion or Contraction	Surveillance	<u>Incremental recommendations for Plant EOL (2020):</u> 1. Complete one-time inspection of seals. 2. Replace the elastomeric seals if required. 3. Replace seals if they show advanced degradation. <u>Incremental recommendations for CO EOL (2024):</u> 1. Replace the elastomeric seals if required. 2. Replace seals if they show advanced degradation.

System 0408 Class 1 & 2 Electrical and Battery Room HVAC

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008260	4	73230 73230	AIR CONDITIONIN G UNIT, Cat 1/2	ACUs maintain the temperature in the Class I Electrical Equipment Rooms including Battery & Rectifier / Inverter Room.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion Fatigue / Mechanical Fatigue		No additional practices are recommended to reach CO EOL (2024).
008265	4	73230 73230	DAMPER, Cat 1/2	These components are electric actuated dampers used for control and isolation of ACU flow of outside air and fan discharge.	Satisfactory	Fatigue / Mechanical Fatigue Corrosion / General Corrosion	Inspection - Visual Functional Test	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete MDP actuator replacements via master NICR 104778. <u>Incremental recommendations for Plant EOL (2020):</u> Revise PMs 92533, 98078, 98079 and 98080 to add tasks for lubrication and inspection every 1 year on 4-73230-MDP2030/2031/2032/2033 (4 components). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
008270	4	73230 73230	FAN, Cat 1/2	This CG consist of supply fans for ACUs, 4-73230-ACU2008/2009/2010/2011	Good	Corrosion / General Corrosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008279	4	73230 67323	SWITCH, Cat 1/2	Generates a contact input signal when the monitored	Satisfactory	Corrosion / General Corrosion	Calibration	<u>Incremental recommendations for Plant EOL (2020):</u>

TETRA TECH

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				differential pressure across the ACU fan or filter drops below or exceeds the predefined set point.			Inspection - Visual Predictive Maintenance - Vibration Monitoring Predictive Maintenance - Thermography	Calibrations to be scheduled at a frequency adequate for each pressure switch. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011436	058,5, 6,7,8	73230	Class 1 & 2 Electrical and Battery Room HVAC Systems-Cable - 4.16 kV, 600V, 125V, 250V - Power Cables	Provide power and control to HVAC equipment.	Good	Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue	Inspection - Condition Monitoring Program	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011451	5,6,7,8	73230 73230	73230 Class 1 & 2 Electrical and Battery Room HVAC Systems - Glycol Pumps - CAT3&4	These pumps circulate glycol/water solution to avoid winter freeze through an air cooled heat exchanger located on the roof	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011452	5,6,7,8	73230 73230	73230 Class 1 & 2 Electrical and Battery Room HVAC Systems-HVAC-ACU's-CAT3&4	Function of these ACUs is to maintain the temperature in the Class I and II Electrical Equipment Rooms between approximately 22.2°C (72°F) and 26.7°C (80°F).	Poor	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging Obsolescence / Immediate		<u>Incremental recommendations for Plant EOL (2020):</u> Resolve obsolescence issues and schedule replacement of ACUs. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Obsolescence Concern		

System 0410 - Common Water Supply

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
08313	012, 034	71100 67111	GAUGE, Cat 1/2	These gauges are used throughout the Common Water Supply System to provide differential pressure, pressure and flow instrumentation (e.g. differential pressure across the travelling screens; discharge pressure and flow indication for pump operation etc.)	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete work order WO 04723427 to replace 034-67112-PG2006 as well as WO 01719247 to replace 034-67111-PG2009. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Initiate a new PM for periodic calibration or replacement of the gauges.
008317	012, 034	71100	MOTOR, Cat 1/2	Motors are used to drive associated pumps	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008322	012, 034	71100 71110 71120	PUMP, Cat 3/4	The pumps serve in two different applications, they are used to pump water to clean / wash the screen and the other application is to act as a sump-pump in the trash bin.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						/ Wear Mechanisms - Wear Fatigue / Mechanical Fatigue Corrosion / General Corrosion		
008323	034	71100	RELAY, Cat 1/2	This relay is part of the control logic for the automatic operation of the bar screen rake (034-71110-SCM7).	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete EC127600 to replace Screen Motor relays. Incremental recommendations for CO EOL (2024): Schedule inspection of the Screen Motor relays 2 years from completion of EC, if needed repair/replace.
008325	012, 034	71100	SCREEN, Cat 1/2	The screens filter out particulate in the Screenhouse. The traveling screens remove debris from intake water. The trash sump bar screens filter water from the trash sump.	Satisfactory	Corrosion / General Corrosion Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Corrosion / Fouling (accumulation of deposits)		Incremental recommendations for Plant EOL (2020): Complete one-time cleaning of screens to remove debris and zebra mussels. Incremental recommendations for CO EOL (2024): 1. Complete one-time inspection and overhaul of the screens. 2. Complete one-time inspection of Screenhouse concrete structures and equipment supports. 3. Re-initiate PMs for zebra mussel cleaning every 104 weeks.
008326	012, 034	71100 71110 71120	SCREEN, Cat 3/4	Bar screens remove debris from intake water.	Satisfactory	Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Corrosion / General Corrosion Corrosion / Fouling (accumulation of deposits)	Lubrication Clean and Inspect	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Repair failed Bar Screens 012-71110-SC11 and 034-71110-SC12 (WO 3249081). Incremental recommendations for Plant EOL (2020):

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
								No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
008327	012, 034	71100 71110	STRAINER, Cat 3/4	These strainers are used to strain screen backwash water.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Corrosion / Fouling (accumulation of deposits)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008328	012, 034	71100 67111	SWITCH, Cat 1/2	These pressure switches sense differential pressure across strainers/ bar screens to operate strainer/ bar screen logic and alarm abnormal conditions.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008335	012, 034	71100	VALVE, MANUAL/HAND OPERATED, Cat 1/2	These manually operated valves function in isolation applications of the common water system.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion		Incremental recommendations for Plant EOL (2020): Obtain adequate spare replacements to reach Plant EOL (2020). Incremental recommendations for Plant EOL (2024): Obtain adequate spare replacements to reach CO EOL (2024).

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CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008339	012	71100	VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2	This is a pneumatic valve used for draining a strainer (012-71110-STR1).	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion		Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Initiate a new PM for AM practices to ensure condition for CO EOL (2024).
008341	012	71100	SOLENOID/SOLENOID OPERATED VAL, Cat 1/2	SV operates strainer 012-71110-STR1 backwash drain valve on high strainer differential pressure.	Good	Mechanical and Thermal Degradation (Deterioration of material - wiring, seals, gaskets, O-rings etc.) / 101		Continue current practice. No additional practices are recommended to reach CO EOL (2024)
011247	056, 078	71100 71110 71120	71100, 71120 Common Water Supply - Screens and Conveyors	The function of the bar screens in the sediment suction system is to protect the pumps from debris. The bar screens remove trash from the intake water. The trash conveyors are installed indoors at the back of the bar screens for the continuous removal of the trash mostly in the form of weeds, fish and debris which are brought up by the cleaning rake. The conveyors discharge the trash into the trash bins. The travelling screens remove trash following the bar screens. The trash removal screens remove trash from water from the travelling screens.	Satisfactory	Corrosion / Fouling (accumulation of deposits) Fatigue / Environmentally-Assisted Fatigue Corrosion / General Corrosion Corrosion / Microbiological Influenced Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Lubrication Clean and Inspect	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Overhaul the trash conveyors, 056, 078-71120 TC1 (WO 2809130 and WO 2795171). Incremental recommendations for Plant EOL (2020): 1. Complete a one-time cleaning of the screens to remove debris and zebra mussels. 2. Resolve spare parts issues. Incremental recommendations for CO EOL (2024): 1. Complete one-time inspection of screenhouse concrete structures and equipment supports. 2. Initiate a new PM for zebra mussel cleaning for the bar screens.
011299	056, 078	71100 53300 54130	54130, 53300 Common Water Supply-Motor	MCCs supply 600Vac power and provide protection and switching for motor operated loads and	Satisfactory	Mechanical and Thermal Degradation / Self-Loosening	Component Replacement Inspection - Visual	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
			Control Centre (MCC) 600V	components in the Screenhouse.		General Corrosion / General Corrosion Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Inspection - Internal Predictive Maintenance - Electrical Testing Predictive Maintenance - Thermography	Complete structural inspection/maintenance activities on remaining CC2 MCCs (056-54130-MCC541, 078-54130-MCC741). Also, accelerate the implementation of all MCC cells replacement as per NK30-ESI-50000-00006. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011300	058,056,078	71100	67127, 71110, 71120, 71270 Common Water Supply - Cables - 4.16 kV, 600V, 125V, 250V	The cables are used to supply power (e.g. 5 kV, 600 V) and for control purposes (e.g. 600V, 300V)	Good	Fatigue / Thermal Fatigue Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Radiation Embrittlement	Inspection - Condition Monitoring Program	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011301	056, 078	71100 71110	71110 Bar Screen, Travelling Screen, Screenwash and Backwash Strainer Motors	The motors are used to operate various equipment such as pumps, strainers, and bar screens.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Thermal Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011302	056, 078	71100 67111 71110	67111, 71110 Common Water Supply-Valves - AOV-Standard-CAT1&2	Screen wash valves are used to supply screen wash water to their respective travelling screens to remove debris. The SV's provide instrument air to their respective MV's. (Example: 056-67111-SV310 for 71110-MV16)	Satisfactory	General Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation		Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Review PM frequency to determine if valve replacement should be more frequent, after completion of 078-71110-MV16 investigation under WO 04735450.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						/ Wear Mechanisms - Wear		
011303	056, 078	71100 71110	71100 Common Water Backwash Strainers	The strainers provide filtered water to clean debris from the travelling screens.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / Microbiological Influenced Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms – Erosion		Incremental recommendations for Plant EOL (2020): Update the Bill of Materials to include a Cat ID for the strainer motor and spare parts. Incremental recommendations for CO EOL (2024): Purchase one additional strainer.
011304	058, 078	71100	Common Water Supply-Mechanical	The carbon steel piping provides a flow path for wash water from the three screen wash pumps to the six travelling screens to remove debris from the screens in order to maintain adequate circulating water flow through the screens.	Good	General Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Corrosion / Microbiological Influenced Corrosion Fatigue / Mechanical Fatigue	Inspection - Condition Monitoring Program	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011531	056, 078	71100 71120	71120 Screenhouse Trash Removal Pump Motors, Bar Screen & Conveyor Motors	These motors operate Trash Removal pumps, Bar Screens and conveyors in the Screenhouse.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current Practices. No additional practices are recommended to reach CO EOL (2024).

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CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011532	058	71100 71270	71270 Sediment Suction Pump Motors (4kV), Bar Screen & Sluice Gate Actuator Motors	These motors are used in the Sediment Suction System which reduces the amount of sediment in the intake water of the P058 Screenhouse. These motors operate the sediment suction pumps at the Screenhouse intake, Bar Screen controls, and sluice gate motors.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

System 0412 – Containment

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008400	4,018	34200 21103 25230	ACTUATOR, Cat 1/2	These actuators are used to operate the personal airlock (AL1) door.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Creep and Stress Relaxation Fatigue / Mechanical Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008401	1,4,0 18	34200 21130 25230	AIRLOCK, Cat 1/2	Provide personnel and equipment access to the Reactor Buildings and Pressure Relief Duct, while maintaining the Containment Boundary.	Satisfactory	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008402	1,4	34200 21103	BLOW IN PANEL, Cat 1/2	Provide atmospheric separation between the RB and the PRD and facilitate isolation from the rest of Containment when required During an Accident: allow overpressure relief so that the structural integrity of containment is not affected.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Creep and Stress Relaxation Fatigue / Mechanical Fatigue		<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Replace a single existing rupture panel and test the old panel to the set point to ensure that the panel ruptures at 15 kPa (d). Depending on test performance, replace additional panels.
008403	1,4	34200 62111	BOX, Cat 1/2	Junction/splice boxes used for the cable splicing or	Good	Mechanical and Thermal Degradation / Deterioration of		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				termination of airlocks, hydrogen igniters, etc.		Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / Galvanic Corrosion		
008405	018	34200 25230	CABLE, Cat 1/2	Electrical conduit seal assembly used to seal cables routed to AL1 which are necessary for airlock operation.	Good	Mechanical and Thermal Degradation / Thermal Aging Radiation Induced Degradation / Radiation Depletion of Material Properties Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008406	1,4	34200 62111	CONTACTOR, Cat 1/2	These contactors used to switch and control electrical power feed to hydrogen igniters	Good	Environmental Degradation/Deterioration of Material / 51 Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008407	1,4, 018	34200 21130	CYLINDER, Cat 1/2	*** 1/4-21130-AL1-CYL1 to -AL6-CYL1: Emergency Air Cylinders for backup air supply to seals and/or emergency operation of an airlock door in the event of a failure of High Pressure Instrument Air. ***018-25230-AL1-PO11, -PO12, -PO21, -PO22:	Good	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Thermal Aging Radiation Induced Degradation / Radiation Embrittlement		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				Personnel Door Locks and Door Safety Gate Latches for PRD Airlock AL1.		Corrosion / General Corrosion		
008408	1,4, 018	34200 25230	DISC, Cat 1/2	Overpressure protection of airlock shell. Relieves to RB side.	Good	Fatigue / Environmentally-Assisted Fatigue Fatigue / Thermal Fatigue Radiation Induced Degradation / Radiation Embrittlement		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008414	1,4, 018	34200 21130 25230	FILTER, Cat 1/2	Particulate filters for Instrument Air supply to airlock controls.	Good	Corrosion / Fouling (accumulation of deposits) Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008415	1,4	34200 62111	FUSE, Cat 1/2	Fuses used to protect the electrical power circuit of the hydrogen igniters against over current.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008416	1,4, 018	34200 21130 25230	GAUGE, Cat 1/2	These Pressure Gauges are used in the airlock seal pressure logic.	Good	Fatigue / Environmentally-Assisted Fatigue Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008418	1,4	34200 62111	IGNITER, Cat 1/2	The Igniters are used to burn hydrogen in order to prevent its accumulation in the FM Vaults and/or the FM Service Rooms following a DBA.	Good	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Radiation Induced Degradation / Radiation Depletion of Material Properties	Component Replacement Inspection - Visual SRST - Functional Test SRST - Logic Test	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete proactive replacement of hydrogen igniters which failed recently in Units 1 and 4 (ref. EC 112707). Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
008419	1,4	34200 62111	IGNITER, Cat 3/4	The Igniters used to burn hydrogen in order to prevent its accumulation in the FM Vaults and/or the FM Service Rooms following a DBA.	Good	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Radiation Induced Degradation / Radiation Depletion of Material Properties		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete proactive replacement of hydrogen igniters which failed recently in Units 1 and 4 (ref. EC 112707). Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
008420	1,4	34200 62111	INDICATOR, Cat 1/2	These indicators are used to indicate the current and voltage of electric power to hydrogen igniters.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Thermal Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008423	1,4	34200 21130	LATCH, Cat 1/2	The safety gate latch prevents operation of the airlock door safety gate	Good	Fatigue / Thermal Fatigue		Continue current practices. No additional practices are recommended to reach CO

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				when the door is closed. (The purpose of the safety gate is to cause a closing door to reopen if an obstruction is encountered.)		Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Radiation Induced Degradation / Radiation Embrittlement		EOL (2024).
008424	1,4	34200 21130	LOCK, Cat 1/2	Airlock personnel door lock or equipment door lock. Provides a physical latch, which prevents opening of the door when engaged.	Good	Fatigue / Thermal Fatigue Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Radiation Induced Degradation / Radiation Embrittlement		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008428	4	34200 34330	RECOMBINATION UNIT	Passive Autocatalytic Recombiners (PARs) provide a backup to the hydrogen igniters. They convert airborne H2/D2 to water, to prevent accumulation of explosive levels of hydrogen within the Containment Envelope following a Design Basis Accident.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / Fouling (accumulation of deposits)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008429	1,4	34200 21130	RECTIFIER, Cat 1/2	These diodes are employed in the lamp test circuit to test the airlock (AL1 thru AL6) outer door status.	Good	Mechanical and Thermal Degradation / Thermal Aging Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach to CO EOL (2024).
008430	1,4, 018	34200 21130 25230	REGULATOR, Cat 1/2	Pressure regulators for instrument air supply and emergency backup air	Good	Fatigue / Thermal Fatigue	Calibration Component Replacement	<u>Incremental recommendations for Plant EOL (2020):</u>

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				supply to the airlock seals, actuators and door latches.		Fatigue / Mechanical Fatigue Radiation Induced Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Wear Mechanisms – Wear	Inspection - Visual SRST - Functional Test	No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Perform one-time inspection, and if degraded repair/replace.
008431	1,4, 018	34200 21130 25230 62111 62113	RELAY, Cat 1/2	These relays and timers are used to control or test airlock doors.	Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Thermal Fatigue		Continue current practices. No additional practices are recommended to reach to CO EOL (2024).
008433	1,4	34200 21130	RESISTOR, Cat 1/2	Resistors employed to test airlock (AL1 thru AL6) outer door status lamps	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Thermal Aging		Continue current practices. No additional practices are recommended to reach to CO EOL (2024).
008435	1,4, 018	34200 21130 25230	SEAL, Cat 1/2	Airlock pneumatic door seals. Provide airtight seal of airlock door perimeter when the door is closed (double seal on each door). Seals are provided for both personnel doors (1/4-21130-AL1 to AL6, 018-25230-AL1) and equipment doors (1/4-21130-AL1 to AL3).	Good	Fatigue / Thermal Fatigue Fatigue / Mechanical Fatigue Radiation Induced Degradation / Radiation Depletion of Material Properties		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Wear Mechanisms - Wear		
008436	1,4, 018	34200 21060 21103 21130 25230	SWITCH, Cat 1/2	These switches used to indicate either open or close status of airlock doors, containment venting bypass valves, and shielding doors.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Calibration SRST - Functional Test SRST - Logic Test Valve Diagnostics	Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Implement a new PM for the position switches of the FM shielding doors to reach CO EOL (2024).
008440	1,4, 018	34200 25230	TANK, Cat 1/2	Air reservoir for instrument air supply to airlock seals, actuators and latches. Original intent was to provide 2 hours of backup air upon failure of instrument air supply, and allow for two complete cycles of airlock use (i.e. four door open/close operations).	Good	Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008443	4	34200 21103	VALVE, AIR RELEASE, Cat 1/2	Provide quick exhaust for actuators of 4-21103-MV1/MV2 (Rupture Panel System Bypass Valves).	Good	Fatigue / Thermal Fatigue Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Radiation Induced Degradation / Radiation Embrittlement		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008444	1,4	34200 21130	VALVE, FLOW CONTROL, Cat 1/2	Speed of actuation of the door lock is by flow control valves FCVs in this CG. The FCVs are installed in tube runs to the actuator. Air is fed into the actuator by one	Good	Corrosion / Fouling (accumulation of deposits) Mechanical and Thermal Degradation		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				tube and exhausted via another.		/ Wear Mechanisms - Wear		
008445	1,4	34200	VALVE, ISOLATION, Cat 1/2	1-21130-AL2-V2006: Test valve on Emergency Backup Instrument Air supply to airlock door seals, actuators and latches. This valve is normally closed. 1/4-21130-AL5-V2013: Isolation valves on Instrument Air supply to airlock door seals. These valves are position assured (locked open).	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008447	1,4, 018	34200 21130 25230	VALVE, MANUAL/HAND OPERATED, Cat 1/2	Isolation valves on instrument air supply lines for airlock door seals, actuators and latches.	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008449	1,4	34200 21130	VALVE, MOTORIZED/MOTOR OPERATE, Cat 1/2	Personnel door actuators for Unit 1 and Unit 4 airlocks.	Satisfactory	Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008450	1,4, 018	34200 21130 25230	VALVE, CHECK/NONR	Check valves for instrument air supply to airlock door seals, actuators and latches	Good	Fatigue / Thermal Fatigue		Continue current practices. No additional practices are recommended to reach CO

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
			ETURN/BACK FL, Cat 1/2	(for Unit 1 and Unit 4 Airlocks).		Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Radiation Induced Degradation / Radiation Embrittlement		EOL (2024).
008452	1,4	34200 21103	VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2	Rupture Panel System (21103) bypass valves (MV1/MV2) for Units 1 and 4. The bypass valves open to relieve RB pressure to the PRD following a Seismic Event, or in the event of Design Basis Accidents (such as Fuel Handling Accidents and blinding break LOCAs) that do not cause rupture of the rupture panels. The valves may be opened either automatically on RB pressure set point, or by Operator action.	Satisfactory	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Creep and Stress Relaxation Fatigue / Mechanical Fatigue		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete 98469 to restore condition of valve actuator. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
008453	1,4	34200 21130	VALVE, PRESSURE REGULATING, Cat 1/2	Pressure regulators for instrument air supply and emergency backup air supply to the airlock seals, actuators and door latches.	Good	Fatigue / Thermal Fatigue Fatigue / Mechanical Fatigue Radiation Induced Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Calibration Component Replacement	<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Perform one-time inspection, and if degraded repair/replace.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008454	1,4, 018	34200 21130 25230	VALVE, PRESSURE RELIEF, Cat 1/2	1/4-21130-AL1 to AL6, RV 2015/RV2016: Provide overpressure protection for the air supply lines to the airlock seals 1/4-21130-AL1 to AL6 - RV3001, 018-25230-AL1-RV3072: Provide overpressure protection for the Emergency Backup Air supply for the U1/U4 airlocks and PRD Airlock AL1. 1/4-21130-AL1 to AL6 - RV3000, 018-25230-AL1-RV3000: Provide overpressure protection for the Airlock Shell for the U1/U4 airlocks and PRD Airlock AL1 1-21130-AL3-RV2095: Provide overpressure protection for the air supply to the RB personnel door actuator for 1-21130-AL3.	Good	Corrosion / General Corrosion Corrosion / Fouling (accumulation of deposits) Fatigue / Mechanical Fatigue	Component Replacement SRST - Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008455	1,4, 018	34200 21103 21130 25230	SOLENOID/SOLENOID OPERATED VAL, Cat 1/2	Solenoid valves used to open or close parent equipment (e.g., MVs, Airlocks, etc.).	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Creep and Stress Relaxation		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011180	5,6,7,8	34200 73110	73110 RB Cooling - Air Recirculation Fans	Cool the remote areas of the concrete structures in the Reactor Building that would otherwise reach	Satisfactory	Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				unacceptably high temperatures (due to lack of air circulation). 5/6/7/8-73110-F501A, F501B, F502A, F502B (Feeder Cabinet Walkway Concrete Cooling Fans): Supply air to Feeder Cabinet walkway, to provide cooling to boiler support concrete beams, channel temperature monitoring RTDs and power cables (two fans in each vault). 5/6/7/8-73110-F505, F506, F507, F508 (Calandria Concrete Face Cooling Fans): Supply air to cool the calandria concrete face. Fans F505 & F506 cool the East face and F507 & F508 cool the West face. The fans are seismically qualified to DBE Category "B". The original fans are also Environmentally Qualified (EQ) for operation under LOCA conditions. 5/6/7/8-73110-F509 (Moderator Room Pipe Chase Concrete Dome Cooling Fan): Cooling the Moderator Room pipe chase dome concrete.		Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		
011181	5,6,7,8	34200 73110	73110 RB Cooling - Boiler Room ACUs	5/6/7/8-73110-ACU1 to - ACU6 are boiler room ACUs whose function is to limit the ambient temperature in the Boiler Room between 36°C (97°F) and 60°C (140°F). These ACUs must operate under normal conditions as well as accident conditions.	Satisfactory	Corrosion / Stress Corrosion Cracking - Carbon and low alloy steels Corrosion / Fouling (accumulation of deposits) Corrosion / General Corrosion	Inspection - Internal Inspection - Visual Lubrication Predictive Maintenance - Vibration Monitoring	Incremental recommendations for Plant EOL (2020): Confirm the integrity of the condensate drain lines (leak-tight and not plugged) for all ACUs. Incremental recommendation for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		
011182	5,6,7,8	34200 73110	73110 RB Cooling-Fuelling Machine Vault ACUs	5/6/7/8-73110-ACU7 to -ACU14 (F/M Vault ACUs): Provide general cooling of the air in the F/M Vaults, to maintain the vault temperature less than 40°C (105°F). These ACUs are required to function under normal conditions as well as accident conditions.	Satisfactory	Corrosion / Stress Corrosion Cracking - Austenitic Stainless Steels Corrosion / Fouling (accumulation of deposits) Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of	Component Replacement Inspection - Internal Lubrication Predictive Maintenance - Vibration Monitoring	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Follow-up OEM and parallel company investigation of premature coil failures. Incremental recommendations for Plant EOL (2020): Perform one-time replacement of coils which have not been replaced since 2009. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Material (Wiring, Seals, Gaskets, O-rings etc.)		
011184	5,6,7,8	34200 73110	73110 RB Cooling - SAA, BR Drier Enclosure & Auxiliary Rooms ACUs	Distribute cooled air and provide general cooling for various accessible areas in the Reactor Building.	Satisfactory	Corrosion / Pitting Corrosion Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Fatigue / Mechanical Fatigue Corrosion / Fouling (accumulation of deposits)		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete coil replacement program. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Perform one-time replacements of all coils which have not been replaced since 2008.
011213	5,6,7,8 ,018	34200 34230 63423	34200, 73110 Containment-TRANSMITTER -ANALYTICAL-CAT 1&2	Temperature transmitters used to, for example, transmit the boiler room temperature and generate an annunciation in the MCR when it exceeds a set point.	Satisfactory	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Thermal Aging		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete outstanding WOs to replace moisture transmitters (WO# 4787973 through 4787976). Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011241	5,6,7,8	34200 21103	21103 Pressure Relief Panel Bypass and Drop Panel Isolation Valves	5/6/7/8-21103-MV1/MV2 (Pressure Relief Panel Bypass Valves): Relieve RB pressure to the PRD in the event of a small LOCA	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Overhaul/Refurbishment SRST - Functional Test SRST - Stroke Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				in which RB pressure does not increase sufficiently to activate the drop panels. Also required to provide a flowpath from the RB to the PRD when operating the Filtered Air Discharge System, in the event of a failed Fuelling Machine following a DBE. 5/6/7/8-21103-MV3/MV4 (Drop Panel Isolation Valves): Provided for non-seismic, small LOCA accidents in which the energy released is insufficient to open blow out panels. Normally open, these valves are closed remotely (by the operator) on non-accident Units, following an accident to prevent instrument air leakage from non-accident Units entering Containment via the drop panel flowpath. Also closed during RB leak testing to prevent pressurizing the drop panels, and closed to create a test space for drop panel setpoint testing.		Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Creep and Stress Relaxation Fatigue / Mechanical Fatigue		
011242	018	34200 34220 34230	34230 Filtered Air Discharge System (FADS) MOVs	The Filtered Air Discharge System provides a filtered and monitored air release path to the environment from the Vacuum Building following an accident. The motor operated valves are closed/opened as necessary to establish the operational flowpath (depending on the FADS mode of operation). Valves MV101 and MV102 are Containment Boundary	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue	Inspection - Visual Overhaul/Refurbishment SRST - Functional Test SRST - Logic Test SRST - Stroke Test Valve Diagnostics	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete the actions listed below: 1. Replace parts of 018-34220-MV12/MV13/M14 (Vacuum Pump Suction Isolation Valves) and 018-34220-MV201/MV203 per EC120350 (WOs 1887012/ 2886975/ 2887049/ 2887432/ 2887431). 2. Replace 018-34230-MV119/MV122 (Vent to U1 Stack Isolation Valves) per EC108218 and EC108257.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				valves.				3. Replace 018-34230-MV124/MV125 (Stack Discharge Isolation Valves) per WR 1663427 and 1663428. 4. Replace 018-34230-MV112 (F102 Discharge Isolation Valve) if required based on results from WO4939713. 5. Complete investigation of MV112 failure, and initiate actions as required to address any AM related findings for the other affected valves. 6. Procure spares/parts to support the above activities, as well as potential valve replacements resulting from inspection PMs. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011257	5,6,7,8,018	34200 21103 21130 25230	25230, 21130 Containment-Valves - Manual - CAT 1&2 And Not Safety Related	These valves function in various isolation applications in the airlock door and seal controls (e.g. isolation, interlock control, control of flow/pressure release, bleed, drain, bypass).	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion	Component Replacement Inspection - Visual SRST - Functional Test SRST - Logic Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011262	018	34200 63421 63423	21100 Containment-ALARM-ALARM UNIT-CAT 1&2	The alarm units 018-63423-P101-PA1 to PA6 are part of the FADS inlet pressure control loops. When FADS inlet pressure reaches the setpoint, the filter cooling air inlet valves are closed and the vacuum pump is stopped to prevent release to atmosphere.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging	Calibration Component Replacement SRST - Functional Test	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Procure a suitable replacement for the FADS pressure alarm units per EC 129463. <u>Incremental recommendations for Plant EOL (2020):</u>

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
								No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011264	5,6,7,8	34200 63420	63420 Containment-ANALYZER-RADIATION-CAT 1&2	The RB ventilation exhaust monitor provides continuous, real-time monitoring of gamma radiation at the Reactor Building ventilation exhaust line.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging	Calibration SRST - Logic Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011277	5,6,7,8	34200 73110	73110 RB Cooling Dampers	Control Dampers 73110-MDP1/2/3/4/5/6 act as discharge dampers for their respective ACU's (73110-ACU1 to ACU6) for RB cooling. Butterfly Valves 73110-MDP305/306/307/308 act as fan discharge dampers for the Calandria Concrete Face Cooling Fans (73110-F505 to F508).	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Radiation Induced Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Obsolescence / Immediate Obsolescence Concern	Surveillance	<u>Incremental recommendations for Plant EOL (2020):</u> 1. Perform a one-time replacement of all dampers. 2. Address obsolescence issues with CatID#151434. 3. Perform a review to determine the availability of spares and stock for at least one complete overhaul of all dampers. <u>Incremental recommendations for CO EOL (2024):</u> Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011333	5,6,7,8 ,018	34200 21060 21103 21130 25230 67311 73110	21000, 25000, 67000, 73000 Containment-Valves - SV-Solenoid Valves-CAT1&2	Solenoid valves used to open or close parent equipment (e.g., doors, dampers, MVs).	Satisfactory	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Component Replacement Overhaul/Refurbishment Inspection - Visual SRST - Functional Test SRST - Logic Test	<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Perform one-time inspection for all the Solenoid Valves that currently have no PMs (e.g. 5-21060-D3–SV1), replace/repair if needed.

TETRA TECH

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		
011334	5,6,7,8 ,018,0	34200 21060 21103 21130 25230 34220 63420 63421	21130, 25230 Containment-Valves - Airlock PRV's	Airlocks (21130, 25230): The PRVs regulate the pressure of the air supply to the airlock seals, actuators and latches, which allows operation of the airlock components, ensuring the doors will open/close and seals will inflate/deflate as required. Shielding Doors (21060): The PRVs regulate the pressure of the instrument air supply to the door air motors and inner/outer seals for the Shielding Doors within the RB (i.e. 21060-D23/D24, D3, D35/D36). PRP System (21103): The PRVs regulate the pressure of the air supply to Bypass Valves 21103-MV1/MV2 and Drop Panel Isolation Valves 21103-MV3/MV4. VB Vacuum System (34220): The PRVs reduce the pressure of instrument air to respective MV actuators. These MVs control opening/isolation of vacuum to the header that controls the IPRVs and allows lift testing of the PRVs in the Pressure Relief System (34210). IPRV System (63421): These PRVs regulate the pressure of instrument air to	Good	General Corrosion / General Corrosion Wear Mechanisms - Wear / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue	Inspection - Visual SRST - Functional Test SRST - Logic Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024)

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				control the Instrumented Pressure Relief Valves in the IPRV system. Reactor Building Pressure Monitoring System (63420): The PRVs regulate the pressure of the instrument air supply to pressure monitoring loop P2, which monitors the wide range pressure of the RB following an accident.				
011335	5,6,7,8,018	34200 21103 21130 25230 63421	21103, 21130, 25230 Containment-Valves - NV-Non-Return - CAT1&2	Airlocks (21130, 25230): The check valves maintain pressure for air supplies to airlock door seals, actuators and latches, PRP System (21103): The USI 21103 check valves maintain pressure of backup air supply to PRP Bypass Valves 21103-MV1/MV2. They are located upstream of the air receiver tank which ensures sufficient stored air for two valve cycles in case of a normal air supply failure. IPRV System (63421): 018-63421-P1-NV1/2/3 prevent instrument air backflow from the instrument air receiver tank (TK1) to the instrument air station. These valves maintain pressure of backup air supply reservoirs for the respective IPRVs (i.e. 63421-IPRV1/IPRV2/IPRV3) in the IPRV control system.	Good	Wear Mechanisms - Wear / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Component Replacement SRST - Functional Test	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete the following work orders: 1. WO 2096010 to remove and replace 018-25230-AL3-NV203. 2. WO 619970 by installing new locking tabs on PRD AL3. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011336	5,6,7,8,018	34200 21060	21130, 25230 Containment-	RB (21130) and PRD (25230) Airlocks form part of	Good	Mechanical and Thermal Degradation		Continue current practices. No additional practices are recommended to reach CO

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
		21130 25230	Major Equipment-Airlocks	the Containment Boundary 5/6/7/8-21130-AL1 to AL6 (Reactor Building Airlocks): Provide controlled entry into the Reactor Buildings so that the containment system integrity is maintained at all times. AL1 to AL3 are Equipment/Personnel Airlocks and AL4 to AL6 are Personnel Airlocks. 018-25230-AL3 (PRD Airlock): Provide personnel access to PRD - Pickering B Side. Access Doors (21060) Access during operation into areas designated as shutdown areas is prevented by the access control interlock system Doors. Included in this system are D56, D57 and D58, each door has an access lock and a quick escape.		/ Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear		EOL (2024).
011339	018,05 6,078, 5,6,7,8	34200	21000,34000,62000,63000,71000,73000 Containment-Power Cables 4.16 kV, 600V, 125V, 250V	The cables used to supply electric power to 5kV, 600V, and 300V loads as well as special definite purposes.	Good	Fatigue / Mechanical Fatigue Radiation Induced Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Thermal Aging	Inspection - Condition Monitoring Program	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011340	058,01 8	34200 34230 71330	34230, 71330 Containment-Valves - Manual-Criticality Category 1 (RS2)	The purge valve (018-34230-V120) allows access to the basement, by flushing the piping that contains radioactive noble gases with outside air.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Inspection - Visual SRST - Functional Test	<u>Incremental recommendations for Plant EOL (2020):</u> Ensure that adequate spares are available to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u>

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				058-71330-V20 is an isolation valve on the ESW supply common for Units 5 - 8. 058-71330-V21, V22, V23, V32, V33, V34, V35 are isolation valves in the emergency water supply/return header.		Corrosion / General Corrosion		Ensure that adequate spares are available to reach CO EOL (2024).
011341	058	34200 71330	71330 Emergency Storage Water Piping - Containment Expansion Joints	These expansion joints allow for movement due to vibration and thermal expansion / contraction in the Emergency Storage Water piping supply and return lines.	Good	Corrosion / Stress Corrosion Cracking - Austenitic Stainless Steels Fatigue / Mechanical Fatigue Corrosion / Pitting Corrosion Fatigue / Environmentally-Assisted Fatigue Corrosion / General Corrosion		Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Complete a baseline inspection, and depending on inspected condition implement a PM to complete recurring inspections at an appropriate frequency to determine internal and external condition.
011356	018	34200 34230	34200 - Containment - Manual Valves - FADS	These manually operated valves serve various isolation purposes. V3000 is a normally closed purge valve/ test valve. Other valves allow measurement of system pressure, isolation of pressure instruments and draining of collected system condensation. The valve V24 (Position Assured and Normally Closed) is provided for pressurization of the interspace between MV101 and MV102, so that a pressure test for leak tightness of these valves can be carried out periodically.	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Inspection - Visual SRST - Functional Test	Incremental recommendations for Plant EOL (2020): Ensure that adequate spares are available to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Ensure that adequate spares are available to reach CO EOL (2024).

System 0417 - Deaerator & Storage Tank

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008601	1,4	43122 43210	EXPANSION JOINT, Cat 1/2	The expansion joints 1, 4-43210-EJ4 are installed on inlet pipe, upstream of auxiliary condensate extraction pumps. The function of expansion joint is to absorb movement, relieve system strain due to thermal change, load stress and compensate for misalignment.	Satisfactory	Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue Corrosion / Flow induced wear of the leading edge Corrosion / General Corrosion	Inspection - Visual	<u>Incremental recommendations for Plant EOL (2020):</u> 1. Perform a one-time proactive external inspection of 1/4-43210EJ4 & 4-43210-EJ3001. 2. Resolve MEL/BOM issues. 4-43210-EJ3001 & 4-43210- EJ4 are linked to incorrect CIDs, identify if CID 606527 (1-43210-EJ4) is appropriate. <u>Incremental recommendations for CO EOL (2024):</u> Conduct an assessment for additional AM practices for CC1/SPV EJ4 and EJ3001 based on expected life after one-time inspection.
008602	1,4	43122 43210	EXPANSION JOINT, Cat 3/4	The function of expansion joint is to absorb movement, relieve system strain due to thermal change, load stress and compensate for misalignment in the condensate pump suction piping.	Satisfactory	Fatigue / Mechanical Fatigue Corrosion / General Corrosion Corrosion / Stress Corrosion Cracking - Carbon and low alloy steels		<u>Incremental recommendations for Plant EOL (2020):</u> Perform a one-time proactive replacement. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
008603	1,4	43122 43122	HEAT TRACING, Cat 1/2	HX4 is the condensate deaerator which receives condensate from the low pressure heaters, vents non-condensable gases and provides condensate to the deaerator storage tank.	Good	Corrosion / General Corrosion Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Corrosion / Stress Corrosion Cracking - Carbon and low alloy steels Mechanical and Thermal Degradation / Thermal Aging Fatigue / Thermal Fatigue	Inspection - Internal Predictive Maintenance - Thermography	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		
008608	1,4	43122 43210	MOTOR, Cat 1/2	The Auxiliary Condensate Extraction Pump discharges directly to the Deaerator via a CV255. The purpose of this pump is to provide a supply of condensate to the Deaerator during Reactor cooldown periods.	Good	Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Lubrication Predictive Maintenance - Electrical Testing Predictive Maintenance - Thermography Predictive Maintenance - Vibration Monitoring SRST - Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008609	1,4	43122 43210	MOTOR, Cat 3/4	Three 50% capacity Main Condensate Extraction Pumps (43210-PI, P2, P3) take condensate from three Condenser shell hotwells and discharge it to the Deaerator. The pump's discharge is also used as a supply of condensate to the BFP glands and LP cylinder exhaust cooling system.	Good	Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Incremental recommendations for Plant EOL (2020): Implement and maintain a bearing lubrication and inspection program. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
008611	1,4	43122 43210	POSITIONER, Cat 1/2	The water level in the Deaerator Storage Tank is controlled by level controllers which modulate the three flow CVs, CV252, CV253 and CV254, to maintain the water level within an operating band. The auxiliary systems use the wide range Deaerator storage tank level measurement (64321-	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				LT252) and modulates CV255.				
008613	1,4	43122 43210	PUMP, Cat 1/2	Auxiliary condensate extraction pumps transfer demineralized water from the condensers to the deaerator during abnormal conditions.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Inspection - Visual Predictive Maintenance - Vibration Monitoring Predictive Maintenance - Thermography SRST - Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008614	1,4	43122 43210	PUMP, Cat 3/4	The transfer of demineralized water from the condensers to the deaerator via the drain coolers is achieved by the use of three main condensate extraction pumps (CEPs) 1-43210-P1, 4-43210-P1, 1-43210-P2, 4-43210-P2, 1-43210-P3, 4-43210-P3 (two duty, one standby) during normal operation.	Good	Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Re-activate PM activities for pumps.
008615	1,4	43122 43210	REGULATOR, Cat 1/2	PRV1 is a pressure regulator for instrument air supply to CV255 for auxiliary condensate pump discharge to the deaerator.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008616	4	43122 43210	RELAY, Cat 1/2	The 43210-PM4-83-R1 relays ensure an alternate 48VDC power supply is available to the control logic	Good	Corrosion / General Corrosion Corrosion / Oxidation		Incremental recommendations for Plant EOL (2020): Schedule periodic testing of 1-43210-PM4-83-R1.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				circuit for the Auxiliary Extraction Pumps for the Condensate System.		Mechanical and Thermal Degradation / Electronic Aging		Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
008618	1,4	43122 43210	STRAINER, Cat 1/2	These strainers pre-filter fluid before it enters the auxiliary pumps in the condensate system. They prevent debris from entering the pumps which could lead to possible damage to the pumps and downstream items.	Good	Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Corrosion / Fouling (accumulation of deposits)		Continue current practices. No additional practices are recommended to reach EOL (2024).
008621	1,4	43122 43122	TANK, Cat 1/2	The deaerator storage tank serves as the head tank, ensuring sufficient NPSH for the Boiler Feed Pumps. The storage tank also acts as a reservoir to provide storage volumes to accommodate transient conditions.	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Fatigue / Mechanical Fatigue	Inspection - Internal Inspection - Visual	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008623	1,4	43122 43210	VALVE, CONTROL, Cat 1/2	Flow control valves CV254, CV255 and CV260 regulate the flow to the deaerator storage tank.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion	Overhaul/Refurbishment SRST - Functional Test Valve Diagnostics	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008624	1,4	43122 43210	VALVE, CONTROL, Cat 3/4	The water level in the deaerator storage tank is controlled by level controllers which modulate the three flow CVs, CV252,	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

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CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				CV253 and CV254, to maintain the water level within an operating band. For normal full load operation any two CVs can be selected from the control room. In the event of a malfunction of one valve, the third (standby) valve will take over automatically.		/ Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear		
008626	1,4	43122 43210	VALVE, MANUAL/HAND OPERATED, Cat 1/2	Manual/ Hand operated valves 1-43210-V12, 4-43210-V12, 1-43210-V301, 4-43210-V301, 1-43210-V32, 4-43210-V32, 1-43210-V4, 4-43210-V4, 1-43210-V69, 4-43210-V69 are used for isolation of piping components for maintenance purposes.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion Obsolescence / Immediate Obsolescence Concern	Surveillance	Incremental recommendations for Plant EOL (2020): 1. Resolve outstanding MEL/BOM (Cat ID should be in approved status) issues on valves 1, 4-43210-V12, 1, 4-43210-V301 & 1, 4-43210-V69. 2. Complete a one-time elastomer replacement and stem lubrication of all valves in this CG. Incremental recommendations for CO EOL (2024): Review the need to initiate a PM for periodic stem lubrication for all valves in this CG, based on condition after one-time Plant EOL (2020) work.
008628	1,4	43122 43210	VALVE, MOTORIZED/MOTOR OPERATE, Cat 1/2	The MV's in this CG are the condensate isolating motorized valves. When closed they isolate their respective LP Heater Banks in the condensate system.	Satisfactory	Corrosion / General Corrosion Mechanical and Thermal Degradation / Lubricant Degradation Mechanical and Thermal Degradation	Lubrication Inspection - Internal SRST - Stroke Test	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete WO 04793232 which requires actuator removal, a valve repack and stem/ drive nut replacement for 1-43210-MV17. Incremental recommendations for Plant EOL (2020):

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						/ Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Erosion		Complete a one-time overhaul & diagnostics of all MV actuators as per instructions in INACTIVE PMs (e.g. 6969). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
008629	1,4	43122 43210	VALVE, CHECK/NONRETURN/BACKFL	Non-return valves 1-43210-NV18, 4-43210-NV18, 1-43210-NV19, 4-43210-NV19 prevent feedwater flow reversal to drain coolers and 1-43210-NV8, 4-43210-NV8 prevent reverse flow to auxiliary condensate pump.	Good	Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Fatigue / Mechanical Fatigue Corrosion / General Corrosion	Inspection - Internal SRST - Functional Test	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete WO 3005442 & 3005443 to inspect and overhaul U1 NV18 & NV19. Based on inspection results, overhaul U4 NVs. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Initiate a PM for in-service testing of 1/4-43210-NV18 & 1/4-43210-NV19 at frequency of 4 years.
008631	1,4	43122 43210	VALVE, PRESSURE REGULATING, Cat 1/2	The PRV's in this CG regulate instrument air pressure to their respective parent valve assemblies (1/4 43210-CV260).	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011211	5,6,7,8	43122 43120	43120 Deaerator & Storage Tank	The Deaerator consists of a heat exchanger 43120-HX4 and a storage tank 43120-	Good	Fatigue / Mechanical Fatigue	Inspection - Internal	<u>Current initiatives that need to be completed for Plant EOL (2020) that are</u>

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				TK1. The DA liberates most of the non-condensable gases from the condensate. The deaerator storage tank serves as the head tank, ensuring sufficient NPSH for the Boiler Feed Pumps and the storage tank also acts as a reservoir to provide storage volumes to accommodate transient conditions.		General Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Corrosion / Flow Assisted Corrosion (Erosion-Corrosion)	Inspection - Non - Intrusive	incremental to current periodic maintenance practices: Complete modifications to the Units 5-8 Deaerator Storage Tanks (DST) supports to increase the seismic capacity beyond the Pickering B Review Level Earthquake (RLE) for applicable deaerators per Master EC124589. Incremental recommendations for Plant EOL (2020): Evaluate the cost/benefits of completing weld inspections (of the two welds on each side of the HX4 of the water box plate to the 43120-HX4 shell) on U2 or U3 and implement additional AM practices on U5-8 if necessary based on inspection results. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

System 0418 – Digital Control Computers[11]

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008634	1,4	66400	CIRCUIT BREAKER, Cat 1/2	Circuit breaker panels (PL175/176) supplying power for DCC system.	Good	Thermal aging, mechanical wear, and corrosion in electronics/connectors, thermo-mechanical fatigues of solder joints, electric contact degradation.	Predictive Maintenance - Thermography	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008638	1,4	66400	DCC MONITOR, Cat 3/4	Provides ANO visual interface to DCC system.	Satisfactory	Thermal aging, mechanical wear, and corrosion in electronics/connectors, thermo-mechanical fatigues of solder joints, electric contact degradation and particularly obsolescence issues due to the fast changing nature of computer technology	Inspection - Visual	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete obsolete monitors replacements. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental Work Required for CO EOL 2024</u> Replace Monitors.
008639	1,4	66400	DIGITAL CONTROL COMPUTER, Cat 1/2	The DCCs provide, via software, many functions for the safe operation of the plant. The main functions of the DCC are: - Reactor Regulating system (RRS) - Boiler Pressure Controller (BPC) Other control functions are: - Zone Thermal Power (ZTP) - Fuel Handling Most functions are duplicated on both channel	Satisfactory	Thermal aging, mechanical wear, and corrosion in electronics/connectors, thermo-mechanical fatigues of solder joints, electric contact degradation and particularly obsolescence issues due to the fast changing nature of computer technology.	Inspection - Visual Overhaul/Refurbishment Predictive Maintenance - Thermography	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete maintenance console PC power supplies and hard drives replacement. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental Work Required for CO EOL 2024</u> • Replace Maintenance Console. • Replace FH Typer. • DCC Cable Replacement

[11] OPG has assessed the Digital Controls Computers under P-CORR-66400-0632085 [37], this table has been extracted from P-CORR-66400-0632085 [37].

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				1 and 2 except Fuel Handling of which is performed on DCC channel 2 only.				
008641	1,4	66400	HOT SPARE [DCC9/10], Cat 3/4	Hot spares for DCCs.	Good	Thermal aging, mechanical wear, and corrosion in electronics/connectors, thermo-mechanical fatigues of solder joints, electric contact degradation and particularly obsolescence issues due to the fast changing nature of computer technology.	Inspection - Visual Predictive Maintenance - Electrical Testing	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008644	1,4	66400	MODULE [UNIT SCANNER], Cat 1/2	Contact scanners are used in the DCC system (via PACE/DAC computer) to scan relay contacts.	Good	Thermal aging, mechanical wear, and corrosion in electronics/connectors, thermo-mechanical fatigues of solder joints, electric contact degradation.	Predictive Maintenance - Thermography	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008645	1,4	66400	MODULE [CONTACT SCANNER HARDWARE], Cat 3/4	Contact scanners SIMs are used in the DCC system (via PACE/DAC computer) to scan relay contacts.	Good	Thermal aging, mechanical wear, and corrosion in electronics/connectors, thermo-mechanical fatigues of solder joints, electric contact degradation.	Predictive Maintenance - Thermography	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008649	1,4	66400	PACE SYSTEM CHASSIS, Cat 1/2	Provides ANO operator interface to DCC system for input and display of DCC system data.	Satisfactory	Thermal aging, mechanical wear, and corrosion in electronics/connectors, thermo-mechanical fatigues of solder joints, electric contact degradation and particularly obsolescence issues due to the fast changing nature of computer technology.	Calibration Inspection - Visual Predictive Maintenance - Electrical Testing Predictive Maintenance - Thermography	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

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CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008650	1,4	66400	PANEL [DCC], Cat 1/2	Provides Class II power to the DCC system.	Good	Normal material degradation.	Predictive Maintenance - Thermography	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008652	1,4	66400	PANEL [DCC], Cat 3/4	Panels for control monitoring and annunciation distributions for the Unit's field instrumentation and also distribute power to the DCC system.	Satisfactory	Normal age-related degradation of the performance characteristics of discrete electronic components.	Calibration Predictive Maintenance - Thermography	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete replacements for panels PL170/171 and the DCC Digital Output cards. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental Work Required for CO EOL 2024</u> Replace or refurbish I/O subsystem components.
008653	1,4	66400	POWER SUPPLY [DCC], Cat 1/2	Various power supplies essential to DCC system operation.	Satisfactory	Thermal aging, mechanical wear, and corrosion in electronics/connectors, thermo-mechanical fatigues of solder joints, electric contact degradation.	Calibration Predictive Maintenance - Electrical Testing Predictive Maintenance - Thermography	<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental Work Required for CO EOL 2024</u> Replace ES1800 DCC power supplies.
008656	1,4	66400	PRINTER, Cat 3/4	DCC system printer.	Good	Thermal aging, mechanical wear, and corrosion in electronics/connectors, thermo-mechanical fatigues of solder joints, electric contact degradation.	Inspection - Visual	<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental Work Required for CO EOL 2024</u> Replace DCC printers.
008664	1,4	66400	SWITCH, Cat 1/2	DCC Channel over temperature switches (TS1/TS2) located in panels PL 26 and PL 40.	Good	Thermal aging, mechanical wear, and corrosion in electronics/connectors, thermo-mechanical fatigues of solder joints, electric contact degradation.	Predictive Maintenance - Thermography	<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental Work Required for CO EOL 2024</u> Replace TS1/TS2 over temperature switches.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008667	1,4	66400	TRANSFORMER, Cat 1/2	The isolating transformers (T1, T2) supply power to the DCC system.	Good	Thermal aging, mechanical wear, and corrosion in electronics/connectors, thermo-mechanical fatigues of solder joints, electric contact degradation.	Calibrations Predictive Maintenance - Thermography	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011228	5,6,7,8	66400	66400 Digital Control Computers (DCCs)-COMPUTER-UNIT DCC CONTROL COMPUTER-CAT 1&2	The DCCs provide, via the software, many functions for the safe operation of the plant. The main functions of the DCC are: - Reactor Regulating System (RRS) - Boiler Pressure Controller (BPC) Other control functions are: - Zone Thermal Power (ZTP) - Moderator Temperature Control (MTC) - Fuel Handling (DCCY only)	Satisfactory	Thermal aging, mechanical wear, corrosion in electronics/connectors, thermo-mechanical fatigues of solder joints, electric contact degradation.	Inspection - Visual Overhaul/Refurbishment Predictive Maintenance - Electrical Testing	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete: • Differential Amplifier replacement. • Replace wire-wrapped IOBIC with CID 643793. • Qualification testing of spare CPUs from Wolsong. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental Work Required for CO EOL 2024 Complete DCC cable replacement.
011229	5,6,7,8	66400	66400 Digital Control Computers (DCCs)-COMPUTER-PERIPHERALS-CAT 1&2	The peripherals in combination with the Varian V72 computers (CG011228) provide all essential computer operations crucial to the control of the reactor. This also includes the operator interfaces (e.g. keyboards, displays).	Satisfactory	Thermal aging, mechanical wear, corrosion in electronics/connectors, thermo-mechanical fatigues of solder joints, electric contact degradation	Inspection - Visual Overhaul/Refurbishment Predictive Maintenance - Electrical Testing	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete: • Replace Megaram with BMSU and BTC. • PIPC computers replacement. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental Work Required for CO EOL 2024 Complete: • Unit Printer replacement.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
								• PTE replacement. • Fuel Handling PC replacement.
011230	5,6,7,8	66400	66400 Digital Control Computers (DCCs)-COMPUTER-PERIPHERALS -CAT 3&4	Cat 3 & 4 Peripherals provide mainly the interfaces between field inputs/outputs and the Varian V72 computers (CG011228) and other devices essential to the computer operations.	Good	Thermal aging, mechanical wear, corrosion in electronics/connectors, thermo-mechanical fatigues of solder joints, electric contact degradation.	Inspection - Visual Overhaul/Refurbishment	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011231	5,6,7,8	66400	66400 Digital Control Computers (DCCs)-POWER SUPPLY - CAT 1&2	The power supplies Cat 1 & 2 in CG 011228 are mainly the power supplies for the Varian V72 computers and for the I/O expansion chassis # 1 and #2. These power supplies are integral parts of the Varian V72 computers.	Good	Thermal aging, mechanical wear, corrosion in electronics/connectors, thermo-mechanical fatigues of solder joints, electric contact degradation	Inspection - Visual Overhaul/Refurbishment Predictive Maintenance - Electrical Testing	Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental Work Required for CO EOL 2024 Refurbish or replace CPU and I/O Expansion chassis power supplies.
011232	5,6,7,8	66400	66400 Digital Control Computers (DCCs)-POWER SUPPLY - CAT 3&4	Cat 3 & 4 power supplies in CG 011228, as listed in Passport, are mainly power supplies for the process I/Os chassis. There are also 500+ power supplies in the 9 DCCs that are not registered in Passport.	Good	Thermal aging, mechanical wear, corrosion in electronics/connectors, thermo-mechanical fatigues of solder joints, electric contact degradation	Inspection - Visual Overhaul/Refurbishment Predictive Maintenance - Electrical Testing	Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental Work Required for CO EOL 2024 Replace CAE/Lambda power supplies.
011233	5,6,7,8	66400	66400 Digital Control Computers (DCCs)-Scanners - CAT 1&2	Contact scanners are used in the DCC to scan relay contacts, limit switches, or other similar types of contacts, and to provide contact alarm annunciation and sequence of events.	Satisfactory	Thermal aging, mechanical wear, corrosion in electronics/connectors, thermo-mechanical fatigues of solder joints, electric contact degradation	Inspection - Visual Overhaul/Refurbishment	Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental Work Required for CO EOL 2024 Complete: • Replace or refurbish scanner power supplies, • Replace Unit scanner CS1 and CS2 wire-wrapped cards on Unit 8 only.

System 0420 - Electrical Systems

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008686	1,2,4	54400 55100	BATTERY, Cat 1/2	As part of the Class I power supply system, the batteries provide the most reliable source of electrical power for a variety of essential loads and safety related systems at the station.	Good	Fatigue / Mechanical Fatigue Corrosion / General Corrosion Mechanical and Thermal Degradation / Thermal Aging	Inspection - Visual Predictive Maintenance - Electrical Testing	Incremental recommendations for Plant EOL (2020): 1. Replace CH 'J' cells of 1/ 4-54400-BY2. 2. U1/U4 Class I battery banks 1/ 4-55100-BY1/BY2 to be replaced. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
008689	1,2,3,4,012,034,014,0	51100 51150 51300 53200 53300 54120 54130 54230 54400 55100 55200	BUS, Cat 1/2	To transfer electrical power throughout the station to the various loads.	Good	Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue Corrosion / General Corrosion	Inspection - Visual Predictive Maintenance - Thermography SRST - Logic Test SRST - Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008692	1	54230	CAPACITOR, Cat 1/2	a) enable quick energy transfer into IGBT circuit b) smooth out DC-bus voltage variation c) prevent ripple from interfering back to DC power source	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Fatigue / Mechanical Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008694	1,2,3,4,012,034,014,0	51200 53200 53300 53400 53500 54120 54130 54150 54230 54400 54500 55100 55200 65440	CIRCUIT BREAKER, Cat 1/2	Circuit breakers are for power transmission and for isolating/clearing faults.	Good	Corrosion / Pitting Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Thermal Aging	Inspection - Internal Inspection - Visual Overhaul/Refurbishment Predictive Maintenance - Thermography SRST - Functional Test SRST - Logic Test	Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Perform a one-time overhaul of all the circuit breakers with no active PM.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008696	1,2,4	54400 55100	CHARGER, Cat 1/2	Rectifier charges Class II TPS battery bank.	Very Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008698	1,2,3,4 ,014	51200 53200 53300 54120 54130 54140 54230 54400 55100	CONTACTOR, Cat 1/2	An electrically controlled switch used in power circuits for motors, transfer schemes etc.	Good	Corrosion / Pitting Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Thermal Aging	Inspection - Visual Predictive Maintenance - Thermography SRST - Functional Test SRST - Logic Test	<u>Incremental recommendations for Plant EOL (2020):</u> Proactive replacement of U1-4 CL II contactors to be completed. <u>Incremental recommendations for CO EOL (2024):</u> Class IV MCC contactors to be maintained along with the recommendation for one-time maintenance for CL III/ IV MCCs to reach CO EOL (2024).
008719	1,3,4	51200 52100 52200	HEAT TRACING, Cat 1/2	Heat exchanger provides cooling for transformer insulating oil, via LPSW.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Clean and Inspect	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> 1. Ensure WOs 3014186, 2332936 are completed for replacement of 1-51200-HX1/2. 2. Complete the proactive replacement of the SST Heat Exchangers (1-52100-HX1, 4-52100-HX1, 1-52100-HX2 and 4-52100-HX2) and the GST Heat Exchangers (1-52200-HX2, 4-52200-HX2, 1-52200-HX1 and 4-52200-HX2 (example WOs are 04939705, 04939724). <u>Incremental recommendations for Plant EOL (2020):</u> Procure spare heat exchangers. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008727	1,4	54230 54400	INVERTER, Cat 1/2	Supply total unit Class II control power.	Very Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging		<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> 1. Re-active the PMs to check and calibrate TPS inverter/battery for 1/4-54400-IN1 every two years. 2. Re-active the 1-65440-PDM-ROUTE “C116” Infrared Thermography for 1-54400-IN2 every year consistent with IQ Review Maintenance Strategy Template. 3. Re-active the 4-65440-PDM-ROUTE “C416” Infrared Thermography for 4-54400-IN2 every year consistent with IQ Review Maintenance Strategy Template.
008731	1,2,3,4 ,014,5, 018,0	53300 54130 54230	MOTOR CONTROL CENTRE, Cat 1/2	These MCCs supply critical loads for safe unit operation.	Good	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Thermal Fatigue Corrosion / Pitting Corrosion	Inspection - Internal Predictive Maintenance - Thermography	<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Perform one-time maintenance for CL III/ IV MCCs to reach CO EOL (2024).
008736	4	51200	MONITOR, Cat 1/2	To detect incipient faults in the MOT by monitoring dissolved combustible gases in the oil and send out warning.	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008738	1,4	67097	MONITOR, Cat 1/2, MONITORING UNITS	Monitoring unit, part of liquid effluent sampling system	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008741	1,4	51200	TRANSFORM R PUMP MOTOR , Cat 1/2	To circulate the oil through the coolers that cools the MOT.	Good	Fatigue / Thermal Fatigue		<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Fatigue / Mechanical Fatigue Corrosion / General Corrosion		Incremental recommendations for CO EOL (2024): Procure MOT pump motor spares.
008742	1,2,3,4 ,0	53300 54130 54140 54230	MOTOR STARTER, Cat 1/2	Device used to start/stop a motor. Starter includes circuit breaker/contactor, control circuit & protective relays.	Good	Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Inspection - Internal Predictive Maintenance - Thermography	Incremental recommendations for CO EOL (2024): 1. Perform one-time inspection of Class II CC1 motor starter cells. 2. Class III/IV MCC motor starters to be maintained along with the recommendation for one-time maintenance for CL III/ IV MCCs to reach CO EOL (2024).
008746	1,2,3,4	54130 54230 55200 65450	POWER SUPPLY, Cat 1/2	To provide regulated DC output voltage to meet system requirements.	Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Component Replacement	Incremental recommendations for Plant EOL (2020): No additional recommendations required to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Perform proactive replacement of the 48VDC power supply (PS) modules for transfer contactors 1/4 -54130-CN101/CN102-B2-PS1, and 4-55200-RF50B.
008748	1,4	51200	PUMP, Cat 1/2	Circulates transformer insulating oil to heat exchanger for cooling.	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Incremental recommendations for Plant EOL (2020): Procure spare MOT pumps. (Note: MOT pump and motor are built as one unit). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
008751	1,2,3,4	55100 55200	RECTIFIER, Cat 1/2	55100-RF1 supply units 250 V DC Class I power from the 600 volt, 60 Hz, Class III System. The 55200-RFs, 600 V AC / 48 V DC power rectifiers, supply the units 48 V DC Class II control power	Good	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging	Calibration Component Diagnostics Component Replacement Inspection - Internal	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete WOs to replace capacitors and address other deficiencies. Incremental recommendations for CO EOL (2024): Resolve obsolescence issues with 55200-RF20A/B/C/D/E/F/G/H/T.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008753	1,2,3,4	54400	REGULATOR, Cat 1/2	To provide regulated power supply, 208/120VAC +/- 1%, to Class II regulated bus.	Poor	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Thermal Aging	Clean and Inspect	Incremental recommendations for Plant EOL (2020): Replacement of regulators to be completed. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
008754	1,2,3,4 ,012,0 34,014 ,0	51200 52100 52200 53200 53300 54120 54130 54140 54230 54400 54500 55100 65010 65400 65414	RELAY (PROTECTIVE), Cat 1/2	A protective relay is a device designed to trip a circuit breaker when a fault is detected.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging Corrosion / General Corrosion	Calibration Overhaul/Refurbishment SRST - Functional Test SRST - Logic Test	Incremental recommendations for Plant EOL (2020): 1. Proactive replacement of ETS relays to be completed. 2. Calibrate U2/U3 spares so they are available if needed Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
008757	1	51200 52100 52200	RELAY, Gas Protection, Cat 1	Detect gas in the transformer and send out alarm at low gas quantity or trip the transformer when pressure surge is detected.	Good	Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008762	1,4,01 2,014	67090	STATION, Cat 1/2	Monitoring unit for Liquid Effluent sampling station.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008769	1,4	51100	SWITCH (GENERATOR N - GND), Cat 1/2	Connect/disconnect generator and Neutral Ground Transformer	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms – Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008778	1,3	54400	UNINTERRUP TABLE POWER SUPPLY, Cat 1/2	TPS UPS system provides an emergency 120 Vac power required by safety-related systems.	Very Good	Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008785	1,2,3,4,014	51100 51200 52100 52200 53200 54130 54140 54230 54400 55100	TRANSFORMERS, Cat 1/2	Transformers reduce higher supply voltages to voltage levels acceptable to common station loads. (E.g. 54130-T21 reduce 4.16KV voltage to feed critical Class III 600 Vac MCC loads.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Thermal Aging Corrosion / Galvanic Corrosion	Calibration Component Diagnostics Inspection - Internal Inspection - Non - Intrusive Inspection - Visual Predictive Maintenance - Vibration Monitoring SRST - Functional Test	Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Implement new PMs for transformers in the group not having a PM, similar to the PMs for the transformers that have PMs.
011159	5,6,7,8	51150	51150 Isolated Phase Bus Coolers	Two 100% duty air coolers remove heat generated by the isolated phase bus.	Satisfactory	Corrosion / Pitting Corrosion		Continue current practices. No additional practices are recommended to reach CO OEL (2024).
011160	5,6,7,8	51220	51220 Main Output Transformer Oil Circulation Pumps	The MOT oil circulation pumps are used to circulate oil through heat exchanger for cooling purposes.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Incremental recommendations for Plant EOL (2020): Replace Unit 5, 7, and 8 MOT circulating oil/pump assemblies (Unit 6 components replaced in 2008). Incremental recommendations for EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011161	5,6,7,8,1,2,3,4,058	52100 52200	52100 52200 Transformer Hydrogen Monitor for SST, GST, and 058-51320-T3	The Calisto 2 Monitor provides protection by identifying onset of incipient faults in a transformer by detecting the presence of H2 or CO in transformer insulating oil.	Good	Corrosion / General Corrosion Radiation Induced Degradation / UV exposure Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practice. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011162	5,6,7,8	54240	54240 Class II Bus Transfer Contactors	To transfer power to transfer buses to ensure power supply to critical loads.	Very Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011283	5,6,7,8	71310	71310 Electrical Systems AOV's	These valves control (modulating or on/off) LPSW to main output transformer oil coolers.	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Wear Mechanisms - Erosion	Component Replacement Overhaul/Refurbishment	<u>Incremental recommendations for Plant EOL (2020):</u> Procure three MV844/845 and three CV826 for replacement, and actuator overhaul kits in case of valve failure. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011284	018,0, 056,07 8,058, 068,5, 6,7,8	53000 54000 55000 65000 79000	53000, 54000, 55000, 65000, 79000 Electrical Systems - Cables - 4.16 kV, 600V, 125V, 250V	The cables used for Pickering GS B can be sub-divided into five (5) basic categories. There are 5 kV power cables, 600 V power cables, 600 V control cables, 300 V control cables and special definite purpose cables. The (major) equipment associated with the cables listed are: Class III / Class IV 4.16kV and 600V Switchgear, Class II / III / IV Motor Control Centers, 600V Distribution Panels and Class III / IV 4.16kV/600V Distribution Transformers.	Good	Mechanical and Thermal Degradation / Radiation Embrittlement Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Inspection - Condition Monitoring Program	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> 1. Complete inspection of the MOT/SST/GST and S/Y underground cables per P-2015-03783. 2. Complete inspection and/or replacement of SES 4kV cable between the P/H and HPECI Building per WO #1681231. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u>

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
								No additional recommendations required to reach CO EOL (2024).
011285	5,6,7,8,058,018,0	53300 54130 54140 54240 54330	54100, 54200, 54300, 53300 Electrical System - Motor Control Centres - 600 V / 208 V	The main function of the motor control centres located throughout the station are to provide 600V / 208V power distribution to process system motors, heaters, distribution panels, for loads rated less than 150 amperes. The centres provide a means of interfacing with process relay logic in order to control the multiple process schemes as well as a centralized location for the termination of the related control and power cables associated with these systems. The MCC's also provide an isolation point for work protection when maintenance is required to be done by Control Maintenance Staff.	Satisfactory	Wear Mechanisms - Wear / Wear Mechanisms - Wear Mechanical Fatigue / Mechanical Fatigue Thermal Fatigue / Thermal Fatigue	Inspection - Visual	<u>Incremental recommendations for Plant EOL (2020):</u> Replace all the CC1 and CC2 MCC cells in P058. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011286	5,6,7,8,056,078,058	53200 54120 54320 54600	53200,54120,54320,54600 Electrical Systems-Breakers-4.16 kV	Circuit breaker is a switch that interrupts electrical power to its load. Breaker utilises spring powered stored energy to instantly interrupt all 3 phases of power supply when required.	Good	Corrosion / Pitting Corrosion Mechanical and Thermal Degradation / Wear Mechanisms – Wear Mechanical and Thermal Degradation / Thermal Aging.	Overhaul/Refurbishment SRST - Functional Test	<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Overhaul nine 4kV breakers not covered by PM program.
011287	5,6,7,8	57100	57100 Electrical Systems-Control Cable	The cables addressed are considered to be all unitized and common system control cables for the complete station currently listed in the Online Wiring database. These cables are routed throughout the plant on dedicated cable trays. The cables are terminated in	Good	Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Radiation Embrittlement Fatigue / Mechanical Fatigue	Inspection - Condition Monitoring Program	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete inspection of the MOT/SST/GST and S/Y underground cables per P-2015-03783. <u>Incremental recommendations for Plant EOL (2020):</u>

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				multi-terminal distribution frames (DF's) and control panels and provide a means to interconnect the various components of process systems. Items interfaced are logic relays, control power supplies, limit switches, annunciation and status indication, instrumentation loops and panel-to-panel multi-connections etc. System identification is not "system specific" for a particular control cable as a multi-conductor cable may contain individual conductors that serve more than one related system index.		Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional recommendations required to reach CO EOL (2024).
011288	5,6,7,8	54250	54250 Electrical Systems-Rectifiers-48 Volt D.C	54230-RFs rectifiers supply units 48 V DC Class II control power from 120/208 V AC Class II power. D.C. outputs of every two main rectifiers are connected in parallel for each distribution panel. This ensures that if one sources or one rectifier should fail, the dc supply will continue to operate without interruption.	Good	Mechanical and Thermal Degradation / Thermal Aging Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Electronic Aging		<u>Incremental recommendations for Plant EOL (2020):</u> Obtain a pair of spare rectifiers and control cards. <u>Incremental recommendations for CO EOL (2024):</u> No additional recommendations required to reach CO EOL (2024).
011289	5,6,7,8 ,058,0 18,0	53300 54130 54240 56200 56210 56220	53300, 54130, 54240, 562XX Electrical Systems - Dry-type Power Transformers; Regulating, 9kVA, 4.16kV/600V	These transformers belong to different groups, those used for lighting and receptacles reduce voltages to acceptable levels for lighting and receptacles. (347/120) Vac. 4.16KV/600 Vac transformers provide power to 600 Vac buses and MCC loads. Regulating Transformers are used to supply loads that cannot tolerate large	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Corrosion / Galvanic Corrosion	Clean and Inspect Predictive Maintenance - Thermography	<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Perform a one-time inspection of transformers used for lighting/receptacles (56210/56220-T10/T14) and if found degraded, repair/replace.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				variations in supply voltages outside of a set percentage band (instrumentation, rectifiers)				
011325	5,6,7,8	65425	65425 - 40V DC Distr.System Power Supplies	The 40 VDC Power Supplies Modules from CL II Electrical Distribution are providing power to instrument loops for various systems for U5, 6,7 & 8 equipment.	Very Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging	Predictive Maintenance - Thermography	Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Perform one-time inspection, and if degraded replace.
011326	0,018	52100 52200 53200 53300 53300 54120 54130 54230 54240 54250 55100 55300	Electrical Systems-Relays-Timing	Timing relays are used in logic control circuits to provide time delay on pickup or drop-out for pumps, motors etc.	Good	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Thermal Aging Obsolescence / Immediate Obsolescence Concern		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011327	5,6,7,8	55100	55100 Electrical Systems-Breakers-250 V D.C	250VDC breaker is used to separate power source from its load in the event of a fault or overcurrent event.	Very Good	Wear Mechanisms - Wear / Wear Mechanisms - Wear Corrosion / General Corrosion Mechanical and Thermal Degradation / Thermal Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011328	5,6,7,8	51200	51200 Electrical Systems - Main Output Transformers (MOT)-24 kV to 230 kV	The primary purpose of MOT's is to step up generator voltage to allow transfer of electrical power to the grid.	Good	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Cracking Due to Settlement Corrosion / General Corrosion		Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Revise existing Unit 6 MOT life management study NK30-ESI-50000-00010 to demonstrate margin for CO EOL (2024) operation.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		
011329	5,6,7,8	52100	52100 Electrical Systems - System Service Transformers (SST)- 230 kV to 4.16 kV	The System Service Transformer steps down the grid voltage from 230 kV to 4kV and supplies station electrical loads from the grid.	Good	Corrosion / General Corrosion Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Thermal Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011330	5,6,7,8	52200	52200 Electrical Systems-Generator System Service Transformer (GST)-24kV to 4.16 kV	The Generator Service Transformer steps down the voltage from 24 kV to 4kV to supplies station electrical loads.	Good	Corrosion / General Corrosion Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Thermal Aging Radiation Induced Degradation / UV exposure		
011331	5,6,7,8,058,018,0	53300 54230 54240 54500	54200 Class II Electrical Systems - UPS / Batteries	Uninterruptible Power Supply (UPS) The function of the Class II UPS system is to provide uninterruptible power supplies to Class II loads. UPS-A & -B supply 600 Vac uninterruptable Class II loads and UPS-C supplies 120/208 Vac for triplicated Class II loads. Batteries 54230-BY1A, BY1B provide dedicated reliable source of electrical power for UPS-A and UPS-B for a minimum period of 40-minute on loss of Class III power. 54240-BY1C provides a backup supply source dedicated to UPS-C solely.	Satisfactory	Thermal Aging / Thermal Aging Corrosion / General Corrosion	Component Replacement Electrical Testing Inspection - Visual Predictive Maintenance - Thermography	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Replace 5-54230-UPSB under WO#4917436. <u>Incremental recommendations for Plant EOL (2020):</u> 1. Replace Units 5, 6, 7, and 8 UPS-Cs. 2. Replace DC and AC capacitors of 54230-UPSA and –UPSB at a frequency of 5 years. 3. Procure one spare for UPSA/UPSB for future maintenance. 4. Procure replacement for GUI for 5/6/7/8-54230-UPSA/UPSB. 5. Replace the Class II batteries within the next 2-3 years. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011332	5,6,7,8,056,078,058,018,0	52100 52200 53200 53300 53380 54120 54130 54240 54250 55100 65400 65510 65530	Electrical Systems Relays-Control & Auxiliary	Control & Auxiliary relays are used in logic control circuits for alarm, annunciation or multiplication of relay contacts.	Good	Thermal Aging / Thermal Aging Corrosion / Environmental Degradation Mechanical and Thermal Degradation / Wear Mechanisms - Wear Obsolescence / Immediate Obsolescence Concern	Calibration Overhaul/Refurbishment	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Procure spare CC1 & CC2 relays (under Relay Obsolescence Project 41042). <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011476	5,6,7,8,056,078,058,018,0	51200 52100 52200 53200 53300 54120 54130 54230 54240 54250 54800 55100 55300	50000 - Protective Relays-250 / 125 / 48 V D.C.	A protective relay is a device designed to trip a circuit breaker when a fault is detected.	Good	Corrosion / General Corrosion	Calibration Overhaul/Refurbishment	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Replace protective relays on U5-8 MOT, SST & GST (under Project 13-40691) and address protective relay obsolescence issues (under I&C Obsolescence Project 41042). Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011523	5,6,7,8,018	53300 54130	53300, 54130, & 54330 600V Metal Clad Switchgear Breakers	Circuit breakers interrupt electrical power to the loads it supplies. The breakers are spring powered stored energy released to instantly interrupt all three phases of the power supply when required.	Good	Wear Mechanisms - Wear / Wear Mechanisms - Wear Corrosion / Pitting Corrosion Mechanical and Thermal Degradation / Thermal Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011524	5,6,7,8	54240	54240 Class II Main Transformer - T1/T2	Transformer steps down the Class II 600V to feed the Class II to 120/208V bus.	Good	Mechanical and Thermal Degradation / Thermal Aging	Inspection - Visual	Incremental recommendations for Plant EOL (2020): Procure a spare transformer. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011537	5,6,7,8	67138	MCCBs	MCCB is a low voltage switch which provides circuit isolation & protective elements for circuit protection.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Incremental recommendations for Plant EOL (2020): Proactive replacement of CC1 & CC2 MCCB's to be scheduled. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

System 0421 - Emergency Coolant Injection System.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008789	1,4	33350 33350	ACTUATOR, Cat 1/2	These valve actuators are used to operate ECI injection and recovery valves.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Creep and Stress Relaxation Fatigue / Mechanical Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008790	1,4	33350 63335	ALARM, Cat 1/2	Indication and Alarm/Trip set points for critical ECI processes	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings, etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008791	1,4	33350 33350 63335	BOX, Cat 1/2	Splice boxes house spliced cable connections in a HARSH environment and must prevent moisture ingress and submergence of cable splices of any accumulating liquids in the event of a Design Basis Accident. Sealed covers and drainage holes provide protection against liquid accumulation.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Incremental recommendations for Plant EOL (2020): Include all splice boxes in this CG in NA44-JBL-60090-00001 & NA44-JBL-60090-00004 for inspection and maintenance as per CMP P-A-CMP- 60090.01 & .09 Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
008793	1,4	33350 63335	COMPUTER, Cat 1/2	In-Core LOCA Level detectors (L9-LD1), via level transmitters (L9-LT1), sense changes in calandria level to	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging	Calibration Inspection - Visual SRST - Logic Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				open 1-32110-CV2 & CV4 on In-Core LOCA signal.		Fatigue / Thermal Fatigue		
008794	1,4	33350 33350 63335	CONTROL, Cat 1/2	Hand controllers (HC) used for automatic control functions during ECIS recovery injection test valve CV450 while flow indication controllers (FIC) used for helium flow control during ECIS in-core LOCA initiation tests.	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Thermal Aging	Calibration Component Diagnostics	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008796	1,4	33350 33350	DISC, Cat 3/4	The FM service rooms and south accessible area are connected by an interconnecting trench. The areas are separated by the rupture disc. The rupture discs provide atmospheric separation.	Good	Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008797	1,4	33350 33350	DOOR, Cat 1/2	Provide effective isolation against leakage of water into the moderator room past the doors D6 and D63.	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008798	014	33350 33350	ELEMENT, Cat 1/2	Thermocouples used to measure the temperature of ECI pipe heaters.	Good	Mechanical and Thermal Degradation / Creep and Stress Relaxation Fatigue / Thermal Fatigue Fatigue / Environmentally-Assisted Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008800	1,4	33350 63335	GAUGE, Cat 1/2	Pressure Gauges are used for local pressure indication.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008802	014	33350 33350	HEATER (GENERIC), Cat 1/2	Trace Heaters provide protection of exposed ECI piping against freezing during winter months.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008810	1,4	33350 33350	POSITIONER, Cat 1/2	Valve positioner, NC1, accurately positions CV450.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008811	1,4	33350 33350 63335	REGULATOR, Cat 1/2	Regulators (PRV1) provide a controlled regulated pressure to valve auxiliary control equipment (AX, positioners).	Very Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue	Calibration Overhaul/Refurbishment SRST - Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008813	1,4	33350 33350 63335	RELAY, Cat 1/2	Relays used for interlock or time delay necessary for valve control.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging	Calibration Valve Diagnostics SRST - Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008814	1,4,01 4	33350 63335	RELAY, Cat 3/4	Relays provide logic and annunciation functions in the ECI system.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Thermal Aging		Continue current practices. No additional practices are recommended to reach CO EOL 2024.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008815	1	33350 63335	RESISTOR, Cat 1/2	Resistors in this CG are used as voltage dividers for low voltage lamps (e.g. 1-63335-CV7-RS1.	Good	Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008817	1,4,014	33350 33350 63335	SWITCH, Cat 1/2	Switches used to indicate control valve positions and to function ECIS piping heat tracing system thermostats.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings, etc.) Fatigue / Thermal Fatigue	Component Replacement Inspection - Internal Inspection - Visual SRST - Functional Test	Incremental recommendations for Plant EOL (2020): 1. Obtain spares for the valve limit switches of 1/4-33350-MV571; 2. Replace the door switches of 1-63335-NS203 & NS204; and 3. Replace the temperature switch of 014-33350-HTR202-TS1. Incremental recommendations for CO EOL (2024): Implement new PMs for the position switch of ECI recovery injection isolation valves (33350-V476/V477) and Moderator Room door limit switches (63335-S201/NS203/NS204).
008818	014	33350 63335	SWITCH, Cat 3/4	Switches used for temperature control of ECI heat tracing.	Satisfactory	Fatigue / Mechanical Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008819	1,4	33350 33350 63335	SWITCH, Cat 1/2	Switches used for ECI tests and panel checks.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008822	1,4	33350 33350 63335	TRANSMITTER, Cat 1/2	Transmitters used to measure the ECI process variables (flow, pressure, level, etc.) and transmit the signals for monitoring and testing.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Creep and Stress Relaxation	Calibration Overhaul/Refurbishment	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008823	1,4	33350 63335	TRANSMITTER, Cat 3/4	Transmitters and transducers used for ECI in-core LOCA initiation testing and transmitter testing	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Creep and Stress Relaxation		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008825	1,4	33350 33350	VALVE, CONTROL, Cat 1/2	33350-CV450 is normally open and is operated when performing NA44-SRS-E-019 to test 1-33350-MV471 & MV571.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue	Calibration Overhaul/Refurbishment Valve Diagnostics	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008828	1,4	33350 33350	VALVE, MOTORIZED/ MOTOR OPERATE, Cat 1/2	1 / 4 -33350-MV154, MV155, MV156 and MV157 provide flow paths for emergency coolant from ECI system to Pickering A Units 1 or 4 during a LOCA.	Satisfactory	Mechanical and Thermal Degradation / Lubricant Degradation Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Corrosion / General Corrosion	Inspection - Visual Lubrication SRST - Functional Test SRST - Logic Test SRST - Stroke Test Valve Diagnostics	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008829	1,4	33350 33350	VALVE, CHECK/NONRETURN/BACKFL, Cat 1/2	Prevent backflow into various portions of the ECI system.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Component Replacement Inspection - Internal	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008830	1,4	33350 33350	VALVE, CHECK/NONRETURN/BACKFL, Cat 3/4	During long term low pressure recovery (after HP injection), NV358 prevents draining of the boiler room piping into the moderator	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring,		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				room piping thereby minimizing mixing of the HTS and moderator fluids.		Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		
008831	1,4	33350 33350	VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2 control and Motorized	Emergency Coolant Injection Valves 1 / 4-33350-CV450, MV471 & MV571 are required to operate during a LOCA event to provide make-up cooling to the Heat Transport System.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Creep and Stress Relaxation Fatigue / Mechanical Fatigue Corrosion / General Corrosion	Calibration Inspection - Internal Inspection - Visual Overhaul/Refurbishment Valve Diagnostics	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008832	1,4	33350 63335	VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2 instrument	MVs are used during SRSTs to verify channel logic, trip parameters and annunciations operate as designed.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Creep and Stress Relaxation Fatigue / Mechanical Fatigue	Surveillance	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008834	1,4	33350 33350	VALVE, PRESSURE RELIEF, Cat 1/2	RVs for CVs: During credited accident conditions, RVs are used to release instrument air to reduce the pressure when it is in excess of the proper pressure for operation of the control valve. RVs for MVs: This equipment is required to provide overpressure protection to the pneumatic circuit.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008835	1,4	33350 33350 63335	SOLENOID/SOLENOID OPERATED VAL, Cat 1/2	SVs are used to control the Control Valve operation or used in test circuits to test operation of transmitters or to initiate channel trip during system tests.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings, etc.) Mechanical and Thermal Degradation / Creep and Stress Relaxation Fatigue / Mechanical Fatigue	Component Replacement	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete Work Orders for a one-time replacement of all test SVs. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Perform one-time replacement of all SVs to ensure all components in this CG reach CO EOL (2024).
011120	5,6,7,8	33350 33350	33350 ECI Quad Pressure Test Valve	This is a normally closed valve used to pressurize the quadrant injection header immediately upstream of the ECI D2O Isolation valves for testing purposes.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011121	056,078	33350 33350	33350 ECI Recovery Heat Exchangers	HX1 provides cooling for ECIS water recovered from the R/B that is then re-injected into the main heat transport system following a LOCA.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Fretting		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Corrosion / Microbiological Influenced Corrosion Corrosion / General Corrosion Mechanical and Thermal Degradation / Creep and Stress Relaxation Fatigue / Mechanical Fatigue Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Corrosion / Fouling (accumulation of deposits)		
011122	058	33350 33350	33350 HPECI Pump Recirculation Heat Exchanger	HX2 removes heat generated by HPECI pump operation to protect the HPECI pumps from damage due to high temperatures during testing as well as during long term cooling of the HPECI pumps following a small LOCA, where sufficient flows through the pump are achieved by valving in the recirculation lines.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Fretting Corrosion / Pitting Corrosion Corrosion / Bio-Fouling Corrosion / General Corrosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Creep and Stress Relaxation		Incremental recommendations for Plant EOL (2020): Perform tube and shell side inspections. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011219	056,058,078,5,6,7,8	33350	33350, 53200, 54120 Emergency Coolant Injection	The cables used for 5kV power cables, 600V power cables, 600V control cables, 300V control cables and	Good	Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue	Inspection - Condition Monitoring Program	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
			System-Cable - 4.16 kV, 600V, 125V, 250V - Power Cables	special definite purpose cables.		Mechanical and Thermal Degradation / Radiation Embrittlement		
011255	5,6,7,8,056,078,058	33350 33350	33350 Emergency Coolant Injection System - Copes-Vulcan AOVs	058-33350-CV177/CV178/CV179/CV180 0 (HPECI Pump Recirculation Valves) valves are required to operate during the high pressure injection and high pressure re-injection phases of ECI to maximize ECI flow for Large breaks and to ensure minimum HPECI for small breaks 058-CV181 and 058-CV182 (HPECI test HX valves) must be capable of being manually positioned to prevent excessive flow diversion during HPECI pump testing and recirculation operation. 058-CV295, 058-CV296, 058-CV301, 058-CV302, 056-CV297, 056-CV298, 078-CV299 and 078-CV300 (chemical recirculation valves) open on a Pickering A or B LOCA conditioning signal to depressurize the HPECI circuit 5-8-33350-CV320 and CV321 (ECI recovery tempering circuit valves) are used to provide tempering water flow control to lower the Reactor Building Sump temperatures.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Lubricant Degradation Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion	Inspection - Internal Overhaul/Refurbishment SRST - Functional Test SRST - Logic Test Valve Diagnostics	<u>Incremental recommendations for Plant EOL (2020):</u> Complete the procurement of replacements for obsolete Copes-Vulcan ECI valves pneumatic positioners (Cat ID: 653079, 627875, and 606006). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011259	5,6,7,8,056,078,058	33350 33350 63335	33350, 63335 Emergency Coolant Injection	To prevent back flow of water to ensure proper ECI operation. To prevent backflow of instrument air to	Satisfactory	Fatigue / Mechanical Fatigue	Inspection - Internal Inspection - Non - Intrusive	<u>Incremental recommendations for Plant EOL (2020):</u>

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
			System - Check Valves	ensure proper air actuated valve operation.		Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / Flow Assisted Corrosion (Erosion-Corrosion)	SRST - Functional Test Valve Diagnostics	Perform one-time inspection and overhaul NV3 as necessary, and complete the replacement of 058-33350-NV3020. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011321	5,6,7,8,058	33350 33350 63335	33350 Emergency Coolant Injection System-Vessels-PV's, Tanks, Drums, etc.-CAT1&2	1) The cyclone separators (058-33350-C3000/C3001/C3002/C3003/C3004/C3005) remove debris from the seal water to ensure mechanical seals of HPECI pumps will not be damaged by foreign material. 2) 058-33350-TK2 is an oil separator. Mechanical seal cooling water leak off is routed to this tank. 3) 5-8-33350-MV70-TK1/MV71-TK1 are instrument air tanks, which provide seismic back-up instrument air to ECI sump isolator valves 33350-MV70/MV71. 4) 058-63335-CYL1 is a pressurized compressed air cylinder used for the HPECI pump discharge pressure and test circuit.	Satisfactory	Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011323	5,6,7,8,058	33350 54120 54320	54120,54320 Emergency Coolant Injection System - 4.16kV Motor Starters	The Motor Starters provide a means of switching the main power at 4.16kV, from either Class III or EPS power supply, so as to control the ECI Recovery Pump Motors 056-33350-PM1/PM2/PM3 and 078-33350-PM1/PM2/PM3.	Satisfactory	Self-Loosening / Self-Loosening Wear Mechanisms - Wear / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Thermal Aging		<u>Incremental recommendations for Plant EOL (2020):</u> Refurbish the contactors by replacing the major internal components i.e. main contactor unit and associated control relays and fuses. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011324	056,078,058	33350 33350	33350 ECIS - Injection Pumps	P1, P2, P3 are low pressure pumps used to recover LOCA water from the RBs	Good	Wear Mechanisms - Wear / Wear Mechanisms - Wear	Lubrication	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u>

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				and return it for re-injection. P5, P6, P7 are high pressure pumps used for high pressure injection/reinjection. P5A, P6A, P7A, P3012, P3021, P3032, P5B, P6B, P7B are lube oil pumps for injection & recovery pumps and motors.		Cracking Due to Vibration / Cracking Due to Vibration Corrosion / General Corrosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Lubricant Degradation	Predictive Maintenance - Thermography Predictive Maintenance - Vibration Monitoring	Ensure examination of wearing rings at higher magnification is included in existing maintenance program (as recommended in NK30-33350-LOF dated 11-Jun-2010). <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011353	5,6,7,8 ,056,0 78,058	33350 33350	33350 Emergency Coolant Injection System-Posi-Seal Valves - AOV-Standard-CAT1&2	056/078-MV114 (ECI recovery HX inlet valve) is used to ensure required flow of ECI coolant can be delivered at sufficient temperature and pressure. 056/078-MV115 (ECI recovery HX bypass isolation valve) is closed to ensure maximum ECI recovery HX heat removal. 058-MV151 (ECI storage tank isolation valve) has been removed from service 058-33350-MV160 (HPECI re-injection bypass) allows gravity feed from the ECI storage tank either directly to the HT system of a LOCA unit or to the ECI recovery sump via the tempering water flow path 056/078-MV23 (ECI recovery test valve) is used to prevent unacceptable ECI	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Obsolescence / Immediate Obsolescence Concern		<u>Incremental recommendations for Plant EOL (2020):</u> Replace recovery test valve 056/078-MV23 positioners with replacement CAT ID 507088. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				recovery pump flow recirculation 056/078-33350-MV7 (ECI recovery HX outlet valve) must be open to allow for ECI recovery to take place. 5-8-33350-MV70 and 5-8-33350-MV71 (Recovery Sump Isolation Valves) open automatically on receipt of a LOCA signal and high recovery sump level or low ECI storage tank level				
011354	5,6,7,8,056,078,058	33350 33350	33350 Emergency Coolant Injection System- Newman Hattersley Valves - AOV- Standard- CAT1&2	5/6/7/8-33350-MV49, MV52, MV53, MV56 & MV102 (D20 check valve interspace vent valves) are used for reducing injection header pressure during check valve testing 058-33350-MV200 (HPECI pump house active drainage sump fill valve) is used to mitigate contamination spread and loss of recovery sump inventory due to reverse leakage into the ECI storage tank inventory. 058-33350-MV284 and 058-33350-MV319 (HPECI Pump House Drainage Pump Discharge Valves) are used to assist in piping leakages back into the ECI system. 056/078-33350-MV110 (Recovery Sump Isolation Valve Test Valves) are no longer used Valves 056/078-33350-MV251, 056/078-33350-MV344, 056/078-33350-MV345 and 5/6/7/8-33350-MV109 are valves for draining, venting, maintenance, and testing.	Good	Mechanical and Thermal Degradation / Wear Mechanisms – Wear Corrosion / General Corrosion		Incremental recommendations for Plant EOL (2020): Replace drain valves and procure adequate spares for 5-33350-MV49 & MV52, 6-33350-MV53 & MV56 and 7-33350-MV56. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				056/078-33350-MV201 and 056/078-33350-MV202 (ECI Recovery Pit Active Drainage Sump Pump Discharge Valves) are used for the transfer of drainage from the ECC Vault recovery tank to the low pressure ECI recovery pump discharge header 5/6/7/8-33350-MV327 & MV328 (Drain Valves) provide connection to miscellaneous D2O collection lines. 5/6/7/8-33350-MV338 & MV339 (ECI Check Valves) are used to test stroke H2O check valves				
011355	5,6,7,8 ,058	33350 63335	63335 Emergency Coolant Injection System- Hoke & Valcor AOVs	This valve group acts as isolation valves to a number of test circuits including, HPECI Pump Discharge Header Pressure Test loop, Moderator High Level Test Circuit loop and ECI Low Pressure Test loop.	Good	Wear Mechanisms - Wear / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Lubricant Degradation	SRST - Functional Test	Incremental recommendations for Plant EOL (2020): Investigate a more robust replacement to mitigate passing Hoke valves. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011528	058	33350 33350	33350 HPECI Pump Motors-4kV	The HPECI Pump Motors are used to drive the HPECI Pumps. The HPECI pumps take suction from the ECI Storage Tank “Golf Ball” which holds demineralised water and pumps this water directly into the heat transport system of the accident unit (including Pickering A).	Good	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011529	056,078	33350 33350	33350 ECI Recovery Pump Motors-4kV	The ECI Recovery Pump Motors are used to drive the ECI Recovery Pumps, vertical two stage, centrifugal pumps. They are an integral part of ECIS whose purpose is to keep the heat transport system refilled after a loss of coolant accident (LOCA).	Satisfactory	Wear Mechanisms - Wear / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Thermal Aging	Calibration Lubrication Predictive Maintenance - Vibration Monitoring SRST - Functional Test	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Procure one (1) spare motor for the ECI Recovery Pump Motor strategy documented in NK30-ESI-05600-0000. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011533	058	33350 33350	33350 HPECI Pump & Motor Lube Oil Pump Motors-575V Motors	HPECI Pump Lube Oil Pump Motors are used to drive their parent pumps. The HPECI Pump contains oil lubricated Kingsbury bearings. Pressure lubrication is supplied by the Aux Lube Oil Pump and/or shaft-driven Oil Pump. The Aux lube oil pump is a back up to the shaft driven oil pump.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Thermal Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011535	5,6,7,8	33350 63330 63332 63336	63330 - ECI - Controller - Hand - CAT 1&2	The hand controllers for the HTS feed/bleed CV's and bleed condenser level CV's, control a process variable at a setpoint by supplying a direct acting control signal to its associated control valve based on HT header pressure and bleed condenser pressure, respectively.	Poor	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear Obsolescence / Immediate Obsolescence Concern	Inspection - Internal SRST - Functional Test	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete strategy to proactively replace hand controllers. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).

TETRA TECH

System 0422 - Emergency Power Supply

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011368	5,6,7,8	54300 54300 54340 54350 65430	54300 - Emergency Power Supply-Rectifiers-48 Volt D.C	The main rectifier panels are fed with 120/208 Volt Class II power. This feeds a 120VAC bus in the distribution panel as well as the rectifier. The rectifier in turn supplies a 48VDC bus in the distribution panel as well as a Ring Bus connecting the other rectifier panels together. The ring bus supplies backup power to the 48VDC bus in the event of a failure of its normal supply. Blocking diodes protect the ring bus in the event of a rectifier failure.	Satisfactory	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Thermal Aging	Inspection - Visual Predictive Maintenance - Thermography Inspection - Visual	<u>Incremental recommendations for Plant EOL (2020):</u> Replace major components (e.g., control cards, blocking diodes and SCRs) to support continued operation to Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011375	058	54300 54800	54800 - Emergency Power Supply-Valves - SV-Solenoid Valves-CAT1&2	The three-way, solenoid-operated pilot valves, 058-54800-EPG1/EPG2-SV50, used to open or close the starter motor shutoff valve, which controls air flow into the starter motors.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Wear Mechanisms - Erosion		<u>Incremental recommendations for Plant EOL (2020):</u> 1. One-time replacement of the SVs. 2. Procure spares for SV50 and SV85. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011376	058	54300 54800	54800 - Emergency Power Supply-Lube and fuel oil Pumps	058-54800-EPG1/2-P5: This pump is part of fuel oil delivery system acting as a booster pump of P4. This pump technically is an ejector, taking its suction from fuel day tank, discharging through filter then feeding suction of Pump 4, which feeds injectors for turbine combustion chambers. 058-54800-EPG1/2-P1/2: Pumps in the lubricating oil system which is providing lubricating oil for the EPG gas turbine and the reduction gear unit bearings, also supplies oil to the variable vane control actuator, regulating engine inlet pressure	Good	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms – Wear Corrosion / General Corrosion	Lube and Inspect SRST - Functional Test	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete WO 3199660 / 3199647 for Proactive replacement of EPG1/2-P5. Incremental recommendations for Plant EOL (2020): Add inspection of P1, P2, and P5 to PM 18249. Incremental recommendations for CO EOL (2024): Procure 3 spare pumps to have one spare for each type of the pumps (058-54800-EPG1/G2-P1, 058-54800-EPG1/G2-P2, 058-54800-EPG1/G2-P5).
011377	058	54300 54800	54800 - Emergency Power Supply Compressors	A compressed air system is used to start each EPG unit. Each compressor delivers air to one of two receiver tanks which supply the pneumatic starting system.	Very Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanisms - Erosion		
011378	058	54300 54800	54800 - Emergency Power Supply - Emergency Power Turbine-driven Generators	TURBINE (TU1): The engines are open cycle, single shaft industrial gas turbines which drive the generators via a star-compound, epicyclic reduction gear. The engine (single shaft) drives the compressor and accessories, and output flange through common shaft. COMPRESSOR (CP3) A compressed air system is used to start each EPS unit. GENERATOR: EPS is an independent, seismically qualified source of power for Class II systems such as Emergency Water System (EWS), Filtered Air Discharge (FADS), UECC, etc. Since this is a safety support system and as such, does not perform any function in normal power generation, nor supplies power to station auxiliaries. It is an independent power source which is separate from normal station service supplies and physically removed from the vicinity of the Powerhouse. The units are designed to be available to supply power to the Class II systems following a design basis earthquake. The units are not expected to operate during such an event, but are required to restart and be fully loaded within twenty minutes of the	Satisfactory	Fatigue / Mechanical Fatigue Corrosion / General Corrosion Fatigue / Thermal Fatigue Corrosion / Fouling (accumulation of deposits) Mechanical and Thermal Degradation / Wear Mechanisms - Erosion	Calibration Inspection - Internal Overhaul/Refurbishment SRST - Functional Test SRST - Logic Test	Incremental recommendations for Plant EOL (2020): Perform inspection and overhaul of EPG1 and EPG2 generators. Also, replace the exhaust stacks for EPG1. Note that all remaining equipment tags are subject to inspection and overhaul. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				event. Each unit is capable of being started, run to full speed, and ready to accept load within two minutes.				
011422	058,5, 6,7,8	54300	Emergency Power Supply-Cable - 4.16 kV, 600V, 125V, 250V - Power Cables	The cables used to supply electric power to 5kV, 600V, and 300V loads as well as special definite purposes.	Good	Fatigue / Thermal Fatigue Fatigue / Mechanical Fatigue Radiation Induced Degradation / Radiation Embrittlement	Inspection - Condition Monitoring Program	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011485	058	54300	54860 - Emergency Power Fuel Oil Supply-Piping	The Fuel Oil System pipes supply fuel oil from the storage tanks to the Combustion Turbines that drive the Emergency Generators. These are divided into three subsystems: Fuel Transfer, Fuel Storage and Fuel Forwarding	Satisfactory	Corrosion / Pitting Corrosion	Inspection - Buried Piping Program	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011491	5,6,7,8 ,058,0 18	54300 54330 54340 54800 65433	54300 Emergency Power Supply-Power Transformers - 600 Volts Secondary and less. 4.16 kV to 600V & 600V to 120/208V	Stepdown transformers (e.g., 058-54330-T1/T2) used to receive 4kV power from 54320-BUA/B, and supply power to 600v busses 54330-BUA/B. They are only energized when the EPG is operating and 600V power is required in any of the 4 UECC MCCs 3 or 4, or to EPS Building MCC30 and/or MCC31.	Good	Corrosion / Galvanic Corrosion Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Fretting	Overhaul/Refurbishment Predictive Maintenance - Thermography	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011492	058	54300 55400	55400 Emergency Power Supply-Battery-125 V	The batteries used to provide backup power to 058-55400-BUA and BUB for EPS generator breaker control,	Good	Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				emergency lighting & protective relaying.				
011534	058	54300 55400	55400 - 125V EPS Rectifiers	The 125 VDC system provides power for EPS generator breaker control, emergency lighting and protective relaying.	Satisfactory	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Thermal Aging		Incremental recommendations for Plant EOL (2020): Complete a proactive replacement of the rectifiers. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

TETRA TECH

System 0423 - Emergency Water Supply (EWS) System

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008836	1,4	71380 71330	ACTUATOR, Cat 1/2	These actuators operate the Emergency Storage Tank make-up valves which are used to fill the storage tank from the high pressure service water system.	Poor	Corrosion / General Corrosion Mechanical and Thermal Degradation / Lubricant Degradation Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Inspection - Internal Lubrication	Incremental recommendations for Plant EOL (2020): Implement a diagnostics program for actuator testing. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
008838	018	71380 67133	ALARM, Cat 1/2	Alarm units provide flow indication and alarm capability on low flow.	Very Good	Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008840	018	71380	ALARM, Cat 1/2	Alarm units provide water level & temperature loop indication, initiates alarms and in some cases have control functions.	Very Good	Mechanical and Thermal Degradation / Electronic Aging		Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Ensure spare transmitters (Cat ID 626506 & 525880) are available.
008843	018	71380 67133	ELEMENT, Cat 1/2	The temperature element is used to detect temperature.	Good	Fatigue / Thermal Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008845	3,4,0	71380 71330	EXPANSION JOINT, Cat 1/2	The expansion joints absorb the heat-induced expansion, contraction and vibration in the service water system emergency storage piping.	Good	Corrosion / General Corrosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Surveillance	Incremental recommendations for Plant EOL (2020): Perform baseline inspection and initiate new PM for routine inspection of Expansion Joints. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
008850	0	71380 71330	HEATER (GENERIC), Cat 1/2	The heat exchangers prevent water from freezing inside the dousing header and the emergency storage water tank. Heat tracing provides heating / prevention of freezing in the supply pipes to the VB.	Very Good	Corrosion / General Corrosion Corrosion / Fouling (accumulation of deposits) Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Clean and Inspect SRST - Functional Test	Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Complete a one-time sample inspection of an HX to ensure condition for CO EOL (2024).
008856	0	71380 71330	MOTOR, Cat 1/2	40HP, 575VAC drive motor for Emergency Water Storage pump.	Good	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Implement PMs (similar to those that are retired 129068 & 129069) for electrical testing of motors (0-71330-PM501 and 0-71330-PM502). Replace if required.
008860	0	71380 71330	PUMP, Cat 1/2	The pumps supply make-up water to the storage tank, recirculate the contents of the storage tank and remove	Good	Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				water accumulation from the vacuum building floor.		Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		
008862	018,0	71380 67133 71330	REGULATOR, Cat 1/2	The PRV's control the Instrument Air supply to actuators on control valves and Instruments in the EWS tank level measurement loops.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Creep and Stress Relaxation		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008865	1,4,0	71380 71330	STRAINER, Cat 1/2	The strainers remove particles and reduce levels of contamination to maintain the quality of the service water emergency water storage system process fluid.	Good	Corrosion / General Corrosion Corrosion / Fouling (accumulation of deposits)	Clean and Inspect	Incremental recommendations for Plant EOL (2020): Initiate a PM to complete regular inspection and cleaning of Unit 0 Strainers. Note: All of the remaining equipment tags are subject to cleaning and inspection. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008867	018,0	71380 67133 71330	SWITCH, Cat 1/2	Temperature switches (TS1's/TS2's) operate Emergency Storage System (ESS) service water piping trace heaters. Pressure switches (L502P–PS1 & L502Q-PS1) senses air receiver pressure for ESS level bubbler air supply PRVs to generate a low air receiver pressure alarm on Units 5, 6, 7 & 8.	Good	Mechanical and Thermal Degradation / Electronic Aging	Calibration Inspection - Visual SRST - Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008873	018	71380 67133	TRANSMITTER, Cat 1/2	RTD elements measure temperature of water in Emergency Water Supply (EWS) system & transmitter output is fed to an alarm/indicator on Vacuum Building panel. Differential pressure transmitter measures water level in EWS Storage Tank.	Good	Mechanical and Thermal Degradation / Electronic Aging		Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): 1. Perform one-time replacement for all transmitters. 2. Ensure spare transmitters are available.
008878	018,0	71380 67133 71330	VALVE, MANUAL/HAND OPERATED, Cat 1/2	Valves function as isolation valves in the EWS. They provide isolation for maintenance purposes, Vacuum building isolation from EWS, instrument isolation etc.	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Surveillance	Incremental recommendations for Plant EOL (2020): Initiate a PM to inspect condition of valves 018-71330-V2007 and V2008. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
008880	1,4	71380 71330	VALVE, MOTORIZED/MOTOR OPERATE, Cat 1/2	Normally closed valves fill the storage tank from the high pressure service water system. During extreme emergency conditions these valves can be opened in the unit to be supplied and the unit supplying, this provides a means of supplying high pressure service water from/to any of the other units.	Poor	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Creep and Stress Relaxation	Inspection - Internal SRST - Stroke Test Valve Diagnostics	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete repair of 1-71330-MV501. Incremental recommendations for Plant EOL (2020): Implement diagnostic testing. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion Mechanical and Thermal Degradation / Lubricant Degradation		
008881	0	71380 71330	VALVE, CHECK/NONRETURN/BACKFL, Cat 1/2	The check valves provide backflow protection for pumps and piping in the EWS system.	Satisfactory	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / Fouling		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						(accumulation of deposits)		
008883	0	71380 71330	VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2	MV2000 is the isolation valve for the EWS supply header from Units 1 - 4.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008884	1,4	71380 71330	VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2	These MV's are normally closed and they isolate the RCW system from the ESW system.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion	Inspection - Internal Overhaul/Refurbishment SRST - Function Test Valve Diagnostics	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008885	0	71380 71330	VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2, Fisher A11 with 1035 ESA600	This valve is an 8 inch diameter by-pass valve on the EWS header to bypass the larger diameter NC MV2000. The valve is normally open.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation /		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Wear Mechanisms - Wear Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Corrosion / General Corrosion		
008886	0	71380 71330	VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2, Jamesbury 815L Hytork XL	MV524 is normally open and allows pumps to recirculate water from the storage tank in the VB main chamber to heat exchangers in the VB basement and back to the tank. With MV524 closed and MV526 open, any water which may have accumulated on the floor of the main chamber is recovered and pumped back up to the storage tank.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion	Overhaul/Refurbishment Valve Diagnostics	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008887	0	71380 71330	VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2, Jamesbury 815L VPVL600SR3	Depending on the arrangement this valve performs two functions: with MV524 open and MV526 closed the pumps recirculate water from the storage tank in the VB main chamber to heat	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Overhaul/Refurbishment	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				exchangers in the VB basement and back to the tank, with MV524 closed and MV526 open, any water which may have accumulated on the floor of the main chamber is recovered and pumped back up to the storage tank.		Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion		
008888	0	71380 71330	VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2	These are normally open valves which allow make up flow from EWS to 0-34220-TK1 & TK2. The diaphragm valves control seal water flow to 3422P4, P5, P6 and P7: The seal and cooling water make up valves 7133-MV544 and MV692 close when both vacuum pumps 3422P4 (P6) and P5 (P7) are stopped.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008890	018,0	71380 67133 71330	SOLENOID/SOLENOID OPERATED VAL, Cat 1/2	These solenoid valves are used during testing of various ESS loops.	Very Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011177	056,078	71380 71380	71380 EWS Recovery Pump Motors	150HP, 575V drive motor for Emergency Water Supply Recovery Pumps.	Good	Mechanical and Thermal	Lubrication	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Predictive Maintenance - Thermography Predictive Maintenance - Vibration Analysis SRST - Functional Test	Procure spare motor as approved in MR #2858124. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Procure spare bearings.
011373	058	71380 71380	71380 Emergency Water System Motors-4.16 kV	Motor - 400HP, 4kV drive motor for Emergency Water Supply pump. Heater - 120VAC, 300W space heater to prevent condensation on motor.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Thermal Aging	Lubrication SRST - Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011421	056,058,078,5,6,7,8	71380	Emergency Water Supply (EWS) System- Cable - 4.16 kV, 600V, 125V, 250V - Power Cables	Cables feed power from source to load.	Good	Fatigue / Thermal Fatigue Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Radiation Embrittlement	Inspection - Condition Monitoring Program	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011440	5,6,7,8,056,078,058	71380 71330 71380	71380 Emergency Water Supply (EWS) System - Valves - Manual - Criticality Category 1 (RS2)	These valves serve various isolation functions in the EWS system such as recovery pump discharge isolation, HT loop isolation and moderator makeup water isolation.	Good	Mechanical Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Inspection - Internal Inspection - Visual Lubrication SRST - Functional Test	Incremental recommendations for Plant EOL (2020): 1. Procure spare parts for maintenance and overhaul. Removed valves to be overhauled and used for future replacements. 2. Resolve obsolescence issues. 3. Initiate a one-time replacement of the shear pins on 056-71380-V3 and 078-

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Obsolescence / Immediate Obsolescence Concern		71380-V3 to prevent reoccurrence of shear pin failure. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011441	5,6,7,8 ,056,0 78,058	71380 71380	71380 Emergency Water Supply (EWS) System-Valves - Manual-Criticality Categories 1 (P1, P2) And 2 (RS3)	These valves serve various isolation functions in the EWS system such as heat exchanger isolation, pump discharge isolation, test loop to inactive drainage isolation, etc.	Good	Mechanical Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Clean and Inspect Lubrication SRST - Functional Test	<u>Incremental recommendations for Plant EOL (2020):</u> Address obsolescence issues and source replacement valves. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011442	5,6,7,8 ,058	71380 71380	71380 Emergency Water Supply (EWS) System-Valves - MOV-Standard-CAT1&2	These valves provide isolation between the EWS and PHT systems. Others are used to equalize the pressure across their respective neighbouring check valve during testing. The final group of valves allow the automatic backwashing system to be used to clean the strainer in EWS system.	Good	Mechanical and Thermal Degradation / Thermal Aging Deterioration of Material (Joint Seals, Gaskets, e / Deterioration of Material (Joint Seals, Gaskets, etc.) Mechanical and Thermal	Calibration Inspection - Internal Inspection - Visual Lubrication SRST - Functional Test SRST - Stroke Test Valve Diagnostics	<u>Incremental recommendations for Plant EOL (2020):</u> Resolve spare parts and obsolescence issues. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Degradation / Wear Mechanisms - Wear Corrosion / Fouling (accumulation of deposits) Radiation Induced Degradation / Radiation Depletion of Material Properties Mechanical and Thermal Degradation / Lubricant Degradation Corrosion / General Corrosion		
011443	5,6,7,8,056,078,058	71380 71380	71380 Emergency Water Supply (EWS) System-Non Return Valves - CAT1&2	The check valves prevent back flow in the EWS system, various applications include: ECI Recovery Heat Exchanger, EWS and recovery pump discharge, Instrument Air Supply to HT Loop etc.	Satisfactory	General Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / Flow Assisted Corrosion (Erosion-Corrosion)	Component Replacement Inspection - Internal Inspection - Non - Intrusive Overhaul/Refurbishment SRST - Functional Test	Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Initiate PMs for NVs in accordance with check valve strategy manual P-MAN-04946 00001 that have no PM practices.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		
011444	056,078,058	71380 71380	71380 - Emergency Water Supply (EWS) System - - PUMPS	The EWS recovery pumps are used to re-circulate water collected in the FM vault service sump back into the HTS and/or the Moderator system, as well as; supply water to the steam generators following BECS Injection, supply water to the fuelling machine vault air coolers, etc.	Poor	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> 1. Complete outstanding pump overhauls from EC 113142 that has two outstanding tasks (WO 01135894 to 056-71380-P1 to overhaul the pump and motor and WO 02278923 to overhaul 078-71380-P1). 2. Complete overhaul of 058-71380-P2 (WO#1135897) and 078-71380-P1 (WO# 2278923). <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011445	058	71380 71380	71380 - Emergency Water Supply (EWS) System - Travelling Screens	The travelling screens are used for removal of fine debris from the raw lake water for EWS.	Good	General Corrosion / General Corrosion Mechanical Fatigue / Mechanical Fatigue	Inspection - Visual Lubrication SRST - Functional Test	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> 1. Resolve issues to enable overhaul of screens (WO 1665303 - SC1, WO 1665305 - SC2). 2. Obtain critical spare parts for traveling screens to be used if required, this will ensure the screens can be replaced

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Microbiological Influenced Corrosion / Microbiological Influenced Corrosion Wear Mechanisms - Wear / Wear Mechanisms - Wear Corrosion / Fouling (accumulation of deposits)		quickly if failure occurs prior to end of life. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011446	058	71380 71380	71380 - Emergency Water Supply (EWS) System- Motorized Strainers	Lake water for EWS is strained through these items before it enters a common 30" header supplying 056 and 078 EWS loads.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear General Corrosion / General Corrosion Mechanical Fatigue / Mechanical Fatigue Microbiological Influenced Corrosion / Microbiological Influenced Corrosion Corrosion / Fouling (accumulation of deposits)	Inspection - Internal Inspection - Visual Lubrication SRST - Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		
011462	5,6,7,8,056,058	71380 67138	67138 - Emergency Water Supply (EWS) System - Temperature Transmitters - CAT 1&2	RTD element measures temperature of water in EWS piping exposed to weather or unheated areas, current output of transmitter feeds an alarm unit to back-up pipe trace heating control & annunciates in CR if low temperature is detected.	Satisfactory	Thermal Aging / Thermal Aging		<u>Incremental recommendations for Plant EOL (2020):</u> Procure: 1. Replacement transmitters. 2. Sufficient spares. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011470	5,6,7,8	71380 67138	67138 Emergency Water Supply (EWS) System- Power Transformers - 600V to 120/208V	Transformer is used to reduce voltage to acceptable levels for lighting and instrumentation power supplies.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Thermal Aging	Calibration Inspection - Visual Predictive Maintenance - Thermography SRST - Functional Test	<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Implement a one-time transformer inspection program, repair/replace if required.
011471	058	71380 54320	54320 Emergency Water Supply (EWS) System- Stand Alone Motor Starters- 4.16 kV	Motor starter serves as a switch to enable starting of 400HP, 4kV, Emergency Water System pump motor.	Good	Wear Mechanisms - Wear / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring,	Inspection - Internal Inspection - Visual Predictive Maintenance - Thermography	<u>Incremental recommendations for Plant EOL (2020):</u> Perform a one-time refurbishment of motor starters. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Seals, Gaskets, O-rings etc.)		
011483	5,6,7,8	71380	71380 Emergency Water Supply (EWS) System-Piping - Cat1&2	EWS piping provides a flow path to supply water to the steam generators, calandria, boiler room, fuelling machine vault air coolers, the ECIS recovery HX and the heat transport system (HTS).	Satisfactory	General Corrosion / General Corrosion Corrosion / Microbiological Influenced Corrosion	Inspection - Buried Piping Program	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011536	058	71380	71380 Emergency Water Supply Strainer Motors	The strainer motor is used to backwash the EWS strainer during flushing process to remove built up debris and sediment.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Thermal Aging		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete replacement of 058-71380-STRM2 under W/O 2935130. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).

TETRA TECH

System 0425 - EWS Intake Structure

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011458	058	21000	23910 - EWS Intake Structure-Foundations - Foundation Slab, Pile Caps	The reinforced concrete foundation slab serves as the floor slab and supports all the equipment in the EPWS building. It was designed to span one way to beams within the slab which in turns carry the floor loads to steel H-piles or to the pile caps then to the steel H-piles.	Good	Corrosion / Chemical Attack Mechanical and Thermal Degradation / Freeze-Thaw Corrosion of Embedded Steel / Corrosion of Embedded Steel Miscellaneous / Leaching Calcium Hydroxide Miscellaneous / Creep		Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Perform a detailed inspection of the underwater concrete foundation of EWS intake structure to ensure integrity.
011459	058	21000	23910 - EWS Intake Structure-Foundations - Steel H-Piles	The steel H-piles support the R/C slab or the pile caps, which in turn is designed to support the EPWS Building.	Good	Fatigue / Fatigue Vibration Corrosion / General Corrosion		Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Collect and analyze recent groundwater samples to validate the non-corrosivity of the area concluded by NK30-CORR-21000-0515625.
011463	058	21000	23910 - EWS Intake Structure-Structural Concrete - Concrete Walls and Slabs of Water Passages	The intended functions of the Emergency Water Pumphouse concrete structures include the following: 1. Channel take water to the EWS pumps. 2. Discharges water from the EWS pumps back to the intake channel. 3. Provide support/shelter for the EWS pumps.	Good	Corrosion / Chemical Attack Abrasion, Erosion and Cavitation / Abrasion, Erosion and Cavitation Corrosion of Embedded Steel	Inspection - Visual	Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Perform a one-time inspection of the concrete walls and slabs of water passages for EWS intake structure.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						/ Corrosion of Embedded Steel Leaching Calcium Hydroxide / Leaching Calcium Hydroxide		

System 0426 - Feedwater and Condensate System

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008895	1	43100	AMPLIFIER, Cat 1/2	Amplifier used in flow measurement circuit for feedheater drains system, takes input from flow meter & outputs an analog alarm signal to MCR.	Good	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Electronic Aging		Incremental recommendations for Plant EOL (2020): Review ability to adopt run-to fail strategy, or reset PM's to Active status. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
008897	1,4	43100 43110	BOOSTER, Cat 1/2	Boosts open/close signal to associated valve actuator.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Functional Test	Incremental recommendations for Plant EOL (2020): Review ability to adopt run-to-fail strategy, or reset PM's to Active status. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
008899	1,4	43100 64311	CONTROL (PRESSURE), Cat 1/2	Electronic controller, used to control main steam pressure to deaerator during start-up & poison prevent modes.	Good	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Electronic Aging		Incremental recommendations for Plant EOL (2020): Reset PM 126285 to Active status. Incremental recommendations for CO EOL (2024): Perform one-time replacement of electronic capacitor.
008901	1,4	43100 64321 64322	CONTROL(LEVEL), Cat 1/2	Digital controller for Deaerator Storage Tank level. Pneumatic controller for normal & emergency make-up to condenser.	Good	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Electronic Aging	Calibration Component Replacement	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete WO's 1709429 to 1709433 to replace controllers. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008905	1,4	43100	ELEMENT, Cat 1/2	Resistance temperature detector measures condensate temperature at designated locations & provides analog Input to digital computer.	Good	Mechanical and Thermal Degradation / Thermal Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008907	1,4	43100 43110 43130	EXPANSION JOINT, Cat 1/2	These expansion joints allow for thermal expansion/contraction at the inlet/outlet piping for the Low Pressure Feedheaters and for piping where rupture disc 4-43110-Y3000 is located.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion Fatigue / Thermal Fatigue Fatigue / Mechanical Fatigue Corrosion / Flow Assisted Corrosion (Erosion-Corrosion)	Surveillance	<u>Incremental recommendations for Plant EOL (2020):</u> 1. Perform one-time inspection of 1-43110-EJ5/6/8/10, and pending inspection findings repair/replace as required. 2. Consider acquisition of spares to address potential replacement depending on inspection result. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
008911	1,4	43100	GAUGE, Cat 1/2	These gauges are suction and discharge pressure gauges for Auxiliary Condensate Extraction Pumps.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008914	1,4	43100 43120	LOW PRESSURE	These Heat exchangers increase overall thermal efficiency of unit, and	Poor	Corrosion / Pitting Corrosion		<u>Incremental recommendations for Plant EOL (2020):</u>

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
			FEEDWATER HEATER	minimize thermal shock to boilers.		Corrosion / Fouling (accumulation of deposits) Corrosion / Stress Corrosion Cracking - Carbon and low alloy steels Mechanical and Thermal Degradation / Wear Mechanisms - Fretting		Perform one-time inspection of heat exchangers, replace if required. <u>Incremental recommendations for CO EOL (2024):</u> Implement new PMs for inspections to be conducted as per P-MAN-04660-10001 Heat Exchanger Strategy Manual with a frequency of 4 years. Replace if required.
008915	1,4	43100 43120	HIGH PRESSURE FEEDWATER HEATER	These heat exchangers heat feedwater and help increase thermal efficiency of a unit, and minimize thermal shock to boilers.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Fretting Corrosion / Pitting Corrosion Corrosion / General Corrosion Corrosion / Fouling (accumulation of deposits)	Clean and Inspect	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008923	1,4	43100 43220	POSITIONER, Cat 1/2, Fisher 3582	These positioners provide control valve position status,	Good	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Corrosion / General Corrosion Mechanical and Thermal Degradation / Thermal Aging		
008925	1,4	43100 43220	POSITIONER, Cat 1/2, Fisher 3570, Hi Temp	These positioners provide control valve position status.	Good	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion Mechanical and Thermal Degradation / Thermal Aging	Calibration Overhaul/Refurbishment Valve Diagnostics	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008931	1,4	43100 43130	REGULATOR, Cat 1/2	The valves in this CG regulate air pressure to condenser level controller, as well as regulate pressure for check valve with air actuator (4-43130-MNV59)	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008934	1,4	43100 43130 43220 64311	REGULATOR, Cat 1/2, regulates air supply pressure	These are pressure regulating valves for air operated control valves in the system.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring,	Calibration Valve Diagnostics	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
			to air operated valve,			Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		
008935	1,4	43100 64311 64313 64321	RELAY, Cat 1/2	Relays and timers used in control circuitry - for Deaerator / Condensate / Boiler Feed System	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging	Calibration SRST - Functional Test Thermography	Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Procure sufficient spares.
008938	1,4	43100 64311 64313 64321 64322	SWITCH, Cat 1/2	Component group comprises various different types of switches which provide control/ alarm function on flow, valve travel, water level & and system pressure.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Calibration Component Diagnostics Functional Test Overhaul/Refurbishment	Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Perform one-time inspection/calibration checks on equipment in this CG not covered under existing PM program.
008940	1,4	43100 64321	SWITCH, Cat 1/2	Hand Switches (HS) used for changing the operating state of equipment (ON/OFF or Auto).	Good	Mechanical and Thermal Degradation / Wear		Continue current practices of run-to–failure. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanisms - Wear		
008945	1,4	43100 64311 64321	TRANSMITTER, Cat 1/2	Level and Pressure Transmitter with Current Transducer for the Deaerator System	Good	Mechanical and Thermal Degradation / Electronic Aging	Calibration Inspection - Visual Overhaul/Refurbishment SRST - Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008951	1,4	43100 43130	VALVE, CONTROL, Cat 1/2, Schutte Koerting	Extraction steam from HP turbine is dumped into condenser via 4313-CV180 under certain conditions (e.g. turbine trip). Water collected from separators is dumped to condenser shell CD1 via CV181 below 50 percent machine load or turbine trip.	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Overhaul/Refurbishment Valve Diagnostics	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008952	1,4	43100 43220	VALVE, CONTROL, Cat 1/2, Fisher 480 actuator model + 9111 valve	These valves are air operated emergency condensate make up level control valves. These valves operate when the normal make-up system cannot maintain hotwell levels.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms – Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008953	1,4	43100	VALVE, CONTROL, Cat 1/2, Fisher 470 actuator model + ES valve	These air operated valves control the flow of start-up steam to the deaerator, to ensure the deaerator is at the proper pressure to prevent air ingress, and remove dissolved gases in the condensate.	Very Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008954	1,4	43100 43220	VALVE, CONTROL, Cat 1/2, Fisher 667 actuator model + AC or ES valve	These are air operated control valves, used to control level in Condensers, by rejecting surplus water to the Condensate storage tank 43220-TK1.	Very Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Ensure PMs for calibration, diagnostics, and actuator overhaul are activated and have authorized MWOs associated with PMs. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
008956	1,4	43100 45310	VALVE, MANUAL/HAND OPERATED, Cat 1/2	These are manual valves, for condensate storage tank drain isolation, isolation of gland supply from condensate storage tank, and for extraction steam instrumentation line isolation/drains.	Good	Corrosion / General Corrosion Corrosion / Flow Assisted Corrosion (Erosion-Corrosion)	Surveillance	<u>Incremental recommendations for Plant EOL (2020):</u> Perform one-time inspection for valves 1/4-45310-V22 (inspection to include valve internals), and if degraded repair/replace. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		
008958	1,4	43100 43110 43130	VALVE, MOTORIZED NON RETURN, Cat 1/2	These check valves prevent backflow of extraction steam from turbine, to heaters or deaerator.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / Flow Assisted Corrosion (Erosion-Corrosion)	Inspection - Internal Overhaul/Refurbishment	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008961	1	43100 43110	VALVE, CHECK/NONRETURN/BACK FL, Cat 1/2	These are air operated check valves for high pressure feedwater heaters extraction steam. Prevents backflow of extraction steam from turbines to HP feedwater heaters.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Inspection - Internal Inspection - Visual Overhaul/Refurbishment SRST - Logic Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Corrosion / General Corrosion		
008963	1,4	43100 43130 64321	VALVE, PRESSURE REGULATING, Cat 1/2	These pressure regulating valves regulate air pressure to actuators for check valves for extraction steam, and PRVs for L11-AX253 (DA storage tank IP transducers)	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms – Wear Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Calibration Inspection - Visual Overhaul/Refurbishment SRST - Functional Test SRST - Logic Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008965	1,4	43100 43150	VALVE, PRESSURE RELIEF, Cat 1/2	These RVs provide over pressure protection to the system (for low pressure feedwater heat exchangers, high pressure feedwater heat exchangers, and deaerator heat exchanger)	Good	Corrosion / General Corrosion Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Component Replacement Inspection - Visual	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Fatigue / Mechanical Fatigue		
008966	1,4	43100 64311 64313 64321	SOLENOID/SOLENOID OPERATED VAL, Cat 1/2	Solenoid valve are energized allowing non-return valve freedom to open and pass extraction steam.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging	Functional Test Inspection - Visual Overhaul/Refurbishment	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008968	1,4	43100 64321 64322	SOLENOID/SOLENOID OPERATED VAL, Cat 1/2, Associated with Valves in the Power Operated Valve Program	A three-way solenoid valve controls the air signal from the positioner to the valve diaphragm	Very Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging	Component Replacement SRST - Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011148	5,6,7,8	43100 43120	43120 Feedwater Heating Drains Coolers	The drain coolers, 43120-HX0A/HX0B, are provided to heat the condensate with the drains from the LP heaters (HX1A/B heaters)	Good	Mechanical and Thermal Degradation / Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanisms - Erosion Mechanical and Thermal Degradation / Wear Mechanisms - Fretting		
011149	5,6,7,8	43100 43120	43120 High Pressure Feedwater Heaters	These heat exchangers are used to pre-heat boiler feedwater using steam from turbine extraction steam and condensate from moisture separators.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Wear Mechanisms - Fretting Corrosion / Fouling (accumulation of deposits)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011150	5,6,7,8	43100 43120	43120 Low Pressure Feedwater Heaters	Condensate from the condensers hotwell it is pumped thorough the tube side of two 100% parallel banks of low pressure feed heaters. Shell side of LP heaters is supplied with extraction steam from the low pressure turbine.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Wear Mechanisms - Fretting Corrosion / Fouling (accumulation of deposits)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011151	5,6,7,8	43100 43130	43130 Heater Drains Pump Motors	Pump motor provides motive power to drive pump.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Thermal Aging		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete proactive one-time motor rewind activities which are currently in progress. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Increment recommendations for CO EOL (2024):</u> Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011152	5,6,7,8	43100 43210	43210 Condensate Extraction Pumps	Three 50% capacity main condensate extraction pumps (43210 -PI, P2, P3) are arranged to take condensate from three condenser shell hotwells and discharge it to the deaerator via the gland steam condenser and two banks of low pressure (LP) feedwater heaters. The pump's discharge is also used as a supply of condensate to the BFP glands and LP cylinder exhaust cooling system (41190).	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / Abrasion, Erosion and Cavitation Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011153	5,6,7,8	43100 43210	43210 Deaerator Level Control Valves	Deaerator level is controlled by three deaerator level control valves (DALCV) CV252, CV253 and CV254	Satisfactory	Corrosion / Pitting Corrosion Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms – Erosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011367	5,6,7,8	43100 43210	43210 Main Condensate Extraction Pump Motors-4kV	Drive motor for one of 3 x 50% duty Main Condensate Extraction pumps, which pump condensate from condenser hotwell to LP feedheaters.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Expedite motor refurbish cycle for remaining un-refurbished motors. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011405	5,6,7,8	43100 64311	64311 - Feedwater And Condensate System-SIGNAL CONDITIONER-CAT 1&2	This signal selector passes the highest signal from the pressure transmitters to the input of a pressure controller (64311-PC190).	Good	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Degradation / Electronic Aging		
011409	5,6,7,8	43100	Feedwater And Condensate System-Cable - 4.16 kV, 600V, 125V, 250V - Power Cables	Cables feed power from source to load.	Good	Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Inspection - Condition Monitoring Program	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011414	5,6,7,8	43100 43110 43210	43210 / 43110 Feedwater And Condensate System-Piping-Expansion Joints	The function of these expansion joints are to allow differential expansion of piping.	Satisfactory	Corrosion / Stress Corrosion Cracking - Nickel-base Alloys General Corrosion / General Corrosion Corrosion / Stress Corrosion Cracking - Austenitic Stainless Steels Fatigue / Thermal Fatigue Fatigue / Mechanical Fatigue	Component Replacement Inspection - Visual	<u>Incremental recommendations for Plant EOL (2020):</u> Procure one spare for each unique CAT ID with priority based on component criticality (i.e. CC1 and CC2). <u>Incremental recommendations for CO EOL (2024):</u> Implement a one-time inspection with priority based on component criticality (i.e. CC1 and CC2).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011415	5,6,7,8	43100 43210	43210 Feedwater and Condensate-Auxiliary Condensate Pump	Auxiliary condensate extraction pumps provide 5% of main condensate extraction pump capacity to supply sufficient feedwater flow to the auxiliary boiler feed pump when needed as a heat sink or for reactor cool down in the event of class IV power loss. Discharge from this pump also provides a backup supply of condensate for the Main Boiler Feed Pump glands.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Predictive Maintenance - Thermography Predictive Maintenance - Vibration Monitoring SRST - Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011416	5,6,7,8	43100 43130	43130 Feedwater And Condensate System-Heater Drains Pumps	The function of the heater drains pumps is to pump the condensate drain water from feedwater heaters HX5A and HX5B (also cascaded from HX6A & HX6B and the turbine moisture separators) to the deaerator to increase cycle efficiency.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011417	5,6,7,8	43100 43220	43220 Feedwater And Condensate System Vessels- Storage Tank	The primary function of the condensate storage tank is to provide sufficient head pressure for the condensate pump mechanical seals, debris filter bearing and instrument supply water and for water sealed valves. The condensate tank also provides swell capacity for the condensers and condensate system.	Good	Corrosion / Stress Corrosion Cracking - Carbon and low alloy steels Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011418	5,6,7,8	43100 43130	43130 Feedwater And Condensate System-Heater Drains Pumps HXs	Heater drains pump mechanical seals require a supply of clean water. The pump discharge, which is condensate, is used for the supply to the mechanical seals. This requires cooling of the hot condensate which is accomplished in the gland coolers. Service water is used to provide the exchange of heat from the gland water.	Satisfactory	Corrosion / General Corrosion Corrosion / Fouling (accumulation of deposits) Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011456	5,6,7,8	43100 64323	64323 - Feedwater And Condensate System -Signal Isolator -CAT 1&2	This signal isolator transfers the current generated by a transmitter measuring boiler level to a secondary instrument loop which	Good	Mechanical and Thermal Degradation / Electronic Aging		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete adding a task to replace signal isolators when active PM to perform loop calibration is executed (ref. CR2013-00974.).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				provides the input to alarm indicators.		Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

System 0428 - Fuel Transfer System

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
009056	012,034	35200 35270	CONVEYOR UNLOADER	The conveyor unloader transfers irradiated fuel bundles from the conveyor to the irradiated fuel storage baskets.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue Radiation Induced Degradation / Radiation Depletion of Material Properties	Functional Test Inspection - Visual	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: 1. Complete the implementation of EC 94748 to modify the IFB-A unloader gate cylinder linkage for 034 Conveyor Unloader. 2. Complete the following work orders: WO's 3249406 and 3249409 for proactive replacement of the unloader pulley bearings. WO 1832501 to install "basket in position" switch for 034 irradiated fuel receiving bay. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024):

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
								Complete one-time replacement of unloader bearings based on condition of bearings replaced in WO's 3249406 and 3249409.
009057	1,4,01 2,034	35200 35200	CYLINDER, Cat 1/2	NGS, NIS, NMS, NRS, NSS, NTS, NUS, NWS-M are spent fuel storage loader hydraulic cylinders. NJR, NKR, NLR-M are Fuel Transfer Mechanism (TM) air cylinders NVR-M is Fuel Transfer air cylinder for conveyor at elevator stops & switches	Poor	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion Radiation Induced Degradation / Radiation Depletion of Material Properties	Component Replacement	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete WOs: 4805164 for replacement of parker fitting/tubing on 4-35200-NLR-M1E and 2456198 to inspect/rebuild/replace conveyor stop cylinder. <u>Incremental recommendations for Plant EOL (2020):</u> Initiate proactive replacement of cylinders on (7 components) 012/034-35200-NGS-M1, 012-35200-NIS/NMS-M1, 4-35200-NVR-M3E, 1/4-35200-NVR-M3W. <u>Incremental recommendations for CO EOL (2024):</u> Complete a one-time replacement of (19 components) 012/034-35200-NGS/NIS/NMS/NRS/NSS/NTS/NUS/NWS-M1, 4-35200-NVR-M3E, 1/4-35200-NVR-M3W.
009062	1,4	35200 35240	ELEVATOR, Cat 1/2	Spent fuel is transferred from the fuelling machine via the transfer mechanism to the elevator. The Elevator carries the spent fuel down to the spent fuel Conveyor, which transports it through the Fuel Tunnel to the Receiving Bay.	Poor	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue Corrosion / General Corrosion		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> 1. Progress ECR# 21176 to modify the carriage. 2. Complete the following outstanding work orders: WO 3160528, 4940901, 4904024 to replace loose tie plate bolts. WO's 4855834 and 4855828 to replace the elevator chain connectors. WO 3001187 to replace the elevator gear box (U4E). WO 4789430 to develop tooling and replace the guide bracket bolts that are missing. WO 4901943 to investigate the issue causing 1-35240-ELEVATOR E to stall while running empty. 3. Conduct a more detailed inspection (resulting from

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
								WO#3067182) for U1W elevator to determine source of stalls. <u>Incremental recommendations for Plant EOL (2020):</u> Add inspection of elevator bottom sprockets to existing PM. <u>Incremental recommendations for CO EOL (2024):</u> Add inspection of elevator chains to existing PM.
009069	1,4,01 2,034	35200 35200 35290	HOSE, Cat 1/2	Catenary hoses allow movement while supplying fuel transfer system components such as the fuel transfer mechanism D2O fill/vent.	Satisfactory	Mechanical and Thermal Degradation / Cracking Due to Cyclic Loading Mechanical and Thermal Degradation / Thermal Aging	Surveillance	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete WO#3148365, 3149052, 3149053, 3149054, 3149055 and 3017012 to replace hoses. <u>Incremental recommendations for Plant EOL (2020):</u> Complete a one-time inspection of hoses which have not been replaced in the last 2 years. <u>Incremental recommendations for CO EOL (2024):</u> 1. Complete a one-time replacement of the D2O flex hoses and all other hoses in the CG. 2. Complete a one-time hose inspection 2 years after replacement.
009080	1,4	35200 35230	MECHANISM, Cat 1/2	Fuel transfer mechanism receives new fuel from the new fuel magazine and transmits it through the fuel transfer port to the fuelling machine. The fuel transfer mechanism then receives irradiated fuel from the fuelling machine and transmits it to the irradiated fuel elevator. The fuel transfer mechanism ram	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> 1. Complete EC (DCR) # 124483 for drawing changes to enable timely ram overhauls. 2. Complete ECR# 19333 to perform the TM U/L Ram position indication change. 3. Complete FH reliability plan work orders not yet done; WOs 2811775 (P1551), 2811812(P1671),

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				pushes the fuel in these operations.				2814749(P1671), 2811816(P1681), 2806566 (P1681), 2808315 (P1681), 2808335 (P1681), 2814794 (P1681). <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011500	5,6,7,8	35200 35230	35230 Fuel Transfer Mechanism Assembly (exclude Ram Assembly)	The Fuel Transfer Mechanism receives new fuel from the new fuel magazine and transmits it through the fuel transfer port to the fuelling machine. The Fuel Transfer Mechanism then receives irradiated fuel from the fuelling machine and transmits it to the irradiated fuel elevator.	Satisfactory	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Radiation Induced Degradation / Radiation Depletion of Material Properties Corrosion / General Corrosion Corrosion / Abrasion, Erosion and Cavitation Obsolescence / Immediate Obsolescence Concern		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete WO#2799773, 2799774, 2799776 to replace Ferguson components and WO#2914769, 2914767 and 2914768 to replace catenaries. <u>Incremental recommendations for Plant EOL (2020):</u> 1. Perform Stress Analysis Assessment of FT Head for pressure cyclic duty, similar to that performed for FM per NK30-REP-35310-00001. 2. Identify and resolve critical spares and obsolescence issues. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011505	5,6,7,8	35200 63524	63524 DC Motor Controllers and Torque Controllers	DC Motor Controller: A DC Motor Controller works in conjunction with a Shunt Wound DC Motor and an Isolation Transformer to make up an adjustable speed drive. These adjustable speed drives are used to provide the following functions: • the transfer mechanism traverse drives; • the transfer mechanism rotary drives; • the elevator drives; • the conveyor drives. Torque Controller: The Torque Controller works in conjunction with a Ferguson Drive providing brake and clutch operation via an external switch for the drive system.	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Obsolescence / Immediate Obsolescence Concern Fatigue / Fatigue due to Vibration Fatigue / Mechanical Fatigue	Surveillance	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Procure spares to facilitate prompt replacement if required. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Perform one-time inspection, and if degraded, repair/replace.
011507	5,6,7,8	35200 35210	35210 New Fuel Magazine Assembly	The new fuel magazine assembly receives new fuel from the new fuel loader and transfers it to the fuel transfer mechanism.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Radiation Induced Degradation / Radiation Embrittlement		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete WO#2799773, 2799774, and 2799776 for Unit 5, 7 & 8 Ferguson Clutch and Brake Replacement. Incremental recommendations for Plant EOL (2020): 1. Resolve spare parts issues. 2. Resolve obsolescence issues from Ferguson Drive Company. 3. Ensure a PdM is in place for yearly oil analysis of new fuel magazine/drives. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Obsolescence / Immediate Obsolescence Concern		
011520	5,6,7,8	35200 63520 63524	63520, 63524 Printed Circuit Boards (PCBs) and Connectors	Differential Voltage Comparator (315059): The comparator module is used to detect the position of the various F/H components such as separator feelers, retractors etc. PCBs have a comparator which receives voltages from the field POTs (fine and coarse). The field voltages are compared to the established setpoints. The voltage comparison is a way to monitor distances travelled by field equipment such the TM UL ram stroke, FT elevator elevation and FT mechanism position. These analog voltage comparator PCBs are connected to the slider of the coarse potentiometers and are used either as window comparator to detect whether a voltage input is within the range of adjusted setpoints in the card or to detect “out of synchronism” of normal and standby potentiometers. The outputs of relay on the board lights the associated status indicators (GI), which in turn indicate the position of the various F/H components such as	Poor	Mechanical and Thermal Degradation / Thermal Aging Corrosion / General Corrosion Corrosion / Oxidation Mechanical and Thermal Degradation / Electronic Aging Obsolescence / Immediate Obsolescence Concern Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Procure spares to facilitate prompt replacement if required. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Perform one-time inspection and functional test, and if degraded, repair/replace.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				separator feelers, retractors and etc. Digital Panel Extension PC Board (253514): The digital panel extension PC board extends the capabilities of the digital panel meter and is used to monitor the positions of the mechanisms in the fuel handling systems. The printed circuit board mates directly with the DPM and its terminals correspond to and represent an extension of the DPM terminals. The range of the meter is varied by means of coarse and fine trimmer potentiometers. The offset voltage of the incoming signal is altered by means of another trimmer potentiometer.				

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011521	5,6,7,8,056,078	35200 63524	63524 Low Voltage DC Power Supplies	Provides a highly stable output voltage for control circuits.	Poor	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Electronic Aging Obsolescence / Immediate Obsolescence Concern Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Procure spares to facilitate prompt replacement if required. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Perform one-time inspection, and if degraded, repair/replace.

System 0430 - Fuelling Machine Ancillaries

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011515	5,6,7,8	35300 63538	35380 FM Ram Force Calibration Assembly	The F/M calibration assembly is used to verify various forces on various components by the Fuelling Machine rams, remotely with a read out in the control room.	Good	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011518	058	35300 35630	35630, 63563 Basket to Module Fuel Transfer Assembly (Module Loader)	The Basket to Module Fuel Transfer Assembly, also referred to as "Module Loader assembly", is used to transfer the irradiated fuel bundles from an irradiated fuel basket into a tube of an irradiated fuel storage Module. The modules are used for storing the irradiated fuel bundles in the Irradiated Fuel Bay (IFB).	Satisfactory	Radiation Induced Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Incremental recommendations for Plant EOL (2020): 1. Prepare a specification for filter 058-35630-FR9. 2. Initiate a review of design discrepancies. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

System 0431 - Fuelling Machine Auxiliary Systems

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
009244	1,4	35300 35390	MOTOR, Cat 1/2	Pump motor provides motive power to drive the pump required to pump out D2O from the FM storage tank to the purification system.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Thermal Aging	Coupling Alignment Inspection - Visual	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete proactive replacement of PM3 under WO #3007230 (Unit 1) and #3007229 (Unit 4). Incremental recommendations for Plant EOL (2020): Procure new assembly (CID 674769). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
009249	1,4	35300 35390	POSITIONER, Cat 1/2	Valve positioners (NC1) accurately position valve stem based on an applied signal.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009254	1,4	35300 35390	PUMP, Cat 1/2	These pumps deliver heavy water from Fuel Machine (FM) storage tank 35390-TK5 to Heat Transport System. FM face seals D2O is collected into this tank. Based on tank level and FM demands, the pump is activated manually or automatically.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue	Coupling Alignment Lubrication	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Ensure existing WO's (1458455, 2811100) for overhaul/replacement are completed. Ensure spare pump parts and spare pump assembly procured and ready for replacement. Incremental recommendations for Plant EOL (2020):

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / High Vibrations		No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
009256	1,4	35300 35390	Regulator, Cat 1/2	These PRV's regulate instrument air pressure for their respective CV valves.	Good	Mechanical and Thermal Degradation / Creep and Stress Relaxation Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Component Diagnostics Inspection - Visual	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009263	1,4	35300 35390	SWITCH, PRESSURE, Cat 1/2	Pressure switches are an integral part of pump (P1/P2) and alarm at low pump lubrication oil pressure.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation /	Surveillance	<u>Incremental recommendations for Plant EOL (2020):</u> Complete a one-time proactive replacement of pressure switches (Unit 1 -WR 854456 & 854463 for PS1 & PS2 respectively) by December 2017. <u>Incremental recommendations for CO EOL (2024):</u>

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Wear Mechanisms - Wear Corrosion / General Corrosion		Add a task in existing PMs for control maintenance to calibrate PS's during pump overhaul.
009268	1,4	35300 35390	TRANSMITTER, Cat 1/2	These devices (AX's) convert an electrical control signal to a pneumatic signal that is used to operate associated control valves	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009271	1,4	35300 35390	VALVE, CONTROL, Cat 1/2	The high pressure supply pumps 35390-P1 and P2 maintain flow via a bypass line (these pumps are constant volume flow). These valves 35390-CV907 and CV944 are the bypass pressure control valves.	Good	Corrosion / General Corrosion Corrosion / Fouling (accumulation of deposits) Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Fatigue / Mechanical Fatigue	Overhaul/Refurbishment Valve Diagnostics	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
009279	1,4	35300 35390	VALVE, PRESSURE RELIEF, Cat 1/2	Fueling machine D2O supply system RVs provide overpressure protection. 35390-RV906 – Relief for pump P2 35390-RV914 – Relief for P1 35390-RV960 – Ajax pumps suction RV	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Fatigue / Mechanical Fatigue Corrosion / Stress Corrosion Cracking - Carbon and low alloy steels Corrosion / General Corrosion		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete existing PMs that have a history of PM deferrals. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
009281	014,0	35300 35610 35630 35670 35680	VALVE, AIFB PRESSURE RELIEF, Cat 1/2	014-35610-FLSK2000-RV1 and 014-35610-FLSK2001-RV1 are relief valves for the irradiated fuel transfer flask. 014-35680-RV2001 is the relief valve for AIFB flask nitrogen purge. 0-35630-RV4 is the pressure relief valve for the h20 module loader in the IFB basket to module fuel transfer system.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Creep and Stress Relaxation		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete EC 114894 to remove RV65 and RV5. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011501	5,6,7,8,058	35300	35390 FM D2O Supply System Piping	This CA covers piping, fittings, and piping supports for the FM D2O supply system. A brief description of the flow path is described below:	Satisfactory	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear	Inspection - Pipe Wall Thinning Program Inspection - Visual	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				Low pressure supply: D2O supply from PHT system is pressurized by Ajax pumps. High pressure supply: discharge from Ajax pumps is filtered, and delivered to either or both FM valve stations. FM magazine return: D2O is received from either or both FM magazines via the FM valve stations, and sent back to Ajax pump suction FM seals return: heavy water received from FM seals (via FM valve stations) is transferred to PHT system, and sent back to Ajax pump suction		Mechanisms - Fretting Mechanical and Thermal Degradation / Wear Mechanisms - Wear		
011502	5,6,7,8	35300 35390	35390 F/M D2O Supply System Heat Exchangers	5/6/7/8-35390-HX5 cools the flow to about 38°C (100°F) or less on the "bypass" (i.e. recirculation) line for D2O supply pumps 35390-P1/P2. The cooling water is taken from the Recirculated Cooling Water System. 6/7/8-35390-HX7 reduces the Fuelling Machine return D2O temperature from approximately 93°C (200°F) to less than 38°C (100°F). This flow then passes to either the suction of the PHT pressurizing pumps (33310-P1/P2), or to the suction of the D2O supply pumps (35390-P1/P2), NOTE: On Unit 5, 5-35390-HX7 has been removed.	Satisfactory	Corrosion / Microbiological Influenced Corrosion Corrosion / Crevice Corrosion Corrosion / Pitting Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> 1. Modify PM activity associated with 5/6/7/8-35390-HX5, to confirm integrity of shell side, as well as tube side (e.g. open valves supplying RCW to the shell side and confirm no leakage). 2. Complete W/O 2723295 - Remove and inspect one HX5 so that confidence is maintained that tubing integrity can meet end of life. <u>Incremental recommendations for Plant EOL (2020):</u> Complete a one-time confirmation of tube integrity for 6/7/8-35390-HX7. <u>Incremental recommendations for CO EOL (2024):</u> Implement a new PM for periodic confirmation of tube integrity for 6/7/8-35390-HX7, or implement a modification to remove these heat exchangers and replace with a spool piece (similar to what was done for 5-35390-HX7).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011503	5,6,7,8	35300 35390 63530 63536	35390, 63530, 63536 D2O Valves	High pressure supply from FM D2O supply pumps (35390-P1/P2) is maintained by controlling bypass flow of the pumps through pressure control valves 35390-CV944 and CV907. Control valve 63536-CV1 is used to control magazine pressure. Control valve 63536-CV2 automatically regulates the differential pressure between the ram supply line and the magazine at set point levels as required. Solenoid valves 63530-P1-SV1 directs and controls flow of air supply which controls the pressure of the fuelling machine magazine. Solenoid valves 63530-N1/2/3/19-SV1/2 are used to control the direction of the D2O flow which operates various components on the FM assembly. 63530-N19-SV1 controls advance or retraction of C-Ram 63530-N3-SV1 controls the separator stop which prevents any undesired movement of the fuel along the reactor tube. 63530-N2-SV1 controls the separator retractor which “pushes” one or a pair of fuel bundles into the fuelling machine magazine.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Fretting Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Calibration Overhaul/Refurbishment Valve Diagnostics	<u>Incremental recommendations for Plant EOL (2020):</u> Ensure spare parts for equipment in this CG are procured as part of FH 100 parts initiative to support run to failure maintenance strategy. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).

TETRA TECH

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				63530-N1-SV1 and SV2 both control the separator feeler which senses the gap between the fuel bundles during fuelling operations.				
011516	5,6,7,8	35300 35390	35390 FM D2O Supply System Strainers	Strainers 3539-STR2 and STR3 are installed at inlets of FM D2O supply pumps 35390-P1 and P2, and serve to protect the pumps from large debris/particles (these strainers removed under MEC 102220 and replaced with spool piece, due to strainer leaks causing D2O supply pump unavailability, replacement strainers are in process of being procured) Strainers 3539-STR4 strains the flow from the FM return line before it is sent back to PHT.	Poor	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Fatigue / Fatigue due to Vibration Corrosion / Fouling (accumulation of deposits)		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Pursue replacement strainers per FH 100 parts initiative. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Assess cost-benefit of proactively replacing strainers given failure trends.
011517	5,6,7,8	35300 35390	35390, 54130 FM D2O Supply Pump & Motor Assembly (Ajax Pumps) and Circuit Breakers	The F/M D2O Supply Pumps supply high pressure D2O for Fuelling Machine on-reactor operation. Only one pump is required to supply the high pressure D2O to both Fuelling Machines. During unit planned outages, both pumps are required to support the UDM operations for the SLAR campaigns.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> 1. Complete replacement of the pump leakage seals under Master NICR 131008. 2. Complete proactive replacement of pump motors (PM1 & PM2) under the following work orders (2705259, 2705262, 2705264, 2705266, 2705979 & 2706024) by 2018. <u>Incremental recommendations for Plant EOL (2020):</u>

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				In the event of failure of both PHT pressurizing pumps, an alternative delivery path for supply of low pressure D2O is ensured from the outlet of the F/M supply pumps. Pump motors (PM1 & PM2) provide motive power to drive the respective attached pumps		Degradation / Wear Mechanisms - Wear Radiation Induced Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Wear Mechanisms - Fretting Fatigue / High Vibrations		No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Implement a vibration monitoring, IR and oil sampling program for these pumps.
011519	5,6,7,8	35300 63530 63535	63530, 63535 FM Oil Hydraulic System Valves	The principal function of the oil hydraulic control systems is to actuate the various drives of the Fuelling Machine including the head, the supporting bridge and the carriage movements during the fuelling process.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear Radiation Induced Degradation / Radiation Embrittlement Fatigue / Mechanical Fatigue	Component Replacement	Incremental recommendations for Plant EOL (2020): Implement PMs for all tags without PMs in the CG. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

System 0432 – Fuelling Machine Carriages and Bridges

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
009284	1,4	35300 35330	ASSEMBLY, Cat 1/2	The Y drive provides vertical positioning for the fuelling Machine Carriage.	Good	Mechanical and Thermal Degradation / Lubricant Degradation Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Lubricant Contamination		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete all outstanding work associated with the FM Bridges Reliability Improvement Project. Incremental recommendations for Plant EOL (2020): Ensure set of spare ball nuts, ball screws, and y-drive gearboxes are available. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
009285	1,4	35300 35330	BEARING, Cat 1/2	The thrust bearings are located at the top of the y-drive ball screw assemblies and facilitate the y-directional movement of the bridge.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Lubricant Degradation Fatigue / Mechanical Fatigue		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete outstanding WO's 2456235 and 2456224 to replace 1-35330-THRUST BRG E and 1-35330-THRUST BRG W respectively. Incremental recommendations for Plant EOL (2020): Implement routine inspection of ball-screws and ball-nuts. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2020).
009287	1,4	35300 35330	BRIDGE, Cat 1/2	The Bridge supports the fuelling machine carriage to bring it to its desired position for loading and unloading fuel in the reactor channels and to pick up new fuel and discharge spent fuel.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal	Surveillance	Incremental recommendations for Plant EOL (2020): Ensure a set of spare ball nuts are available. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2020).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Degradation / Lubricant Degradation		
009288	1,2,3,4	35300 35320	CARRIAGE, Cat 1/2	The FM carriage function is to A) support the fueling machine and move it to a desired position; B) support the electrical and control conductors, and oil hydraulic and D2O hoses.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Lubricant Degradation		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Resolve obsolescence issues relating to 1/4-35320-CARRAIGE - Fine Y Bevel Gearbox CID 553949. (Added to the FH Top 100 Parts list). <u>Incremental recommendations for Plant EOL (2020):</u> Ensure a set of spare ball nuts are available. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
009289	1,4	35300 35320	DRIVE(GENERIC), Cat 1/2	The Z drive supports fueling machine actions by driving the fueling machine head to the desired position in the Z direction (horizontal and perpendicular to the X direction).	Good	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Lubricant Degradation		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011513	5,6,7,8	35300 35320 35321 35323	35320 FM Carriage, Trolley, Upper Gimbal and X Drive Assembly	The FM carriage function is to A) support the fueling machine and move it to a desired position; B) support the electrical and control conductors, and oil hydraulic and D2O hoses.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue		<u>Incremental recommendations for Plant EOL (2020):</u> Obtain sufficient spare part inventory necessary to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Lubricant Degradation Mechanical and Thermal Degradation / Lubricant Contamination		
011514	5,6,7,8	35300 35333	35333 FM Bridge Vertical Drive	The FM bridge drive mechanism enables the Fuelling Machine to access all the fuel channels for fuelling the reactor.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Obsolescence / Immediate Obsolescence Concern		Incremental recommendations for Plant EOL (2020): 1. Implement routine inspection of ball-screws and ball-nuts. 2. Obtain sufficient spare part inventory. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

System 0433 - Fuelling Machine Head

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
009292	1,4	35300 35316	GUIDE, Cat 1/2	The Guide Sleeve (GS) is inserted into the snout passage via Guide Sleeve Insertion Tool (GSIT) in order to facilitate the delivery of fuel bundles.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion		Incremental recommendations for Plant EOL (2020): Procure new assemblies (GS and GSIT) and resolve spare part issues. Incremental recommendations for CO EOL (2024): No additional practice are recommended to reach CO EOL (2024).
009293	1,4	35300 35313	MAGAZINE, Cat 1/2	The magazine assembly provides suitably shaped tubes for the storage of a variety of plugs, tools and fuel bundles.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Radiation Induced Degradation / Radiation Depletion of Material Properties	Calibration Inspection - Internal Overhaul/Refurbishment Supply Monitoring	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete replacement of FM Magazine Ferguson drive bearings and FM Magazine rear shaft bearings per outstanding Work Orders (WO # #2453453, 2453452, 2416636, 2416634). Incremental recommendations for Plant EOL (2020): Initiate proactive one-time replacement of the Ferguson drive assemblies at the end of expected design life of Ferguson drive assemblies. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
009295	1,2,3,4,014	35300 35314	RAM, Cat 1/2	The Ram assembly performs the function of opening and closing the fuel channels, and pushing the fuel bundles into the fuel channel.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring,		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Seals, Gaskets, O-rings etc.) Corrosion / Fouling (accumulation of deposits)		
009296	1,4	35300 35312	SEPARATOR, Cat 1/2	The separators assembly senses the position of gaps between fuel bundles, ram adaptor, or shield plug. It also prevents axial movement of the fuel column, and pushes fuel bundles through the last one half inch into the magazine tube.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009297	1,4	35300 35311	SNOUT, Cat 1/2	The snout assembly provides a means for accurately aligning the head with an end fitting. It also forms a leak-proof seal between the heavy water in the head housing and the reactor PHT system.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion Radiation Induced Degradation / Radiation Embrittlement		<u>Incremental recommendations for Plant EOL (2020):</u> Increase current PM frequency for Hydrodyne seal and O-ring replacement which is currently set at once every 4 years to address OPEX of recurring seal failures. <u>Incremental recommendations for CO EOL (2024):</u> Revise NA44-CALC-35310-00001 “Fatigue Analysis of PA FM Pressure Boundary Components” analysis to extend CO EOL from 2020 to 2024.
011504	5,6,7,8	35300 63530 63534	63530, 63534 Printed Circuit Boards (PCBs) and Connectors	These Printed Circuit Boards provide a variety of logic functionality for the operation of the Fuelling Machines (e.g. help control Ram acceleration).	Satisfactory	Mechanical and Thermal Degradation / Creep and Stress Relaxation Mechanical and Thermal		Continue current Practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Degradation / Thermal Aging Corrosion / General Corrosion		
011506	5,6,7,8	35300 63534	63534 Low Voltage DC Power Supplies	Provides a highly stable power supply required for operation of Fuelling Machines.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging		Continue current Practices. No additional practices are recommended to reach CO EOL (2024).
011508	5,6,7,8	35300 35310 35311 35315 35316	35310 FM Head Assembly	35310-FM HD ASSY E/W – Major component of fuel handling system, two operational heads for each reactor unit, one at each end of reactor (east & west). The fuelling machine heads are designed to fuel the reactor while it is at full power, or any level of power, including shutdown. 35316-GS TOOL EAST/WEST – Guide Sleeve Insertion Tool, used to move guide sleeve between the magazine and fuelling machine snout by the fuelling machine ram assembly using this tool. 35316-GS-E/W – Provides a smooth bore for the passage of fuel bundles and shield plugs between the magazine and the fuel channel. 35311-SNOUT E/W –The snout assembly forms an extension of the pressure boundary between the magazine housing and the channel end fitting, and contains the mechanisms	Satisfactory	Fatigue / Mechanical Fatigue Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Wear Mechanisms - Fretting Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Creep and Stress Relaxation		<u>Incremental recommendations for Plant EOL (2020):</u> Resolve spare / obsolescence issues. <u>Incremental recommendations for CO EOL (2024):</u> Revise existing analysis (Reports 30-35310-SR-001 Rev/ 01, 30-35310-ASD-001 Rev. 0) to demonstrate adequacy to CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				for clamping, locking and sealing the head to the end fitting. Snout assembly senses position of the end fitting and provides a positive clamp. 35315-SNOUT PLUG E/W – Function is to seal the fuelling machine head snout, while the head is in the “park” or “off-reactor” condition, to allow the head to be filled with D2O and pressurized.				
011509	5,6,7,8	35300 35310 35313	35310, 35313 FM Magazine DR Gearbox & FM Magazine Assembly	The Magazine assembly components and functions are: The Magazine Rotor is a twelve-chamber magazine rotor assembly to store a variety of plugs, tools and fuel bundles. Magazine D2O Flows provides cooling of the magazine housing and irradiated fuel bundles Magazine Shaft Seal maintains the pressure boundary of the magazine assembly. Magazine Drive System permits accurate alignment of the magazine sites with the snout bore.	Satisfactory	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Calibration Inspection - Internal Inspection - Visual Overhaul/Refurbishment	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> 1. Complete WO 04826430 for 7-35313-MAGAZINE W, to repair / replace a large Graylok seal leak. 2. Complete WO 02625577 for 7-35313-MAGAZINE W magazine seal replacement. <u>Incremental recommendations for Plant EOL (2020):</u> Resolve obsolescence and issues. <u>Incremental recommendations for CO EOL (2024):</u> Revise existing analysis (Reports 30-35310-SR-001 Rev. 01, 30-35310-ASD-001 Rev. 0) to demonstrate adequacy to CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Obsolescence / Immediate Obsolescence Concern		
011510	5,6,7,8	35300 35312	35312 Fuelling Machine Separator	The separators assembly senses the position of gaps between fuel bundles, ram adaptor, or shield plug. It also prevents axial movement of the fuel column, and pushes fuel bundles through the last one half inch into the magazine tube.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Incremental recommendations for Plant EOL (2020): Resolve spare part issues. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011511	5,6,7,8	35300 35314 35317	35314, 35317 FM Main Ram Assembly and Ram Adaptor	The ram assembly performs opening and closing of the coolant channels, and pushes the fuel bundles into the reactor. The ram adaptor centralizes the 'C' ram and prevents sagging while operating inside the fuel channel.	Poor	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear	Component Monitoring Overhaul/Refurbishment	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete AR#28153602-06 for new RAM ball screw seals. Incremental recommendations for Plant EOL (2020): Resolve obsolescence/spares issues. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanisms - Erosion Obsolescence / Immediate Obsolescence Concern		
011512	5,6,7,8	35300 35319	35319 FM Cradle Assembly	The cradle and gimbal support allows the FM head to pivot in both horizontal and vertical directions while engaging an end fitting.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue	Surveillance	Incremental recommendations for Plant EOL (2020): Resolve spare parts issues. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

System 0436 - HPECI Storage Tank

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011206	058	57439	33350 HPECI Storage Tank	The HPECI Storage Tank maintains water inventory required for high pressure injection following a loss of coolant accident.	Satisfactory	Corrosion / Environmental Degradation Corrosion / General Corrosion	Inspection - Visual	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011216	058	57439	33350 HPECI Storage Tank - Foundations, Steel	The steel H-piles support the storage tank foundation concrete.	Good	Corrosion / General Corrosion Fatigue / Environmentally-Assisted Fatigue	Sampling Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011217	058	57439	33350 HPECI Storage Tank - Foundations, Concrete	The reinforced concrete base supports the HPECI storage tank.	Good	Corrosion / Chemical Attack Corrosion of Embedded Steel / Corrosion of Embedded Steel Mechanical and Thermal Degradation / Cracking due to Expansion/Creep Fatigue / Environmentally-Assisted Fatigue	Sampling Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

System 0437 - HPECI Supply & Recirculation

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
009377	1,4	71260 63335	ALARM, Cat 1/2	The indicating alarm units provide HT pressure indication and alarm annunciation during ECIS HT pressure tests (channels E and F), which are mounted on the panel 66100-PL7C of the control room.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings, etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009379	1,4	71260 63335	TRANSMITTER, Cat 1/2	The pressure transmitter is to monitor boiler room ECIS piping pressure continuously by an indicator, 63335-P38-PIA, located on the CR panel 7C.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings, etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011169	058	71260 71260	71260 HPECI Storage Tank Fill & Recirculation Heat Exchanger	Maintains HPECI storage tank water temperature below the maximum temperature limit during hot summer months. Uses LPSW to cool recirculation system demineralized water.	Good	Corrosion / Pitting Corrosion	Surveillance	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011207	058	71260 71260 71310	71260 HPECI Supply & Recirculation-Valves - NV-Non-Return - CAT1&2	Check valves prevent reverse flow in the HPECI storage tank supply & recirculation circuit. Failure of these check valves could result in: -no temperature control of the storage tank. -reverse flow through non-operating centrifugal recirculation pumps 058-71260-P1 & P2. -unavailability to make up water to the storage tank -heat exchanger 058-71260-HX2000 over pressurization	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion	Surveillance	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Repair 058-71310-NV2140. <u>Incremental recommendations for Plant EOL (2020):</u> Perform one-time internal inspection of 058-71260-NV11. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).

TETRA TECH

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011208	058	71260 71260	71260 HPECI Supply & Recirculation-Piping - Expansion Joints	The primary function of these two expansion joints is to accommodate the thermal expansion/contraction between the HPECI tank and the embedded piping that runs underground to the ECI auxiliary services building.	Satisfactory	Corrosion / General Corrosion Fatigue / Mechanical Fatigue		Incremental recommendations for Plant EOL (2020): Procure spare expansion joints. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011209	058	71260 71260	71260 HPECI Supply & Recirculation Pumps	The two 100 percent duty pumps are provided to fill the HPECI Storage Tank with filtered service water and subsequently recirculate it on a continuous basis.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Abrasion, Erosion and Cavitation / Abrasion, Erosion and Cavitation		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Procure spares to facilitate prompt replacement if required. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011218	058	71260	71260, 71310 HPECI Supply & Recirculation-Cable - 4.16 kV, 600V, 125V, 250V - Power Cables	These cables are 1000 volt rated and are supplying 600V power to the two (2) ECIS Storage Tank Heaters (rated 100 kW each.) and the two (2) closed loop circulation pumps.	Good	Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Radiation Embrittlement	Inspection - Condition Monitoring Program	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011522	058	71260	71260 HPECI Supply & Recirculation-Piping - Cat1&2	The 4" and 6" carbon steel piping recirculate demineralised water which is part of the temperature control loop for the ECIS storage tank. The 2" stainless steel piping provides demineralised water from the plant to the storage tank building to	Satisfactory	Corrosion / General Corrosion	Inspection - Buried Piping Program	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				provide filling and make-up water to the storage tank.				

System 0439 - Instrument Air

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
009436	034	75120 75130	COMPRESSOR, Cat 1/2	The function of the compressor is to produce high pressure instrument air.	Good	Mechanical and Thermal Degradation / Lubricant Degradation Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		<u>Incremental recommendations for Plant EOL (2020):</u> Install physical barriers (directly above compressors) to prevent FME material from entering compressor intakes. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
009443	018	75120 67513	DISC, Cat 1/2	This rupture disc is for overpressure protection for tank 018-67513-TK2000	Satisfactory	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Inspection - Visual SRST - Functional Test	<u>Incremental recommendations for Plant EOL (2020):</u> Complete a one-time replacement of the rupture disc. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
009444	034	75120 75130	DRYER, Cat 1/2	The function of this component is to dry the air for the high pressure compressors (CP2044/CP2045/CP2046).	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete execution of approved ECs to replace the gearbox and motor of the air dryers. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
								Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
009446	034	75120 67513	ELEMENT, Cat 1/2	Generate and transmit signals representing instrument air dew points, which correspond to moisture levels.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Electronic Aging	Calibration	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete replacement of 034-67513-ME2142 as per WO#1975449. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Implement new PM for routine calibrations.
009449	1,4,03 4	75120 75130	FILTER, Cat 1/2	These components filter instrument air prior to final use.	Very Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Incremental recommendations for Plant EOL (2020): Initiate new PMs (annual filter replacement) for: 4-75130-FR3030/FR3031/FR3033/FR3034/FR3035/FR3037 and 1-75130/FR3070/FR3071/FR3072/FR3073/FR3074/FR3075. The quantities of spare parts should be reviewed and potentially increased to ensure the quantities are adequate to meet the requirements of the new PMs. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
009451	4,018	75120 67510	GAUGE, Cat 1/2	4-67510-PG2001/PG2002 provide local indications of the instrument air pressures at the central air cylinders stations. 018-67513-PG2012 provides local indication of the pressure of the instrument air supplied	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Implement new PM for routine calibrations.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				to 018-34220-MV49/MV2005.		Corrosion / General Corrosion		
009455	018	75120 67513	HOSE, Cat 1/2	These Flex Hoses are used to provide instrument air to the Vacuum Header Pressure CVs in the PRD Weather Enclosure.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Surveillance	Incremental recommendations for Plant EOL (2020): Perform a one-time inspection of the hoses to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
009460	034	75120 75130	MOTOR, Cat 1/2	Drive compressors and dryers for the Instrument Air System.	Poor	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Lubricant Degradation Obsolescence / Immediate Obsolescence Concern	Lubrication Predictive Maintenance - Electrical Testing Predictive Maintenance - Thermography Predictive Maintenance - Vibration Monitoring	Incremental recommendations for Plant EOL (2020): Implement a new PM to perform inspection and oil change for 034-75130-DRM2047/DRM2048/DRM2049 every year. Incremental recommendations for CO EOL (2024): Resolve spares issues for 034-75130-CPM2044/ CPM2045/ CPM2046.
009463	018	75120	PANEL, Cat 1/2	Mount instruments and associated isolation valves and tubing for the instrument air supply to 018-34220-MV49 and -MV2005 of the Negative Pressure Containment System.	Satisfactory	Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
009466	1,4,01 2,034	75120 75120 75130	RECEIVER, Cat 1/2	These receivers are used with the HP and LP instrument air compressors to provide a supply of instrument air.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024)
009467	1,4	75120 75130	REGULATOR, Cat 1/2	The pressure regulating valve provides specific pressure to an instrument air vessel (i.e. TK512, TK513).	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue	Calibration Overhaul/Refurbishment	Incremental recommendations for Plant EOL (2020): Perform a one-time overhaul of these pressure regulating valves. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
009469	012,03 4	75120 75130	RELAY, Cat 1/2	Implements part of the control logic for the Instrument Air compressors.	Poor	Mechanical and Thermal Degradation / Thermal Aging Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Obsolescence / Immediate Obsolescence Concern	Predictive Maintenance - Thermography SRST - Logic Test	Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): 1. Implement new PM for routine calibrations/tests. 2. Resolve obsolescence and spares issues.
009475	1,4	75120 75130	STATION, Cat 1/2	The stations are used for connections to and distribution of station instrument air.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Inspection - Visual	Incremental recommendations for Plant EOL (2020): Implement a TOV replacement program. Incremental recommendations for CO EOL (2024):

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		No additional practices are recommended to reach CO EOL (2024).
009477	1,4	75120 75130	STRAINER, Cat 1/2	These strainers are used to filter air used for the operation of actuators used for steam release valves.	Satisfactory	Corrosion / Pitting Corrosion	SRST - Functional Test	Incremental recommendations for Plant EOL (2020): Update maintenance procedures to inspect the strainers when SRV maintenance is being performed. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
009482	1,4,01 8	75120 67513 75130	TANK, Cat 1/2	These tanks are used for backup instrument air supply.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion	Inspection - Radiography SRST - Functional Test	Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Implement new PMs for internal/external inspection of 018-67513-TK2000, 1-75130-TK2001, 4-75130-TK2001, and 1-75130-TK2002.
009485	034	75120 75130	TRAP, Cat 1/2	These traps are used with the air receiver filer to contain particulates.	Very Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Overhaul/Refurbishment	Continue current practices. No additional practices are recommended to reach CO EOL (2024)
009488	1,012	75120 75130	VALVE, CONTROL, Cat 1/2	The function of the control valve 1-75120-CV3033 is to tie-in LP instrument air to pneumatic devices on the vacuum building emergency water storage system, function of 012-75130-CV3044 is tie-in of HP and LP instrument air.	Satisfactory	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue	Functional Test	Incremental recommendations for Plant EOL (2020): Implement a one-time inspection and/or overhaul program for these control valves. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
009490	1,034	75120 75120 75130	VALVE, ISOLATION, Cat 1/2	The function of these valves are to isolate instrument air for station 2094/2095 or for commissioning and testing purposes.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Surveillance	Incremental recommendations for Plant EOL (2020): Implement one-time inspections/replacements of these isolation valves. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
009499	1,4,01 2,034, 018	75120 67513 75100 75120 75130	VALVE, PRESSURE RELIEF, Cat 1/2	These RVs provide over pressure protection to the Instrument Air system.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009505	1,4	75120 75100	CYLINDER, Cat 1/2	These cylinders are used to supply back up instrument air.	Satisfactory	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion	SRST - Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011198	5,6,7,8 ,058	75120 67512 75120	75120 Instrument Air System Dryers	Dryer function is to dry instrument air to an outlet dewpoint of -40C. Changeover valve function is to enable switch-over between dryer chambers. Solenoid valve function is to control dryer purge/changeover functions.	Good	Corrosion / General Corrosion Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Obsolescence / Immediate Obsolescence Concern	Calibration Functional Test Inspection - Visual	Incremental recommendations for Plant EOL (2020): Obtain sufficient spares for Instrument Air CV's. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011199	5,6,7,8	75120 75120	75120 Instrument Air Check Valves - Airlocks	Prevents backflow of instrument air supply.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion		Complete WO 2871081 to replace 7-75120-NV1505. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011200	5,6,7,8	75120 75120	75120 Instrument Air Compressors	Instrument Air Compressors provide clean, dry air at pressure (~800 kPa) to plant Instrumentation and Control pneumatic equipment.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / Fouling (accumulation of deposits) Fatigue / Fatigue due to Vibration Mechanical and Thermal Degradation / Wear Mechanisms - Fretting Obsolescence / Immediate Obsolescence Concern Mechanical and Thermal Degradation / Lubricant Contamination		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> 1. Obtain replacement items and procure sufficient quantities for CIDs 221426,186292, 184473, 553410, to ensure timely repairs. 2. Complete compressor replacement program (e.g. WO 4937597). <u>Incremental recommendations for Plant EOL (2020):</u> 1. Review and reinstate the following PMs (sample PM 12564): • Calibrate Panel pressure gauge PG1601 (yearly) • Perform Vibration Analysis (quarterly). • Collect Compressor Performance Data (weekly). • PM 112966 Oil samples every 13 weeks 2. Perform a one-time maintenance service (reference sample retired PM 12564) for the following: • Calibrate Panel temp. Gauge at 10,000 hrs • Grease Compressor Motor Bearings at 10,000 Hour Service • PM Electrical Testing (Off-Line) (sample PM 117067) <u>Incremental recommendations for CO EOL (2024):</u> Review and reinstate the following PMs (sample PM 12564):

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
								• Calibrate Panel temp. Gauge at 10,000 hrs (every 2 years) • PM Electrical Testing (Off-Line) every 2 years (sample PM 117067 Perform a one-time maintenance service (reference sample retired PM 12564) for the following: • Grease Compressor Motor Bearings at 10,000 Hour Incremental Service
011202	5,6,7,8	75120 75120	75120 Instrument Air Manual Valves Inside Containment	These Isolation valves are used to meet instrument air (IA) requirements inside each Reactor Building (RB).	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Radiation Induced Degradation / Radiation Embrittlement	Component Replacement SRST - Functional Test	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete current diaphragm valve replacement campaign. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011220	018	75120	75120 Instrument Air - Cables 4.16 kV, 600V, 125V, 250V	Provide power to the motors, heaters and dryers of Instrument Air System.	Good	Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue	Inspection - Condition Monitoring Program	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011226	058	75120 75120	75120 Instrument Air Pressure Regulating Valves	Provide pressure regulation for various air cylinders providing back-up air supplies.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Obsolescence / Immediate Obsolescence Concern	Surveillance	<u>Incremental recommendations for Plant EOL (2020):</u> Resolve obsolescence by procuring an adequate replacement PRV. <u>Incremental recommendations for CO EOL (2024):</u> Complete a one-time replacement of all PRVs.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011227	5,6,7,8 ,056,0 78,058 ,018	75120 75120	75120 Instrument Air Check Valves - Excluding Airlocks	These check valves are used to maintain back up air supply to critical equipment.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Component Replacement Inspection - Non - Intrusive SRST - Functional Test	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete work orders 02906169, 02906171, 02964221, 02908772, 02964232, 02964234, and 02723248, to inspect and replace specified check valves. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

System 0440 - Irradiated Fuel Bay Auxiliaries

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
009511	0	34400 35710	FILTER, Cat 1/2	Two shielded cartridge filters (each designed for 50% duty) remove suspended solids down to 10µm from the bay water for cooling and purification.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Fretting Corrosion / Fouling (accumulation of deposits)		Incremental recommendations for Plant EOL (2020): Reactivate PMs for filter replacement (114597-01, 114598-01). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
009514	0	34400 35710	HEAT EXCHANGER, Cat 1/2	These two HXs remove heat from re-circulated bay water.	Satisfactory	Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Corrosion / Fouling (accumulation of deposits) Corrosion / Microbiological Influenced Corrosion Corrosion / Pitting Corrosion		Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Initiate new PM tasks to clean HX and conduct Eddy Current Testing (ECT) on HXs every 4 to 5 years per Equipment Strategy Manual.
009521	0	34400 35710	PUMP, Cat 1/2	Two 100% duty horizontal end-suction centrifugal pumps re-circulate bay water through a train of filters, ion exchangers and heat exchangers for cooling and purification.	Good	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Wear Mechanisms – Wear		Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Complete a one-time external inspection.
011127	058	34400 34410	34410 IFB Cooling Heat Exchanger	Three 50% duty HXs remove heat from circulated bay water.	Good	Corrosion / General Corrosion Corrosion / Microbiological Influenced Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Corrosion / Pitting Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Fretting Corrosion / Fouling (accumulation of deposits) Corrosion / Stress Corrosion Cracking - Austenitic Stainless Steels		
011423	058	34400	34400 Irradiated Fuel Bay Auxiliaries- Cable - 4.16 kV, 600V, 125V, 250V - Power Cables	The cables used for P014 can be sub-divided into five (5) basic categories. There are 5 kV power cables, 600 V power cables, 600 V control cables, 300 V control cables and special definite purpose cables.	Good	Fatigue / Thermal Fatigue Fatigue / Mechanical Fatigue Radiation Induced Degradation / Radiation Embrittlement	Inspection - Condition Monitoring Program	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011447	058	34400 73170	73170 Irradiated Fuel Bay Ventilation - Motor Operated Dampers - CAT1&2	These components are air control dampers in the spent fuel bay ventilation system. MDP3 is an inlet isolation damper for filter FR501. MDP4 is a by-pass damper for filter FR501.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms – Wear		<u>Incremental recommendations for Plant EOL (2020):</u> Resolve spare parts issues. Specifically, there are no spare parts for 058-73170-MDP3/4 (actuator or damper). Recommend procuring P.V.C. flexible damper and jamb seals, damper nylon bearings, linkage brass bearings and actuator elastomer seals. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011457	058	34400 34410 34420 71310 71620	34410, 71310 Irradiated Fuel Bay Auxiliaries- Valves - Manual - CAT 3&4	These valves are isolators for heat exchangers 058-34410-HX1, 2 & 3. The isolators are required to allow inspection / repairs of the heat exchangers	Good	Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Corrosion / Microbiological Influenced Corrosion		<u>Incremental recommendations for Plant EOL (2020):</u> Resolve spare parts issue. Specifically, recommendation for 058-34410-V25-28, 31 & 33 is to purchase spare valves (Cat ID 148575).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

System 0441 - Irradiated Fuel Bays

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
009532	0	34400 63440	ALARM, Cat 3/4	Alarm units provide flow/temperature indication and alarm annunciation for the purification and cooling circuits.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009533	0	34400 63440	ELEMENT, Cat 1/2	Temperature elements are used to monitor recirculated bay water temperature for the purification system.	Good	Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Radiation Induced Degradation / Radiation Depletion of Material Properties		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009536	0	34400 34420	FILTER, Cat 1/2	Filters remove recirculated bay water suspended solids down to 40µm from the recirculated bay water upstream of ion exchangers in the purification and cooling circuit.	Poor	Mechanical and Thermal Degradation / Wear Mechanisms - Fretting Corrosion / Fouling (accumulation of deposits)		<u>Incremental recommendations for Plant EOL (2020):</u> Reactivate PMs for filter replacement (114602-01 / 114603-01). <u>Incremental recommendations for CO EOL (2024):</u>

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
								No additional practices are recommended to reach CO EOL (2024).
009539	0	34400 34410	HEAT EXCHANGER, Cat 3/4	These 3 HX cool recirculated bay water.	Good	Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Corrosion / Fouling (accumulation of deposits) Corrosion / Microbiological Influenced Corrosion Corrosion / Pitting Corrosion Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Fretting Corrosion / Stress Corrosion Cracking - Austenitic Stainless Steels		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009541	0	34400 63440	INDICATOR, Cat 1/2	Temperature indicator is used to provide temperature indication of the bay water purification circuit.	Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Radiation Induced Degradation / Radiation Depletion of Material Properties		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011318	058	34400	21500 – IFB-(Irradiated Fuel Bay)	The Irradiated Fuel Bay (IFB-B):	Good	Mechanical and Thermal Degradation / Cracking Due to	Inspection - Leak Rate Monitoring	<u>Current initiatives that need to be completed for Plant EOL (2020) that are</u>

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				- Stores and cools fuel from all 4 P058 reactors. - Provides transparent shielding to allow storage and safeguarding operations to be carried out conveniently and safely. - Protects the environment from any leakage of water.		Expansion or Contraction Mechanical and Thermal Degradation / Shrinkage Radiation Induced Degradation / Radiation Embrittlement		<u>incremental to current periodic maintenance practices:</u> Complete repairs to liner cracks under Project 13-40703. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).

System 0444 - Liquid Zone System

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
009613	1,4	34800 34810	COMPRESSOR, Cat 1/2	These compressors provide helium cover gas for LZC system. Suction is taken from the Delay Tank and discharge is sent to the Helium Storage Tank. The compressors maintain pressure of the Helium Storage tank at set point.	Satisfactory	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue	Lubrication Inspection - Internal Predictive Maintenance - Vibration Monitoring	Incremental recommendations for Plant EOL (2020): Address compressor seal flow conditions through MV31/33. This is a known cause for compressor degradation. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
009631	1,4	34800 34810	PUMP, Cat 1/2	These 100 percent duty pumps provide flow for the closed circuit demineralized water for the LZC system.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Incremental recommendations for Plant EOL (2020): Add lubrication/oil sampling to super route PM for this CG. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
009654	1,4	34800 34810	VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2	1/4-34810-MV105 provides emergency water supply from the Helium Storage tank to the Helium Supply Header when header pressure falls below 80 psi. 1/4-34810-MV15 is normally open, and fails closed to prevent the zones from draining.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009659	1,4	34800 34810	SOLENOID/SOLENOID OPERATED VAL, Cat 1/2	Solenoid valves are used to actuate valves 34810-MV105 & 34810-MV15. The water supply header distributes water from the supply pumps to the fourteen valves controlling the flow each into the zone compartments. Should the pressure fall due to pump or motor failure, pipe fracture or any other circumstance,	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation		Continue current practice. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				the change in pressure is detected by pressure switches. The low pump discharge pressure will cause closure of valve 34810-MV15 and opening of valve 34810-MV105.		/ Wear Mechanisms - Wear Radiation Induced Degradation / Radiation Depletion of Material Properties Mechanical and Thermal Degradation / Thermal Aging		
011129	5,6,7,8	34800 34810	34810 Liquid Zone Control Compressor Motors	The LZC CP Motors drive one of two 100% duty helium compressors. During normal operation one is selected on Auto 1 while the other selected to Auto 2. The duty compressor (Auto 1) maintains storage tank pressure by operating in On/Off mode. If a compressor fails, the standby compressor (Auto 2) is selected to Auto 1 duty and represents a total loss of redundancy. Failure of both LZC compressor motors would result in the loss of helium storage tank pressure and likely a forced outage.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Radiation Induced Degradation / Radiation Depletion of Material Properties Mechanical and Thermal Degradation / Cracking Due to Cyclic Loading	Component Replacement Lubrication Predictive Maintenance - Vibration Monitoring	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete CR2012-01045 to re-instate PM for motor replacement every 4Yr. Incremental recommendations for Plant EOL (2020): Implement proactive replacement of compressor motors to address concerns regarding premature winding failures (ref. SCR P-2012-16220). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011130	5,6,7,8	34800 34810	34810 Liquid Zone Control Coolers	These Heat Exchangers cool the recirculated demineralized LZCS water.	Good	Corrosion / Pitting Corrosion Corrosion / Fouling (accumulation of deposits) Fatigue / Thermal Fatigue		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Based on OPEX showing poor condition, execution of PMs should not be deferred without detailed consideration. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020).

TETRA TECH

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
								Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011131	5,6,7,8	34800 34810	34810 Liquid Zone Control H2O Supply Valves	These valves are used to control the level in each of the 14 zone control compartments for the purpose of spatial flux control. These control valves fail open, filling the compartments.	Good	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Overhaul/Refurbishment SRST - Functional Test Valve Diagnostics	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Continue replacing positioners with smart positioners as per ECR 15052. Incremental recommendations for Plant EOL (2020): Address vendor quality issues for Units 1&4 valves leaking through the O-rings Incremental recommendations for CO EOL (2024): Perform a one-time overhaul of the U6 actuators to reach CO EOL (2024).
011132	5,6,7,8	34800 34810	34810 Liquid Zone Control Return Header Isolating Valves	These valves are air to close, spring to open. These valves close and isolate zone outlets in case of low H2O supply header pressure (due to loss of supply pumps) thus preventing zones from draining.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion Corrosion / Pitting Corrosion	Overhaul/Refurbishment SRST - Functional Test SRST - Stroke Test	Incremental recommendations for Plant EOL (2020): Follow-up on BOM for MVs and ensure spare parts procured. Incremental recommendations for CO EOL (2024): Re-initiate PMs for diagnostics every 4 years e.g. PM 97834-01.
011374	5,6,7,8	34800 34810	34810 Liquid Zone System-Helium Bubbler & Balance Header Pressure And Storage Tank Level Control Valves	These Air Operated Valves have various functions: CV53, CV71, CV64, CV70, CV68 and CV69 are used to control the Helium Bubbler Header and Helium Balance Header pressure at set point to ensure continuous bleed to bubbler tubes for zone level measurement. CV171 is used to control Helium Storage Tank level	Poor	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Erosion	Overhaul/Refurbishment SRST - Functional Test Valve Diagnostics	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete outstanding work orders to proactively replace the Bubbler Header Control Valves and the Back-up Balance Header Control Valves. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
								Incremental recommendations for CO EOL (2024): Implement one-time internal inspections of valves.
011438	5,6,7,8	34800	34810, 63480 Liquid Zone System-Cable - 4.16 kV, 600V, 125V, 250V - Power Cables	The cables used for Pickering GS B can be sub-divided into five (5) basic categories. There are 5 kV power cables, 600 V power cables, 600 V control cables, 300 V control cables and special definite purpose cables.	Good	Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue Radiation Induced Degradation / Radiation Embrittlement	Inspection - Condition Monitoring Program	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011453	5,6,7,8	34800 34810	34810- Liquid Zone System-Non-Return Valves - CAT 2	These check valves are used to prevent reverse flow through the LZ pumps (34810-NV16, NV55, NV58), maintain an emergency water inventory in the Helium Storage Tank (34810-NV225) and prevent backflow from the H20 Supply Header to the Helium Storage Tank (34810-NV37).	Good	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / Flow Assisted Corrosion (Erosion-Corrosion)	SRST - Functional Test	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Continue to execute recommended actions from previous CA NK30-REP-34810-00012 R002 i.e. procure spares for NV225, one-time internal inspection of one of 5-8-34810-NV225 (see AR 1646265, WO 4814904), and inspect one of NV16, NV55, NV58 from U7 due to these valves being in service the longest. Incremental recommendations for Plant EOL (2020): Complete one-time inspection of 7-34810-NV37. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011465	5,6,7,8	34800 34810	34810 - Liquid Zone System-Tanks	The delay tank 34810-TK1 provides a delay time of approximately 5 minutes for return flow from components, to allow for decay of O-19 and N-16. The storage tank TK2 is sized to handle two fill/drain cycles of the zones between on/off cycles of the	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	SRST - Functional Test Tank Level Monitoring	Incremental recommendations for Plant EOL (2020): Implement periodic tank inspections. Incremental recommendations for CO EOL (2024): Depending on inspected condition, establish procurement route to purchase a new tank.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				compressors. This tank also receives compressor liquid ring seal and mechanical seal coolant flow. 34810-FA1/2 are flame arrestors upstream and downstream of recombination unit. They are designed to prevent any flame propagation from the recombination unit into parts of the circuit.		Fatigue / Mechanical Fatigue Radiation Induced Degradation / Radiation Embrittlement		
011474	5,6,7,8	34800 34810	34810 Liquid Zone System-CONTROLLER -HAND-CAT 1&2	The function is to provide a bumpless transfer from Auto to Manual control of the Liquid Zone Level control valves and also a bumpless transfer from Manual to Auto. The normal control signal is via the DCC computer and through the current generator to the zone CV I/P transducer (AUTO mode).	Good	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging Obsolescence / Immediate Obsolescence Concern		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Maintain spare part inventory as required per AR#28117445. Incremental recommendations for Plant EOL (2020): Perform one-time replacement of all hand controllers with new or refurbished units. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

System 0445 - Moderator System

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
008250	1,4	31200 63241 63243	ALARM, Cat 1/2	Flow indicating alarm meters, located in the Main Control Room panel, provide indication and low flow annunciation for the Calandra/Dump Tank spray.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging	Calibration Component Replacement	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008252	1,4	31200 63241	CAPACITOR, Cat 1/2	These capacitors provide electronic filtering within the Selector Switches which are part of the Calandria Dump Tank logic.	Good	Mechanical and Thermal Degradation / Electronic Aging	Calibration Component Replacement	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008253	1,4	31200 63241	CONTROL, Cat 1/2	Flow indicating controller for the Calandria Spray system.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging	Calibration	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
008254	1,4	31200 63241	CONVERTER, Cat 1/2	Current to current converters for the Calandria Spray system.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation	Calibration Component Replacement	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						/ Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging		
008257	1,4	31200 63241 63243	TRANSMITTER, Cat 1/2	These flow transmitters convert process flow parameters to an analog current signal for use in the Calandria & Dump Tank control logic.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Creep and Stress Relaxation Radiation Induced Degradation / Radiation Embrittlement	Calibration Component Replacement	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009664	4	32000 32210	Actuator, Cat 1/2	MV15 isolates the moderator purification system outlet from the moderator pump suction headers. The motorized valve is normally kept open when moderator poison removal Is required, otherwise the MV is closed.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Creep and Stress Relaxation		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
009666	1, 4	32000 32310	Actuator, Cat 1/2, Moderator Dump Valves	These actuators allow for control of the Moderator Cover Gas System Dump Valves. Under accident conditions, the Moderator Dump Valves must not open spuriously for a mission time of five (5) minutes and then they must open on demand and remain open for up to thirty (30) minutes.	Very Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation - Creep and Stress Relaxation Fatigue - Mechanical Fatigue		<u>Incremental recommendations for Plant EOL (2020):</u> Procure spare actuator/parts. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
009667	1, 4	32000 32310	Actuator, Cat 1/2, Large Moderator Level CV	These valves are the moderator level regulating CV's which provide coarse level control between the dump tank and Calandria cover gas.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Creep and Stress Relaxation		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete work orders WO 1505860 and WO 1505859 to overhaul actuators. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
009668	1, 4	32000 32310	Actuator, Cat 1/2, Small Moderator Level CV	The actuators control the small moderator level regulating control valves.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Creep and Stress Relaxation		
009669	1, 4	32000 32310	Actuator, Cat 1/2, Helium Storage Tank CV	The actuators control valves that regulate the flow of helium to the dump tank.	Good	Mechanical and Thermal Degradation / Creep and Stress Relaxation Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue		Continue current practice. No additional practices are recommended to reach CO EOL (2024).
009670	1, 4	32000 32510	Actuator, Cat 1/2, Mod D2O Collection	The pneumatic actuators operate the ECI Recovery Injection Drain Motorized Valve MV43. MV43 is an isolation valve from ECI D2O recovery to moderator D2O collection. The valves are normally closed, and facilitate testing the operation of valves 33350-CV450, 33350-MV471, and 33350-MV571.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Creep and Stress Relaxation		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009671	1, 4	32000 32110	ACTUATOR, Cat 1/2	CV 7 & 8 are the dump tank outlet control valves. They maintain level by balancing inflow with outflow in the dump tank. CV 2, 4, 26, 27 are the Calandra outlet valves and	Poor	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Overhaul/Refurbishment SRST - Functional Test SRST - Logic Test Valve Diagnostics	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> 1. Complete work orders to perform one-time valve and actuator replacement/overhaul: WO

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				together with the Calandria inlet valves are used to modulate and balance the inflow with the outflow. MV9 &10 are the motorized dump tank outlet valves. They are opened during moderator pump-up and closed to prevent draining of the dump tank. CV48, 50, 54, 55 are used to control the spray flow to the Calandria by regulating the Calandria inlet flow.		Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Creep and Stress Relaxation		04976469, 04976480, 04976489, 4967469, 04974812, and 1677304. 2. Complete outstanding work order 03175399 "4-32110-CV27 Replace Calandria Outlet Actuator / Valve". <u>Incremental recommendations for Plant EOL (2020):</u> 1. Ensure that there are adequate spare parts for the moderator outlet valve actuators. 2. Review EQA N-EQA-04944-00036 for the CV actuators and complete a one-time replacement of components which have not been replaced for 30 years. 3. Review EQA NA44-EQA-04944-00009 for MV9 and complete a one-time replacement of components which have not been replaced for 40 years. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
009672	1, 4	32000 32110	ACTUATOR, Cat 1/2	These actuators operate the dump tank outlet valves. They are used to pump back D2O from the dump tank to the Calandria.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion Mechanical and Thermal Degradation / Lubricant Degradation Mechanical and Thermal Degradation / Wear Mechanisms - Wear		<u>Incremental recommendations for Plant EOL (2020):</u> Replace actuators 1/4-32110-MV9-ACT1/MV10-ACT1 prior to 2019. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
009673	1, 4	32000 63210	Level Switch, Cat 1/2	Generate alarms when the Calandria moderator levels	Good	Mechanical and Thermal Degradation / Thermal Aging	Calibration	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				exceed predefined setpoints.		Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		
009674	1, 4	32000 63210	Alarm, Cat 1/2	Provide indications of moderator dump tank levels and generate alarms when the measured levels exceed or drop below predefined setpoints.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete proactive replacement per WO#2073795 and WO#2073796. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
009675	1, 4	32000 63210	Alarm, Cat 1/2	Provide contact inputs for control or alarm when the moderator dump tank levels exceed predefined setpoints.	Poor	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Incremental recommendations for Plant EOL (2020): Complete proactive replacement of 1/4-63210-L7-LS1/LS3. Incremental recommendations for CO EOL (2024): Perform periodic calibrations on 1/4-63210-L7-LS1/LS2.
009676	1, 4	32000 63210 63220	Alarm, Cat 1/2	1/4-63210-P1-PIA1 provides indication of the moderator pump suction pressure and generate an alarm when the	Very Poor	Mechanical and Thermal Degradation / Thermal Aging		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				pressure drops below predefined setpoint. 1/4-63220-P1-PIA1 provides a contact input for generating an alarm when the moderator Ion Exchanger differential pressure exceeds the predefined setpoint.		Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear Obsolescence / Immediate Obsolescence Concern		1. Replace 4-63220-P1-PIA1 per WO#4724999. 2. Complete proactive replacement of 4-63210-P1-PIA1 per WO#1549255. Incremental recommendations for Plant EOL (2020): Resolve obsolescence and spares issues for 1/4-63220-P1-PIA1. Incremental recommendations for CO EOL (2024): No additional recommendations required to reach CO EOL (2024).
009677	1, 4	32000 63230	Alarm, Cat 1/2	These temperature indicating alarm units provide indication of the moderator cover gas recombiner catalytic bed temperature and generates an alarm when the temperature exceeds the predefined setpoint.	Poor	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Surveillance	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete proactive replacement per WOs 1549266, 1549267, 1549251, and 1549269. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Perform one-time calibrations.
009680	1, 4	32000 32310	Arrestor, Flame/Fire, Cat 1/2	Flame arrestors are installed before and after the recombination units to stop any spread of burning gases that would result from ignition of the deuterium/oxygen mixture in the recombination unit.	Poor	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion	Surveillance	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete WO#3005445 to complete a visual inspection of 1-32310-FA2 to determine the integrity of the flame arrestor. The inspection should include a review of internal screen material to reach Plant EOL (2020). Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024):

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
								Perform a one-time inspection of 1-32310-FA2 to determine the integrity of the flame arrestor if necessary based on inspection results of WO#3005445 to reach CO EOL (2024).
009681	1, 4	32000 63211	Box, Electrical (Junction Box), Cat 1/2	These junction boxes are used for field connection of cables, including terminal jumpering, in moderator main circulation system.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion	Surveillance	Continue current practices. No additional recommendations required to reach CO EOL (2024).
009685	1, 4	32000 32310	Compressor, Gas, Cat 3/4	Moderator cover gas compressors circulate cover gas through recombination units to prevent build-up of deuterium in cover gas	Satisfactory	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue Corrosion / General Corrosion Fatigue / High Vibrations		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete the following work orders: WO 02628617 “4-32310-CP1 Inspect & Lube Coupling“, WO 2836859 “Replace Moderator Cover Gas Comp. Motor”. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
009686	1, 4	32000 63210	Controller, Cat 1/2	Provide indication and control of dump port moderator levels.	Good	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Electronic Aging	Calibration	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009688	1, 4	32000 32310	Controller, Cat 1/2, I/P, 546NS	This controller receives either a voltage (Vdc) or a current (mAdc) input signal and transmits a proportional pneumatic output pressure to the control valve.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Fatigue due to Vibration		
009689	1, 4	32000 32110 32310	Controller, Cat 1/2, I/P, Large Moderator Level	This control valve transducer receives either a voltage (Vdc) or a current (mAdc) input signal and transmits a proportional pneumatic output pressure to the associated control valve.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Fatigue due to Vibration		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete the WOs for replacing 1-32310-CV1/2-AX1 and 1-32110-CV55-AX1. Refer to WOs 3099938, 3099937, 1505860, 1505859, 2851238-14, etc. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Increase the frequency of the existing PMs for 32310-CV1/2/3/4 to refurbish and calibrate the actuator and sub-components including AX1 from 10 years to 6 years, example PM 9954-01.
009691	1, 4	32000 63210 63230	Controller, HC1/PIC1/TIC1	1/4-63210-L8-HC1 provides control for moderator dump tank level; 1/4-63230-P2-PIC1 provides control and indication for moderator dump tank pressure; and 1/4-63210-T1/T2-TIC1 provides control and indication for moderator temperature.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Electronic Aging Obsolescence / Immediate Obsolescence Concern	Calibration Component Replacement Overhaul/Refurbishment	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete proactive replacement of 1/4-63210-L8-HC1 and 1/4-63230-P2-PIC1 per WOs 2073755, 2073756, 3003138, and 3001713. Incremental recommendations for Plant EOL (2020): Resolve obsolescence and spares issues. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
009692	1, 4	32000 63210	Converter, Current To Current, Cat 1/2	These Temperature Modules are used to generate 4-20mA bias signals for moderator temperature control.	Good	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009694	1	32000 32110	AMPLIFIER, Cat 1/2	These amplifiers are used to generate signals representing vibration of main moderator pumps.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Electronic Aging Obsolescence / Immediate Obsolescence Concern		Incremental recommendations for Plant EOL (2020): Resolve obsolescence and spares issues. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
009699	1, 4	32000 32310	Disc, Rupture, Cat 1/2	These rupture discs limit the peak overpressure in the Calandria following a burst pressure/Calandria tube accident/emergency condition.	Satisfactory	Corrosion / General Corrosion Fatigue / Mechanical Fatigue Corrosion / Pitting Corrosion Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Cracking Due to Expansion or Contraction	Component Replacement	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete outstanding W/O's to replace 1-32310-Y6 and 4-32310-Y8/Y9. Incremental recommendations for Plant EOL (2020): Perform a one-time replacement of 1/4-32310-Y7 rupture discs. Note that all the remaining components are subject to replacement. Incremental recommendations for CO EOL (2024): Perform a one-time replacement of 1/4-32310-Y7 rupture discs. Note that all the remaining components are subject to replacement.
009701	1, 4	32000 63230	Element (temp), Cat 2	These temperature elements are used to generate signals representing the moderator temperatures downstream of the heat exchangers or the moderator cover gas	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of	Surveillance	Incremental recommendations for Plant EOL (2020): No additional recommendations required to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024):

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				recombine catalytic bed temperatures.		Material (Wiring, Seals, Gaskets, O-rings etc.)		Perform one-time calibrations on the temperature loops associated with 1-63230-T3/T4-TE1.
009707	1, 4	32000 63210 63230	Element, Cat 1/2	These temperature elements are used to generate signals representing various temperatures of the moderator system. In addition, T11-TE1 serves as part of the moderator temperature control loop during ECI recovery.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Calibration SRST - Functional Test SRST - Logic Test	Incremental recommendations for Plant EOL (2020): No additional recommendations required to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Perform one-time calibrations on the temperature loops associated with 4-63230-T3/T4-TE1. Note that all the remaining components are subject to calibration.
009708	1, 4	32000 63230	Element, Cat 2	These moisture elements are used to detect a build-up of condensate on the closed vanes of the dump valves, which could slow down the valve opening.	Poor	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Obsolescence / Immediate Obsolescence Concern	Surveillance	Incremental recommendations for Plant EOL (2020): Resolve obsolescence and spares issues. Incremental recommendations for CO EOL (2024): Perform calibrations at a frequency consistent with the approved IQ Review Maintenance Template.
009710	1, 4	32000 63210	Element (FE), Cat 1	These flow elements are used to generate a signal representing the total D20 flow through the two moderator heat exchangers.	Poor	Mechanical and Thermal Degradation / Thermal Aging Corrosion / General Corrosion Obsolescence / Immediate Obsolescence Concern	Surveillance	Incremental recommendations for Plant EOL (2020): Resolve obsolescence and spares issues. Incremental recommendations for CO EOL (2024): Inspect and clean the flow elements and verify their operation every 6 years.
009712	1, 4	32000	Gauge, Cat 1/2	These pressure gauges provides local indication of the moderator dump tank pressure.	Good	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						/ Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion		
009715	1, 4	32000 32310	Heater (Generic), Cat 1/2	These heaters, strapped on to the pipe run, function as a Moderator Cover Gas pre-heater to dry the moderator cover gas mixture to prevent the catalyst in the recombination unit from becoming moist and losing its effectiveness.	Good	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Surveillance	Continue current practice. No additional practices recommended to reach CO EOL (2024).
009717	1, 4	32000 32210	Ion Exchanger (Column), Cat 1/2	Remove boric acid, corrosion products and ionic impurities from the Moderator D2O Circuit for system chemistry control.	Satisfactory	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Corrosion / Fouling (accumulation of deposits)	Surveillance	<u>Incremental recommendations for Plant EOL (2020):</u> Perform a one-time inspection of Ion exchange columns to determine their condition (pressure boundary and strainer integrity). <u>Incremental recommendations for CO EOL (2024):</u> Perform a one-time inspection of Ion exchange column (vessel) to address effects of aging if necessary based on results of the last inspection.
009719	1, 4	32000 32310	Motor, Compressor, Cat 3/4	These motors are part of Helium circulation compressors 3231-CP1 & -CP2, which are used to drive the compressors for Helium circulation.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Thermal Aging		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Replace the compressor motors (1/4-32310-CPM1 to CPM4) per WOs 2836800, 2836801, 2836860, & 2836859. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Corrosion / General Corrosion Fatigue / Thermal Fatigue		Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
009726	1, 4	32000 32310	Positioner, Pneumatic, Cat 1/2	These positioners receives a pneumatic signal from a transducer and modulates the actuator position of the associated control valve accordingly. The large moderator level regulating valves 32310-CV1/2/3/4 are controlled by the Reactor Regulating System to adjust Calandria moderator level.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / High Vibrations		Incremental recommendations for Plant EOL (2020): No additional recommendations required to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Ensure adequate spares are available.
009727	1, 4	32000 32310	Positioner, Pneumatic, Cat 1/2, Small Moderator Level CV	These positioners modulate the actuator position of a control valve based on a pneumatic signal from a transducer. The small moderator level regulating valves 32310-CV104/105 are controlled by the Reactor Regulating System for fine control of the moderator level.	Very Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Fatigue / High Vibrations Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009728	1, 4	32000 32110 32310	Positioner, Pneumatic, Cat 1/2, Helium Level Tank CV	These positioners are used to modulate control valves during operations.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Fatigue due to Vibration		
009730	1, 4	32000 32310	Recombination Unit, Cat 1/2	The function of the recombination unit is to maintain the cover gas deuterium concentration (produced by moderator radiolysis) to <2% under normal operation.	Very Poor	Corrosion / Environmental Degradation Corrosion / General Corrosion	Surveillance	Incremental recommendations for Plant EOL (2020): Implement a one-time replacement of the catalyst and perform an internal and external inspection. Incremental recommendations for CO EOL (2024): Perform a one-time inspection to verify the internal and external condition of the recombination unit if necessary based on the results of the last inspection.
009733	1, 4	32000 32310	Regulator, Pressure, Cat 1/2, For Control Valve, Fisher 67CFR	PRV regulates instrument air pressure used for control of moderator CVs.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Creep and Stress Relaxation		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009734	1, 4	32000 32310	Regulator, Pressure, Cat 1/2, For Control Valve, Dresser 77-N	PRV regulates instrument air pressure used for moderator CV control.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring,	Component Replacement	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Creep and Stress Relaxation		
009735	1, 4	32000 32310	Regulator, Pressure, Cat 1/2, For Dump Valves	PRV regulates instrument air pressure used for moderator MV control.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Creep and Stress Relaxation		<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Perform a one-time refurbishment / replacement of the PRVs in this CG.
009736	1, 4	32000 32310	Relay, Booster, Cat 1/2	These boosters (AF1) are used to amplify the pneumatic signals used to modulate the control valve actuator position.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						/ Creep and Stress Relaxation Mechanical and Thermal Degradation / Wear Mechanisms - Wear		
009737	1, 4	32000 63210	Relay, Time, Cat 1/2	These time delay relays are used to generate a high/low level annunciation for the moderator dump tank.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion	Calibration	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009738	1, 4	32000 63220	Recorder, Cat 1/2	These recorders are used to log various parameters such as moderator purification outlet conductivity, moderator dump tank level, and moderator outlet temperature.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging	Calibration Clean and Inspect	<u>Incremental recommendations for Plant EOL (2020):</u> No additional recommendations required to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> 1. Calibrate the moderator purification conductivity CRs as part of the loop calibration. 2. Perform one-time replacement of the capacitors in all CRs.
009740	1, 4	32000 63210	Selector, High, Cat 1/2	This Signal Selector module is used to select the higher of two input signals for moderator temperature control.	Very Good	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

TETRA TECH

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
009743	1, 4	32000 32310	Station, Current Output, Cat 1/2	These current output stations are used to modulate moderator level control valves and to display the control output signals.	Poor	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Surveillance	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete outstanding WO #3004764 to replace 1-3231-CV104-HC1. . Incremental recommendations for Plant EOL (2020): No additional recommendations required to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): 1. Perform calibrations at a frequency consistent with the approved IQ Review Maintenance Template, and, 2. Complete replacement of 4-32310-CV104-HC1.
009744	1, 4	32000	Switch, Pressure Cat 2	Provide contact inputs to generate alarms when the moderator dump tank pressures exceed or drop below predefined set points.	Very Poor	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Obsolescence / Immediate Obsolescence Concern		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete outstanding WO #03238410 to restore condition of 4-63230-P2-PS1. Incremental recommendations for Plant EOL (2020): Resolve obsolescence and spares issues. Incremental recommendations for CO EOL (2024): No additional practices recommended to reach CO EOL (2024).
009754	1, 4	32000 32310	Valve, Control, Cat 1/2, Helium Tank Level CV,Hammel Dahl	Feed and Bleed control valves in main helium circuit, maintain dump tank pressure by regulating transfer of helium between dump tank and storage tank.	Good	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Calibration Overhaul/Refurbishment SRST - Functional Test Valve Diagnostics	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Radiation Induced Degradation / Radiation Depletion of Material Properties Fatigue / Thermal Fatigue		
009756	1, 4	32000 32310	Valve, Control, Cat 1/2, Small Moderator Level CV	Cover gas control valves, these are the Small (1") Moderator Level Regulating CV's used in moderator level control.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Component Replacement Inspection - Internal Overhaul/Refurbishment	Incremental recommendations for Plant EOL (2020): Obtain spares either by purchasing or from harvesting Unit 2/3. Incremental recommendations for CO EOL (2024): No additional practices are recommend to reach CO EOL (2024).
009757	1, 4	32000	Valve, Control, Cat 1/2, Large Moderator Level CV, Hammel Dahl	Cover gas control valves, these are the Large (4") Moderator Level CV's used in moderator level control.	Satisfactory	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009762	1, 2, 3, 4	32000 34930	Valve, Motorized/Motor Operate, Cat 1/2	U2/U3 MV2026 is a valve from moderator system and radioactive filter/resin handling systems (abandoned) to Sulzer feed.	Satisfactory	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Surveillance	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Corrosion / General Corrosion		
009764	1, 4	32000	Valve, Check/Nonreturn/Back Fl, Cat 1/2	NVs prevent backflow. 32310-NV31 – compressor discharge NV 32310-NV32 – helium tank to compressor NV	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Fatigue due to Vibration Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Fatigue / Mechanical Fatigue Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009766	1, 4	32000	Valve, Check/Nonreturn/Back Fl, Cat 1/2, Large Regulating CV BUIA NVs	32310-CV1/CV2/CV3/CV4-NV1: NV for back-up air supply to CVs, prevent back flow of air supply when the normal air supply is not available.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Creep and Stress Relaxation Fatigue / Mechanical Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Corrosion / General Corrosion		
009767	1, 4	32000 32310	Valve, Check/Nonreturn/Back Fl, Cat 1/2, Helium Compressor NV	32310-NV2008, NV2009 are the helium circulation compressor discharge NVs, prevent backflow of helium	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Creep and Stress Relaxation		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009768	1, 4	32000 32510	Valve, Pneumatic/ Pneumatic Ac, Cat 1/2, Ball CV	These MVs are used for Moderator D2O collection isolation (normally closed).	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024)
009770	1, 4	32000 32310	Valve, Pneumatic/ Pneumatic Ac, Cat 1/2, Dump Tank MVs	The moderator Cover Gas System Dump valves are air operated butterfly valves which provide rapid equalization of pressure between the helium in the	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation	Calibration SRST - Functional Test Valve Diagnostics	Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024):

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				dump tank and the Calandria.		/ Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Thermal Fatigue		Implement a one-time seat replacement.
009771	1, 4	32000 32210	Valve, Pneumatic/ Pneumatic Ac, Cat 1/2, Mod Purification	These valves are moderator purification outlet isolation valves.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue Corrosion / General Corrosion Fatigue / Thermal Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024)
009772	1, 4	32000 32510	Valve, Pneumatic/ Pneumatic Ac, Cat 1/2, Drain CV	These valves are used for testing the operation of double process valves (34310-MV4/MV5 and 34310-MV1/MV2).	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue Corrosion / General Corrosion	Overhaul/Refurbishment SRST - Functional Test SRST - Logic Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Creep and Stress Relaxation		
009773	1, 4, 012, 034	32000 32300 32310	Valve, Pressure Regulating, Cat 1/2	These pressure regulating valves are used to support bulk helium supply distribution and flow including the gas bottle station.	Very Good	Fatigue / Mechanical Fatigue Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009776	1, 4	32000 32310	Valve, Pressure Relief, Cat 1/2	32310-RV47 – dump tank pressure relief.	Good	Corrosion / General Corrosion Fatigue / Mechanical Fatigue Fatigue / Environmentally-Assisted Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009778	1, 4	32000 32110 32310	VALVE, PRESSURE RELIEF FOR DUMP VALVE	Relief valves provide over pressure protection to actuator instrument air supply.	Satisfactory	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation		Incremental recommendations for Plant EOL (2020): Replace CV8-RV1. Incremental recommendations for CO EOL (2024): Implement a one-time RV replacement for all MV-RV's that have not been replaced since 2017.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						/ Creep and Stress Relaxation Fatigue / Mechanical Fatigue Corrosion / General Corrosion		
009783	1, 4	32000 32310	Solenoid/Solenoid Operated Val, Cat 1/2, Dump MVs	These solenoid valves are used to control the valve actuator of the moderator dump valves (32310-MV7 to MV12).	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Creep and Stress Relaxation Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Calibration Component Diagnostics SRST - Functional Test	<u>Incremental recommendations for Plant EOL (2020):</u> No additional recommendations required to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Refurbish the SVs as part of the MV PMs.
009784	1, 4	32000 32110 32210 32310 32510	Solenoid/Solenoid Operated Val, Cat 1/2, Valcor V70900-98-07	These solenoid valves are used to control the valve actuators of the Calandria outlet CV's (32110-CV2 through CV2) to regulate the outflow from the Calandria to the moderator pumps.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Creep and Stress Relaxation Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Calibration Component Diagnostics Overhaul/Refurbishment SRST - Functional Test	<u>Incremental recommendations for Plant EOL (2020):</u> No additional recommendations required to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Implement new PMs for the solenoid valves of 32210-MV15 and 32510-MV43.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
009785	1, 4	32000 63210	Indicator, Cat 2	Provide indications of the moderator temperature downstream of the heat exchangers and the moderator flow at the heat exchanger effluent header.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete proactive replacement per WOs (#01546185, #01549256, #01549257, #01549250). Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices recommended to reach CO EOL (2024).
009786	1, 4	32000 63220	Transmitters (CTs), Cat 2	These transmitters are used to convert and transmit a moderator ion exchanger outlet conductivity signal to an electrical signal for annunciation and recording.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Thermal Aging		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Replace 1-63210-C1-CT1 per WO# 02677961 to restore condition. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
009787	1, 4	32000 63210	Transmitters (FT), Cat 1/2	Transmit signal representing the moderator flow through the effluent header for the heat exchangers.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Calibration	Incremental recommendations for Plant EOL (2020): No additional recommendations required to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Complete one-time proactive replacement.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
009788	1, 4	32000 63210	Transmitters (LT), Cat 1	These transmitters are used to transmit signals representing moderator dump port or tank levels.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Calibration Component Replacement	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete one-time replacement for 1/4-63210-L8-LT2 and 1-63210-L1-LT1/LT2 (per WO#02073776 and WO#02073779). Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices recommended to reach CO EOL (2024).
009789	1, 4	32000 63210 63250	Transmitter (LT), Cat 1/2	These transmitters are used to transmit signals representing moderator narrow-range calandria levels (1/4-63210-L5A/L5B-LT1), narrow-range dump tank levels (1/4-63210-L8-LT1) and D2O collection tank levels (1/4-63250-L1-LT1).	Poor	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Obsolescence / Immediate Obsolescence Concern		Incremental recommendations for Plant EOL (2020): No additional recommendations required to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Complete one-time proactive replacement, resolve obsolescence and spare issues.
009790	1, 4	32000 63210	Transmitter (LT), Cat 1/2	Transmit the signals representing wide-range Calandria and dump tank moderator levels.	Good	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring,		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Seals, Gaskets, O-rings etc.)		
009791	1, 4	32000 63210 63230	Transmitter (PT), Cat 1/2	Transmitters are used to transmit signals representing moderator circulating pump suction pressures and dump tank pressures.	Very Good	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Calibration Component Replacement Functional Test Inspection - Visual	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009792	1, 4	32000 63210	Transmitter (Temp), Cat 1/2	Transmit signals representing the moderator temperatures downstream of the heat exchangers. In addition, transmitters serve as part of the moderator temperature control loop during ECI recovery.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Electronic Aging		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete life cycle replacement for 1/4-63210-T11-TT1 per WO#02953863/4. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
009793	1, 4	32000 63210	Transmitter (Temp), Cat 1/2	Transmitters used to transmit and isolate signals representing moderator temperatures at the heat exchange inlets.	Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
009795	1, 4	32000 32110	CONTROL, Cat 1/2	These controllers are used to control the moderator dump tank level by modulating associated dump tank outlet control valves.	Good	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009796	1	32000 32110	CONVERTER, Cat 1/2	These voltage-to-current converters are an integral part of the control circuit used to modulate the Calandria inlet CVs and dump tank outlet CVs.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009797	1,4	32000	Alarm, Cat 1/2	These alarm units provide indication and generate alarms when the Calandria wide-range moderator levels drops below predefined setpoints.	Good	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009799	1, 4	32000 32110	ELEMENT, Cat 1/2	These temperature sensing elements are used to generate main moderator pump motor winding and bearing temperature signals.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		<u>Incremental recommendations for Plant EOL (2020):</u> No additional recommendations required to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Implement new PMs to perform calibrations on the associated temperature loops at a frequency consistent with the approved IQ Review Maintenance Template.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
009801	1, 4	32000 32110	EXPANSION JOINT, Cat 1/2	Bellows expansion joints 3211-EJ1 and 3211-EJ2 allow thermal, or shock, movement of the dump tank. Bellows expansion joints 3211-EJ3 and 3211-EJ4 allow the piping for pumps 3211-P4 and 3211-P5, and for pumps 3211-P1, 3211-P2 and 3211-P3, Respectively, to expand north. Bellows expansion joint 3211-EJ5 is used to seal the Calandria vault atmosphere from the moderator room and allow movement of the dump tank return line. 3211-EJ506 is expansion joint for LPSW outlet at HX2.	Satisfactory	Mechanical and Thermal Degradation / Cracking Due to Cyclic Loading Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Thermal Aging	Surveillance	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete the inspection's scheduled in WO#03005464 and WO#3003872 for Units 1 and 4 EJ3. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Perform a one-time inspection of all expansion joints based on inspection results from WO#03005464 and WO#3003872.
009804	1, 4	32000 32110	HEAT EXCHANGER, Cat 1/2	Two 50% heat exchangers serve as heatsink to cool the moderator heavy water in the moderator main circuit.	Satisfactory	Corrosion / Pitting Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Fretting Corrosion / Fouling (accumulation of deposits) Mechanical and Thermal Degradation / Creep and Stress Relaxation Corrosion / Microbiological Influenced Corrosion		<u>Incremental recommendations for Plant EOL (2020):</u> Complete shell side and LPSW inlet/ outlet nozzles inspections. These inspections should be added to the current PM instructions. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommend to reach CO EOL (2024).
009805	1, 4	32000 32110	HEATER (GENERIC), Cat 1/2	These heaters are used to heat the drive end (DE) bearing part of the pump	Good	Corrosion / General Corrosion	Inspection - Visual Predictive Maintenance - Electrical Testing	<u>Incremental recommendations for Plant EOL (2020):</u> No additional recommendations required to reach Plant EOL (2020).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				motor when the motor is not running.		Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Predictive Maintenance - Lubrication Analysis Predictive Maintenance - Thermography	Incremental recommendations for CO EOL (2024): Perform periodic heater inspection with a contingency task to replace as required.
009810	1	32000 32110	MOTOR, Cat 1/2	These actuator motors are used to drive the moderator dump tank outlet motorized valves (MVs) on demand in order to control the moderator level in the dump tank.	Very Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Lubricant Degradation Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continued current practices. No additional practices are recommended to reach CO EOL (2024).
009811	1, 4	32000 32110	MOTOR, Cat 1/2	These motors drive the main moderator pumps which circulate the moderator heavy water through the heat exchangers to maintain the moderator temperature within acceptable limits.	Very Poor	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Restore condition of 4-32110-PM5 which is currently unavailable (ref. WO#03176544 currently scheduled in APPROVED P1641 forced outage scope). Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						/ Lubricant Degradation Radiation Induced Degradation / Radiation Embrittlement		
009815	1, 4	32000 32110	PUMP, Cat 1/2	The Main Moderator Circulation Pumps 1/4-32110-P1-P5 circulate moderator between calandria and moderator heat exchanger for heat removal to ensure that the moderator is available as a heat sink. They are also used for ECIS recovery operation by injecting water in the RB into the HTS following a LOCA.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion Corrosion / Abrasion, Erosion and Cavitation Fatigue / High Vibrations	Overhaul/Refurbishment Predictive Maintenance - Lubrication Analysis Predictive Maintenance - Thermography Predictive Maintenance - Vibration Monitoring	<u>Incremental recommendations for Plant EOL (2020):</u> Complete a one-time inspection of one pump internals to assess degradation. <u>Incremental recommendations for CO EOL (2024):</u> Perform further pump inspections/overhaul as required based on previous sample inspection.
009816	1, 4	32000 63231	RECEIVER, Cat 1/2	The Air Receivers are used to provide air for the dump valve actuators.	Good	Corrosion / General Corrosion		<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommend to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Perform a one-time internal inspection of receivers to ensure corrosion is not a significant problem to reach CO EOL (2024).
009818	1, 4	32000 32110	REGULATOR, Cat 1/2 for Calandria inlet/outlet	The Pressure Regulating Valves are used to regulate instrument air pressure for the proper operation of control valves.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Inspection - Visual SRST - Functional Test SRST - Logic Test	<u>Incremental recommendations for Plant EOL (2020):</u> Replace all PRV's that have not been replaced since 2015. <u>Incremental recommendations for CO EOL (2024):</u>

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Creep and Stress Relaxation Fatigue / Mechanical Fatigue		Replace all PRV's that have not been replaced by 2019.
009819	1, 4	32000 32110	RELAY, Cat 1/2	These boosters (AF1) are used to boost pneumatic signals to modulate the control valves during operations.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Creep and Stress Relaxation Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Component Diagnostics Component Replacement SRST - Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009825	1, 4	32000 32110	SWITCH, Cat 1/2, Moderator pump motor oil	These level switches generate alarms when the upper bearing lube oil levels of main moderator pump motors drop below predefined setpoints.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Thermal Aging	Calibration	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009830	1	32000 32110	TRANSMITTER, Cat 1/2	Generate signals representing vibration of the main moderator pumps.	Good	Mechanical and Thermal Degradation / Thermal Aging Fatigue / Fatigue due to Vibration Radiation Induced Degradation / Radiation Embrittlement		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
009831	1, 4	32000 32110	VALVE, CONTROL, Cat 1/2, Calandria inlet/outlet, DT outlet	CV2/4/26/27 are used to regulate the outflow from the calandria to maintain moderator level. CV48/50/54/55 are used to control the spray flow to the calandria by regulating the calandria inlet flow. CV7/8 are used to regulate the dump tank outlet flow to control the dump tank level. All the valves have functions in the ECI system.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Fatigue due to Vibration	Overhaul/Refurbishment SRST - Functional Test Valve Diagnostics	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: 1. Perform a one-time replacement/overhaul of Calandria outlet valves (1/4-32110-CV2/4/26/27) before EOL per WOs 4976443, 4976469, 4976480, 4976489, 4967469, 4974812, 4974813, 3175399. Note that all of the remaining equipment tags are subject to overhaul. 2. Obtain required spare parts for the Calandria outlet valves 1/4-32110-CV2/4/26/27. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommend to reach CO EOL (2024).
009834	1, 4	32000 32110	VALVE, MOTORIZED/MOTOR OPERATE, Cat 1/2	The valves are dump tank outlet valves, normally closed. They open when pumping up the calandria from the dump tank.	Very Good	General corrosion, degradation of lubrication, wear of brass stem nut, wear, degradation of electrical wiring, degradation of packing. / 134	Inspection - Visual Valve Diagnostics	Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Revise EQ assessment NA44-EQA-04940-00020 to allow extending the life of the existing valves seats to reach CO EOL (2024), or perform replacement.
009835	1, 4	32000 32110	VALVE, CHECK/NONRETURN/BACKFL, Cat 1/2	32110-NV14, NV15, NV16, NV17, NV18 – moderator main circulation pump discharge NV preventing backflow when a pump is not running.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation	Functional Test Inspection - Non - Intrusive Valve Diagnostics	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete WOs 03180892, 02791486, 02791484, 02791483 for overhaul and internal inspections. Incremental recommendations for Plant EOL (2020):

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						/ Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion		No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommend to reach CO EOL (2024).
011100	5, 6, 7, 8	32000 32110 32210 32310 32510 32710	32110 Moderator Head Tank	Moderator Head Tank 32110-TK1 accommodates moderator swell and limits moderator level in the Calandria to normal setpoint. Ion exchange columns in the moderator purification system (32210-IX1 to -IX5) remove moderator poisons, corrosion products and ionic impurities. Moderator purification filters (32210-FR1/2) remove any insoluble debris from the purification flow before it passes through the ion exchangers. Recombination units (32310-RU1/2) maintain the deuterium concentration in the moderator cover gas (helium) to less than 2% under normal operation. Flame Arrestors (32310-FA1 to FA5) stop any spread of burning gases that would result from ignition of the deuterium/oxygen mixture in the Recombination Unit.	Poor	Corrosion / Microbiological Influenced Corrosion Corrosion / Stress Corrosion Cracking - Austenitic Stainless Steels Corrosion / Chemical Attack Corrosion / Fouling (accumulation of deposits) Corrosion / General Corrosion Mechanical and Thermal Degradation / Thermal Aging Fatigue / Thermal Fatigue	Inspection - Internal Inspection - Visual Sampling Test	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: 1. Complete outstanding work order WO#3056287 (internal inspection of moisture separator 5-32310-TK1). 2. Complete work order WO#3005445 to inspect Pickering A flame arrestor 1-32310-FA2. Based on findings, determine if inspection of Pickering B flame arrestors is warranted, and establish work orders for inspection/replacement. Incremental recommendations for Plant EOL (2020): 1. Address spare parts supply issues with 32210-FR1/FR2. 2. Complete a one-time inspection of selected components from all component groups. Incremental recommendations for CO EOL (2024): Perform a one-time inspection of selected components from each of the component groups if required based on results from previous inspections.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				32310-TK1 is the moisture separator to remove D2O from the Moderator Cover Gas system (at the compressor inlet). The Moderator D2O Collection Tank (32510-TK1) holds heavy water collected from various parts of the Moderator System. Liquid poison mixing tanks 32710-TK2/TK3 are used for gadolinium mixing and boron mixing, respectively.				
011101	5, 6, 7, 8	32000 32110	32110 Moderator Heat Exchangers	Two 50% heat exchangers serve as the heatsink to cool the moderator heavy water in the moderator main circuit.	Satisfactory	Corrosion / Pitting Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Fretting Corrosion / Fouling (accumulation of deposits) Mechanical and Thermal Degradation / Creep and Stress Relaxation Corrosion / Microbiological Influenced Corrosion	Inspection - Internal	Incremental recommendations for Plant EOL (2020): Complete shell side and LPSW inlet/ outlet nozzles inspections. These inspections should be added to the current PM instructions. Incremental recommendations for CO EOL (2024): No additional practices are recommend to reach CO EOL (2024).
011102	5, 6, 7, 8	32000 32110 32310	32110 Moderator HX Relief Valves	Relief valves 32110-RV18 & RV19 provide overpressure protection to moderator HX2 & HX1, respectively. 32310-RV42, RV43 – downstream PRV2 & PRV3, respectively, provides pressure relief. 32310-RV49 – upstream PRV5 relief valve.	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				32310-RV50 – PRV5 downstream relief valve. 32310-RV66 - PRV1 downstream pressure relief. 32310-RV72 – instrument air purge relief.		/ Wear Mechanisms - Wear Corrosion / Fouling (accumulation of deposits) Mechanical and Thermal Degradation / Wear Mechanisms - Erosion		
011103	5, 6, 7, 8	32000 32110 32310	32110 Moderator Main Circuit Energy Dissipator	The energy dissipators (OR) act on D2O fluid flow to prevent Calandria inlet line vibration and cavitation. Y7 and Y8 are cooling jackets which cool the moderator cover gas after it passes through the recombination units to protect downstream equipment from the high temperatures.	Poor	Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Corrosion / Fouling (accumulation of deposits) Corrosion / Abrasion, Erosion and Cavitation		Incremental recommendations for Plant EOL (2020): 1. Modify the existing PM for moderator piping inspections to include the energy dissipater. 2. Complete a one-time inspection of a sample cooling jacket. Determine if any further AM practices are required on other cooling jackets based on inspection results. Incremental recommendations for CO EOL (2024): No additional practices are recommend to reach CO EOL (2024).
011104	5,6,7,8	32000	32110 Moderator Main Circuit Piping	This piping provides a flow path and acts as a pressure boundary for the moderator system.	Good	Corrosion / Abrasion, Erosion and Cavitation Mechanical and Thermal Degradation / Radiation Embrittlement Corrosion / Chemical Attack Corrosion / Stress Corrosion Cracking - Austenitic Stainless Steels Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation	Inspection - Condition Monitoring Program	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						/ Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Wear Mechanisms - Fretting Fatigue / High Vibrations		
011105	5, 6, 7, 8	32000 32110	32110 Moderator Main Circuit Pumps	These pumps are used to circulate Moderator D2O through the two parallel heat exchangers and then back to the Calandria.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / Abrasion Erosion and Cavitation Fatigue / High Vibrations Radiation Induced Degradation / Radiation Depletion of Material Properties	Inspection - Visual Predictive Maintenance - Vibration Monitoring	Continue current practices. No additional practices are recommended to reach CO EOL (2024)
011106	5, 6, 7, 8	32000 32110 32210 32310 32510 32710	32110 Moderator Pipe Supports	The hangers provide support for main moderator and auxiliary system pipe lines.	Satisfactory	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Fretting Mechanical and Thermal Degradation / Distortion Due to Settlement Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Creep and Stress Relaxation		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011107	5, 6, 7, 8	32000 32110 32510 32710	32110 Moderator Pump Discharge Non-Return Valves	Function of check valves is to prevent back flow. 32110-NV3, NV4, NV5, NV6, NV7 – moderator circulating pump discharge pump NVs, 32710-NV14 – gadolinium addition pump discharge NV. 32510-NV18 – moderator D2O collection pump discharge NV. 32710-NV20 – boron addition pump discharge NV.	Satisfactory	Corrosion / General Corrosion Corrosion / Pitting Corrosion Corrosion / Crevice Corrosion		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: 1. Complete work associated with replacement/inspection of Unit 5 NVs (WO 2574684, 2572581, 2572529, 2542782). 2. Complete WO 02542781 for 5-32110-NV5 Inspect NV and WO 02542780 for 5-32110-NV4 Inspect NV. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommend to reach CO EOL (2024).
011108	5, 6, 7, 8	32000 32110	32110 Moderator Pump Motors-4.16kV	These 4kV motors drive the main moderator pumps which circulate the moderator heavy water through the heat exchangers to maintain the moderator temperature within acceptable limits.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Lubricant Degradation	Inspection - Visual Predictive Maintenance - Lubrication Analysis Predictive Maintenance - Vibration Monitoring	Incremental recommendations for Plant EOL (2020): Resolve obsolescence and spares issues. Incremental recommendations for CO EOL (2024): No additional practices are recommend to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011109	5, 6, 7, 8	32000 32110	32110 Moderator to EWS Tie Non-Return Valves	32110-NV37 – main moderator system is connected to the emergency water supply system via 32110-NV37. Its purpose is to prevent backflow from moderator system to EWS during normal operation and to open during EWS recovery following a seismic event.	Satisfactory	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear	SRST - Functional Test	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete WO 1580499 to replace 5-32110-NV37. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Initiate WOs to inspect 50% of this valve group.
011110	5, 6, 7, 8	32000 32310	32310 Calandria Rupture Disc	Four calandria rupture discs are provided to limit the peak overpressure in the calandria following a burst pressure/calandria tube accident/emergency condition.	Good	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Cracking Due to Expansion or Contraction Fatigue / Thermal Fatigue Corrosion / Pitting Corrosion Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011111	5, 6, 7, 8	32000 32310 32510 32710	32310 Moderator Cover Gas Compressor Motors	These compressors are used to drive the moderator cover gas compressors	Very Good	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear Radiation Induced Degradation / Radiation Embrittlement		<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommend to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011112	5, 6, 7, 8	32000 32710	32710 Moderator Liquid Poison Addition Pumps	The pumps are used to inject Boron or Gadolinium to the Moderator D2O to add negative reactivity.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Fatigue due to Vibration	SRST - Functional Test	Incremental recommendations for Plant EOL (2020): Purchase one spare pump and perform BOM verification of subcomponent CIDs to ensure spare parts are available. Incremental recommendations for CO EOL (2024): No additional practices are recommend to reach CO EOL (2024).
011163	5, 6, 7, 8	32000 63210 63220 63230 63250 63270	63210 Moderator Narrow Range Level Transmitters - Rosemount 1152DP4N	These transmitters transmit level signals used by the narrow-range Calandria moderator level control circuit.	Good	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Calibration Component Replacement SRST - Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011164	5, 6, 7, 8	32000 63210 63230	63210 Main Moderator and Moderator Cover Gas Temperature Transmitters	Transmit moderator and moderator cover gas temperature signals.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Electronic Aging		Incremental recommendations for Plant EOL (2020): Complete proactive replacement. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011165	5, 6, 7, 8	32000 63210	63210 Moderator Wide Range Level Transmitters - Rosemount 1152DP5N	Transmit wide-range Calandria moderator level signals.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Incremental recommendations for Plant EOL (2020): Complete proactive replacement. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011172	5, 6, 7, 8	32000 32210 32310 32510 32710 71310	71310 Moderator HX Temperature Control Valves	These air operated valves are used to control the following Moderator system functions: 71310-CV509/CV512/CV542/CV543 - LPSW supply to Moderator heat exchangers 32210-MV39 – Purification Ion Exchange columns bypass 32210-MV40/MV44 – Flow to Ion Exchange columns IX1-IX5 32310-CV1/CV2 – Cover gas purge control 32310-CV1-ACT1/CV2-ACT1 – Cover gas purge control valve actuators 32310-MV11/MV12 – Cover Gas compressor discharge 32310-MV78 – Cover gas O2 addition 32310-MV79/MV81 – Helium supply isolators 32510-MV12 – Moderator D2O Collection pump discharge to purification 32710-MV7 – Boron addition 32710-MV8 – Gadolinium addition	Satisfactory	Corrosion / General Corrosion Corrosion / Abrasion, Erosion and Cavitation Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Calibration Component Replacement Inspection - Visual Overhaul/Refurbishment SRST - Functional Test SRST - Stroke Test Valve Diagnostics	<u>Incremental recommendations for Plant EOL (2020):</u> Purchase spares for valves: 5/8-32210-MV39/MV40, 5/8-32310-CV1/CV2, & 5/8-32710-MV7/MV8. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommend to reach CO EOL (2024).
011173	5, 6, 7, 8	32000 71310	71310 Moderator LPSW Relief Valves	71310-RV626 – LPSW relief from HX2 (moderator heat exchanger). 71310-RV628 - LPSW relief from HX1 (moderator heat exchanger).	Good	Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Corrosion / General Corrosion Corrosion / Fouling (accumulation of deposits) Mechanical and Thermal Degradation / Deterioration of Material (Wiring,		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		
011357	5, 6, 7, 8	32000 32310	32310 Moderator Cover Gas Check Valves	Moderator cover gas check valves, 5-8-32310-NV16 and NV18, prevent back-flow of D2O to the cover gas compressors and recombination units, respectively.	Satisfactory	General Corrosion / General Corrosion Wear Mechanisms - Wear / Wear Mechanisms - Wear	Surveillance	Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): 1. Initiate a sampling strategy to inspect one valve per valve group and replace based on inspection results. 2. Procure spare NV's.
011358	5, 6, 7, 8	32000 32510	32510 Moderator D2O Collection Pumps	D2O collection pumps return reactor grade contents of D2O collection tank via the purification system to the moderator system.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion Corrosion / Abrasion, Erosion and Cavitation		Incremental recommendations for Plant EOL (2020): 1. Perform a one-time inspection of one D2O collection pump and modify practices as required based on inspection results. 2. Procure a spare moderator D2O collection Pump. Incremental recommendations for CO EOL (2024): No additional practices are recommend to reach CO EOL (2024).
011359	5, 6, 7, 8	32000 32310	32310 Moderator Cover Gas Compressors	Moderator cover gas compressors circulate cover gas through recombination units to prevent build-up of deuterium in cover gas.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Fatigue / High Vibrations Corrosion / General Corrosion	Clean and Inspect Lubrication Overhaul/Refurbishment	Incremental recommendations for Plant EOL (2020): Complete replacement of compressors under Master EC# 129405 prior to 2018. Incremental recommendations for CO EOL (2024): No additional practices are recommend to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011360	5, 6, 7, 8	32000 32110	32110 Moderator System-Valves - HX Manual Isolators	Manual valves are used to isolate or drain processes to allow for maintenance and other functions. 5,6,7,8-32110-V10/V11/V12/V13 are moderator heat exchanger isolators. 5,6,7,8-32110-V20 are moderator purification isolators. 5,6,7,8-32110-V36 and V40 isolate moderator from EWS. They are opened during EWS recovery following an accident.	Satisfactory	Corrosion / Pitting Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion	Inspection - Internal Inspection - Visual SRST - Functional Test	Incremental recommendations for Plant EOL (2020): Perform a one-time test/inspection of 5/6/7/8-32110-V20 to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Perform a one-time test/inspection of 5/6/7/8-32110-V20 to reach CO EOL (2024), if required based on previous inspection/test.
011361	5, 6, 7, 8	32000 32210	32210 Moderator System-Valves - Purification Filters Manual Isolators	Manual valves are used to isolate various equipment associated with Moderator D2O purification system.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue Corrosion / General Corrosion		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete the following outstanding work orders; WO 01598752 5-32210-V2 REPLACE VALVE WO 01598756 5-32210-V3 REPLACE VALVE WO 01598757 5-32210-V4 REPLACE VALVE WO 01598750 5-32210-V1 REPLACE VALVE WO 01598761 REPLACE VALVE 6-32210-V1 WO 01598764 REPLACE VALVE 6-32210-V2 WO 01598767 REPLACE VALVE 6-32210-V3 WO 01598769 REPLACE VALVE 6-32210-V4 WO 03218140 7-32210-V1 SUSPECTED PASSING - REPLACEMENT REQUIRED WO 03218149 7-32210-V3 SUSPECTED PASSING - REPLACEMENT REQUIRED Incremental recommendations for Plant EOL (2020):

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
								Add valves 5/6/7/8-32110-V3, V4, V14 to SRE walkdowns. Incremental recommendations for CO EOL (2024): No additional practices are recommend to reach CO EOL (2024).
011362	5, 6, 7, 8	32000 32110 32210 32310 32710	32110 Moderator System-Valves - Pump Suction Manual Isolators	These manual valves are used to isolate associated moderator system equipment.	Satisfactory	Pitting Corrosion / Pitting Corrosion Mechanical and Thermal / Deterioration of Material (Joint Seals Gaskets etc.) Wear Mechanisms - Wear / Wear Mechanisms - Wear		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete outstanding WOs for replacement, e.g.: WO# 01581304, 01581304. Incremental recommendations for Plant EOL (2020): 1. Complete a one-time stroke test of the CC1/CC2 valves. 2. Resolve spares issues for Cat ID 120574. Incremental recommendations for CO EOL (2024): Initiate a new PM to stroke test CC1/CC2 valves every 104 weeks (or at minimum during unit outages).
011363	5, 6, 7, 8	32000 32210 32710	32000 Moderator System-Valves - Manual Isolators	These manual valves are used to isolate and/or drain associated equipment (check valves, pumps, integral orifice flow transmitters, ion exchangers, spool piece isolators and drain valves).	Satisfactory	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion	Inspection - Visual	Incremental recommendations for Plant EOL (2020): Perform a one-time inspection of the CC2 components in this CG and repair / replace as required to reach Plant EOL (2020) (excluding V15 & V21, which have associated PMs in place to replace diaphragm). Incremental recommendations for CO EOL (2024): Perform a one-time inspection/replacement for the CC2 valves in this CG to reach CO EOL (2024), (excluding V15 & V21, which have associated PMs in place to replace diaphragm).
011364	5,6,7,8	32000	32110, 32310, 32510, 32710, 63230, 63260	These cables are used to supply electric power to 5kV, 600V, and 300V loads	Good	Fatigue / Thermal Fatigue	Inspection - Condition Monitoring Program	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
			Moderator System - Power Cables - 4.16 kV, 600V, 120V	as well as specially defined purposes.		Fatigue / Mechanical Fatigue Radiation Induced Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		
011365	5, 6, 7, 8	32000 32310	32310 - Moderator Cover Gas Heater - 120 Volt	These heaters, strapped on to the pipe run, function as a Moderator Cover Gas pre-heater to dry the moderator cover gas mixture to prevent the catalyst in the recombination unit (RU) from becoming moist and losing its effectiveness.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Surveillance	<u>Incremental recommendations for Plant EOL (2020):</u> No additional recommendations required to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Periodically conduct heater inspection with a contingency task to replace as required.

System 0447 - Powerhouse Emergency Venting System

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011166	5,6,7,8	73200 67324	67324 Powerhouse Emergency Venting Panels	The vent panels are normally closed panels in the Turbine Hall (TH) & Turbine Auxiliary Bay (TAB) walls. These panels are opened rapidly in a steam line accident to maintain acceptable ambient conditions.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Overhaul/Refurbishment SRST - Functional Test SRST - Logic Test	Incremental recommendations for Plant EOL (2020): Replace rusted latching plates on all panels. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011424	058	73200 67322	67322 - Powerhouse Emergency Venting System-SIGNAL CONDITIONER -CAT 1&2	Execute logic for Powerhouse Emergency Ventilation System.	Satisfactory	Thermal Aging / Thermal Aging Mechanical and Thermal Degradation / Electronic Aging Obsolescence / Immediate Obsolescence Concern	SRST - Functional Test	Incremental recommendations for Plant EOL (2020): Resolve obsolescence and spares issues. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

System 0449 - Primary Heat Transport System

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
009884	1,4	33000 33120	ACTUATOR, Cat 1/2	Drive associated motorized HTS interconnect valves.	Satisfactory	Mechanical and Thermal Degradation / Lubricant Degradation Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Thermal Aging Radiation Induced Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Component Diagnostics Inspection - Internal Inspection - Visual Lubrication Overhaul/Refurbishment SRST - Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009885	1,4	33000 33610 33710	ACTUATOR, Cat 1/2, Liquid Relief Valve	Actuate the associated HTS CVs or MVs.	Poor	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear Radiation Induced Degradation / Radiation Embrittlement		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete EC#104737 to address low margin issue. Incremental recommendations for Plant EOL (2020): Resolve obsolescence and spares issues. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Obsolescence / Immediate Obsolescence Concern		
009886	1,4	33000 33120 63331 63332 63333 63334 63360	ALARM, Cat 1/2	Generate alarms when the monitored parameters exceed predefined limits. Indicating alarms also provide indications of the parameters monitored.	Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009888	1,4	33000 63310	AMPLIFIER, Cat 3/4	Amplify signals from main HTS pump speed meters.	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009895	1,4	33000 33320	CONDENSER, Cat 1/2	Bleed circuit consists of two bleed control valves (one per loop) which remove heavy water from the west pump suction headers. Flashed vapor is condensed via the bleed condenser. The bleed condenser is a carbon steel pressure vessel, 3 ft. 7-3/4 in. outside diameter by approximately 19 ft. long, designed to the ASME BPVC III as a class A vessel. The free internal volume of the tank is 140 ft3. (Refer to NA44-DM-33300-00001 & P-SPM-33000-0558424)	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Fretting Corrosion / Fouling (accumulation of deposits) Corrosion / Stress Corrosion Cracking - Nickel-base Alloys Fatigue / Thermal Fatigue Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring,	Inspection - Internal Inspection - Eddy Current Test Inspection - Pressure Test Inspection - Visual	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Seals, Gaskets, O-rings etc.)		
009897	1,4	33000 63331 63332 63336	CONTROL, Cat 1/2	Control level, pressure, or temperature.	Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging		<u>Incremental recommendations for Plant EOL (2020):</u> Complete proactive replacement of controllers. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
009899	1,4	33000 33210 63333	CONVERTER, Cat 1/2	1-33210-CV22-AX1 converts current signal to pressure signal for a control valve; 1/4-63333-L1-LT2 coverts voltage signal to current signal for D2O tank level control.	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging		<u>Incremental recommendations for Plant EOL (2020):</u> Complete proactive replacement. <u>Incremental recommendations for CO EOL (2024):</u> Calibrations to be conducted at a frequency adequate for each device.
009900	1,4	33000 33120	DETECTOR, Cat 1/2	Generate signals representing the speed of main HTS pumps.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Electronic Aging	Surveillance	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009901	1,4	33000 33330	DISC, Cat 1/2	Provide overpressure protection for D2O Storage Tank 33330-TK1.	Good	Corrosion / General Corrosion		<u>Incremental recommendations for Plant EOL (2020):</u> Perform one-time replacement to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
009903	1,3,4	33000 33120 33310 63312 63332 63333 63334	ELEMENT (TEMP.), Cat 1/23	Provide signals representing temperatures to temperature transmitters for HTS components.	Good	Fatigue / Thermal Fatigue Fatigue / Environmentally-Assisted Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring,	Calibration Overhaul/Refurbishment	<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Initiate new PMs for Boiler Outlet, Pump Motor & Pump Gland Circuit TEs to perform calibrations at a frequency adequate for each temperature loop.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Seals, Gaskets, O-rings etc.)		
009905	1,4	33000 33320 33610 63320 63331 63332 63334	ELEMENT (FLOW), Cat 1/2	Generate signals representing flow rates for various HTS flow paths.	Good	Corrosion / Flow induced wear of the leading edge	Calibration	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009910	1,4	33000 33210 33340	FILTER (GLAND/BLEED), Cat 2/3	Filters remove soluble and insoluble impurities from the PHT Bleed Circuit (33210) and Gland Seal Circuit (33340). Impurities include activated corrosion products, fission products, non-active ionic impurities and particulates.	Good	Corrosion / Fouling (accumulation of deposits) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Incremental recommendations for Plant EOL (2020): Perform one-time inspection of filter housings, visual if possible (with remote camera), and ultrasonic for thickness measurement. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
009921	1,4	33000 33320	HEAT EXCHANGER(BLEED COOLER), Cat 1/2	Cool the hot D2O condensate from the bleed condenser prior to condensate entering the bleed purification circuit, to prevent IX resin breakdown.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Fretting Corrosion / Fouling (accumulation of deposits) Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion	Clean and Inspect	Incremental recommendations for Plant EOL (2020): Perform one-time inspection of heat exchangers, and if degraded repair / replace. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
009923	1,4	33000 33340	HEAT EXCHANGER (GLAND), Cat 1/2	These heat exchangers are heat transport pump gland seal water coolers.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Fretting Corrosion / Fouling (accumulation of deposits) Corrosion / General Corrosion	Inspection - Internal	Incremental recommendations for Plant EOL (2020): 1. Remove an HX from U2 or U3, inspect to confirm good condition, then use it to replace an HX from Unit 4. The HX from U4 to be used to determine the current condition of the active gland cooler heat exchangers. 2. Based on findings, create WOs for replacement of HXs.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Wear Mechanisms - Erosion		3. Purchase spare HXs as needed based on findings. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
009924	1,3,4	33000 33810	HEAT EXCHANGER (COLLECTION) , Cat 1/2	This HX cools D2O collected in the D2O collection tank to prevent vapor locking/cavitation of the D2O collection pumps and to protect resin beads in the purification system.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Fretting Corrosion / Fouling (accumulation of deposits) Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Fatigue / Thermal Fatigue	Surveillance	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete WO 1418348 &1645320 to replace HXs. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Based on the condition of the replaced HXs, implement PMs to inspect/clean HX at frequency of 4 years, if required.
009926	1,4	33000 33320	HEATER (Bleed Cond), Cat 1/2	Provide heating to HTS bleed condensers.	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Thermal Aging		Incremental recommendations for Plant EOL (2020): One-time inspection of heater, and if degraded repair/replace. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
009927	1,4	33000 33120	HEADER / MANIFOLD (33120), Cat 3/4	33120-HD1, HD6, HD7, HD12: Reactor Outlet Headers. 33120-HD3, HD4, HD9, HD10: Reactor Inlet Headers. 33120-HD2, HD5, HD8, HD11: PHT Pump Discharge Headers.	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Fatigue / Thermal Fatigue		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Inspection of selected locations for headers and supports per PIP (NA44-PIP-03641.2-00001 for U1 and NA44-PIP-03641.2-00007). Based on inspection results, generate additional WOs for further inspection and/or repair, as required.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Thermal Aging		Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Ensure the schedule for inspection of selected locations for headers and supports per PIP (NA44-PIP-03641.2-00001 for U1 and NA44-PIP-03641.2-00007 for U4 is updated to accommodate CO EOL (2024).
009931	1,4	33000 63320	INDICATOR, Cat 1/2	Provide indications of bleed filter flows.	Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009941	1,4	33000 63332 63333	MODULE, Cat 1/2	Execute a part of the logic to control the levels of bleed condenser and D2O storage tank.	Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009943	1,4	33000 33120	MOTOR (MCP), Cat 2	Drive main HTS pumps which circulate heavy water (D2O) through the boilers.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Lubricant Degradation Radiation Induced Degradation /		Incremental recommendations for Plant EOL (2020): Refurbish sufficient number of removed motors to serve as spares. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Radiation Embrittlement		
009945	1,4	33000 33310	MOTOR (PRESS PUMP), Cat 2	Drive HTS pressurizing pumps.	Good	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Lubricant Degradation Radiation Induced Degradation / Radiation Embrittlement	Lubrication Predictive Maintenance - Thermography	<u>Incremental recommendations for Plant EOL (2020):</u> Include vibration monitoring in Super Route PM, to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
009946	1,4	33000 33910	MOTOR (RECOVERY), Cat 2	Drive D2O recovery pumps.	Good	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Lubricant Degradation	Lubrication Overhaul/Refurbishment Predictive Maintenance - Vibration Monitoring	<u>Incremental recommendations for Plant EOL (2020):</u> Complete proactive replacement of motors. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Radiation Induced Degradation / Radiation Embrittlement		
009947	1,4	33000 33710	MOTOR (TRANSFER), Cat 1	Drive D2O transfer pumps.	Good	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Lubricant Degradation Radiation Induced Degradation / Radiation Embrittlement	Surveillance	<u>Incremental recommendations for Plant EOL (2020):</u> Implement new PM to perform electrical testing every 2 years and vibration analysis to be performed every year. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
009952	1,4	33000 33210 33310 33320 33340	POSITIONER, Cat 1/2, 3582D	Adjust the supply pressures to miscellaneous HTS control valve actuators to position the valves at the desired points corresponding to the input signals.	Very Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Radiation Induced Degradation / Radiation Embrittlement		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
009953	1,4	33000 33320	POSITIONER, Cat 1/2, 3582i	Adjust the supply pressures to the HTS bleed control valves to position the valves at the desired points corresponding to the input signals.	Very Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging Radiation Induced Degradation / Radiation Embrittlement		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009954	1,4	33000 33320	POSITIONER, Cat 1/2, 3573	Adjust the supply pressures to the Bleed Condenser level control valve actuators to position the valves at the desired points corresponding to the input signals.	Very Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Radiation Induced Degradation / Radiation Embrittlement		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009957	1,4	33000 63336	PROCESSOR, Cat 1/2	Select median signals and send them to the HTS pressure hand controllers.	Good	Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Thermal Aging		
009959	1,4	33000 33120 33310 33710 33910	PUMP, Cat 1/2	1/4-33710-P1 (D2O Transfer Pump) Transfers D2O to other Units via D2O Transfer Header. Required to automatically trip upon receipt of an ECI LOCA signal, or manually from the MCR. 1/4-33910-P1 (D2O Recovery Pump) Delivers heavy water from the D2O Recovery Tank (33910-TK1) back to the HT system at the pressurizing pump suction header. 1/4-33120-P1 to -P16 (Main Primary Heat Transport Pumps) Circulate D2O through the reactor. The pumps draw from the boilers via a suction header and pump to the Reactor Inlet Header which distributes flow to individual inlet feeders of the reactor 33310-P1 (PHT Pressurizing Pumps) Provide HT pressurizing flow to the reactor outlet headers to counteract HT shrinkage. The pumps also supply gland flow to the HT main circulating pumps and shutdown cooling pumps, cooling flow to the bleed condenser reflux or spray circuits and pressurizing flow to the fuelling machines when they require relatively low pressure.	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / Fouling (accumulation of deposits) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Corrosion / Flow induced wear of the leading edge	Inspection - Visual Lubrication Predictive Maintenance - Infrared Analysis Predictive Maintenance - Vibration Monitoring SRST - Functional Test	Incremental recommendations for Plant EOL (2020): Perform a one-time inspection of pump internals (impeller, casing, shaft etc.). Inspect one primary heat transport circulation pump, pressurizing pump and D2O transfer pump. Based on inspection results create supplemental WOs to inspect/overhaul other pumps within this CG. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
009960	1,4	33000 33810 34910 34940 34960	PUMP, Cat 3/4	1/4-33810-P1 and -P2 (D2O Collection Pumps) Transfer D2O from collection tank 33810-TK1 to the PHT Purification System, Downgraded D2O System or the IXCU. 1/4-34910-P1 (Misc. D2O Collection Pumps) Transfer D2O from collection tank 34910-TK1 to the PHT System, or Downgraded D2O System. 1/4-34940-P2 (Ion Exchange D2O Sampling Pump) Transfer D2O from Ion Exchange system to Moderator Dump Tank Return. 1/4-34960-P1 (Vapour Recovery Transfer Pump) Transfer D2O from D2O vapour recovery to D2O Storage or Oil Separators.	Very Good	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Lubricant Contamination Corrosion / Fouling (accumulation of deposits) Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Wear Mechanisms - Erosion		<u>Incremental recommendations for Plant EOL (2020):</u> Perform a one-time inspection, if degraded repair/replace (for 1/4-34910-P1, 1/4-34940-P2 and 34960-P1). <u>Incremental recommendations for CO EOL (2024):</u> Proactive replacement of seals to be completed at a frequency of 4 years (for 1/4-34910-P1, 1/4-34940-P2 and 34960-P1).
009961	1,4	33000 63333 63336	RECORDER, Cat 1/2	Record D2O tank level or HTS pressure.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete existing review to determine if Heat Transport trend recorders should be replaced. <u>Incremental recommendations for Plant EOL (2020):</u> 1. Perform one-time replacement of recorder capacitors. 2. Ensure spare recorders are available. <u>Incremental recommendations for CO EOL (2024):</u>

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
								1. Perform one-time replacement of1/4-63333-L1-LR1. 2. Add calibration of recorders 1/4-63336-P17A-PR1/P18A-PR1 to PM's for loop calibration.
009962	1,4	33000 63320 63332 63336 63381	RECORDER, Cat 3/4	Record conductivity, level, or pressure.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging		<u>Incremental recommendations for Plant EOL (2020):</u> Perform one-time replacement of recorder capacitors. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
009965	1,4	33000 63312 63330	RELAY, Cat 1/2	Implement a part of valve and pump control logic.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Calibration Predictive Maintenance - Thermography SRST - Functional Test SRST - Logic Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009967	1,4	33000 33320	RELAY (BOOSTER), Cat 1/2	Provide boosting actions to the positioners for HTS bleed and reflux control valves.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Creep and Stress Relaxation Fatigue / Mechanical Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Wear Mechanisms - Wear		
009968	1,4	33000 33610	SHOCK ABSORBER OR SNUBBER ASSE, Cat 1/2	Heat Transport LRV pipe support (shock absorber) for 6" line 33610-L2 from Bleed Condenser to Liquid Relief Valves.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Fretting Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009970	1,4	33000 33340	STRAINER, Cat 1/2	Gland filter bypass strainers capture resin when filter bypass used. Failure of strainer can allow debris flow to pump seals if bypass line used (i.e. 33340-MV341 opened) without confirmation of strainer integrity.	Good	Corrosion / General Corrosion Corrosion / Fouling (accumulation of deposits) Mechanical and Thermal Degradation / Wear Mechanisms - Erosion	Surveillance	<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Complete a one-time strainer removal, inspection and replace or clean as necessary. Based on condition of strainers, supplemental WOs to be generated.
009976	1,4	33000 33330 33810	TANK (33810), Cat 1/2	1/4-33330-TK1: D2O storage tank Provides feed to the pressurizing pumps and receives heavy water from the purification system. During shutdown conditions (i.e. maintenance outages) the storage tank provides a continuous pressure to the core via the level control header and the shutdown cooling pump loops. 1/4-33810-TK1: D2O collection tank. Collects D2O recovered by	Good	Corrosion / General Corrosion	Inspection - Visual SRST - Functional Test Tank Level Monitoring	<u>Incremental recommendations for Plant EOL (2020):</u> Perform a one-time inspection, and if degraded repair/replace (for all tanks). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				the D2O Collection System, for return to the PHT (via the PHT Purification Circuit).				
009978	1,4	33000 63360	TANK (BUIA 33610), Cat 1/2	Instrument Air Reservoir for PHT Liquid Relief Valves 33610-CV1 to -CV4, which provide overpressure protection for the PHT System Under accident conditions, the valves are required to operate on demand to relieve HTS overpressure (30 minute mission time). The tanks provide the air reserve for the required mission time if instrument air is lost.	Good	Corrosion / General Corrosion	Inspection - Visual SRST - Air Holding Test	<u>Incremental recommendations for Plant EOL (2020):</u> Perform a one-time inspection, and if degraded repair/replace (for all tanks). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
009982	1,4	33000 33210 33310 33320 33340	TRANSDUCER, Cat 1/2	Convert current signals to pressure signals to modulate HTS control valves.	Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009984	1,2,4	33000 63312 63320 63331 63332 63333 63334 63336 63360	TRANSMITTER, Cat 1/2/3	Transmit pressure, flow, temperature, and level signals	Satisfactory	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging Fatigue / Environmentally-Assisted Fatigue Mechanical and Thermal Degradation / Creep and Stress Relaxation	Calibration	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete life cycle replacements and procure spares. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Perform one-time replacement of all transmitters without replacement PM.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
009986	1,4	33000 33210 33310 33320 33340	VALVE, CONTROL, Cat 1/2, 657/667 w/ A or DBQ	33310-CV3, CV6 – Flow control for PHT feed. 33310-CV10 – Pressure-reducing control valve for D2O supply from PHT Feed to Fuelling Machine. 33310-CV9 – Pressure-reducing control valve for D2O return from Fuelling Machine to PHT Feed. 33320-CV102, 104 – Control bleed flow from PHT Pump Suction Headers to Bleed Condenser. 33320-CV111 – Bleed Condenser pressure control 33320-CV113 – Bleed Condenser spray nozzle flow control 33210-CV22 – HT Purification bypass flow control.	Very Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Calibration Overhaul/Refurbishment Valve Diagnostics	<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Perform a one-time internal inspection of a representative control valve sample from each functional group. Raise supplemental WO's for overhaul or replacement as a result of internal inspection findings.
009987	1,4	33000 33610	VALVE, CONTROL, Cat 1/2, Liquid Relief Valves (Actuator, see CG490021)	Instrumented Liquid Relief Valves (LRVs) control discharge of heavy water from the heat transport system to the bleed condenser during HTS over-pressure transients.	Very Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Overhaul/Refurbishment SRST - Air Holding Test SRST - Functional Test SRST - Stroke Test Valve Diagnostics	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009988	1,4	33000 33320	VALVE, CONTROL, Cat 1/2, 473 HSV	Used in bleed condenser level control. The valves regulate flow through the bleed cooler from the PHT purification system, to maintain a constant level in the bleed condenser when bleed flow is being discharged into the bleed condenser.	Very Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Calibration Inspection - Visual SRST - Functional Test SRST - Leak Tightness Test SRST - Logic Test Valve Diagnostics	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
009991	1,4	33000 33120 33210 33320 33330 33340 33540 33910 34960 63336	VALVE, MANUAL/HAND OPERATED, Cat 1/2	Manual hand operated valves used for isolation purposes.	Satisfactory	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Thermal Aging	Overhaul/Refurbishment	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Perform scheduled replacements of elastomeric gaskets/packing and/or valves with materials vulnerable to aging (e.g. diaphragm valves). Incremental recommendations for Plant EOL (2020): 1. Resolve obsolescence issues with 1/4-33320-V110/V119, 1/4-33540-V21/V2. 2. Update SPMP to include all manual valves in CG. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
009993	1,4	33000 33120	VALVE, MOTORIZED/ MOTOR OPERATE, Cat 1/2	1/4-33120-MV1 to MV24: Boiler Isolating Valves (48): Permit isolation of a boiler with failed tubes. Closed for the boiler to be drained. 1/4-33120-MV25 to MV40: Main PHT Pump Discharge Valves (32): Prevent reverse rotation of a stopped pump, and also permit starting of pumps with closed discharge, if desired. 1/4-33120-MV42 and MV43: PHT Interconnect Valves (4): Provide isolation for 4" interconnect lines connecting north and south Reactor Outlet Headers.	Poor	Corrosion / General Corrosion Mechanical and Thermal Degradation / Lubricant Degradation Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Inspection - Visual Inspection - Internal Lubrication Valve Diagnostics	Incremental recommendations for Plant EOL (2020): 1. Proactively refurbish a contingency of actuators for fast swapping with malfunctioning actuators during outages (for Boiler Isolating Valves and Pump Discharge Valves). 2. Perform one-time inspection, and if degraded repair/replace. 3. Replace worn out stem nuts (for all valves). Incremental recommendations for CO EOL (2024): Perform recurring inspection/lubrication at a frequency of 2 years for all valves/actuators.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				Closed automatically on high Boiler Room pressure plus LOCA signal.				
009996	1,4	33000 33610	VALVE, CHECK/NONRETURN/BACKFL, Cat 1/2	Check valve on instrument air supply of PHT Liquid Relief Valve actuators.	Very Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009998	1,4	33000 33310 33330	VALVE, CHECK/NONRETURN/BACKFL, Cat 1/2, Swing HTS-05, 11	33310-NV14, 33310-NV18 – prevent back flow to heavy water pressurizing pumps 33310-P1 and 33310-P2, respectively. 3310-NV202 – prevent back flow to pressurizing pumps.	Satisfactory	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Inspection - Internal Inspection - Radiography	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Inspect/repair/replace 4-33310-NV14. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
009999	1,4	33000 33310	VALVE, CHECK/NONRETURN/BACKFL, Cat 1/2, Swing HTS-06	33310-NV16 – in parallel with pressurizing pumps is a 4 inch bypass line equipped with NV16. This line is provided as a low-resistance path to the heat transport system for water from D2O recovery system.	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Inspection - Radiography Inspection - Visual SRST - Logic Test SRST - Stroke Test	<u>Incremental recommendations for Plant EOL (2020):</u> Perform a one-time inspection of valve internals and if degraded, repair/replace valve. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
010000	1,4	33000 33910	VALVE, CHECK/NONRETURN/BACKFL, Cat 1/2, Swing HTS-30, 30A, 33, 34	33910-NV6, NV7 – Check valves for discharge of D2O recovery pumps 33910-P1 and P2. 33910-NV1, NV2003 – check valves for pump discharge valves MV2 and MV2002.	Very Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010001	1,4	33000 33340	VALVE, CHECK/NONRETURN/BACKFL, Cat 1/2, Ball&Piston, HTS-01, 01A	33340-NV527 – Pressurizing Header to Main Primary Heat Transport Pumps Gland Filter Supply Check Valve	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Corrosion / General Corrosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of	Surveillance	<u>Incremental recommendations for Plant EOL (2020):</u> Perform one-time inspection, and if degraded repair/replace valves. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).

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CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Material (Wiring, Seals, Gaskets, O-rings etc.)		
010002	1,4	33000 33310	VALVE, CHECK/NONRETURN/BACKFL, Cat 1/2, Ball & Piston, HTS-03, 07	1/4-33310-NV1, NV2 – prevent backflow from the Main Heat Transport System to PHT Feed System (lines 33310-L1 and 33310-L2, respectively). 1/4-33310-NV30, NV31 - prevent backflow on the lines from the PHT Feed pressurizing pumps (33310-P1/P2) to the Fuelling Machine Supply.	Satisfactory	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Incremental recommendations for Plant EOL (2020): Perform a one-time inspection of valve internals and if degraded repair/replace (for 1/4-33310-NV30/NV31). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010003	1,4	33000 33340	VALVE, CHECK/NONRETURN/BACKFL, Cat 1/2, Ball&Piston, HTS-14	33340-NV348 – check valve downstream of 33340-MV308 (Gland Return Valve on return line to HT Purification).	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Inspection - Radiography Inspection - Visual	Incremental recommendations for Plant EOL (2020): Perform one-time internal inspection, and if degraded repair/replace valves. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010004	1,4	33000 33320 33330 33340	VALVE, CHECK/NONRETURN/BACKFL, Cat 1/2, Ball & Piston, HTS-09, 10, 12, 13	D2O Storage Circuit: 33320-NV114 – prevents accidental backflow to the pressurizing pump header 33320-NV112 – prevents backflow through the reflux condenser. 33320-NV216 – helium supply to D2O storage tank, prevents backflow of helium	Very Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Fatigue / Mechanical Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				supply. Gland Seal Circuit 33340-NV311 – PHT Main Pumps 33120-P13-P16 gland seal supply. 33340-NV312 – PHT Main Pumps 33120-P5-P8 gland seal supply. 33340-NV313 – PHT Main Pumps 33120 P1-P4 gland seal supply. 33340-NV315 – PHT Main Pumps 33120 P9-P12 gland seal supply.		Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		
010005	1,4	33000 33210	VALVE, CHECK/NONRETURN/BACKFL, Cat 1/2, Swing HTS-02	33210-NV36, NV37 – check valves prevent back flow from the HT System through the filters and ion exchangers of the PHT Purification Circuit.	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Surveillance	Incremental recommendations for Plant EOL (2020): Perform a one-time inspection of a valve from this CG. Based on the findings generate WOs for inspection/overhaul/replacement. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010006	1,4	33000 33610	VALVE, CHECK/NONRETURN/BACKFL, Cat 1/2, NVs to CVs	33610-NV4002, NV4006, NV4010, NV4014 – check valves prevent backflow on instrument air supply to the actuators of the PHT Liquid Relief Valves (1/4-33610-CV1 to CV4).	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Fatigue / Mechanical Fatigue		Continue current practice. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Creep and Stress Relaxation		
010007	1,4	33000 33610	VALVE, CHECK/NONRETURN/BACKFL, Cat 1/2, Ball & Piston, HTS-23	33610-NV3001, NV3002, NV3003, NV3004, NV3005, NV3006, NV3007, NV3008, NV3009, NV3010, NV3011, NV3012 – boiler over pressure relief non-return valve.	Satisfactory	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Creep and Stress Relaxation	Surveillance	Incremental recommendations for Plant EOL (2020): Perform a one-time sample inspection of 1-33610-NV3001. Determine if further inspections are required from the results of inspection. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010008	1,4	33000 33540	VALVE, CHECK/NONRETURN/BACKFL, Cat 1/2, Ball & Piston, HTS-18	33540-NV5, NV7 – (in Gas Control D2/H2 Addition System, 33540) prevents backflow during on power operation to provide continuous injection of normal supply of D2/H2, to remove oxygen from the HTS.	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Fatigue / Mechanical Fatigue	Inspection - Radiography	Incremental recommendations for Plant EOL (2020): Perform one-time inspection for one representative valve in each unit. From inspection results, determine if any further inspections and/or replacements are required. Incremental recommendations for CO EOL (2024):

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Creep and Stress Relaxation		No additional practices are recommended to reach CO EOL (2024).
010009	1,4	33000 33120	VALVE, CHECK/NONRETURN/BACK FL, Cat 1/2, Ball&Piston, HTS-16	33120-NV47, NV48, NV49, NV50, NV53, NV54, NV55, NV56, NV57, NV58, NV59, NV60, NV63, NV64, NV65, NV66 – are non-return valves from SDCS warmup circuit to the PHTS.	Satisfactory	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Fatigue / Mechanical Fatigue	Surveillance	<u>Incremental recommendations for Plant EOL (2020):</u> Inspect 1/4-33120-NV49, 1/4-33120-NV54, 1/4-33120-NV58 & 1/4-33120-NV64 and determine if further inspections or replacements are required based on inspection results. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
010010	1,4	33000 33120 33310 33320 33330 33340	VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2, Martonair Actuators, w newmen Hender valves	33120-MV45, MV51, MV61, MV67 – Main PHT pump warm up valves 33120-MV46, MV52, MV62, MV68 – Main PHT cool down valves 33320-MV142 – D2O Collection return isolation valve 33330-MV217 – Isolation valve on Helium supply to D2O Storage Tank 33340-MV308 – Gland return valve 33340-MV341 – Gland Filter bypass valve 33340-MV530 – HT test circuit isolating valve	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Calibration Inspection - Visual Overhaul/Refurbishment SRST - Functional Test SRST - Logic Test SRST - Stroke Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				33310-MV5, MV22 – Isolation valves on Heat Transport to fueling machine supply		Mechanical and Thermal Degradation / Wear Mechanisms - Erosion		
010012	1,4	33000 33910	VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2, AOV Program Valve, D2O Recovery Return	33910-MV2/MV2002: D2O recovery isolating valves – under normal operation valves are poised closed. Valves are open in “normal operation” condition, when requirement for D2O recovery is established. Valves are operated in parallel.	Very Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Inspection - Radiography Inspection - Visual Overhaul/Refurbishment SRST - Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010013	1,4	33000 33710	VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2, AOV Program Valve, D2O TRANSFER P1 BYPASS MV	D2O transfer pump bypass valves are normally in the closed position. The bypass valve is only open to receive D2O from another unit. The bypass valve has a safety function to open to permit receiving D2O from other units, to prevent heat transfer piping voids as a result of shrinkage during cool down. The bypass valve also has a safety function to close, while another unit suffers a LOCA.	Very Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue	Surveillance	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010014	1,4	33000 33320	VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2, Persta Valve w/ Actuator	These are isolation valves for: 33320-MV103 inlet to bleed condenser, valve is normally open. 33320-MV107 bypass around bleed condenser, valve is normally closed. 33320-MV109 bleed condenser	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Air Holding Test Inspection - Internal Inspection - Visual Overhaul/Refurbishment Stroke Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				outlet valve, normally open.		Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue		
010019	1,4,01 2,034	33000 33210 33310 33320 33330 33530 33540 33610	VALVE, PRESSURE RELIEF, Cat 1/2	These RVs provide overpressure protection to the PHT System.	Very Good	Corrosion / General Corrosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / Fouling (accumulation of deposits)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010020	1,4	33000 33120 33310 33320 33330 33340 33610 33710 33910 34960	SOLENOID/SO LENOID OPERATED VAL, Cat 1/2	Operate associated CVs or MVs of the HTS system.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Thermal Aging Radiation Induced Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Creep and Stress Relaxation	Overhaul/Refurbishment SRST - Functional Test	Incremental recommendations for Plant EOL (2020): Obtain sufficient spares for 1/4-33310-MV5-SV1, 1/4-33330-MV217-SV1 & 1/4-34960-MV3008-SV1. Incremental recommendations for CO EOL (2024): 1. Perform one-time replacement of associated SV's for 1/4-33910-MV2002 & 1/4-34960-MV3008. 2. Perform one-time replacement of associated SVs for 1/4-33610-CV1-SV1/2, 1/4-33610-CV2-SV1/2, 1/4-33610-CV3-SV1/2, and 1/4-33610-CV4-SV1/2.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011113	5,6,7,8	33000 33120	33120 HT Pump Preheat and Cooling Valves	HT pump warmup valves, 33120-MV45, 51, 61, 67, must be opened during start-up/warm-up of HT system, remain open while operating to establish warming flow to the non-operating pumps, and closed during HT cooldown. The main circulating pumps cool down valves, 33120-MV46, 52, 63, 68, will be opened to provide cooling of the main pumps by the shutdown cooling circuits.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Fatigue / Mechanical Fatigue	Surveillance	<u>Incremental recommendations for Plant EOL (2020):</u> 1. Perform a one-time inspection of sample valves from each unit. 2. Provide BOM in Passport and ensure new cast urethane disc material for SVs is included in BOM and update reorder points and target maximums to ensure adequate supplies of parts are available and stocked. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011114	5,6,7,8	33000	33120 PHT Pump Motor Stator Coolers	The heat exchangers in this CG are coolers for the PHT pump motors 5/6/7/8-33120-PM1 to PM16. There are two coolers per pump motor utilized to cool the air circulating to the motors.	Good	Corrosion / Pitting Corrosion Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Replace all coolers to ensure condition to CO EOL (2024).
011115	5,6,7,8	33000	33320 Bleed Condenser Bypass Isolating Valve	5/6/7/8-33320-MV107 is the normally closed bleed condenser bypass isolating valve.	Good	Corrosion / Crevice Corrosion Corrosion / General Corrosion Mechanical and Thermal Degradation		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete documentation of replacement model Cat ID 559647. <u>Incremental recommendations for Plant EOL (2020):</u>

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						/ Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011116	5,6,7,8	33000 33320	33320 Bleed Condenser Inlet Isolating Valve	The valve function in HT Bleed System is to open to put the Bleed Condenser into service. The valve can also bypass the bleed condenser by closing, while MV107 opens.	Good	Corrosion / Crevice Corrosion Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Inspection - Visual Overhaul/Refurbishment Stroke Test	Incremental recommendations for Plant EOL (2020): 1. Add a task to the current PM actuator overhaul to replace the limit switch. 2. Create separate PMs for 5-8-33320-MV103 stroke tests. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011117	5,6,7,8	33000 33320	33320 Heat Transport Bleed Condensers	Bleed flow is flashed to a substantially lower pressure in the bleed condenser, which is equipped to condense the flashed vapor. The use of this vessel also provides a means for degassing. The safety related function is to provide alternate heat sink (maintain HTS pressure control). The Heat Transport System (HTS) pressure is controlled by injection and bleeding of HTS D2O. Part of the hot and high pressure HTS D2O is bled and condensed in the bleed condensers. There are two modes of operation. Under Normal Mode, cold HTS D2O flows through the tubes of the	Poor	Mechanical and Thermal Degradation / Wear Mechanisms - Fretting Fatigue / Thermal Fatigue Corrosion / General Corrosion Corrosion / Fouling (accumulation of deposits) Corrosion / Stress Corrosion Cracking - Nickel-base Alloys Fatigue / Mechanical Fatigue	Inspection - Eddy Current Test Inspection - Pressure Test Inspection - Visual SRST - Functional Test SRST - Logic Test	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Inspection of Unit 7 tube bundle is scheduled under WO 4819201 during P1671. Further inspections for P058 side tube bundles to be scheduled based on the outcome of this inspection. Incremental recommendations for Plant EOL (2020): Procure spares as required to support inspection/tube bundle replacement. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				reflux condenser condensing the hot vaporized HTS D2O that is bled. Under Spray Mode, cold HTS D2O is sprayed through the spray nozzles to cool & condense the hot D2O.				
011118	5,6,7,8	33000 33320	33320 Heat Transport Bleed Cooler	Cool the hot D2O condensate from the bleed condenser prior to condensate entering the bleed purification circuit, to prevent IX resin breakdown.	Good	Corrosion / Pitting Corrosion Mechanical and Thermal Degradation / Creep and Stress Relaxation Corrosion / Fouling (accumulation of deposits) Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Wear Mechanisms - Fretting	Clean and Inspect	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011119	5,6,7,8	33000 33340	33340 Heat Transport Pump Gland Seal Coolers	One gland recirculation heat exchanger is supplied for each PHT pump and is used to supply D2O for cooling and lubrication of the PHT pump bearings and mechanical seals.	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Fretting Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Corrosion / Fouling (accumulation of deposits)	Surveillance	<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Perform sample inspections (similar to U2 planned inspections) to ensure condition is acceptable to reach CO EOL (2024). Based on inspection results, as well as U2 inspection results, determine any further AM activities to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011203	5,6,7,8	33000 33910	33910 - D2O Recovery Pumps	Pump recovered D2O from the D2O recovery sump to the suction of the PHT Pressurizing Pumps, as required in emergency conditions.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Lubricant Contamination Corrosion / Fouling (accumulation of deposits) Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion	Lubrication Inspection - Visual SRST - Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011204	5,6,7,8	33000 33310	33300 HT Pressurizing Pumps	Provide HT pressurizing flow to the reactor outlet headers to counteract HT shrinkage. The pumps also supply gland flow to the HT main circulating pumps and shutdown cooling pumps, cooling flow to the bleed condenser reflux or spray circuits and pressurizing flow to the fuelling machines when they require relatively low pressure.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation	Inspection - Visual Logic Test Predictive Maintenance - Vibration Monitoring	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						/ Lubricant Contamination Corrosion / Fouling (accumulation of deposits) Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear		
011214	5,6,7,8	33000 33120 33310 63332	33000 Primary Heat Transport System-TRANSMITTER-ANALYTICAL-CAT 1&2	Transmit temperature signals from various HTS components.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Electronic Aging Obsolescence / Immediate Obsolescence Concern	Calibration SRST - Functional Test	<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Resolve obsolescence and spares issues for transmitters.
011243	5,6,7,8	33000 33120	33120 HT Pump Discharge & Boiler Inlet/Outlet Isolation Valves	MV1 to MV24 isolate boilers for drainage and maintenance. MV25 to 40 provide isolation for pump maintenance during outages and to prevent reverse rotation.	Satisfactory	Corrosion / Fouling (accumulation of deposits) Mechanical and Thermal Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Thermal Aging	Inspection - Visual Lubrication Stroke Test	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete planned work orders for gasket/packing replacement and actuator overhaul. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011244	5,6,7,8	33000 33710	33710 HT D2O Inter-Unit Transfer Pump	Permit cold shutdown of a reactor and to transfer D2O from its associated unit to provide for the expansion	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Inspection - Visual SRST - Logic Test	<u>Incremental recommendations for Plant EOL (2020):</u>

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				volume created during start up.		Mechanical and Thermal Degradation / Lubricant Degradation Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Acquire one spare pump to enable replacement in the event of component failure. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011258	5,6,7,8	33000 33210 33310 33320 33330 33340 33610 33710 33910	33000 Primary Heat Transport System - Check Valves - Swing Type	Prevent backflow in circuits of various PHT systems.	Satisfactory	Mechanical Fatigue / Mechanical Fatigue Wear Mechanisms - Wear / Wear Mechanisms - Wear Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Obsolescence / Immediate Obsolescence Concern	Inspection - Internal Inspection - Visual SRST - Functional Test SRST - Logic Test	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Inspect valve internals of selected check valves: 5-33340-NV312 (WO# 2719342) Inspect NV, 8-33340-NV312 (WO# 2878274) Inspect NV, 5-33710-NV8 (WO# 2722790) to inspect and overhaul NV. <u>Incremental recommendations for Plant EOL (2020):</u> Resolve obsolescence issues and obtain necessary spare parts. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011266	5,6,7,8	33000	33000 Primary Heat Transport System- Power and Control Cable - 4.16 kV, 600V, 125V, 250V	Provide power and control for HTS equipment.	Good	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Radiation Embrittlement	Inspection - Condition Monitoring Program	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Fatigue / Thermal Fatigue		
011270	5,6,7,8	33000 33120 33210 33910	33000 Primary Heat Transport System-Valves - MOV-Standard-CAT1&2	5/6/7/8-33120-MV41 to MV44: PHT Interconnect Valves The function of the PHT Interconnect valves is to isolate the circuits from one another at the interconnection point in the event of a Loss of Coolant Accident. 5/6/7/8-33910-MV1: D2O Recovery Pump Discharge Isolator During normal reactor operation, the D2O Recovery System is in the poised state and is isolated from the PHT System through D2O Storage Tank Pump Discharge Isolator. 5/6/7/8-33210-MV15 and MV31: Bleed Filter Inlet Isolators The Bleed Filter Inlet Isolators remain open until the bleed cooler outlet temperature exceeds 60°C. A high purification temperature initiates an alarm and causes the motorized Bleed Filter Inlet Isolators that are in service to close.	Satisfactory	Corrosion / Fouling (accumulation of deposits) Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear Radiation Induced Degradation / Radiation Embrittlement Corrosion / General Corrosion Obsolescence / Immediate Obsolescence Concern	Inspection - Internal Inspection - Visual Lubrication SRST - Functional Test SRST - Stroke Test Valve Diagnostics	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete WOs for actuator overhauls. <u>Incremental recommendations for Plant EOL (2020):</u> Resolve valve obsolescence issues. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011271	5,6,7,8	33000 33120 33210 33310 33320 33330 33340 33610 33710 63103	33000 Primary Heat Transport System - AOVs	This covers a large number of HT system AOVs which are bellows sealed globe valves. The most significant AOVs in the group are: 1) HT Feed Valves - 5/6/7/8-33310-CV3 and CV6	Satisfactory	Corrosion / General Corrosion Mechanical and Thermal Degradation / Thermal Aging Fatigue / Thermal Fatigue	Calibration Inspection - Internal Overhaul/Refurbishment SRST - Functional Test SRST - Logic Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
		63336 71320		2) HT Bleed Valves - 5/6/7/8-33320-CV102 and CV104 3) Bleed Condenser Reflux Valve - 5/6/7/8-33320-CV111 4) Bleed Condenser Level Control Valves - 5/6/7/8-33320-CV122 and CV123 5) HT Relief Valves - 5/6/7/8-33610-CV1/CV2/CV3/CV4		Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Wear Mechanisms - Erosion		
011272	5,6,7,8	33000 33310 33320 33340 33610 71320 71330	33000 Primary Heat Transport System - PRVs	Instrument air pressure regulators for various MVs & CVs.	Good	Corrosion / General Corrosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Creep and Stress Relaxation	Inspection - Visual Overhaul/Refurbishment SRST - Functional Test SRST - Logic Test Valve Diagnostics	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011273	5,6,7,8	33000 33120 33330 33340 33610	33000 Primary Heat Transport System - Flow Orifices	33340-FR1/FR2 Gland supply filters remove from the high pressure water supplied to the glands of the heat transport pumps particulate matter which might otherwise damage the mechanical seals housed in the gland cavity. 33330-RD1 Rupture disc in parallel with the relief valve will burst at 50 psig and will relieve heavy water or steam if all the helium gas is expelled from the tank or if hot water	Good	General Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Fatigue / Thermal Fatigue Corrosion / Fouling (accumulation of deposits)		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete the following WO's to assess condition: • Radiography inspection of 5 - 33120-RO1 (WO #2710154). • Radiography inspection of 5-33610-RO3000 (WO #2710161). • Radiography inspection of 5-33340-RO1 (WO #2710231). • Radiography inspection of 5-33340-RO2 (WO #2710232).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				is inadvertently admitted to the tank. 33120-Y3/Y4/Y5/Y6 Four corrosion-monitoring autoclaves are installed in each unit at Pickering G.S – one on the reactor inlet and one on the reactor outlet of each loop of the PHT system. 33120-RO1/RO2/RO3/RO4 Allow warm/cool D2O to enter the discharge of the shutdown HT circulating pumps to keep HT piping temperature consistent. 33610-RO3000/RO3001/RO3002/RO3003 Allow expansion of D2O out of the Shut Down cooling circuit to the Reactor Inlet headers. 33340-RO1 Breakdown the D2O pressure used for testing of SDS1 & SDS2 parameters. 33340-RO2 Breakdown the D2O pressure used for testing of ECI parameters.				<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011274	5,6,7,8	33000 33120	33120 Primary Heat Transport Pumps	The PHT Pumps circulate D2O through the reactor. The pumps draw from the boilers via a suction header and pump to the Reactor Inlet Header which distributes flow to individual inlet feeders of the reactor.	Good	Wear Mechanisms - Wear / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring,	Inspection - Ultrasonic Test Predictive Maintenance - Vibration Monitoring	<u>Incremental recommendations for Plant EOL (2020):</u> 1. Complete a one-time proactive replacement of the HTS pump seals. 2. Spares/Parts to be procured to support repair/replacement of the seals. (In addition, safe storage of 3 assembled seal cartridges is recommended). 3. Spares/Parts to be procured to support overhaul of the pump.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Lubricant Contamination Corrosion / Fouling (accumulation of deposits) Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Corrosion / Flow induced wear of the leading edge		Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011275	5,6,7,8	33000 33810	33810 PHT D2O Collection Pumps	5,6,7,8-33810-P1/P2 These pumps transfer D2O from collection tank TK1, back into the HT system. 5,6,7,8-33810-P3/PM3 This pump returns D2O from the PHT D2O Collection Tank (TK2) to the PHT Purification System.	Good	Wear Mechanisms - Wear / Wear Mechanisms - Wear Mechanical Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Thermal Aging Corrosion / General Corrosion		Incremental recommendations for Plant EOL (2020): Procure one spare pump for P1 and P2 and one spare pump for P3. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011276	5,6,7,8	33000 33330 33610 33810	33810 Primary Heat Transport System-Vessels-PV's, Tanks, Drums, etc.-CAT1&2	5/6/7/8-33330-TK1 (D2O Storage Tank) Provides feed to the pressurizing pumps and receives heavy water from the purification system. During shutdown conditions (i.e. maintenance outages) the storage tank provides a	Good	Mechanical and Thermal Degradation / Radiation Embrittlement Corrosion / General Corrosion	Air Holding Test Inspection - Visual	Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Complete a one-time inspection, and if degraded repair/replace (for all tanks).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				continuous pressure to the core via the level control header and the shutdown cooling pump loops. 5/6/7/8-33810-TK1 and -TK2 (D2O Collection Tanks) Collect D2O recovered by the D2O Collection System, for return to the PHT (via the PHT Purification Circuit). 5/6/7/8-33610-CV1-TK1, -CV2-TK1, -CV3-TK1, -CV4-TK1 Instrument Air Reservoirs for PHT Liquid Relief Valves 33610-CV1 to -CV4, which provide overpressure protection for the PHT System. Under accident conditions, the valves are required to operate on demand to relieve HTS overpressure (30 minute mission time). The tanks provide the air reserve for the required mission time if instrument air is not available.		Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		
011291	5,6,7,8	33000 33810	33810 PHT Leakage Collection - Heat Exchangers	Cool the hot D2O collected in the D2O collection tank to prevent vapour locking/cavitation of the D2O collection pumps and to protect the resin beads in the purification system when the D2O is pumped back to system.	Good	Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Fretting Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Corrosion / General Corrosion	Surveillance	<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> 1. Perform a one-time inspection and if degraded, repair/replace. 2. Proactively procure replacement heat exchangers for all units.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Corrosion / Fouling (accumulation of deposits)		
011292	5,6,7,8	33000 33110 33120 33200 33210 33310 33320 33330 33340 33710 33810 33910 63103 63330 63331 63336 63381	33000 Primary Heat Transport System - Manual Valves - D2O Inside Containment	Manual valves used for isolation or drain purposes.	Poor	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Calibration Lubrication	<u>Incremental recommendations for Plant EOL (2020):</u> 1. One-time replacement/repair of manual valves with known negative OPEX. 2. Establish a performance monitoring program to trend of failure rates of Boiler Drain Valves to determine if a future maintenance strategy is required. 3. Resolve outstanding spare parts issues. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011338	5,6,7,8	33000 33120	33000 Primary Heat Transport System-Piping - CAT1&2	HD9 is a major PHTS pressure boundary piping component serving as a reactor inlet header. It receives flow from the PHTS pumps and delivers the flow to the reactor inlet feeder pipes in the south loop	Good	Fatigue / Thermal Fatigue Corrosion / General Corrosion Mechanical and Thermal Degradation / Thermal Aging Corrosion / Flow Assisted Corrosion (Erosion-Corrosion)		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Implement an inspection strategy to determine current design margin and thinning rates at the most corrosion/erosion susceptible locations. Complete the outstanding WOs for wall thickness inspections. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011352	5,6,7,8	33000 33120 33310 33320 33330 33340 33540	33000 Primary Heat Transport System - Check Valves - Ball & Piston Type	Prevent backflow in various parts of the PHT System Circuits. Several valves act as Seismic Boundary (33310-NV1/NV2, 33320-NV112,	Satisfactory	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Inspection - Internal Inspection - Non - Intrusive Overhaul/Refurbishment	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete planned Work Orders to inspect valve internals of selected check valves

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
		33610 33710 33810		33340-NV527, 33540-NV28/NV29).		Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Obsolescence / Immediate Obsolescence Concern	SRST - Functional Test	(i.e. 5-33330-NV216, 8-33330-NV216, 7-33310-NV2 and 8-33310-NV2). Incremental recommendations for Plant EOL (2020): Obsolescence issues to be resolved. Procure replacement valves and spare parts to support valve inspection and potential replacement/overhaul activities. Incremental recommendations for CO EOL (2024): Perform additional inspections if currently scheduled inspections show degraded condition to ensure condition is maintained to CO EOL (2024).
011525	5,6,7,8	33000	33120 Primary Heat Transport (PHT) Main Circ. Pump (MCP) Motors-4kV	Drive the main HTS Pumps which circulate heavy water (D2O) through the boilers.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Radiation Embrittlement Corrosion / Microbiological Influenced Corrosion Mechanical and Thermal Degradation / Lubricant Degradation		Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Resolve obsolescence and spares issues for the pump motors.
011526	5,6,7,8	33000	33310 Heat Transport (HT) Pressurizing Pump (HTPP) Motors-4kV	Drive HTS pressurizing pumps.	Good	Corrosion / Flow Assisted Corrosion (Erosion-Corrosion)		Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Lubricant Degradation Obsolescence / Immediate Obsolescence Concern		Incremental recommendations for CO EOL (2024): Resolve obsolescence and spares issues for the pump motors.

System 0452 - Reactor Building

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
010106	1,4	21000 21262	BLOW IN PANEL, Cat 1/2	The blow out panels activate in emergency conditions (e.g. LOCA) to relieve pressure in the overpressure areas in the Reactor Building.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue		Incremental recommendations for Plant EOL (2020): Revise PM 11083 to include inspection of panel 4-21262-BOP2053. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010111	1,2,3,4	21000 21030	EMBEDDED PART, Cat 1/2	Embedded Parts (EPs) are leak-tight sealed penetration through Reactor Building concrete walls or floors for cable, pipe or other component entries.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Fretting Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010113	1,2,4	21000 21032	EMBEDDED PART, Cat 1/2	Embedded Parts are opening in concrete structures for piping to penetrate into the Reactor Building structure.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Wear Mechanisms - Fretting Mechanical and Thermal Degradation / Wear Mechanisms - Erosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010114	1,4	21000 21270	HATCH, Cat 3/4	The hatches or openings in reactor building concrete floors are used for the installation and maintenance of equipment.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Radiation Induced Degradation / Radiation Depletion of Material Properties		
010118	1	21000 21270	MOTOR, Cat 3/4	HDM1 is a pneumatic motor which operates the R/B shielding hatch.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Lubricant Degradation Mechanical and Thermal Degradation / Wear Mechanisms – Wear		Incremental recommendations for Plant EOL (2020): Implement a new PM to Lube and inspect the hatch itself or motor to ensure adequate operation and seal inflation. Refer to PM 11169 - 01. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010122	1,4	21000 62127	REGULATOR, Cat 1/2	These components regulate pressure from the instrument air system to the R/B shielding hatch seal inflation supply circuit.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete EC 117700 BOM verification on 1, 4 to ensure spare parts are be readily available. (Current CID for regulator replacement (0000673200) is: H/USER). Incremental recommendations for Plant EOL (2020): Implement a PM to check the set point of the Regulators (Reference PM 117098 - 01). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010129	1,4	21000 21270	VALVE, PRESSURE RELIEF, Cat 1/2	These RVs provide overpressure protection for instrument air supplied to devices in the hatch drive motor controls.	Good	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Creep and Stress Relaxation		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear		
011237	5,6,7,8	21000	21000 Reactor Building-Structural Concrete - Concrete Dome, Exterior Walls, Pressure walls, Columns, Slabs	The reactor building structural concrete components serves to form part of the containment structure to provide the airtight enclosure and shields staff and the public from radiation hazards.	Good	Corrosion of Embedded Steel / Corrosion of Embedded Steel Corrosion / Chemical Attack Mechanical and Thermal Degradation / Freeze-Thaw Corrosion / General Corrosion Mechanical and Thermal Degradation / Thermal Aging Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011316	5,6,7,8	21000	21000 Reactor Building - Foundations, Concrete	The foundation slab carries all loads from reactor building and forms part of the containment structure providing the bottom of the airtight enclosure.	Good	Corrosion of Embedded Steel / Corrosion of Embedded Steel Miscellaneous / Leaching Calcium Hydroxide	Sampling Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Cracking Due to Expansion/Contraction Corrosion / Chemical Attack Mechanical and Thermal Degradation / Freeze-Thaw		
011317	5,6,7,8	21000	21000 Reactor Building-Foundations, Steel H-Piles	The steel H-piles support the pile cap, which in turn is designed to support the entire RB structure.	Good	Corrosion / General Corrosion Fatigue / Mechanical Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011348	5,6,7,8	21000 21260	21000 Reactor Building - Seals & Sealants	Elastomeric seals and sealants provide leak tightness for air and/or water penetrations.	Satisfactory	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO (2024).
011351	5,6,7,8	21000	21000 Reactor Building - Steel Liners	The steel liners on both sides of the shielding wall provide additional shielding to the maintenance personnel in the Fueling Machine Service Room. The steel liners on the East and West Fuelling Machine Vault roofs, through which the embedded re-circulating cooling water piping penetrate. The two steel liners also served as formwork for concrete at the time of construction.	Good	Corrosion / General Corrosion Radiation Induced Degradation / Radiation Embrittlement		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Repair/replace the damaged insulations which are attached to the fueling machine vault roof steel liners. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

System 0453 - Reactor Regulating System

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
010131	1,4	63711 63174 63715	AMPLIFIER, Cat 1/2	Amplifier for Reactor Regulating system ion chamber signal. Amplifier for Shutdown System A in-core flux detector signal.	Very Good	Mechanical and Thermal Degradation / Electronic Aging		Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Implement new life-cycle capacitor replacement PM's.
010132	1,4	63711 63174	AMPLIFIER, Cat 3/4	Amplifiers provide local indication in Main Control Room from In-Core flux detectors.	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging Obsolescence / Immediate Obsolescence Concern		Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Complete one-time amplifier replacement to reach CO EOL (2024).
010135	1,4	63711 63174	DETECTOR, Cat 1/2	The In-core Flux Detectors (ICFD) output a current proportional to the localized fission rate which is used by the RRS for spatial control of the reactor power.	Satisfactory	Radiation Induced Degradation / Radiation Depletion of Material Properties	Calibration	Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Replace all detectors that were last replaced before 2004.
010138	1,4	63711 63101 63102	ELEMENT, Cat 1/2	Resistance temperature detectors are used to measure 22 channel inlet temperatures and 390 channel outlet temperatures. Temperature signals are fed to DCC1/2 and used for thermal power calculation & power control.	Good	Mechanical and Thermal Degradation / Thermal Aging Radiation Induced Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Elevated Temperature		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010141	1,4	63711 63715	ION CHAMBER, Cat 1/2	Out-of-core ion chamber outputs a current proportional to reactor flux, signal is fed to an amplifier	Satisfactory	Radiation Induced Degradation / Radiation Depletion of Material Properties	Calibration Functional Test	Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				for Reactor Regulating system.				Incremental recommendations for CO EOL (2024): Replace all detectors that were last replaced before 2004.
010142	1,4	63711 63174	MODULE, Cat 1/2	Amplifier module receives input signal from in-core flux detector & outputs to Reactor Regulating system.	Very Good	Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010147	1,4	63711 63171 63174 63715	POWER SUPPLY, Cat 1/2	Reactor Regulating system power supplies for adjuster rod position, In-core Flux Detector amplifiers & AC power transfer panel Channel A/B/C ion chambers.	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging	Inspection - Visual SRST - Functional Test	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete replacement of AC Transfer Panels 4-63715-R1A/R1C-PS1 (ref. EC#30104, WO's 1377367 & 1377370), ensure spares are available for 1/4-63715-R1A/B/C-PS1. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Complete the rest of the lifecycle replacements of power supplies to reach CO EOL (2024).
010153	1,4	63711 63101 63171	TRANSMITTER, Cat 1/2	Differential pressure transmitters produce mA output signal proportional to channel flow, used by ZOTPR program to calculate reactor thermal power. Position sensing potentiometer for Adjuster Rod position provides an input to the DCC.	Good	Mechanical and Thermal Degradation / Electronic Aging		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete Differential Pressure Transmitter replacement under NICR's 77661 & 77549 for U1/4 respectively. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Procure spares for Adjuster Rod position sensing potentiometers (Cat ID 293908).
010154	1,4	63711 63101	VALVE, MANUAL/HAN	These are RRS flow transmitter manifold valves.	Good	Mechanical and Thermal Degradation		Current initiatives that need to be completed for Plant EOL (2020) that are

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
			D OPERATED, Cat 3/4	The flow signals are used by the ZOTPR (Zone Thermal Power Routine) program to calculate the reactor thermal power		/ Wear Mechanisms - Wear		incremental to current periodic maintenance practices: Ensure NICR's 77549, 77661, and 108733 for replacement of transmitters/manifold valves are completed and closed out. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011407	5,6,7,8	63711 31700 31710 31770	31710/31770 Reactor Regulating System - Major Equipment - Reactivity Control Units	Adjuster Units provide excess reactivity needed to overcome Xe-135 transients following a power reduction, flux shaping for optimum reactor power and fuel burnup and to maintain Liquid Zone levels within limits. Control Absorber Units are normally fully withdrawn which provides poised reactivity to be inserted in the core when the RRS initiates a Stepback.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Radiation Induced Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Creep and Stress Relaxation		Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Procure spares for Control Absorber Reactivity Mechanism (Cat ID 208826).
011408	5,6,7,8	63711	31710, 31770 Reactor Regulating System-Cable - 600V, 125V - Power Cables	Cables feed power from source to load.	Good	Fatigue / Thermal Fatigue Fatigue / Mechanical Fatigue Radiation Induced Degradation /	Inspection - Condition Monitoring Program	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Radiation Embrittlement		
011455	5,6,7,8	63711 63171 63177	63171/63177 - Reactor Regulating System-SIGNAL CONDITIONER -CAT 1&2	End Stop & Position Monitoring (ESPM) module which, via relay logic, stops power to AA or CA drive motor when rod has reached a predetermined position.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging	Calibration Logic Test Predictive Maintenance - Thermography SRST - Functional Test	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete replacement program for AA & CA ESPM modules. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011472	5,6,7,8	63711 63102	63102 - Reactor Regulating System - Channel Temperature RTD's - CAT 1&2	Resistance Temperature Detectors (RTD) are used by RRS to monitor inlet and outlet channel temperatures for thermal power calculation/power control and to provide indication to monitor compliance with operating limits and to infer channel flow.	Good	Radiation Induced Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Elevated Temperature Corrosion / Oxidation		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011499	5,6,7,8	63711 31741 60432 63104 63170	63170 Reactor Regulating System (RRS) - Reactivity Deck Receptacles / Connectors-Various Voltages A.C. and D.C.	Connectors are used to connect following components to plant wiring: Adjuster Rod motors, Adjuster Rod position potentiometers, Control Absorber motors, Control Absorber electrical clutches, Control Absorber position potentiometers and RRS Vertical Flux Detectors, cable/connector for VFD assemblies are environmentally qualified.	Good	Mechanical and Thermal Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Thermal Aging	SRST - Functional Test	Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach CO EOL (2020). Incremental recommendations for CO EOL (2024): Perform one-time replacement of the VFD cable and ensure spare VFD cable assemblies are available.

System 0454 - Recirculating Cooling Water (RCW)

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
010155	1,4	71320 67132	ALARM, Cat 1/2	The flow and temperature indicating alarm meters are used to measure flow and temperature and provides annunciation when low flow or High/Low temperatures are detected in the Recirculated Water System.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010156	1,4	71320 67132	CONTROL, Cat 1/2	The recirculated cooling water temperature at heat exchangers 7132-HX501 and HX502 is controlled by the 67132-T501-TIC1 via associated low pressure service water control valves 7131-CV583 and CV585.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010158	1,4	71320 67132	ELEMENT, Cat 1/2	Temperature and flow elements for the Recirculated Cooling Water system. Temperature elements provide a low voltage signal proportional to process temperature.	Good	Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Electronic Aging	Calibration	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Perform a one-time inspection of 1/4-67132-T501-TE1 and if degraded replace.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				These flow elements are orifice plates used in conjunction with transmitters that measure the differential pressure across the orifice plate to determine the process flow.		Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms – Erosion		<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Implement a PM program to test and inspect these TEs.
010163	1,4	71320 71320	HEAT EXCHANGER, Cat 1/2	To serve as heat sink to cool the demineralized water in the closed loop circuit of the RCW System during unit operation and unit shutdown. The safety function is to provide cooling water supply to support the operation of safety-related systems. Tube failure will result in lake water leak into demineralized water system. Isolation of one HX during summer could result in unit derating.	Poor	Corrosion / Pitting Corrosion Corrosion / Fouling (accumulation of deposits) Corrosion / Stress Corrosion Cracking - Nickel-base Alloys Corrosion / Stress Corrosion Cracking - Austenitic Stainless Steels Mechanical and Thermal Degradation / Wear Mechanisms - Fretting Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Creep and Stress Relaxation	Clean and Inspect	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Maintain established PM frequency (cessation of repetitive PM deferral history), and complete outstanding WO#2164403 to obtain a spare floating end cap from Unit 2 and WO#03088812 to replace cover gasket. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
010169	1,4	71320 71320	POSITIONER, Cat 1/2	Positioners for Control Valves CV553 & CV554. These valves are modulated by temperature indicating controller's 63332-T19A/B-TIC1 as part of the Bleed	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				Cooler Temperature Control System.		Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Radiation Induced Degradation / Radiation Depletion of Material Properties		
010180	1,4	71320 67132 71320	TRANSMITTER, Cat 1/2	1/4-71320-CV553/CV554-AX1 are used to convert the current signal from T19A/T19B-TIC to a pressure signal to control the position of valves CV553/CV554 for the bleed cooler 3332-HX1. 1/4-67132-F501-FT1 are flow transmitters used to determine the gross recirculated water flow. The output is connected to 67132-F501-FIA1 which alarms in the event of low flow (AN-58). 1/4-67132-T501-TT1 are temperature transmitters used to monitor the temperature of the recirculated water system. The output is connected to 67132-T501-TIC1 which modulates CV583 and CV585.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Electronic Aging Radiation Induced Degradation / Radiation Embrittlement		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010189	1,4	71320 71320	VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2	These Air Operated Valves close automatically to isolate the RCW supply to the Bleed Cooler following activation of the service water load shedding circuit	Satisfactory	Corrosion / General Corrosion Fatigue / Mechanical Fatigue	Overhaul/Refurbishment SRST - Functional Test SRST - Stroke Test	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete NICRs to replace Unit 4 MV531/MV560.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				in response to a Class IV Power failure.		Mechanical and Thermal Degradation / Creep and Stress Relaxation Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms – Wear Obsolescence / Immediate Obsolescence Concern	Valve Diagnostics	Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010191	1,4	71320 71320	VALVE, PRESSURE REGULATING, Cat 1/2	Pressure regulating valves for air supply to 1/4-71320-CV553/CV554 and 1/4-71320-MV531/MV560. The parent valves control or isolate RCW flow to the PHT Bleed Coolers.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Creep and Stress Relaxation Fatigue / Mechanical Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010193	1,4	71320 71320	VALVE, PRESSURE RELIEF, Cat 1/2	Provide overpressure relief for the heat exchangers/coolers of the various systems supplied by the RCW System.	Very Good	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Component Replacement Inspection - Visual	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011174	5,6,7,8	71320 71320	71320 RCW Heat Exchangers	These Heat Exchanges serve as a heat sink to cool the demineralized water in the closed loop circuit of the RCW System during unit	Satisfactory	Corrosion / Pitting Corrosion	Clean and Inspect	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				operation and unit shutdown. The safety function is to provide cooling water supply to support the operation of safety-related systems. Tube failure will result in lake water leak into demineralized water system. Isolation of one HX during summer could result in unit derating.		Corrosion / Fouling (accumulation of deposits) Corrosion / Stress Corrosion Cracking - Austenitic Stainless Steels Mechanical and Thermal Degradation / Wear Mechanisms - Fretting Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Creep and Stress Relaxation Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Maintain established PM frequency (to address deferred PMs). Monitor wall losses and leakages, and raise Work Orders as necessary to address aging management issues. <u>Incremental recommendations for Plant EOL (2020):</u> 1. Procure sufficient spare parts to support potential gasket replacements, as needed. 2. Proactively establish procurement requirements for new HX, tube bundle and channel cover to reduce procurement time, should a replacement be required. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011401	5,6,7,8	71320 71320	71320 Recirc Cooling Water (RCW)-Valves - AOV-Standard-CAT1&2	Following a loss of Class IV Power, RCW flow is automatically reduced via a water reduction control circuit. Valves 71320-MV561/MV597 (on the inlet to the motor bearing and pump gland cooling for the HT circulating pumps 33120 P1 to P16) are closed when service water reduction is initiated. Solenoid valves MV561-SV1/MV597-SV1 control the air supply to the actuators of the parent valves.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / Environmental Degradation Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

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CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				Pressure regulators MV561-PRV1/MV597-PRV1 regulate the pressure of the air supply to the actuators of the parent valves.		Mechanical and Thermal Degradation / Thermal Aging		
011402	5,6,7,8	71320 71320	71320 Recirc Cooling Water (RCW)-Expansion Joints	These expansion joints are installed on the suction and discharge of the RCW circulating pumps 5/6/7/8-71320-P501/P502/P503, to prevent stresses due to temperature expansion and contraction, to insulate against the transfer of noise and pressure (waterhammer) and to compensate for misalignment.	Satisfactory	Mechanical and Thermal Degradation / Cracking Due to Cyclic Loading Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Fatigue due to Vibration	Inspection - Visual	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: 1. Investigate an alternate style/type/material of expansion joint that can withstand the piping misalignment, or a way of aligning the piping to prevent the expansion joint from leaking. Once a solution is found, complete replacement of discharge expansion joints EJ544/EJ545/EJ546 under Master NICR 124073. 2. Complete open WOs to replace both suction and discharge expansion joints and initiate WOs to replace the remaining suction side expansion joints, if necessary, based on the results of visual inspection during pump maintenance. 3. Complete outstanding corrective WOs to address leakage problems. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011403	5,6,7,8	71320 71320	71320 Recirc Cooling Water (RCW)-Strainers	Suction Strainers for 5/6/78-71320-P501/P502/P503 to protect the pumps. Strainer was required during commissioning to remove large particles only and is no longer required for normal operations as per RCW System Engineer.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion	Surveillance	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete removal of the filter element from 6-71320-STR501. Incremental recommendations for Plant EOL (2020):

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Corrosion / Fouling (accumulation of deposits)		No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011439	5,6,7,8	71320	Recirc Cooling Water (RCW) - Cable - 4.16 kV, 600V, 125V, 250V - Power Cables	The cables used for Pickering GS B can be sub-divided into five (5) basic categories. There are 5 kV power cables, 600 V power cables, 600 V control cables, 300 V control cables and special definite purpose cables. The (major) equipment associated with the cables listed are: Class III / Class IV 4.16kV and 600V Switchgear, Class II / III / IV Motor Control Centers, 600V Distribution Panels and Class III / IV 4.16kV/600V Distribution Transformers	Good	Fatigue / Thermal Fatigue Fatigue / Mechanical Fatigue Radiation Induced Degradation / Radiation Embrittlement	Inspection - Condition Monitoring Program	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011473	5,6,7,8	71320 67132	67132 - Recirc Cooling Water (RCW)-CONTROLLER - ELECTRONIC-CAT 1&2	The function of temperature controller's 67132-T501-TIC1 and T502-TIC1 is to maintain the temperature of the RCW at the outlet of RCW heat exchangers, 71320-HX501 and HX502, at a set temperature of 29C via manipulation of the LPSW control valves 71310-CV583 and CV585, which provide cooling water to the HXs.	Good	Mechanical and Thermal Degradation / Thermal Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011478	5,6,7,8	71320 71329	71329 Recirc Cooling Water (RCW)-Piping-Hoses-CAT3&4	These RCW flex hoses connect the PHT pump gland cooling and motor bearing cooling circuits (33120) to the RCW supply and return headers (71320).	Poor	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Incremental recommendations for Plant EOL (2020): Implement a one-time replacement of the hoses. Incremental recommendations for CO EOL (2024):

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Wear Mechanisms - Fretting Mechanical and Thermal Degradation / Thermal Aging		Implement a one-time inspection of the hoses.
011480	5,6,7,8	71320 67132	67132 Recirc Cooling Water (RCW)-SIGNAL CONDITIONER -CAT 1&2	Signal conditioner 67132-T501-TM1 is a high signal select used in conjunction with 67132-T501-TIC1 to control Recirculating Cooling Water (RCW) temperature through 71320-HX501. Its main function is to limit the cooling flow to HX501 upon a load reduction of the RCW system during a failure of Class IV power by selecting control signal from 67132-T501-TIC1 or 67132-T501-HC1. The same can be said for 67132-T502-TM1 and 67132-T502-TIC1 temperature control of RCW through 71320-HX502. The same signal from 67132-T501-HC1 supplies both loops.	Good	Mechanical and Thermal Degradation / Thermal Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011489	5,6,7,8	71320 67132	67132 Recirc Cooling Water (RCW)-TRANSMITTE R-ANALYTICAL-CAT 1&2	The temperature transmitters, via a RTD temperature element, measures the temperature of the outlet of Recirculated Cooling Water (RCW) heat exchangers 71320-HX501 and HX502. The current output of the transmitter is	Good	Mechanical and Thermal Degradation / Thermal Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

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CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				the input to digital controller's 67132-T501-TIC1 and T502-TIC1 which is used to control the cooling water through tubes of 71320-HX501 and HX502 and maintains RCW temperature at a setpoint of 29C.				
011493	5,6,7,8	71320 71330	71330 ESW To Bleed Cooler Isolation AOVs	These Motorized Valves close automatically to isolate the RCW supply to the Bleed Cooler following activation of the service water load shedding circuit in response to a Class IV Power failure.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

System 0455 - Relief Ducts

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
010197	4	34200 62520	RELAY, Cat 1/2	The relay is used in the negative pressure containment bulkhead isolation doors control circuit. Its contact energizes solenoid 62520-SV1 used to unlatch the door.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Creep and Stress Relaxation Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Electronic Aging Radiation Induced Degradation / Radiation Depletion of Material Properties		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010198	0	34200 25200	SWITCH, Cat 3/4	These thermostat switches are part of the electric heat tracing that protects the Vacuum duct from freezing.	Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010201	1,4	25100 62520	SOLENOID/SOLENOID OPERATED VAL, Cat 1/2	The SV is used to unlatch the negative pressure containment bulkhead isolation door.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						/ Creep and Stress Relaxation Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging Radiation Induced Degradation / Radiation Depletion of Material Properties		
011239	018	25200 25200	25200 PRD Structure and Seals	Concrete Structure: The PRD structure connects the four P058 units and four P014 units to the common VB via 'bulkheads'. In an accident event, a large RB pressure occurs and contaminants such as radioactive fission products including tritium could be released into the PRD to be transferred and contained in the VB. Seals and Sealants: The joints in the PRD box section are designed to allow differential settlement, longitudinal thermal contraction and expansion, and facilitate construction. Steel Parts: The steel parts, including embedded structural steels, clamping plates, bolts, washers, and tendons, are a portion of the joint assembly. They are designed for clamping the rubber seals to be firmly	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Self-Loosening Mechanical and Thermal Degradation / Freeze-Thaw Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Cracking Due to Expansion or Contraction		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				held in place without leakage.				
011240	018	25200	25200 PRD - Bearing Plates & Sliding Joints	The intended function of bearing plates/sliding joints is to allow for bearing support, as well as East-West movement of the Pressure Relief Duct.	Good	Corrosion / General Corrosion Corrosion / Fouling (accumulation of deposits)	Inspection - Visual	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011312	018	25200	25200 Relief Ducts - Foundations - Concrete Footings	The foundations are designed to support the PRD piers and frames which support the main relief duct structure.	Good	Corrosion / Corrosion of Embedded Steel Mechanical and Thermal Degradation / Freeze-Thaw Mechanical and Thermal Degradation / Leaching Calcium Hydroxide Mechanical and Thermal Degradation / Cracking Due to Expansion or Contraction Corrosion / Chemical Attack		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011313	018	25200	25200 Relief Ducts - Foundations, Steel	The steel H-piles are designed to support the pile caps, which in turn support the PRD piers, frames and the main relief duct.	Good	Corrosion / General Corrosion Fatigue / Fatigue due to Vibration	Sampling Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

System 0456 - Service Water Systems

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
010203	1,4	71300 71340	ACTUATOR, Cat 1/2	Actuators are used to drive 71340-MV2005/MV2008 which deliver water as a long-term heat sink source to the boilers following depletion of the boiler and Boiler Emergency Cooling System (BECS) inventories.	Satisfactory	Corrosion / General Corrosion Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Lubricant Degradation		Incremental recommendations for Plant EOL (2020): No additional recommendations required to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Revise the scope of the current PM to include diagnostics per P-AB-CMP-60460.52 Diagnostic Testing LIMITORQUE AND ROTORK Rising Stem Motor Operated Valves.
010205	1,4	71300 67131 67134	ALARM, Cat 1/2	The alarm units are used to provide annunciation when HPSW/LPSW Header pressures are below set points and when HPSW temperature is low.	Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging	Calibration SRST - Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010206	1,4,01 4	71300 67165	ANALYZER, Cat 1/2	The analyzer is used to measure the Total Residual Chlorine (TRC) levels in the service water.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Electronic Aging Fatigue / Environmentally-Assisted Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Fatigue / Mechanical Fatigue		
010207	1,4	71300	ANALYZER (GAS), Cat 1/2	This analyzer measures chlorine vapour levels in the Chlorine Building.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Electronic Aging Fatigue / Environmentally-Assisted Fatigue Fatigue / Mechanical Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010208	1,4	71300	ANUNCIATOR, Cat 1/2	The annunciator is used to alarm when Total Residual Chlorine (TRC) levels in the service water exceed setpoint.	Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010210	014	71300 67165	CIRCUIT BREAKER, Cat 1/2	This circuit breaker is used to supply or disconnect 120 Vac power to chemical injection pump 014-67165-P2065.	Good	Corrosion / Pitting Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Thermal Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010211	1,4	71300 67131	CONTACTOR, Cat 1/2	This contactor is used to energize / de-energize the backwash arm driven motor of the Low Pressure Service Water Strainer 1/4-71310-STR3.	Good	Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging	Inspection - Visual	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Wear Mechanisms - Wear		
010212	1,2,3,4	71300 67131	CONTROL, Cat 1/2	These controllers provide 3 to 13 PSIG (water) control signals to control valves 1/2/3/4-71310-CV825 and 1/4-71310-CV826.	Satisfactory	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Calibration Functional Test	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Implement late/deferred Predefine WO# 2432900 to overhaul/diagnose 4-71310-CV825. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010217	1,4	71300 71310	EXPANSION JOINT, Cat 1/2, Moderator	The expansion joints allow for movement caused by vibration and expansion/ contraction between the moderator heat exchangers and the cooling fluid inlet and outlet piping. The equipment is required to be available as a flow path for normal operation as well as in accident scenarios.	Very Poor	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Thermal Aging	Surveillance	Incremental recommendations for Plant EOL (2020): Perform a one-time replacement of all expansion joints. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010219	1,4	71300 71310 71340	EXPANSION JOINT, Cat 1/2	The expansion joints allow for movement caused by vibration and expansion/ contraction between the low pressure service water pumps (71310) and high	Poor	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation	Surveillance	Incremental recommendations for Plant EOL (2020): Complete a one-time inspection of the EJs in this CG to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024):

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				pressure service water pumps (71340) outlet piping.		/ Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue		Complete a one-time inspection of the EJs in this CG to reach CO EOL (2024), depending on outcome of last inspection.
010222	1,4	71300 71340	FILTER, Cat 1/2, Valve Filter	This CG filters air supply to MVs.	Very Good	Corrosion / Fouling (accumulation of deposits) Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Component Replacement	Continue current practices no additional practices are recommended to reach CO EOL (2024).
010224	1,4	71300	GAUGE, Cat 1/2	These pressure gauges measure pump pressure.	Good	Corrosion / General Corrosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Incremental recommendations for Plant EOL (2020): Perform a one-time calibration and function check for the following gauges 4-67134-PG2268, PG2269 to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Perform an additional one-time calibration and function check for the following gauges 4-67134-PG2268, 2269 to reach CO EOL (2024).
010227	1,4	71300 71310 71340	HEATER (GENERIC), Cat 1/2, CLIII Pumps	These heaters are used to prevent motor condensation.	Good	Corrosion / General Corrosion Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Thermal Aging		
010228	1,4	71300 71310 71340	HEATER (GENERIC), Cat 1/2, CLIV Pumps	These heaters are used to protect motors by preventing condensation inside the motors.	Good	Corrosion / General Corrosion Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Thermal Aging	Predictive Maintenance - Electrical Testing Predictive Maintenance - Thermography	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010235	1,4	71300 71310	LUBRICATOR, Cat 1/2	Lubricators inject oil to air supply to provide lubrication to the internal working parts of the downstream pneumatic actuator (1/4-71310-MV194).	Poor	Corrosion / Fouling (accumulation of deposits) Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Lubricant Degradation Mechanical and Thermal Degradation / Lubricant Contamination	Component Diagnostics Overhaul/Refurbishment	<u>Incremental recommendations for Plant EOL (2020):</u> Complete a one-time inspection of the lubricators to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Complete a further one-time inspection of the lubricators to reach CO EOL (2024), depending on previous inspection condition.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
010237	1,4	71300 71310 71340	MOTOR, Cat 1/2, CLIII Pumps	These pump motors are used to drive the Emergency Low Pressure and High Pressure Service Water Pumps.	Good	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Lubricant Degradation		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010240	4	71300 71340	MOTOR, Cat 1/2, MOVs	These Actuator motors drive the Emergency Boiler Water Supply motorized valves.	Satisfactory	Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Lubricant Degradation	Component Diagnostics Lubrication Inspection - Internal Inspection - Visual SRST - Functional Test	<u>Incremental recommendations for Plant EOL (2020):</u> No additional recommendations required to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Change PM frequency for associated MV diagnostic testing from every 8 years to every 4 years, to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Electronic Aging		
010241	1,4	71300 71310 71340	MOTOR, Cat 1/2, CLIV Pumps	These pump motors are used to drive Low Pressure and High Pressure Service Water Pump sets (71310-P3,-P4,-P5 and 71340-P3, -P4).	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion Mechanical and Thermal Degradation / Lubricant Contamination	Lubrication Overhaul/Refurbishment Predictive Maintenance - Thermography Predictive Maintenance - Vibration Monitoring	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Execute PMs that have not been performed in the last 4 years (e.g. PM #126075). Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010246	1,4	71300 71310	POSITIONER, Cat 1/2, Fisher 3570	Valve positioners are used to accurately position valve stem based on an applied control signal.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010247	1,4	71300	POSITIONER, Cat 1/2, Fisher 3582D	These valve positioners are used to accurately position	Very Good	Mechanical and Thermal Degradation		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				valve stem based on an applied control signal.		/ Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		
010248	1,4	71300 71310	POSITIONER, Cat 1/2, Neles NP724A	These valve positioners (NCs) are used to accurately position valve stem based on applied control signal.	Very Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010249	1,4	71300 71310 71340	POSITIONER, Cat 1/2, Fisher 3610J	These valve positioners (NC) are used to accurately positions valve stem based on the applied control signal.	Very Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010250	1,4	71300 71310 71340	PUMP, Cat 1, ELPSW/EHPSW Pumps	The pumps are two 100% capacity deep well pumps (1, 4-71310-P1 and P2), operating on Class IV / III power. These pumps take suction from the intake channel and discharge to the low pressure service water header which supplies LPSW loads as well as the HPSW pumps.	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Wear Mechanisms - Erosion		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete EC127792, EC131259, EC130608, EC130606, EC124661, EC125155, EC124683 and EC127528 for pump replacements. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Corrosion / Microbiological Influenced Corrosion Corrosion / Fouling (accumulation of deposits)		Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010251	1,4	71300 71310	PUMP, Cat 1, LPSW Pumps	Low pressure service water is supplied by three vertically mounted pumps, 7131-P3, P4, P5. These pumps supply all the water from intake channel to the reactor unit equipment and to auxiliary equipment in other buildings.	Satisfactory	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / Fouling (accumulation of deposits) Corrosion / Microbiological Influenced Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion	Overhaul/Refurbishment Predictive Maintenance - Vibration Monitoring	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: 1. Complete outstanding weld repair of pump top case and suction bell (WO# 2933902). 2. Significant negative OPEX (SCR's) suggests that deferred PM activities have resulted in the downgrade of the condition. The PM activities need to be completed as per their approved frequency to reach Plant EOL (2020). Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010252	1,4	71300 71340	PUMP, Cat 2, HPSW Pumps	High pressure service water (HPSW) is taken from the low pressure outlet header, pressurized and delivered via two vertical centrifugal pumps (71340-P3 and P4).	Poor	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Inspection - Visual Lubrication Vibration Monitoring SRST - Functional Test	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete the following WOs: WO 2850568 to overhaul pump 1-71340-P3, WO 1667952 to overhaul pump 4-71340-P3, WO 2826402 to replace 1-71340-P4, WO 02854187 to perform daily IR & vibration monitoring 4-71340-P4. Incremental recommendations for Plant EOL (2020):

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / Microbiological Influenced Corrosion Mechanical and Thermal Degradation / Lubricant Degradation		Complete a one-time task for pump monitoring, inspection and lubrication to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Complete a one-time task for pump monitoring, inspection and lubrication to reach CO EOL (2024), depending on condition from last completion.
010253	1,4	71300 67131	RECEIVER, Cat 1/2	These air receivers provide local back up instrument air for control valves 71310-CV509/CV583/CV512/CV585.	Good	Corrosion / Fouling (accumulation of deposits) Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Wear Mechanisms – Erosion	Inspection - Internal	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010256	1,4	71300 67131 67134 71310 71340	RELAY, Cat 1/2	These relays are used as logic relays, interlocking relays and as timer relays in control circuits of pumps, strainers & valves in LPSW system.	Good	Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Component Replacement Predictive Maintenance - Thermography SRST - Functional Test	Incremental recommendations for Plant EOL (2020): No additional recommendations required to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Validate status of spare relays to ensure between 5 &10% spares are available to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Electronic Aging		
010261	014	71300 67165	STATION, Cat 1/2	This 400 Gallon Air Driven Tempered Water System provides water for an emergency shower and eyewash.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / Fouling (accumulation of deposits)		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete WO 4832096 to replace hose connection to air bottles. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010262	1,4	71300 71310 71340	STRAINER, Cat 1/2	Pump discharge strainer for LPSW and HPSW systems. Used to strain or filter out debris in the water system.	Satisfactory	Corrosion / General Corrosion Corrosion / Fouling (accumulation of deposits) Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Corrosion / Microbiological Influenced Corrosion Mechanical and Thermal Degradation	Inspection - Internal Lubrication SRST - Functional Test	Incremental recommendations for Plant EOL (2020): 1. Complete a one-time internal inspection of 1-71310-STR3310, 1-71310-STR3311, and 1-71310-STR3312 to reach Plant EOL (2020). 2. Increase the frequency of existing PM (e.g. 10938) execution from every 3 years to every to 2 years. If performance does not improve, increase the frequency further. Incremental recommendations for CO EOL (2024): Complete a further one-time internal inspection of 1-71310-STR3310, 1-71310-STR3311, and 1-71310-STR3312 to reach CO EOL (2024).

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CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						/ Wear Mechanisms – Wear		
010266	1,4,014	71300 67131 67134	SWITCH, Cat 1/2, Pressure Switch	Pressure Switches detect High/Low Pressure of Service Water Systems and alarm and/or set logic based on these conditions.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion Fatigue / Mechanical Fatigue	Calibration SRST - Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010269	1,4	71300 67131 67134	SWITCH, Cat 1/2, Flow Switch	These Flow switches alarm on low flow to indicate inadequate bearing cooling supply for service Water pumps.	Satisfactory	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Electronic Aging	Surveillance	Incremental recommendations for Plant EOL (2020): No additional recommendations required to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Initiate PM to calibrate and function check CC2 items 67131-FS195, FS196 & FS197 (ref. PM 117096).
010270	4	71300 67131	SWITCH, Cat 1/2, 71310-MV15	These Switches are used to test Solenoid Valves.	Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010276	1,4	71300 67131	TIMER, Cat 1/2	These timers are used to activate strainer backwash valves.	Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation	Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						/ Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Thermal Aging		
010278	1,4	71300 71310 71340	TRANSMITTER, Cat 1/2, AOV AX Transducers	These transducers are used to convert electrical signals to pneumatic signals for operation of control valves in the service water system.	Very Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Thermal Aging Fatigue / Mechanical Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010280	1,4	71300 67131 67134	TRANSMITTER, Cat 1/2, LPSW/HPSW Header Pressure	These Pressure Transmitters are used to measure and transmit system pressure for Service Water.	Very Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging Corrosion / General Corrosion		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete outstanding work orders to replace transmitters (ref. WO # 1593600 and 1596556). <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional recommendations required to reach CO EOL (2024).
010287	1,4	71300 71310	VALVE, CONTROL, Cat 1/2, Mod HX TCV	The valves in this CG are service water/shell side control valves for Heat Exchanger (32110-HX1/2).	Good	Fatigue / Mechanical Fatigue	Calibration Inspection - Visual	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

TETRA TECH

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				CV509 controls flow for HX2, and CV512 for HX1.		Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Mechanical and Thermal Degradation / Wear Mechanisms - Erosion	Overhaul/Refurbishment Valve Diagnostics	
010288	1,4	71300 71310	VALVE, CONTROL, Cat 1/2, Quarter Turn Cooling CVs	The valves in this CG are control valves for Heat Exchanger (71320-HX501/502). CV583 controls flow from HX501, and CV585 from HX502.	Satisfactory	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Corrosion / General Corrosion Mechanical and Thermal Degradation / Creep and Stress Relaxation		Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Complete a one-time replacement of the valves in this CG to reach CO EOL (2024).
010289	1,4	71300 71310	VALVE, CONTROL, Cat 1/2, Diaphragm Actuator, Rising Stem Cooling CVs, Hammel Dahl	The valves in this CG are control valves for Heat Exchanger. CV591 controls flow from 34110-HX4, CV594 controls flow from 34110-HX3, CV690 controls flow from 34110-HX5, CV603 controls flow from 34130-HX1, and CV606 controls flow from 34130-HX2.	Good	Corrosion / General Corrosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Creep and Stress Relaxation	Calibration Inspection - Internal Inspection - Visual Overhaul/Refurbishment Valve Diagnostics	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		
010290	2,0	71300 71310	VALVE, CONTROL, Cat 1/2, Diaphragm Actuator, Rising Stem Cooling CVs, Fisher	Two of the valves in this CG are control valves for Heat Exchanger. CV697 controls temperature of 34410-HX1, and CV698 controls temperature of 34410-HX2. The other two are control valves (CV810 / CV813) for HTG steam condensation overflow.	Satisfactory	Corrosion / General Corrosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Creep and Stress Relaxation Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Calibration Overhaul/Refurbishment Valve Diagnostics	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: 1. Complete WO for replacement of 2-71310-CV697 and 2-71310-CV698 (WO# 2480901, 2480903). 2. Complete WO 04861286 for one-time actuator overhaul of 0-71310-CV810. Incremental recommendations for Plant EOL (2020): Complete a one-time diagnostic/calibration of 0-71310-CV813. Incremental recommendations for CO EOL (2024): 1. Complete a one-time overhaul of 0-71310-CV813. 2. Complete a one-time diagnostics/calibration 2 years after CV813 overhaul.
010291	1,4	71300 71310	VALVE, CONTROL, Cat 1/2, Diaphragm Actuator, Rising Stem Motor Cooling CVs, Fisher	The valves in this CG are control valves for Boiler Feed Pump Motor. These CVs control flow of motor cooling outlet.	Satisfactory	Corrosion / General Corrosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Creep and Stress Relaxation Mechanical and Thermal Degradation		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						/ Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		
010292	1,3,4	71300 71310	VALVE, CONTROL, Cat 1/2, Diaphragm Actuator, Butterfly, Fisher	The valves in this CG are service water control valves for loads in Transformer Insulating Oil System.	Good	Fatigue / Mechanical Fatigue Corrosion / General Corrosion Mechanical and Thermal Degradation / Creep and Stress Relaxation Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Inspection - Internal Overhaul/Refurbishment Valve Diagnostics	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010293	1,4	71300 71340	VALVE, CONTROL, Cat 1/2, Diaphragm Actuator, Ball, Fisher	The valves in this CG are control valves for Heat Exchangers (33410-HX1/2/3/4). CV504 and 507 control flow of HX outlet temperature, and CV521 and 524 control flow of HX service water.	Satisfactory	Fatigue / Mechanical Fatigue Corrosion / General Corrosion Mechanical and Thermal Degradation / Creep and Stress Relaxation Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> 1. Complete WO for replacement valve / actuator assembly (WO 4815118). 2. Complete WO for replacement hand wheel (WO 3175506). <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u>

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Wear Mechanisms - Wear		No additional practices are recommended to reach CO EOL (2024).
010300	1,3,4,0	71300 71310 71340	VALVE, MOTORIZED/MOTOR OPERATE, Cat 1/2	(71340) MV2005 and MV 2008: These MOVs are part of the Emergency Boiler Water Supply (EBWS). Following the occurrence of a main steam line break and low boiler water level, one valve would be opened to inject emergency water to the boilers. The valves can be throttled to maintain the desired water level to the boilers.	Poor	Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Lubricant Degradation	Inspection - Internal Inspection - Visual Lubrication SRST - Functional Test SRST - Stroke Test Valve Diagnostics	Incremental recommendations for Plant EOL (2020): 1. Complete a one-time diagnostic testing of 0-71310-MV401 and 3-71310-MV2012. Note that the remaining equipment tags are subject to diagnostic testing. 2. Complete a one-time inspection and lubrication of 1/4-71310-MV2070, 1/4-71340-MV2191, MV2332, 0-71310-MV459, MV401 and 3-71310-MV2012 to reach Plant EOL (2020). Note that the remaining equipment tags are subject to inspection. Incremental recommendations for CO EOL (2024): Complete a further one-time inspection and lubrication of 1/4-71310-MV2070, 1/4-71340-MV2191, MV2332, 0-71310-MV459, MV401 and 3-71310-MV2012 to reach CO EOL (2024).
010302	1,4	71300 71340	VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2, Jamesbury 815W ES005	The valves are located in the High Pressure Service Water System, MV2115 and MV2116 modulate to provide cooling to RB ACU's. They also OPEN to increase flow to Vault ACU's on high pressure in the RB.	Poor	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue Corrosion / General Corrosion		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete EC 125925. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Fatigue / Environmentally-Assisted Fatigue		
010303	1,4	71300 71310	VALVE, CHECK/NONRETURN/BACKFL, Cat 1/2, LPSW-05, 05A	Protects pumps from backflow.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion Fatigue / Mechanical Fatigue Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	SRST - Functional Test	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Implement a one-time inspection/overhaul/repair/replacement of 1-71310-NV20. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
010305	1,4	71300 67131	VALVE, CHECK/NONRETURN/BACKFL, Cat 1/2, CA-52, 53	Prevents backflow of air supply to control valves.	Very Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion Fatigue / Mechanical Fatigue Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
010306	1,4	71300 71340	VALVE, CHECK/NONRETURN/BACKFL, Cat 1/2, HPSW-01, 01A	Prevents backflow from EHPSW pumps 71340-P1 and 71340-P2.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion Fatigue / Mechanical Fatigue Corrosion / Flow Assisted Corrosion (Erosion-Corrosion)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010307	1,4	71300 71340	VALVE, CHECK/NONRETURN/BACKFL, Cat 1/2, HPSW-02	These valves prevent backflow of high pressure service water from pumps 71340-P3 & -P4.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue Corrosion / General Corrosion Corrosion / Flow Assisted Corrosion (Erosion-Corrosion)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010308	1,4	71300 71340	VALVE, CHECK/NONRETURN/BACKFL, Cat 1/2, HPSW-12	These check valves prevent backflow from the boiler and steam water system into the emergency boiler water system.	Very Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion Fatigue / Mechanical Fatigue Corrosion / Flow Assisted Corrosion (Erosion-Corrosion)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010309	1,4	71300 71310	VALVE, CHECK/NONRETURN/BACKFL, Cat 1/2, LPSW-03, 03A	These NVs are LPSW pump discharge check valves, whose function is to prevent backflow.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u>

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Fatigue / Mechanical Fatigue Corrosion / General Corrosion Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		PM deferral will result in further degradation of the pumps. PM's need to be completed at their scheduled frequency to reach Plant EOL (2020). <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
010310	1,4	71300 71310	VALVE, CHECK/NONRETURN/BACKFL, Cat 1/2, HPSW-04, 04A	The valves prevent back flow / pump reverse rotation of the low pressure service water pumps (P1, P2).	Satisfactory	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete WO's for NV16 inspection to determine condition (WO # 02831926 and WO # 02832190). <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
010311	1,4	71300 71310	VALVE, CHECK/NONRETURN/BACKFL, Cat 1/2, HPSW-08, 09	The check valves prevent coolant (LPSW) back-flow from the heat exchanger / turbine oil coolers.	Satisfactory	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Surveillance	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete outstanding work orders WO 2786118, WO 2968105, WO 2967008 and WO 2967009 to inspect/repair the respective valves. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Corrosion / Fouling (accumulation of deposits)		Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010314	1,4	71300	VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2, Backwash Strainer MVs	These are pneumatically operated valves which control the LPSW backwash through the LPSW pump strainers.	Poor	Corrosion / Fouling (accumulation of deposits) Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms – Wear		Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Complete a one-time replacement of the MV's to reach CO EOL (2024).
010315	1,2,3,4,0	71300 71310	VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2, Piston, Butterfly Keystone MVs	The valves in the CG are motorized / pneumatic valve to supply cooling. MV124 is for Gen. stator cooling, MV142 is for oil coolers, MV176 is for Isolated Phase Bus (IPB) cooling, MV186, 187 are for HX on Vacuum pump, MV194 is for service transformer, MV432 is for main oil cooler, MV54 is for LPSW supply, and MV94 is for Gen. hydrogen cooling.	Satisfactory	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion	Overhaul/Refurbishment SRST - Functional Test	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete a one-time overhaul/refurbishment of valve, actuator and instruments for valves 1,4-71310-MV186, MV187. Note that all remaining equipment tags are subject to overhaul/refurbishment. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010316	1,4	71300 71310	VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2,	The valves in the CG are pneumatic operated valves to control / isolate flow to service water loads such as	Good	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation	Inspection - Internal Inspection - Visual Overhaul/Refurbishment	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
			Piston, Butterfly Jenkins MV	generator stator cooling and Isolated Phase Bus (IPB).		/ Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion	SRST - Functional Test Valve Diagnostics	
010317	1	71300 71310	VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2, Piston, Butterfly Jamesbury MV	The valve in this CG is a pneumatically operated MV on the LPSW pump discharges header. It provides remote isolation for LPSW loads such as isolated phase bus.	Satisfactory	Corrosion / General Corrosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Creep and Stress Relaxation Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Overhaul/Refurbishment	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete outstanding WO to refurbish valve 1-71310-MV176 and instruments (WO# 02254655). Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010318	1,4	71300 71310 71340	VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2, Piston, Centerline Butterfly MV	These valves are pneumatic motorized valves (MV502, 505) used for isolation of service water to RB Air Condition Unit (ACU), and for isolation of D2O vapour recovery (MV660).	Satisfactory	Corrosion / General Corrosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Creep and Stress Relaxation Mechanical and Thermal Degradation / Deterioration of	Overhaul/Refurbishment SRST - Stroke Test Valve Diagnostics	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: 1. Complete replacement of valve assembly 1-71310-MV502 (ref. NICR 105583; WO# 2878189). 2. Overhaul valve and actuator of 1-71310-MV505 (WO 939721). 3. Replace valve assembly of 71340-MV660 (WO #02417971). Incremental recommendations for Plant EOL (2020):

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010319	4	71300 71310	VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2, Piston, ITT Grinell Diaphragm MV	These valves are pneumatic operated valves for service water supply to RB Air Condition Unit (ACU).	Good	Corrosion / General Corrosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Creep and Stress Relaxation Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete WO 1918140. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010320	4	71300 71340	VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2, Piston, Butterfly Keystone MVs Associated UPP Supply MV and High Pressure pump discharge MV	These valves are pneumatic valves for service water isolation of pump discharge (MV28, 29), and UPP supply (MV30, 31).	Satisfactory	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Inspection - Visual SRST - Functional Test	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete WOs for valve replacement (1561412, 1561410). Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Complete a one-time actuator overhaul for 4-71340-MV28/29.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
010321	1,4	71300	VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2, Fisher 657 ES	The valves are located in the High Pressure Service Water System, which supplies cooling lake water to heat exchangers, cooling units and other equipment in the Reactor Building. MV510 and MV513 modulate to provide cooling to R/B ACU's. They also OPEN to increase flow to Vault ACU's on high pressure in the R/B.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010323	1,3,4,0	71300 67131 71310	VALVE, PRESSURE REGULATING, Cat 1/2, Fisher 67CFR Associated with Program Valves	These valves regulate instrument air pressure for MVs and CVs.	Very Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Component Diagnostics Inspection - Internal Inspection - Visual Overhaul/Refurbishment SRST - Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010324	1,4	71300 71310 71340	VALVE, PRESSURE REGULATING, Cat 1/2, Fisher 64R Associated with Program Valves	The PRV's regulate instrument air supply pressure to the actuators of their respective control valves.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue	Calibration Overhaul/Refurbishment SRST - Functional Test Valve Diagnostics	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010325	4	71300 71340	VALVE, PRESSURE REGULATING, Cat 1/2, Fisher 67FR Associated UPP Supply MV and High Pressure	These valves regulate instrument air pressure for pneumatic MVs.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Surveillance	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
			pump discharge MV			Mechanical and Thermal Degradation / Wear Mechanisms - Wear		
010326	1,2,3,4	71300 67131 71310 71340	VALVE, PRESSURE RELIEF, Cat 1/2	Relief valves are credited with over pressure protection for components and system piping, for example: 67131-RV2021 – relief for back-up air receiver 71340-RV725/726 – OPP for recombination units 71310-RV136/147/588/626 – OPP for LPSW to HX	Satisfactory	Corrosion / Fouling (accumulation of deposits) Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Incremental recommendations for Plant EOL (2020): Complete a one-time replacement of 4-71340-RV764 and 4-71340-RV766 to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Complete a further one-time replacement of 4-71340-RV764 and 4-71340-RV766 to reach CO EOL (2024).
010327	1,4	71300 67131 71340	SOLENOID/SOLENOID OPERATED VAL, Cat 1/2, ASCO associated with Program AOVs	Electro-mechanical operated valve, position of mechanical plunger is controlled by electric current flowing through a solenoid to open or close valve.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging	Component Diagnostics Overhaul/Refurbishment SRST - Functional Test	Continue with current practices. No additional practices are recommended to reach CO EOL (2024).
010329	4	71300 67134	SOLENOID/SOLENOID OPERATED VAL, Cat 1/2, Associated UPP Supply MV and High Pressure	These solenoid valves are used to operate the HPSW P3 & P4 discharge motorized valves (i.e. MV28, 29, 30 & 31).	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	SRST - Functional Test	Incremental recommendations for Plant EOL (2020): No additional recommendations required to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024):

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
			pump discharge MV					Implement PM to overhaul/replace SVs during MV overhaul PM which is currently set at 208 week interval.
010330	1,4	71300 67131 71310	SOLENOID/SOLENOID OPERATED VAL, Cat 1/2, Valcor SV associated with Program AOVs	These solenoid valves control operation of associated control valves (SV1) or, provide backup air to associated control valves (SV2044 & SV2049)	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Calibration Component Diagnostics Function Test Inspection - Visual Overhaul/Refurbishment	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010331	1,4	71300 71310 71340	SOLENOID/SOLENOID OPERATED VAL, Cat 1/2, ASCO NPX associated with Program AOVs	These solenoid valves are used to control the operation of associated MVs.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010332	1,4	71300 67131	SOLENOID/SOLENOID OPERATED VAL, Cat 1/2, Schader Bellows SV associated with Program AOVs	This Solenoid Valves SV424 operates seal oil cooler MV432.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010337	1,4	71300 67131	TRANSFORMER, Cat 1/2	These transformers reduce 600 Vac supply voltage to an acceptable level (120Vac) for strainer control logic circuits.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Predictive Maintenance - Thermography	<u>Incremental recommendations for Plant EOL (2020):</u> No additional recommendations required to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Perform a one-time inspection of transformers and if degraded repair/replace to ensure equipment reaches CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011170	5,6,7,8	71300 71310	71310 Emergency LPSW Pumps	In the event of class IV power failure, three 50% capacity emergency low pressure pumps (71310-P4, P3, P4), operating on class III power, these pumps discharge to the low pressure service water header to supply essential equipment . ECI Recirculation HX 056-33350-HX1 cooling is provided by dedicated Class III power LPSW pumps 5-71310-P1 and 6-71310-P1. ECI Recirculation HX 078-33350-HX1 cooling provided by dedicated Class III LPSW pumps 7-71310-P1 and 8-71310-P1.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Corrosion / General Corrosion Mechanical and Thermal Degradation / Lubricant Contamination Corrosion / Microbiological Influenced Corrosion Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Corrosion / Fouling (accumulation of deposits)	Functional Test Inspection - Ultrasonic Test Lubrication Overhaul/Refurbishment Predictive Maintenance - Vibration Monitoring	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> PM deferral will result in further degradation of the pumps. PM's need to be completed at their scheduled frequency to reach Plant EOL (2020). <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Complete a one-time detailed inspection of the concrete supporting pads.
011171	5,6,7,8	71300 71310	71310 LPSW Pumps	These pumps are 3 X 50% capacity pumps drawing water from the intake channel and discharging to a common LPSW header which supplies cooling loads and also supplies HPSW.	Poor	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Corrosion / Microbiological Influenced Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring,		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> PM deferral will result in further degradation of the pumps. PM's need to be completed at their scheduled frequency to reach Plant EOL (2020). <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Initiate a one-time inspection of equipment concrete support pads.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion Mechanical and Thermal Degradation / Lubricant Degradation		
011175	5,6,7,8	71300 71340	71340 HPSW Pumps	High pressure service water (HPSW) is taken from the low pressure outlet header and raised in pressure by high pressure pumps discharging into the high pressure discharge header. High pressure water is delivered by two vertically-mounted pumps (71340-P3 and P4) operating on class IV power and two vertically-mounted pumps (71340-P1 and P2) operating on class III power.	Satisfactory	Corrosion / Microbiological Influenced Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion Mechanical and Thermal Degradation / Lubricant Degradation	Inspection - Internal Lubrication Overhaul/Refurbishment	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> PM deferral will result in further degradation of the pumps. PM's need to be completed at their scheduled frequency to reach Plant EOL (2020). <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011370	5,6,7,8	71300 71310 71340	71310 Emergency LPSW Pump Motors-4kV	The Emergency Class III LPSW Pump Motors provide power for associated pumps.	Satisfactory	Thermal Aging / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms – Wear	Inspection - Visual Lubrication Overhaul/Refurbishment Predictive Maintenance - Electrical Testing	<u>Incremental recommendations for Plant EOL (2020):</u> No additional recommendations required to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Refurbish all ELPSW pump motors before 2020 to restore condition of all ELPSW motors to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
							Predictive Maintenance - Thermography Predictive Maintenance - Thermography	
011371	5,6,7,8	71300 71340	71340 Emergency CLIII HPSW Pump Motors-4kV	The Emergency Class III HPSW Pump Motors provide power for the associated pumps.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms – Wear Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Lubrication Overhaul/Refurbishment Predictive Maintenance - Electrical Testing Predictive Maintenance - Thermography Predictive Maintenance - Vibration Monitoring	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Accelerate completion of motor refurbishment program that was started in 2009, to support continued operation. <u>Incremental recommendations for Plant EOL (2020):</u> No additional recommendations required to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011372	5,6,7,8	71300 71310	71310 Main LPSW Pump Motors-4kV	Class IV LPSW Pump Motors are used to drive LPSW pumps, which supply cooling water to various nuclear and conventional loads using one of three 100% duty pumps.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms – Wear Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Corrosion / Microbiological Influenced Corrosion Mechanical and Thermal Degradation / Thermal Aging		<u>Incremental recommendations for Plant EOL (2020):</u> Complete off-site motor refurbishment program. <u>Incremental recommendations for CO EOL (2024):</u> Implement off-line electrical testing PM's.
011383	5,6,7,8	71300 71310 71340	71310, 71340 Service Water Systems-Valves – Manual Diaphragm – Cat 1&2	These manual valves provide isolation in various applications; ACU inlet and outlet isolators, HPSW pumps gland supply isolation valve, AUX BFP5 B/V supply isolation valve etc.	Poor	Mechanical and Thermal Degradation / Wear Mechanisms – Wear Mechanical and Thermal Degradation / Wear Mechanisms – Erosion	Calibration Inspection - Visual Overhaul/Refurbishment SRST - Functional Test	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete Work Orders; 2928072, 3108125, 2928086, 2928089, 2928091, 2928092 and 2940512.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / Fouling (accumulation of deposits)		Incremental recommendations for Plant EOL (2020): Replace valves as they reach their 10 year life expectancy. Incremental recommendations for CO EOL (2024): Replace the valves that have not been replaced for 10 years.
011384	5,6,7,8	71300 67134 71310 71340 71650	71310, 71430 Service Water Systems – Valves – Manual Butterfly – CAT 3&4	The valves in this CG are manual isolators used within the main system and branch systems of the LPSW.	Poor	Mechanical and Thermal Degradation / Wear Mechanisms – Erosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms – Wear Corrosion / Fouling (accumulation of deposits)		Incremental recommendations for Plant EOL (2020): Complete a one-time replacement or refurbishment of valves that have not been replaced since 2010. Incremental recommendations for CO EOL (2024): Complete a one-time replacement or refurbishment of valves that have not been replaced since 2014.
011385	5,6,7,8,018	71300 71310 71340	71310, 71340 Service Water Systems – Valves – Manual Diaphragm – CAT 3&4	These manual valves provide isolation in various applications; LPSW to reheat drains pumps, isolation for LPSW to H2 dryer, ACU inlet isolation valve etc.	Poor	Mechanical and Thermal Degradation / Wear Mechanisms – Wear Corrosion / Fouling (accumulation of deposits) Corrosion / General Corrosion Mechanical and Thermal Degradation		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete Work Orders; 2692189 and 2692188. Incremental recommendations for Plant EOL (2020): Replace valves as they reach their 10 year life expectancy. Incremental recommendations for CO EOL (2024):

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						/ Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / Fouling (accumulation of deposits)		Replace valves that have not been replaced for 10 years.
011386	5,6,7,8 056, 078, 058, 018	71300 71310 71340	71310, 71340 Service Water Systems – Valves – Manual Gate & Globe – CAT 3&4	These manual valves provide isolation in various applications; HX2 LPSW inlet isolation valve, HPSW P3 discharge isolation valve, RB ACU outlet isolation etc.	Poor	Mechanical and Thermal Degradation / Wear Mechanisms – Wear Mechanical and Thermal Degradation / Wear Mechanisms – Erosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion Corrosion / Fouling (accumulation of deposits)		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete work orders; 2789153, 4809944, 3017545, 2795881, 2872114, 3233306, 3212335 and 3217769. <u>Incremental recommendations for Plant EOL (2020):</u> Replace valves as they reach their 10 year life expectancy. <u>Incremental recommendations for CO EOL (2024):</u> Replace valves that have not been replaced for 10 years.
011387	056, 078, 058	71300 71310	71310 Service Water Systems-Valves – MOV – Butterfly – CAT1&2	MV401 is a normally open valve used to throttle flow for frazzle ice protection. MV402/403 and MV404/405 are normally closed valves used to allow RBSW discharge flow to the Screenhouse.	Satisfactory	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms – Wear Mechanical and Thermal Degradation	Inspection - Internal Inspection - Summerization Program Inspection - Visual Inspection - Winterization Program	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete inspections of 056-71310-MV402/ MV403 and 078-71310-MV404/405. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u>

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						/ Wear Mechanisms – Erosion Corrosion / Fouling (accumulation of deposits) Corrosion / Microbiological Influenced Corrosion		Complete a one-time actuator inspection for 056-71310-MV402/403 and 078-71310-MV404/405 similar to PM#18360.
011388	5,6,7,8,056,078	71300 71310 71340	71310 / 71340 Service Water Systems-Valves - AOV-Standard-CAT1&2	The majority of valves in this CG control the flow to the Heat Exchangers (HXs). CV505 is a control valve for generator H2 cooling water, and MV657 is a motorized valve for HP service water.	Poor	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Corrosion / Fouling (accumulation of deposits)	Calibration Overhaul/Refurbishment SRST - Functional Test SRST - Logic Test SRST - Stroke Test	<u>Current initiatives that need to be completed for Plant EOL (2020) that are Incremental to current periodic maintenance practices:</u> Complete outstanding WOs: WO for 8-71310-CV585 replacement (WO 4907606), WO for 5-71310-CV583 monitoring temp changes (WO 4907649), WO for 5-71310-CV585 replacement (WO 2508842), WO for 7-71310-CV585 troubleshooting (WO 3067647). Generate WR for 6-71310-CV585 replacement (no WO). <u>Incremental recommendations for Plant EOL (2020):</u> Complete a sample inspection of one CV, based on results repair/replace as necessary and implement further PM activities for remaining CVs. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011389	5,6,7,8	71300 67131 71310 71340	71340 Service Water Systems-Valves - SV-Solenoid Valves-CAT1&2	Electro-mechanical operated valve, position of mechanical plunger is controlled by an electric current flowing through a solenoid to open or close flow of HPSW to LPSW pump motor bearings.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Corrosion / Microbiological Influenced Corrosion	Surveillance	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011390	5,6,7,8	71300 71310 71340	71310 / 71340 Service Water Systems-Valves	Low Pressure Service Water (71310): The pressure regulating valve regulates	Good	Corrosion / Microbiological Influenced Corrosion	Calibration Component Diagnostics	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
			- PRV's- CAT1&2	supply of the ABFP with backup cooling water from the fire protection system. High Pressure Service Water (71340): Pressure regulating valve for supply of backup cooling water to HPSW, CCW & LPSW pump glands and bearings from the domestic water system. High Pressure Service Water (71340): Pressure regulating valve for supply of HPSW to CCW & LPSW pumps for gland and bearing cooling and lubrication.		Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / Fouling (accumulation of deposits) Corrosion / General Corrosion Fatigue / Mechanical Fatigue	Component Replacement SRST - Functional Test SRST - Stroke Test	
011391	5,6,7,8 ,056, 078	71300 71310 71340	71310 / 71340 Service Water Systems-Non Return Valves - CAT1&2	The function of the check valves is to protect against back-flow in the LPSW and HPSW systems. Application examples include; LPSW Strainer discharge, LPSW Pump outlet, HPSW gland supply.	Satisfactory	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Corrosion / Fouling (accumulation of deposits)	Component Replacement Inspection - Internal Inspection - Non - Intrusive SRST - Functional Test	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> 1. Complete a one-time practice per the check valve strategy manual P-MAN-04946-00001 for all equipment tags that are not covered by an active PM to reach Plant EOL (2020). 2. Complete all Work Orders and Change Requests in the Action List of the CA. 3. Complete WO's: 2853616, 04781418 02736496, 02733802, 02626810, 03060255, 01703138, 02626810, 02736588 and 2735252. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u>

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
								No additional practices are recommended to reach CO EOL (2024).
011392	5,6,7,8	71300 67131 71340	71340 Service Water Systems-Cyclone Separators	The cyclone separators remove particulates from the cooling water for the HPSW, Main LPSW, and CCW pumps glands and motor bearing's.	Very Poor	Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Corrosion / General Corrosion		Incremental recommendations for Plant EOL (2020): Implement outstanding actions from ECR 16816, and complete a one-time replacement of all separators. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011393	5,6,7,8	71300 71349	71349 Service Water Systems-Piping-Hoses-CAT3&4	Flexible hoses used on the inlet/outlet of PHT Pump (33120) Stator Cooling.	Poor	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Wear Mechanisms - Fretting Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Thermal Aging		Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Complete a one-time replacement of all hoses.
011394	5,6,7,8,056,078	71300 71310	71310 Service Water Systems-Valves - Cast Iron Manual Butterfly - CAT1&2	The valves in this CG are manual isolation valves for LPSW pumps and strainers. V370 is a normally closed isolator for MV15 bypass.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion Corrosion / Fouling (accumulation of deposits)	Inspection - Visual SRST - Functional Test	Incremental recommendations for Plant EOL (2020): Replace 5, 7, 8-71310-V370. Replace or refurbish six other valves in this CG based on OPEX. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011395	5,6,7,8	71300 71310	71310 Service Water Systems-Valves - Cast	These valves are manual isolators for service water loads such as 73110-	Good	Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
			Iron Manual Gate - CAT1&2	ACU30/ACU31, 32110-HX1/HX2.		Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Wear Mechanisms - Erosion		
011396	5,6,7,8	71300 71310 71340	71310, 71340 Service Water Systems - Valves - Cast Iron Manual Diaphragm - CAT3&4	Manual diaphragm valves in this CG are main and branch isolators.	Poor	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / Fouling (accumulation of deposits) Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Erosion		<u>Incremental recommendations for Plant EOL (2020):</u> Perform a one-time replacement of valves as they approach their 10 year service life. <u>Incremental recommendations for CO EOL (2024):</u> Perform a one-time replacement of valves that have not been replaced for 10 years.
011397	5,6,7,8	71300 71310 71340	71310, 71340 Service Water Systems - Valves - Cast Iron Manual Butterfly - CAT3&4	Manual valves in this CG are used for isolation purposes.	Poor	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / Fouling (accumulation of deposits) Mechanical and Thermal Degradation / Deterioration of Material (Wiring,		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete outstanding WOs WO#: 3129325, 1478301, and 03129325). <u>Incremental recommendations for Plant EOL (2020):</u> Perform a one-time replacement of valves as they approach their 10 year service life. <u>Incremental recommendations for CO EOL (2024):</u> Perform a one-time replacement of valves that have not been replaced for 10 years.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Erosion		
011398	5,6,7,8 058,01 8	71300 71310	71310, 71340 Service Water Systems - Valves - Cast Iron Manual Gate & Globe - CAT3&4	Manual gate & globe valves are used in the LPSW system for isolation purposes.	Poor	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / Fouling (accumulation of deposits) Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / Fouling (accumulation of deposits) Corrosion / General Corrosion		Incremental recommendations for Plant EOL (2020): Perform a one-time replacement of valves as they approach their 10 year service life. Incremental recommendations for CO EOL (2024): Perform a one-time replacement of valves that have not been replaced for 10 years.
011399	5,6,7,8	71300 71310 71340	71310, 71340 Service Water Systems-Valves - Manual Butterfly - Criticality 1 & 2	The valves in this CG are main and branch isolators for the service water high & lower pressure systems.	Poor	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion	Component Replacement	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete WO 02945929, WO 02945931, WO 02657191. Incremental recommendations for Plant EOL (2020): Perform a one-time replacement of valves as they approach their 10 year service life. Incremental recommendations for CO EOL (2024):

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Corrosion / Fouling (accumulation of deposits) Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Perform a one-time replacement of valves that have not been replaced for 10 years.
011400	5,6,7,8	71300 71340	71310, 71340 Service Water Systems - Valves - Manual Gate & Globe - Cat 1 & 2	Valves provide isolation as follows: 5/6/7/8-71310-V522 LPSW to RB ACU'S & FM HX1E 5/6/7/8-71340-V13 Fire Header Water Supply 5/6/7/8-71340-V30 HPSW to Cyclone Separator Isolation Valve 5/6/7/8-71340-V42 HPSW P1 Gland Supply Isolation Valve 5/6/7/8-71340-V43 HPSW P2 Gland Supply Isolation Valve 6-71340-V851 Emergency Boiler Water System Isolation Valve 7-71340-V851 Emergency Boiler Water System Isolation Valve	Poor	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Corrosion / Fouling (accumulation of deposits) Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Inspection - Visual Lubrication	<u>Incremental recommendations for Plant EOL (2020):</u> 1. Perform a one-time replacement of valves as they approach their 10 year service life. 2. Procure spare CAT ID 123565. <u>Incremental recommendations for CO EOL (2024):</u> Perform a one-time replacement of valves that have not been replaced for 10 years.
011410	058	71300	Service Water Systems-Cable - 4.16 kV, 600V, 125V, 250V - Power Cables	Cables feed power from source to load.	Good	Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue Radiation Induced Degradation / Radiation Embrittlement	Inspection - Condition Monitoring Program	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011482	058	71300	71650 Service Water Systems - Chlorine Piping - Cat3&4	The buried chlorine piping carries lake water for Low Pressure Service Water (LPSW), Emergency Water System, and Screen Wash	Satisfactory	Corrosion / Pitting Corrosion Corrosion / General Corrosion	Surveillance	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				(SW) systems. The chlorinated piping is for protection against Zebra mussels and biofouling.		Mechanical and Thermal Degradation / Wear Mechanisms - Erosion		
011530	5,6,7,8	71300 71340	71340 Main CLIV HPSW Pump Motors-4kV	The CLIV High Pressure Service Water (HPSW) Pump Motors drive the CLIV HPSW Pumps and are used to take water from the LPSW pumps to loads requiring cooling water above 274' elevation primarily in the reactor building. The HPSW pumps also supply water to the fire protection system.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Thermal Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

System 0458 - Shield Cooling System

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
010353	1,4	34100 34110	EXPANSION JOINT	The expansion joints allow for movement due to vibration and thermal expansion/ contraction between the inlet piping and the Calandria end shield.	Satisfactory	Radiation Induced Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Cracking Due to Cyclic Loading Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Thermal Aging	Surveillance	Incremental recommendations for Plant EOL (2020): Add a task to routine RB operator rounds to perform a visual inspection of the expansion joints. Look for signs of mechanical/thermal degradation of the joints, as well as material deterioration resulting in cracking and failure of the pressure boundary. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010354	1	34100	EXPANSION JOINT	The expansion joints allow for movement due to vibration and thermal expansion / contraction between the inlet piping and the calandria end shield.	Satisfactory	Radiation Induced Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Cracking Due to Cyclic Loading Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Thermal Aging		Incremental recommendations for Plant EOL (2020): Add a task to routine RB operator rounds to perform a visual inspection of the expansion joints. Look for signs of mechanical/thermal degradation of the joints, as well as material deterioration resulting in cracking and failure of the pressure boundary. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010355	1,4	34100 34110	VALVE, PRESSURE RELIEF	These RVs provide over pressure protection for the shell side of the main end shield cooling heat exchangers.	Very Good	Mechanical and Thermal Degradation / Creep and Stress Relaxation Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010356	1,4	34100 34310 63431	VALVE, CONTROL, Cat 1/2	CV provides bubbler flow for sump level measurement.	Satisfactory	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Overhaul/Refurbishment SRST - Functional Test	Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Incremental recommendations for CO EOL (2024): 1. Re-activate PMs to calibrate the actuator & components every 728 days. 2. Complete a one-time actuator overhaul / replace sub-components for all control valves.
010357	1,4	34100 34310	RELAY, Cat 1/2	Timer relays used in the Emergency Coolant Injection system FM Sump isolation valve Control Circuits.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging Obsolescence / Immediate Obsolescence Concern	Predictive Maintenance - Thermography	Incremental recommendations for Plant EOL (2020): 1. Implement a PM to periodically replace the relays. 2. Implement a PM to perform periodic inspection and testing. 3. Populate model information and Cat ID in Asset Suite for 1/4-34310-MV5-R61. 4. Contact vendor to confirm spare part availability for 1/4-34310-MV5-R61 following identification of the model(s). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010360	1,4	34100 34310	VALVE, PRESSURE REGULATING, Cat 1/2	The PRV's regulate instrument air supply pressure to their respective MVs.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Creep and Stress Relaxation Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue	Overhaul/Refurbishment SRST - Functional Test SRST - Logic Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010361	1,4	34100 34310	VALVE, PRESSURE	The relief valves provide overpressure protection to their respective MV.	Good	Mechanical and Thermal Degradation / Deterioration of	Inspection - Visual Overhaul/Refurbishment	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
			RELIEF, Cat 1/2			Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Creep and Stress Relaxation Mechanical and Thermal Degradation / Wear Mechanisms - Wear		
010362	1,4	34100 34310	SOLENOID/SOLENOID OPERATED VAL, Cat 1/2	SVs actuate valves MV1 and MV5 as part of the Emergency Coolant Injection Systems Sump Isolation control scheme.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Creep and Stress Relaxation Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue Radiation Induced Degradation / Radiation Depletion of Material Properties		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010363	1,4	34100 34310	VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2	These valves are required to OPEN automatically and remain in that position when ECIS recovery is initiated to allow the recovery of water collected in the fuelling machine sumps.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Calibration SRST - Functional Test SRST - Logic Test Valve Diagnostics	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Ensure 1-34310-MV1/MV5 limit switches are replaced per EC 67465 and EC 67464.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Creep and Stress Relaxation Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue		Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010364	1	34100 34310	MOTOR, Cat 1/2	Actuator motor for valves 34310-MV2 / MV4 are a part of the Fuelling Machine Service Room Sump Drain Valves. These valves open when ECI Recovery mode is initiated.	Good	Radiation Induced Degradation / Radiation Depletion of Material Properties Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Cracking Due to Vibration Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010365	1,4	34100 34310	VALVE, MOTORIZED/MOTOR OPERATE, Cat 1/2	34310-MV2, MV4 – motorized valve D2O recovery isolator	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring,	Lubrication Inspection - Internal Inspection - Visual SRST - Functional Test Valve Diagnostics	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Lubricant Degradation Corrosion / General Corrosion		
010370	1,4	34100 63431	TRANSMITTER, Cat 1/2	This CG contains Level Transmitters (LTs) and Current Isolators (AXs)	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging Radiation Induced Degradation / Radiation Depletion of Material Properties		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010372	4	34100 63431	VALVE, CHECK/NONRETURN/BACKFL, Cat 1/2	These check valves prevent backflow of instrument air to the recovery sump level transmitters.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	SRST - Functional Test	Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Implement PMs for NV replacements.

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CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
010373	1,4	34100 63431	ALARM, Cat 1/2	The alarm units are used to provide annunciation to the MCR when RB flood and FM vault level exceeds its set point.	Poor	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Calibration Component Replacement SRST - Functional Test	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete WO 2796608 to correct deficiencies with 1-63431-L1-LIA1 and L4-LIA1. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Perform proactive replacement of Alarm Units.
011126	5,6,7,8	34100 34110	34110 End Shield Cooling Heat Exchangers	These heat exchangers are used to cool recirculated water from the shield cooling system.	Good	Corrosion / General Corrosion Corrosion / Microbiological Influenced Corrosion Corrosion / Pitting Corrosion Corrosion / Fouling (accumulation of deposits)	Clean and Inspect	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: 1. Complete outstanding WOs 02868142, 02868136, 02868103 to replace rear channels. 2. Complete WOs 2320095, 02839923 for inspection and cleaning activities. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011425	5,6,7,8	34100	Shield Cooling System-Cable - 4.16 kV, 600V, 125V, 250V - Power Cables	The cables used for Pickering GS B can be sub-divided into five (5) basic categories. There are 5 kV power cables, 600 V power cables, 600 V control cables, 300 V control cables	Good	Fatigue / Thermal Fatigue Fatigue / Mechanical Fatigue Radiation Induced Degradation /	Inspection - Condition Monitoring Program	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				and special definite purpose cables.		Radiation Depletion of Material Properties		
011479	5,6,7,8	34100 34110	34110 End Shield Cooling System-Non Return Valves - CAT1&2	34110-NV19, NV20, NV21 - These check valves are located on the shield tank inlet line to maintain inventory and on the ESC pump discharge to prevent backflow. 34110-NV3, NV4 – ESC pump discharge NV, water circulating pumps	Poor	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Wear Mechanisms - Erosion	Surveillance	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete WOs 02719289, 02733800, 02737230, 02734283, 02525539, 02734357 to overhaul/replace NVs. Incremental recommendations for Plant EOL (2020): Resolve spare parts issue associated with Cat ID 118501. Incremental recommendations for CO EOL (2024): Perform one-time NIT for valve groups consistent with P-MAN-04946-00001.
011488	5,6,7,8	34100	34100 - Shield Cooling System-Piping - Cat1&2	Piping is used to recirculate demineralized water from the shield tank to external heat exchangers.	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion	Inspection - Pipe Wall Thinning Program	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete WO 01463596 for piping inspections. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

TETRA TECH

System 0459 - Shutdown Cooling System

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
010375	1,4	33410 33410	ACTUATOR, Cat 1/2	These motorized actuators are used to open and close the shutdown cooling System isolation valves 1/4-33410-MV1, -MV2, -MV4 and -MV5, or -MV7, -MV8, -MV10 and -MV11.	Very Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Lubricant Contamination Mechanical and Thermal Degradation / Lubricant Degradation Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Overhaul actuators for EQ qualification that have not been overhauled prior to 1984.
010377	1,4	33410 63340	CONTROL, Cat 1/2	Control the shutdown cooling heat exchangers outlet temperature through modulating the service water valve and also provide indication of the temperature at the main control room.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Electronic Aging		Incremental recommendations for Plant EOL (2020): Perform proactive replacement of controller display screens. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010378	4	33410 63340	CONVERTER, Cat 1/2	Convert potentiometer signal to current signal to enable operators to manually control the outlet temperature of shutdown cooling heat exchanger.	Very Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging	SRST - Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
010379	1,4	33410 33410 63340	ELEMENT, Cat 1/2	Measure the temperatures of shutdown cooling loops and pump seals.	Good	Fatigue / Thermal Fatigue Fatigue / Environmentally-Assisted Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Calibration SRST - Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010380	1,4	33410 33410	HEAT EXCHANGER, Cat 1/2	The heat exchangers act as a shutdown heat sink for the PHT system residual heat when HTS is below 350degF Additional functions include: - Emergency cooldown of the heat transport system. - Provide a flowpath for emergency coolant injection/recovery.	Good	Corrosion / Fouling (accumulation of deposits) Mechanical and Thermal Degradation / Creep and Stress Relaxation Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Fretting	Inspection - Ultrasonic Testing Inspection - Visual SRST - Functional Test	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete the implementation of EC's 107659 and 108155 to install new leak off valves for the isolators on 1, 4-33410-HX-1, 2, 3, 4. <u>Incremental recommendations for Plant EOL (2020):</u> 1. Perform a one-time inspection of 1, 4-33410-HX1, 2, 3, 4 to include inspection of tubes for cleanliness and eddy current inspection. 2. Complete internal and external inspections of the shell for MIC on 1, 4-33410-HX1, 2, 3 and 4 every outage. 3. Initiate a one-time inspection and tube cleaning of 1, 4-33410-HX5, 6, 7, 8. <u>Incremental recommendations for CO EOL (2024):</u> 1. Initiate new PMs for tube cleaning and internal inspections of 1, 4-33410-HX5, 6, 7, 8 every 4 years. 2. Complete shell replacements on 1-33410-HX1, 2.
010381	1,4	33410 63340	INDICATOR, Cat 1/2	Provide indication of shutdown cooling pump suction pressure.	Good	Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Thermal Aging		
010385	1,4	33410 33410	MOTOR, Cat 1/2	The pump motors are used to drive the shutdown cooling pumps which circulate heavy water (D2O) in the Heat Transport System (HTS) following unit shutdown.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Radiation Induced Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Thermal Aging	Component Diagnostics Lubrication Overhaul/Refurbishment Predictive Maintenance - Vibration Monitoring	<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Reinstate Baker Analysis and Megger Check for pump motors (e.g. PM #6709-/10/3).
010388	1,4	33410 33410	PUMP, Cat 1/2	The shutdown cooling system pumps are used to circulate main heat transport system fluid to the shutdown cooling system heat exchangers when HTS temperature is below 350degF. Additional functions include: - Emergency cooldown of the heat transport system. - To provide a flowpath for emergency coolant injection/recovery.	Very Good	Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Inspection - Visual Lubrication Predictive Maintenance - Thermography Predictive Maintenance - Vibration Monitoring	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010395	1,4	33410 33410	SWITCH, Cat 1/2	Operates associated pumps and valves of the Shutdown Cooling System.	Very Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010396	1,4	33410 33410	SWITCH, Cat 3/4	Operates associated MVs of the Shutdown Cooling System.	Very Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
010397	1,4	33410 63340	TRANSMITTER, Cat 1/2	Transmits signals for shutdown cooling loop temperatures and pump suction pressures.	Very Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging Radiation Induced Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010400	1,4	33410 33410	VALVE, MOTORIZED/MOTOR OPERATE, Cat 1/2	These valves function as isolation for the Shutdown Cooling Inlet (33410-MV1/4/7/10) and the Shutdown Cooling Outlet (33410-MV2/5/8/11). They are also opened to supply ECIS water to the HTS after a LOCA.	Very Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Lubricant Degradation Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion	Inspection - Internal Inspection - Visual SRST - Functional Test SRST - Logic Test Valve Diagnostics	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010403	1,4	33410 33410	VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2	3341-MV29, 30, 31, 32 are depressurization/ pressurization valves. They control flow from shutdown cooling circuits to the	Very Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Inspection - Internal Inspection - Visual Overhaul/Refurbishment	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				purification line downstream of the bleed condenser. 3341-MV3, 6, 9, 12 are used to preheat the SDCS.		Fatigue / Mechanical Fatigue Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Thermal Aging	SRST - Functional Test SRST - Stroke Test	
010404	1,4	33410 33410	SOLENOID/SOLENOID OPERATED VAL, Cat 1/2	Operates the associated MVs of Shutdown Cooling System.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Radiation Induced Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Inspection - Internal Overhaul/Refurbishment SRST - Stroke Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011123	5,6,7,8	33410	33410 Heat Transport Shutdown Cooling HXs	The function of the Shutdown Cooling System heat exchangers is to remove heat from the primary heat transport system following reactor shutdown.	Good	Corrosion / Microbiological Influenced Corrosion Mechanical and Thermal Degradation		Incremental recommendations for Plant EOL (2020): Replace the shells of all shutdown cooling heat exchangers to mitigate extensive MIC and frequent leaks.

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CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						/ Wear Mechanisms - Fretting Corrosion / General Corrosion Fatigue / Mechanical Fatigue Corrosion / Fouling (accumulation of deposits) Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Corrosion / Pitting Corrosion Mechanical and Thermal Degradation / Creep and Stress Relaxation		Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011124	5,6,7,8	33410 33410	33410 Heat Transport Shutdown Cooling Pumps	The function of the pumps is to circulate the PHT flow through the shutdown cooling heat exchanger when the SDC system is operating.	Satisfactory	Mechanical and Thermal Degradation / Elevated Temperature Corrosion / Crevice Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation	Inspection - Visual Overhaul/Refurbishment Predictive Maintenance - Lubrication Analysis Predictive Maintenance - Thermography Predictive Maintenance - Vibration Monitoring	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						/ Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / Microbiological Influenced Corrosion		
011125	5,6,7,8	33410 33410	33410 Shutdown Cooling Pump Motors	Drive the shutdown cooling pumps which circulate heavy water (D2O) in the Heat Transport System (HTS) following unit shutdown.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Thermal Aging	Component Replacement Predictive Maintenance - Thermography Predictive Maintenance - Vibration Monitoring SRST - Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011176	5,6,7,8	33410 71340	71340 Shutdown Cooling HX Temperature Control Valves	The function of these valves is to modulate the service water flow to the Shutdown Cooling Heat Exchangers in order to maintain the required HX temperature control.	Satisfactory	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / Microbiological Influenced Corrosion Corrosion / Pitting Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Erosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011293	5,6,7,8	33410	33410 Shutdown Cooling System - 600V - Power Cables	Provide power to valves and pumps of the Shutdown Cooling System.	Good	Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Radiation Embrittlement	Inspection - Condition Monitoring Program	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011294	5,6,7,8	33410 33410	33410 Shutdown Cooling System - AOV's	The warmup valves (MV3, MV6, MV9 and MV12) are used for warming up the respective Shutdown Cooling (SDC) circuits prior to placing in service. The depressurization valves (MV29 to MV32) are used to depressurize the Primary HT System via the SDC circuits. The valves connect the SDC circuits to the bleed purification line downstream of the bleed condenser.	Satisfactory	Thermal Fatigue / Thermal Fatigue General Corrosion / General Corrosion Thermal Aging / Thermal Aging Wear Mechanisms - Wear / Wear Mechanisms - Wear Mechanical and Thermal / Deterioration of Material (Joint Seals Gaskets etc.)	Overhaul/Refurbishment SRST - Functional Test	<u>Incremental recommendations for Plant EOL (2020):</u> Perform one-time overhauls of the outstanding depressurization and warm-up valve actuators. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011295	5,6,7,8	33410 33410	33410 Shutdown Cooling System - Check Valves	The function of the NV is to prevent reverse flow in the SDC Gland Supply backup.	Satisfactory	General Corrosion / General Corrosion Mechanical Fatigue / Mechanical Fatigue Wear Mechanisms - Wear / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Wear Mechanisms - Erosion	Surveillance	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011347	5,6,7,8	33410 33410	33410 Shutdown Cooling System - Solenoid Valves	Operate associated MVs of the Shutdown Cooling System.	Satisfactory	Wear Mechanisms - Wear / Wear Mechanisms - Wear	SRST - Functional Test	<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Thermal Aging		Incremental recommendations for CO EOL (2024): Perform one-time proactive replacement.

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System 0460 - Shutdown System 1 (SDS1)

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011419	058	31730	Shutdown System 1 (SDS1)-Cable - 125V, 250V - Power Cables	The cables used for Pickering GS B can be sub-divided into five (5) basic categories. There are 5 kV power cables, 600 V power cables, 600 V control cables, 300 V control cables and special definite purpose cables. Cables having a serial number -101 to -399 are sourced from a 120v or 208v supply. Cables having a serial number -601 to -799 are supplied at 575V from a motor control centre. Cables having a serial number -501 to -599 are supplied from a 600v switchgear source. Cables having a serial number -401 to -499 will have a 4.16kV power sources.	Good	Fatigue / Thermal Fatigue Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Inspection - Condition Monitoring Program	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011434	5,6,7,8	31730 31730	31730 Shutdown System 1 (SDS1) - Major Equipment - Shut-off Units	The shut-off units drop a shut off rod into the reactor that is used to shut down the reactor rapidly when a trip condition occurs. They are also used to drive the rod back out and hold it in the ready position.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Fretting Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Creep and Stress Relaxation Mechanical and Thermal Degradation / Deterioration of Material (Wiring,	Inspection - Visual SRST - Functional Test	<u>Incremental recommendations for Plant EOL (2020):</u> Replace SA Rod Ready Reed Switch assemblies on the SA Mechanisms in Units 5, 6, 7 and 8. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Seals, Gaskets, O-rings etc.)		
011454	5,6,7,8	31730 63173	63173 - Shutdown System 1 (SDS1)-SIGNAL CONDITIONER -CAT 1&2	The ESPM, via relay logic, stops power to S/A drive motor when the rod has reached a predetermined position limit when driving in either the "IN" or "OUT" direction. As well, it provides analog position indication of the shut off rod to a panel meter in the control room and the DCC's.	Poor	Thermal Aging / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear Obsolescence / Immediate Obsolescence Concern		Incremental recommendations for Plant EOL (2020): Perform proactive replacements on Unit 7 & 8. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011468	5,6,7,8	31730 63721	63721 Shutdown System 1 (SDS1)-Independent PCB-S/R	The Rod Ready circuit displays the number of SORs outside the core, ready for a reactor trip. The circuit causes an alarm if the number is less than 26. The rod ready circuit is a duplicated system consisting of two identical rod ready chassis and two identical digital panel meters with alarms.	Poor	Thermal Aging / Thermal Aging Obsolescence / Immediate Obsolescence Concern	Inspection - Indication Checks SRST - Functional Test	Incremental recommendations for Plant EOL (2020): Perform proactive replacements on Unit 7 & 8. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011469	5,6,7,8	31730 31741 63170 63173	31741, 63170, 63173 Shutdown System 1 (SDS1)-Reactivity Deck Receptacles / Connectors-Various Voltages A.C. and D.C.	Cables and connectors in this CG are used to connect the following components to plant wiring: Shutoff Rod motors, Shutoff Rod position potentiometers, Shutoff Rod "Rod Ready Switches", Shutoff Rod electrical clutches and SDS1 Vertical Flux Detectors.	Good	Mechanical and Thermal Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

System 0461 - Shutdown System 2 (SDS2)

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011128	5,6,7,8	34710 34710	34710 LISS Poison Addition & Sampling Pumps	Pumps used to mix, transfer, sample and adjust the gadolinium concentration in the main LISS injection Tanks.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Obsolescence / Immediate Obsolescence Concern Corrosion / General Corrosion	SRST - Functional Test	Incremental recommendations for Plant EOL (2020): Acquire a spare pump for 34710-P1. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011212	5,6,7,8	34710 63470 63738	63730 Shutdown System 2 (SDS2)-TRANSMITTER-ANALYTICAL-CAT 1&2	Fisher Temperature Transmitters are part of measurement loop for Heat Transport High Temperature Trip.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Radiation Embrittlement		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011245	5,6,7,8	34710 63470	63470 Shutdown System 2 (SDS2) Conductivity Probes	Conductivity probes/cells are used to monitor the conductivity of the liquid in each of the six injection lines from the LISS poison injection tanks during normal operation.	Good	Galvanic Corrosion / Galvanic Corrosion Wear Mechanisms - Erosion / Wear Mechanisms - Erosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011256	5,6,7,8	34710 34710	63730 SDS2 Check Valves	The NV's prevent instrument air back-flow into their associated LISS MV's in the event of a loss of instrument air supply. -NV13, 14 and 15 function as non-return valves and prevent the loss of helium from the helium header through the vent line. - NV217 and NV54 prevent reverse flow in the P2 discharge line and RB ventilation respectively. - NV9 is a check valve which prevents depressurization of the helium supply tank 3470-TK10 if the line upstream of	Satisfactory	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / Fouling (accumulation of deposits) Corrosion / General Corrosion	Component Replacement Inspection - Internal Inspection - Non - Intrusive Leak Test Overhaul/Refurbishment	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete all active work orders to replace and/or inspect valve internals. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Complete one-time replacement of 6-34710-NV217 and 8-34710-NV15.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				the tank is depressurized accidently. Failure of this NV to open can delay start-up due to the inability to repoise the LISS after maintenance.				
011279	5,6,7,8	34710 34710 63734 63736 63739 63753 63754 63756	34710 Shutdown System 2 (SDS2)-Valves - Manual-Criticality Category 1 (RS2)	This commodity group contains a number of manually operated isolation valves that perform the following functions: Drain, Vent, Recirculation, Suction and Supply.	Good	Corrosion / Chemical Attack - Aggressive Mechanical and Thermal Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Component Replacement Leak Test SRST - Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011280	5,6,7,8	34710 34710 63734 63736 63737 63739 63753 63754 63756	63730, 34710 Shutdown System 2 (SDS2) PRVs - CAT1&2	The PRV's are required to supply SDS2 (Liquid Injection Shutdown System) with a regulated air supply for SDS2 AOV operation and testing of process trip instrumentation. PRV7 supplies regulated helium pressure from bottle supplies to TK10.	Good	Thermal Aging / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion Fatigue / Mechanical Fatigue	Logic Test Overhaul/Refurbishment	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011281	5,6,7,8	34710 31760 34710 63739 63754	31760 Shutdown System 2 (SDS2)-Piping-Piping Components-CAT1&2	Components in this CG includes instrument tubing, rupture disks, restricting orifices, manifold and filter assemblies. The header feeds helium through 1" SS piping to the six poison tanks in turn connected via 2-1/2" stainless steel pipes to the core horizontal injection nozzles. These components also facilitate the movement of helium and pressurized injection of D2O poison into the reactor core.	Good	Environmentally-Assisted Fatigue / Environmentally-Assisted Fatigue Thermal Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Creep and Stress Relaxation Corrosion / Fouling (accumulation of deposits)	Inspection - Internal Inspection - Visual SRST - Functional Test	Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Complete mechanical design evaluation of the LISS Piping for continued operation and implementation of any findings.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011282	5,6,7,8	34710 34710 63753	34710 Liquid Injection Shutdown System (LISS) Tanks	The LISS tanks store the poison solution in readiness for injection. These tanks have a Polyethylene ball used for sealing. Other tanks and vessels in this CG are the helium supply tanks, instrument air tanks, gadolinium sample cabinet tanks, etc.	Good	Corrosion / General Corrosion Wear Mechanisms Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Inspection - Pressure Test Inspection - Sampling Test Inspection - Visual Leak Test	<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> 1. Perform internal inspections of one of the LISS Tanks in each of the Units (5/6/7/8-34710-TK1, TK2, TK3, TK4, TK5, and TK6). 2. Perform an internal inspection of the mixing tanks in each of the units (5/6/7/8-34710-TK11). 3. Perform an internal inspection of the Helium Supply Tank (5/6/7/8-34710-TK10) in each of the units.
011296	058	34710	34710 Shutdown System 2 (SDS2) - 600 Volt - Power Cables	Major Equipment associated with these cables are: Liquid Injection Poison Pumps (Unit-3471-PM1 / PM2).	Good	Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Radiation Embrittlement	Inspection - Condition Monitoring Program	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

System 0462 - Shutdown System A (SDSA)

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
010405	1,4	63721 63104	ACTUATOR, Cat 1/2	This CG contains pneumatic actuators that retract the Ion Chamber Shutter Assembly. In its normal position, the shutter partially shields the Ion Chamber and thus reduces the thermal neutron flux at its location. The movement of each of the neutron-absorbing shutters creates a local neutron flux disturbance which is used to monitor the static and dynamic response of the associated Ion Chamber and its associated electronics.	Satisfactory	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Radiation Induced Degradation / Radiation Depletion of Material Properties		Current practices are adequate. No additional practices are recommended to reach CO EOL (2024).
010406	1,4	63721 63101 63312 63336 63721 63723 63725 63744 64323	ALARM, Cat 1/2	Most of the alarm units (including indication) responsible for process and neutronic trip coverage. HTLGF and some HTHP/LP units are for alarm only (not indication).	Good	Mechanical and Thermal Degradation / Electronic Aging		<u>Incremental recommendations for Plant EOL (2020):</u> Complete proactive replacement of obsolete/problematic Versatile meters. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
010407	1,4	63721 63725	AMPLIFIER, Cat 1/2	Generate linear, Log N and Log N Rate signals which are used for annunciation and trip logic.	Good	Mechanical and Thermal Degradation / Electronic Aging	Calibration SRST - Functional Test	<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach CO EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Initiate EQ PM to replace the amplifiers, as the amplifiers will be reaching their qualified life.
010409	1,4	63721 63716	ANALYZER, Cat 1/2	Single Channel Analyser is part of start-up instrumentation loop which measures neutron flux from when the reactor power is below the rationality limits of the Ion Chambers.	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging	SRST - Functional Test	<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u>

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
								Perform one-time replacement.
010412	4	63721 63720	CAPACITOR, Cat 1/2	4-63720-CAP/Dx/Ex/Fx reduce or eliminate relay 63720-R81 contact bounce when R81 is energized/de-energized. 4-63720-CAP1/2/3 is part of the RC circuits which provide the rate of change of the linear power to the trip alarm unit. The capacitance of the RC circuit is provided with capacitor (external to the amplifier), and the resistance by the internal impedance of the trip alarm unit.	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging	SRST - Functional Test SRST - Logic Test	Incremental recommendations for Plant EOL (2020): Implement a comprehensive capacitor monitoring (thermography) PMs. Incremental recommendations for CO EOL (2024): Implement proactive replacement PMs.
010413	1	63721 63720	CAPACITOR, Cat 3/4	The capacitor is part of the RC circuits which provide the rate of neutron power changes to the SDSA High Linear N Rate trip alarm unit. The capacitance of the RC circuit is provided with capacitor (external to the amplifier and the trip alarm unit), and the resistance is by the internal impedance of the trip alarm unit.	Very Good	Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010417	1,4	63721 63721	COMPUTER, Cat 1/2	The function of Dump Arrest Unit (DAU) is to prevent moderator dump following successful shutoff rod operation or to open the dump valves to allow the moderator to dump when the reactor power rundown is inadequate.	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010418	1,4	63721 63744	CONTROL, Cat 1/2	Control the valves 1/4-63744-CV1 to perform Boiler Feedline Low Pressure trip tests.	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
010419	1,4	63721 63721	CONVERTER, Cat 1/2	The signal converters are used to convert the 0 to -10 V output signal from the SDSA ion chamber amplifier to a 0 to +5 V signal for Dump Arrest Unit to use.	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging		Incremental recommendations for Plant EOL (2020): Perform one-time proactive replacement to address the EPRI recommended 10 year age related limit. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010420	1,4	63721 63721 63725	DETECTOR, Cat 1/2	The flux detectors provide an analog signal representative of the neutron flux in the vicinity of each detector in the reactor core. These localized neutron flux signals are used to determine reactor power when the unit is operating at Full Power and trip the reactor when reactor power exceeds neutron overpower trip setpoint. The ion chambers detect thermal neutrons and generate a signal proportional to the Thermal neutron flux at their location, which is sent to the amplifier.	Satisfactory	Radiation Induced Degradation / Radiation Depletion of Material Properties	Calibration SRST - Logic Test	Incremental recommendations for Plant EOL (2020): Inspect all detectors, if needed repair/replace. Incremental recommendations for CO EOL (2024): Replace all ICFD.
010423	1,4	63721 63101 63312	ELEMENT, Cat 1/2	Generate signals representing reactor channel flows or outlet header temperatures.	Good	Mechanical and Thermal Degradation / Thermal Aging Radiation Induced Degradation / Radiation Embrittlement Corrosion / Flow induced wear of the leading edge		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010425	1,4	63721 63101 63173	INDICATOR, Cat 1/2	Provide indications of HTS flows, shutoff rod positions, or shutoff rod clutch currents.	Good	Mechanical and Thermal Degradation / Electronic Aging	Calibration SRST - Functional Test	Incremental recommendations for Plant EOL (2020): Complete one-time replacement of meters and position indicators.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Thermal Aging		Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010428	1,4	63721 63173	MODULE, Cat 1/2	Transmit signals representing the positions of shutoff rods.	Poor	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging	Calibration Inspection - Visual	Incremental recommendations for Plant EOL (2020): Complete one-time replacement of all modules. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010429	1,4	63721 31730	MOTOR, Cat 1/2	The Shut-off Rod (SOR) motors are used to withdraw the Shutoff (SA) Rods out of the reactor core.	Satisfactory	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Lubricant Degradation Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging Radiation Induced Degradation / Radiation Embrittlement	Component Replacement SRST - Functional Test	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Proactive replacement to be completed: Backlogged Work Orders to replace all Unit 1, 4 aged SOR motor to be completed. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010433	1,4	63721 63173 63720	POTENTIOMETER, Cat 1/2	Potentiometers with SCI 63173 provide resistance signals representing Shutoff Rod (SOR) position. Potentiometers with SCI 63720 are used to test moderator dump valves and dump arrest units (DAUs).	Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging		Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Implement one-time replacement of DAU potentiometers.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Wear Mechanisms - Wear		
010434	1,4	63721 63173 63721 63744	POWER SUPPLY, Cat 1/2	Supply power to various SDSA devices.	Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Thermal Aging	SRST - Functional Test SRST - Logic Test	<u>Incremental recommendations for Plant EOL (2020):</u> Implement a PM for proactive life cycle replacement. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
010437	4	63721 63170	RECEPTACLE, Cat 1/2	To connect shutoff rod clutch with the power supply circuit.	Satisfactory	Corrosion / Oxidation Corrosion / Pitting Corrosion Radiation Induced Degradation / Radiation Embrittlement		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010445	1,4	63721 31730 63104	SWITCH, Cat 1/2	Provides contact inputs indicating whether the shutoff rods are ready or whether the movements of the ion chamber shutters are completed.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Thermal Aging Radiation Induced Degradation / Radiation Embrittlement	SRST - Functional Test SRST - Logic Test	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Replace all hand switches/pushbuttons when performing SDS tests. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
010446	1,4	63721 63720 63721 63744	SWITCH, Cat 1/2	These handswitches and pushbuttons are used to change the operating state of associated SDSA devices.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u>

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CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
								Replace all hand switches/pushbuttons when performing SDS tests. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010451	1,4	63721 63101 63173 63336 63723 63744 64323	TRANSMITTER, Cat 1/2	Measure HT coolant flow for Heat Transport Low Gross Flow (HTLGF) Trip circuits; Measure reactor outlet headers pressures, boiler feedlines pressure for Heat Transport High Pressure (HTHP) Trip, Heat Transport Low Pressure (HTLP) and Heat Transport Very Low Pressure (HTVLP) Trip and Boiler Feedline Low Pressure (BFLP) Trip circuits; Measure the levels of the boilers for Boiler Low Level (BLL) Trip circuits.	Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Creep and Stress Relaxation Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Environmentally-Assisted Fatigue Corrosion / Fouling (accumulation of deposits)	Calibration Component Replacement Predictive Maintenance - Thermography SRST - Functional Test	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Implement outstanding WOs for the pro-active replacement of Boiler Room High Pressure (BRHP) Pressure Transmitters. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010452	1,4	63721 63744	VALVE, CONTROL, Cat 1/2	These valves control flow through the adjustable test pressure instrument loop for the SDSA Boiler Feedline Low Pressure (BFLP) Trip System.	Satisfactory	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010455	1,4	63721 63104	VALVE, PRESSURE	The pressure-reducing valve limits the high pressure instrument air that enters	Satisfactory	Mechanical and Thermal Degradation / Deterioration of		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
			CONTROL, Cat 1/2	the pneumatic shutter control device.		Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		
010456	1,4	63721 63101 63336 63744	VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2	This CG contains pneumatic isolation valves which provide isolation for various test circuits. Example: The SDSA boiler feedline pressure trip measurement loop, the Loss of coolant trip and test circuits and the process side HTS high, low and very low pressure trips.	Satisfactory	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear	SRST - Functional Test SRST - Logic Test	<u>Incremental recommendations for Plant EOL (2020):</u> Replace all valves with USI 63101 and USI 63744 that have not been replaced since 2012. <u>Incremental recommendations for CO EOL (2024):</u> Perform one-time replacements of all valves which have not been replaced since 2016.
010457	1,4	63721 63104	VALVE, SPEED CONTROL, Cat 1/2	The speed with which the shutter changes position is controlled by the air Flow adjustment through the flow control valves in this CG. One of these valves is located on each side of the piston, the first valve adjusts the air flow and the second valve adjusts the exhaust air / shutter speed to generate a rate Log N.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Radiation Induced Degradation / Radiation Depletion of Material Properties		<u>Incremental recommendations for Plant EOL (2020):</u> Complete a one-time replacement of the following equipment in this CG: 1-63104-R1-FC1, 4-63104-R1-FC1, 1-63104-R1-FC2, 4-63104-R1-FC2, 1-63104-R1-FC3, 4-63104-R1-FC3, 1-63104-R1-FC4, 4-63104-R1-FC4, 4-63104-R1-FC5 and 4-63104-R1-FC6. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
010458	1,4	63721 63101 63336 63723 63744	SOLENOID/SOLENOID OPERATED VAL, Cat 1/2	Solenoid valves are used to operate associated valves to perform SDSA tests.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Calibration Component Replacement SRST - Stroke Test	<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Implement one-time replacement.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Creep and Stress Relaxation		
010460	1,4	63721 63716	DETECTOR, Cat 1/2	Detector is part of Start-up instrumentation loop, which measures neutron flux at low power levels until ion chambers are rational and on-scale.	Satisfactory	Radiation Induced Degradation / Radiation Depletion of Material Properties	Inspection - Visual SRST - Logic Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010461	1,4	63721 63716	MODULE, Cat 1/2	Measure low level neutron flux during the reactor start-up, send the measurement signals to the SDSA trip logic, and generate alarms when the measurements exceed the predefined setpoints.	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Thermal Aging Radiation Induced Degradation / Radiation Embrittlement	Calibration Inspection - Spectrum Checks SRST - Functional Test SRST - Logic Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010462	1,4	63721 63716	POWER SUPPLY, Cat 1/2	Supply power to SDSA start-up instrumentation.	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Thermal Aging	Inspection - Spectrum Checks SRST - Functional Test	<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> 1. Procure sufficient spares. 2. Perform one proactive replacement. 3. Initiate PM to Monitor AC Ripple and DC Output Voltage before start-up.
010463	1,4	63721 63104	SOLENOID/SOLENOID OPERATED VAL, Cat 1/2	Operate the ion chamber shutter withdrawal mechanism to perform Neutron Overpower and High Rate Log N trip test.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation /		<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Creep and Stress Relaxation Radiation Induced Degradation / Radiation Embrittlement		Incremental recommendations for CO EOL (2024): Perform one-time replacement of the SVs to reach CO EOL (2024).
010464	1	63721 63716	AMPLIFIER, Cat 1/2	Amplifier/pre-amplifier is part of start-up instrumentation loop which measures neutron flux when reactor power levels fall below the rationality limits for the ion chambers.	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010465	1,4	63721 63716	RECEPTACLE, Cat 1/2	Receptacle is located in panel 63716-PL173 on 6610-PL4E in control room connecting preamplifier signal from detector through high voltage power supply module.	Satisfactory	Corrosion / Pitting Corrosion	Inspection - Internal SRST - Logic Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

System 0463 - Shutdown System E (SDSE)

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
010466	1,4	63732 63736	ALARM, Cat 1/2	SDSE HTS High/Low Pressure Trip logic. The instrument loop for each pressure measurement consists of pressure transmitter, current alarm unit, isolation amplifier, indicating meter and dedicated Class II channelized 45VDC power supply to the pressure transmitter.	Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010467	1,4	63732 63732	AMPLIFIER, Cat 1/2, ICFD	Vertical Flux Detector Amplifier. The input current signal is derived from the In-Core Detector. The Signal is fed to the In-Core Amplifier Circuit Board and it passes through the Comparator Circuit Board. When the input current signal exceeds the Trip Set-point (alarm mode), the Comparator Output will de-energize the relay and through this relay contacts, it will transfer the alarm signal to the other modules in the system.	Good	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	t	Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Perform one-time proactive replacement.
010468	1,4	63732 63735	AMPLIFIER, Cat 1/2, Fission Chamber	SDSE fission chamber (FC) amplifiers used to condition and amplify the FC detector signals for displays and input to the SDSE trip logic.	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Component Replacement SRST - Logic Test	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete installation of modified amplifiers per Design ECs 102583 and 102590. Incremental recommendations for Plant EOL (2020): Resolve spare part issues. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
010469	1,4	63732 63733	ARRESTOR (Dump Arrest Unit), Cat 1/2	SDSE dump arrest units (DAU) used to initiate a dump if power rundown, following a reactor trip caused by any SDSE trip parameter, is not adequate.	Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	SRST - Functional Test	Incremental recommendations for Plant EOL (2020): Procure sufficient spares. Incremental recommendations for CO EOL (2024): Implement new PMs for calibration.
010470	1,4	63732 63735 63736	BOX, Cat 1/2	Junction boxes used for SDSE cable splicing or termination.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / Galvanic Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010474	1,4	63732 63736	CONTROL, Cat 1/2	Controllers used for SDSE HTS pressure high and low trip tests.	Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Thermal Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010477	1,4	63732 63735	CYLINDER, Cat 1/2	Air cylinders (shutters), a part of the fission chamber unit are used to test the operation of the ex-core neutron flux monitoring.	Good	Mechanical and Thermal Degradation / Thermal Aging Radiation Induced Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Deterioration of		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Material (Wiring, Seals, Gaskets, O-rings etc.)		
010478	1,4	63732 63732	DETECTOR, Cat 1/2	Flux detectors are used to monitor the thermal neutron flux and generate a signal to the associated amplifier to trip.	Good	Radiation Induced Degradation / Radiation Depletion of Material Properties Fatigue / Environmentally-Assisted Fatigue	Calibration Predictive Maintenance - Electrical Testing SRST - Functional Test SRST - Logic Test	<u>Incremental recommendations for Plant EOL (2020):</u> Inspect all detectors, if needed repair/replace. <u>Incremental recommendations for CO EOL (2024):</u> 1. Perform prompt fraction checks as required prior to CO EOL (2024). 2. Replace all ICFD.
010479	1,4	63732 63735	DETECTOR, Fission Chamber, SPV	Fission chambers (detectors) used to measure and generate a trip condition for the SDSE high Log N Rate.	Good	Fatigue / Environmentally-Assisted Fatigue Radiation Induced Degradation / Radiation Depletion of Material Properties	Calibration SRST - Functional Test SRST - Logic Test	<u>Incremental recommendations for Plant EOL (2020):</u> 1. Perform Fission Chamber detector Plateau Checks every 2 years. 2. Based on the results of the Plateau Checks, replace the detectors if required. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
010483	1,4	63732 63736	FILTER, Cat 1/2	This equipment filters instrument air that is used for valve control in various SDSE process trip test circuits.	Very Good	Corrosion / Fouling (accumulation of deposits) Radiation Induced Degradation / Radiation Embrittlement		Continue current practices. No additional practices are recommended to reach CO EOL (2024)
010496	1,4	63732 63736	POWER SUPPLY, Cat 1/2	The power supply used to supply power to the instruments of SDSE HTS high/low pressure trip loops (G/H/J).	Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Component Replacement Inspection - Visual SRST - Functional Test	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Proactive replacements to be completed for the power supplies (e.g. WO# 3003011). <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u>

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
								Implement a new PM for monitoring AC ripple and output voltage
010499	1,4	63732 63735 63736	REGULATOR, Cat 1/2	The equipment in this CG regulates the pressure of the instrument air supply to valve actuators. The actuators control valves which are part of the SDSE Process Trip Test Circuit.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Radiation Induced Degradation / Radiation Embrittlement Corrosion / Fouling (accumulation of deposits) Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Inspection - Set-point Check SRST - Functional Test SRST - Logic Test	Incremental recommendations for Plant EOL (2020): Initiate a one-time replacement of all valves in this commodity group. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010500	1,4	63732 63230 63731 63732 63733 63735 63736	RELAY, Cat 1/2	Relays used for SDSE control, trip and test logic.	Good	Corrosion / Oxidation Mechanical and Thermal Degradation / Electronic Aging	Calibration Predictive Maintenance - Thermography SRST - Functional Test SRST - Logic Test	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Replace the relays 1-63731-R93G/R94G/R96G per WO# 3261833. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010508	1,4	63732 63731 63732 66700	SWITCH, Cat 1/2	Hand switches (e.g., 63732-HS1G/H/J) used for SDSE-NOP setpoint switching.	Very Good	Mechanical and Thermal Degradation / Electronic Aging Corrosion / Oxidation		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010510	1,4	63732 63731 63733	TIMER, Cat 1/2	These timers are used for testing the Loop Response time. They measure shutter stroking time which is then	Good	Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				verified against the requirements.		Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		
010511	1,4	63732 63736	TRANSMITTER, Cat 1/2	For Channel G, each of the 4 pressure transmitters feeds a signal to a dual setpoint Alarm Unit. If the high trip setpoint is exceeded on any one of the alarm units, this will result in de-energizing of HT high pressure trip relay, opening contacts in the main trip chain.	Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010516	1,4	63732 63736	VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2	The air operated globe valves in this CG provide process control / isolation in the SDSE process trip circuits for the applicable channel.	Good	Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Corrosion / Fouling (accumulation of deposits) Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete the following Work Orders 4857423 and 4880224 to replace passing valves. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010518	1,4	63732 63735 63736	SOLENOID/SOLENOID OPERATED VAL, Cat 1/2	If the reactor power rundown, as sensed by the SDSE dump arrest units is not adequate, the Moderator	Good	Corrosion / Fouling (accumulation of deposits)	Component Replacement SRST - Logic Test	Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				Dump Valve solenoids will be de-energized, opening the dump valves and the Moderator will be dumped.		Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Electronic Aging		Incremental recommendations for CO EOL (2024): Implement new PMs for the solenoid valves for use in SDSE HTS High/Low Pressure Trips.

System 0464 - Site Electrical System

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CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
011411	058	53200	53200 Site Electrical System-Cable - 4.16 kV - Power Cables	The cables are associated with the 4KV Site Electrical System.	Good	Fatigue / Thermal Fatigue Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Radiation Embrittlement	Inspection - Condition Monitoring Program	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

System 0465 - Standby Generators

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
010563	012, 034	54600 54660	ACCUMULATOR, Cat 3/4	The accumulators dampen pressure fluctuations in the fuel oil system to prevent vibrations that can result in breaks in the piping and potential fuel release into the environment.	Good	Fatigue / Mechanical Fatigue Corrosion / Galvanic Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / Environmental Degradation		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010564	012, 034	54600 54600	ACTUATOR, Cat 1/2	012-54600-SG2-GVA21, 012-54600-SG3-GVA31, 034-54600-SG1-GVA11, 034-54600-SG2-GVA21, 034-54600-SG3-GVA31 actuators control the position of the governor valve based on signals from the governor controller. All other tags in this CG are actuators that control louvers to admit light and air into the SG enclosure, while keeping rain and direct sunshine out. The louvers also mitigate outside noise.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Functional Test	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> 1. Complete implementation of Master EC 114279 and WO's 02620689, 02620691 for governor valve replacement. 2. Complete WO 04802166 to ensure that the louver failure was not a result of actuator failure. <u>Incremental recommendations for Plant EOL (2020):</u> Complete a one-time inspection of governor actuator. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
010565	012	54600 54600	ACTUATOR, Cat 3/4	The actuators control louvers to admit light and air into the SG enclosure, while keeping rain and direct sunshine out. The louvers also mitigate outside noise.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						/ Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		
010568	012, 034	54600 54600	ASSEMBLY, Cat 1/2	Valve drain assembly. These valves drain unused fuel from combustion chambers.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	SRST - Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010569	012, 034	54600 54600	BATTERY, Cat 1/2	BY1 supplies 28V DC to the gas turbine starter system. BY2 supplies 125V DC to the DC lube oil pump motors, the fuel system servo-actuator motor and the gas generator governor actuator motor. DC power is also supplied to the control panels, switchgear and static inverter.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Corrosion / General Corrosion		Incremental recommendations for Plant EOL (2020): Based on the condition of battery bank BY2, discharge tests to be completed yearly until planned battery bank replacement. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010570	012, 034	54600	BOX, Cat 1/2	The junction box is a container for electrical connections, intended to conceal the connections from sight and provide protection.	Good	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010571	012, 034	54600 54600 65460	BOX, Cat 3/4	The junction box is a container for electrical connections, intended to conceal the connections from sight and provide protection.	Good	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010572	012, 034	54600 54600 54630	CIRCUIT BREAKER, Cat 1/2	The circuit breakers protect electric circuits from damage caused by	Good	Corrosion / Pitting Corrosion	Inspection - Internal Overhaul/Refurbishment	Current initiatives that need to be completed for Plant EOL (2020) that are

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				overcurrent/overload or short circuit by interrupting the current flow.		Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Thermal Aging	SRST - Functional Test	incremental to current periodic maintenance practices: EC#118495 and WOs# 4857973, 3272375, 44675553, 44675558, 44675563, 44675567, 44675571 to be completed. Incremental recommendations for Plant EOL (2020): 1. Implement PMs on the CBs that are not subject to already under a PM. 2. Resolve obsolescence issues on applicable breakers. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010573	012, 034	54600 54600	CIRCUIT BREAKER, Cat 3/4	The circuit breakers protect electric circuits from damage caused by overcurrent/overload or short circuit by interrupting the current flow.	Good	Corrosion / Pitting Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Thermal Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010574	012, 034	54600 54600	CHARGER, Cat 1/2	The battery charger puts energy into the battery by forcing electric current through it.	Good	Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Electronic Aging Corrosion / General Corrosion Corrosion / Oxidation		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete overhaul of chargers. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

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010575	012, 034	54600 54600	CHAMBER, Cat 1/2	Six combustion chambers encircle the gas generator downstream of the axial flow compressor. The gases resulting from combustion in the chambers are used to power the SG turbines.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / Environmental Degradation Corrosion / Fouling (accumulation of deposits) Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete WO 04848091. Incremental recommendations for Plant EOL (2020): Complete a one-time inspection of the combustion chamber to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Complete a further one-time inspection of the combustion chamber to reach CO EOL (2024).
010576	012, 034	54600 54600	COMPRESSOR, Cat 1/2	These compressors increase air pressure for combustion of fuel oil.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Thermal Fatigue Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Lubricant Degradation Mechanical and Thermal Degradation / Lubricant Contamination Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring,	SRST - Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Seals, Gaskets, O-rings etc.)		
010578	012, 034	54600 54600 54660 65466	CONTACTOR, Cat 1/2	A contactor is an electrically controlled switch used for switching an electrical power circuit, similar to a relay except with higher current ratings.	Good	Corrosion / Pitting Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Thermal Aging		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: 1. WO# 2804770 to be completed to restore condition of 034-54600-CN303-E. 2. Procure spares to facilitate prompt replacement if required. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010579	012, 034	54600 54600	CONTACTOR, Cat 3/4	A contactor is an electrically controlled switch used for switching an electrical power circuit, similar to a relay except with higher current ratings.	Good	Corrosion / Pitting Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Thermal Aging		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: 1. WO# 3170958 to be completed to restore condition of 034-54600-CN202-E. 2. Procure spares to facilitate prompt replacement if required. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010580	012, 034	54600 54600 65460 65462	CONTROL, Cat 1/2	The controller sends/transmits output signals related to the governor valve control, exhaust gas, lube oil, or cubicle temperatures.	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging		Incremental recommendations for Plant EOL (2020): Proactive replacement of all 014-SGs louver controllers (C2038) to be completed per SCR P-2013-08067. Incremental recommendations for Plant EOL (2020):

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								No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010582	012, 034	54600 54600	CONVERTER, Cat 1/2	The temperature transmitter converts a Type K thermocouple sensor input signal to 4-20mA output for the interfacing controllers.	Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging		Continue current practices. No additional practices are recommended to reach EO EOL (2024).
010586	012	54600 54600	DAMPER, Cat 1/2	Dampers are used to provide ventilation and cooling for generator/drive cubicles.	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach EO EOL (2024).
010587	012	54600 54600	DAMPER, Cat 3/4	These dampers provide air flow to generator/drive cubicles, and prevent overheating.	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024)
010588	012, 034	54600 54600	DETECTOR, Cat 1/2	The fire detector detects and responds to the presence of a flame or fire.	Good	Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

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						Mechanical and Thermal Degradation / Thermal Aging		
010591	012, 034	54600 54600	ELEMENT, Cat 1/2	The temperature element (Type J thermocouple) is used for the lube oil monitoring circuit. It measures temperature and the output signal from this thermocouple is fed to the controller.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010592	012, 034	54600 54600 65460	ELEMENT, Cat 3/4	The temperature element (Type J thermocouple) is used to measure temperature, the output signal from this thermocouple is fed to the controller.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

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010593	012, 034	54600 54600	EXCITER, Cat 1/2	The exciter supplies current and generates the rotor magnetic field, and thus controls the terminal voltage of the generator.	Good	Fatigue / Environmentally-Assisted Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Thermal Aging Fatigue / Mechanical Fatigue Corrosion / Oxidation Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010594	012, 034	54600 54600	FAN, Cat 1/2	These fans provide air flow for lube oil cooling, and for controlling air supply and exhaust for: - combustion air - exhaust excess heat from engine surfaces and generator - maintaining reasonable temperature within the SG powerhouse - preventing negative pressure in the SG powerhouse.	Good	Corrosion / Fouling (accumulation of deposits) Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Overhaul/Refurbishment	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Ensure spare parts/spare fans are available to facilitate repair/replacement/corrective action if fans fail regular function test. Incremental recommendations for Plant EOL (2020): Perform a one-time lubrication of fan/motor bearings, and inspection of fan for wear and proper alignment to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Perform a further one-time lubrication of fan/motor bearings, and inspection of fan for wear and proper alignment to reach CO EOL (2024).
010595	012, 034	54600	FAN, Cat 3/4	These fans (including filter) remove air particulate/dirt for air ventilation of SG control cubicle.	Satisfactory	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Ensure spare parts/spare fans are available to facilitate

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CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Fatigue / Mechanical Fatigue Corrosion / Fouling (accumulation of deposits) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		repair/replacement/corrective action if fans fail regular function test. <u>Incremental recommendations for Plant EOL (2020):</u> Perform a one-time lubrication of fan/motor bearings, and inspection of fan for wear and proper alignment to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Perform a further one-time lubrication of fan/motor bearings, and inspection of fan for wear and proper alignment to reach CO EOL (2024).
010596	012, 034	54600 54600	FILTER, Cat 1/2	Filters are used in lube oil and fuel system for SGs, to filter out solid debris.	Good	Mechanical and Thermal Degradation / Lubricant Contamination Mechanical and Thermal Degradation / Lubricant Degradation Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / Fouling (accumulation of deposits) Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010600	012, 034,0	54600 54600 65466	GAUGE, Cat 3/4	These pressure gauges provide various pressure readings for:	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging		<u>Incremental recommendations for Plant EOL (2020):</u> Perform proactive replacement of the gauges, priority given to CC3 equipment.

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				-differential pressure across strainers -pump suction and discharge pressure -tank temperature gauge (thermometer) -metering valve inlet and outlet pressure gauge For lube oil and fuel oil flow.		Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Obsolescence / Immediate Obsolescence Concern		Incremental recommendations for CO EOL (2024): Perform calibration of gauges (covered during major outage SG work), priority given to CC3 equipment.
010601	012, 034	54600 54600	GAUGE, Cat 3/4	Bourdon type pressure gauges, providing local pressure indication for gas producer compressor discharge.	Good	Mechanical and Thermal Degradation / Electronic Aging Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010602	012, 034	54600 54600	GEARBOX, Cat 1/2	These gearboxes transfer power from power turbine to drive the generator, and also provides mounting for mechanical overspeed trip mechanism.	Satisfactory	Mechanical and Thermal Degradation / Lubricant Contamination Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Lubricant Degradation Fatigue / Mechanical Fatigue	Overhaul/Refurbishment Predictive Maintenance - Vibration Monitoring SRST - Functional Test	Incremental recommendations for Plant EOL (2020): Complete cost/benefit for obtaining one spare. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

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						Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		
010603	012, 034	54600 54600	GENERATOR, Cat 1/2	The Brush generator is a device which conducts current between stationary wires and moving parts in a rotating shaft.	Satisfactory	Corrosion / Fouling (accumulation of deposits) Fatigue / Environmentally-Assisted Fatigue Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Lubricant Contamination Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Predictive Maintenance - Vibration Monitoring SRST - Functional Test	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete WOs 4701963, 2834659, 3141105 to restore condition. <u>Incremental recommendations for Plant EOL (2020):</u> Perform major generator inspection (electrical and mechanical testing) on 012-SGs and 034-SGs <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
010604	012, 034	54600 54600	GENERATOR, Cat 1/2	The exciter supplies current and generates the rotor magnetic field, and thus controls the terminal voltage of the generator.	Good	Fatigue / Environmentally-Assisted Fatigue Fatigue / Mechanical Fatigue	Inspection - Internal Inspection - Visual SRST - Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / Oxidation Corrosion / General Corrosion		
010606	034	54600 54600	GENERATOR, Cat 3/4	These pulsation dampeners provide vibration mitigation and reduce mechanical fatigue in the fuel oil piping system.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010608	012, 034	54600 54600	HEAT EXCHANGER, Cat 1/2	Lube Oil Cooler for SG1, SG2 & SG3.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Lubricant Contamination Mechanical and Thermal Degradation / Lubricant Degradation Corrosion / Fouling (accumulation of deposits)	SRST - Functional Test SRST - Logic Test	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Ensure adequate spares are available to facilitate replacement if required. Incremental recommendations for Plant EOL (2020): Complete a one-time cleaning, internal inspection, NDE inspection and leak testing. Incremental recommendations for CO EOL (2024): Complete a further one-time cleaning, internal inspection, NDE inspection and leak testing.

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CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Corrosion / General Corrosion		
010609	012	54600 54600	HEAT EXCHANGER, Cat 3/4	Lube Oil Cooler for SG3.	Good	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Lubricant Contamination Mechanical and Thermal Degradation / Lubricant Degradation Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Corrosion / General Corrosion Corrosion / Fouling (accumulation of deposits)		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Ensure adequate spares are available to facilitate replacement if required. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010610	012, 034	54600 65466	HEAT TRACING, Cat 1/2	Heat tracing is run in physical contact along the length of the pipe and is used to maintain or raise the temperature of the pipe, in order to avoid an increase of the fuel oil viscosity and reduction of flow to the gas producer.	Poor	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete EC134232 (includes replacement of heat tracing) to restore condition. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020).

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						Mechanical and Thermal Degradation / Thermal Aging		Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010611	012, 034	54600 54600 54660 65466	HEATER (GENERIC), Cat 1/2	The heaters ensure that the fuel is warmed as it enters the gas turbine fuel system.	Poor	Corrosion / Oxidation Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete EC134232 (includes replacement of heaters) to restore condition. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010612	012, 034	54600 54600 65466	HEATER (GENERIC), Cat 3/4	The subject heaters ensure optimal operating temperature for equipment in the Control/Generator/Drive/CB Cubicle.	Poor	Corrosion / Oxidation Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Perform a one-time replacement of heaters.
010613	012, 034	54600 54600	HEATER (GENERIC), Cat 1/2	The subject heaters are used to maintain the temperature of the lube oil, in order to avoid increase of the lube oil viscosity and reduction of flow to the power turbine.	Poor	Corrosion / Oxidation Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete replacement of defective/damaged heaters. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024):

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
								No additional practices are recommended to reach CO EOL (2024).
010614	012, 034	54600 54600	INDICATOR, Cat 1/2	These indicators/meters measure and record speed/frequency (GPSIND/UI) of a periodic electrical signal or electrical voltage/power (VI/WI) through the circuit, and provide input signal for indication in the MCR. This allows for early detection of equipment failure and electrical balance of the power system.	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Perform a one-time replacement.
010618	012, 034	54600 65460	ISOLATOR REPEATER, Cat 3/4	The current isolator is used to reduce ground loop problems, which can cause inaccurate sensor readings by negatively affecting the signal from the temperature transmitter.	Good	Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010622	012, 034	54600 54600	MOTOR CONTROL CENTRE, Cat 3/4	The power supply for charging the batteries (28VDC and 125VDC) is supplied from the 600V bus in the MCC.	Good	Fatigue / Mechanical Fatigue Fatigue / Fatigue due to Vibration Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010623	012, 034	54600 54600	METER, Cat 1/2	These indicators/meters measure and record power VAR (XI) / Watt (WI) / Watt-hour (WZ) through the circuit and the Synchroscope (SY) is used for SG synchronization.	Satisfactory	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
010624	012, 034	54600 54600	METER, Cat 3/4	These indicators/meters measure and record Peak Time (PTM), Elapsed Time (ETM) and Base load Time (BTM) of electrical equipment/processes or Volt/Amp (VI/AI) through the circuit.	Satisfactory	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010625	012, 034	54600 54600 65460	MODULE, Cat 1/2	The vibration control unit (auxiliary relay) has output contacts for SG trip and high vibration annunciation.	Good	Mechanical and Thermal Degradation / Electronic Aging		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Project #13-40972 (SG Reliability Project) and EC# 131260 (for vibration system and SG controls upgrades) to be completed to restore condition. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010626	012	54600 65460	MODULE, Cat 3/4	The vibration monitors are used to monitor vibration at the Gear Box, AC Generator, Gas Producer and Power Turbine.	Poor	Mechanical and Thermal Degradation / Electronic Aging		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete Project #13-40972 (EC#131260) to upgrade the 012/034-54600-SG1/2/3 vibration monitoring system. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
010627	012	54600 65462	MODULE, Cat 1/2	The valve drive unit/module sends input signals to control the governor valve.	Good	Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010628	0	54600 65466	MODULE, Cat 3/4	The level/leak module is used to monitor the waste oil tank level for leak detection.	Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010629	012, 034	54600 54600	MONITOR, Cat 1/2	This generator voltage relay is used as an input to energize the governor controller.	Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010630	012, 034	54600 54600	MONITOR, Cat 3/4	The overspeed switch protects the power turbine from damage and safety hazard conditions of overspeed.	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Proactive replacement to be completed per Master NICR EC#111098. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
010631	012, 034,0	54600 54600 54660	MOTOR, Cat 1/2	These motors drive various loads in the SG system (e.g. pumps, fans, governor actuator).	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Thermal Aging		<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Perform a one-time overhaul of motors to last until CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Lubricant Degradation Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue Corrosion / Fouling (accumulation of deposits)		
010633	012, 034	54600 54600	MOTOR STARTER, Cat 1/2	The motor starter protects the motor from current overload.	Good	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Thermal Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010634	012, 034	54600 54600	MOTOR STARTER, Cat 3/4	The motor starter protects the motor from current overload.	Good	Fatigue / Mechanical Fatigue Fatigue / Fatigue due to Vibration Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Thermal Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010635	012, 034	54600 54600	ORIFICE, Cat 3/4	Restriction orifices used to ensure proper lube oil flow to power turbine and	Good	Mechanical and Thermal Degradation		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				generator bearings and other rotating elements.		/ Wear Mechanisms - Wear Fatigue / Mechanical Fatigue Corrosion / General Corrosion Corrosion / Flow Assisted Corrosion (Erosion-Corrosion) Corrosion / Fouling (accumulation of deposits)		
010636	012, 034	54600 54600 54630 54660 65460	PANEL, Cat 3/4	These panels are used to control the SGs.	Good	Mechanical and Thermal Degradation / Electronic Aging Fatigue / Environmentally-Assisted Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Thermal Aging Fatigue / Mechanical Fatigue Corrosion / Oxidation Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Project# 13-40972 (SG Reliability Project) to be completed to upgrade the 012/034-54600-SG1/2/3 control logic. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
010637	012, 034	54600 65460	PANEL, Cat 3/4	These relay protection panels control the SGs.	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging Fatigue / Environmentally-Assisted Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Thermal Aging Fatigue / Mechanical Fatigue Corrosion / Oxidation Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Project# 13-40972 (SG Reliability Project) to be completed to upgrade the 012/034-54600-SG1/2/3 control logic. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
010638	012, 034	54600 54600	PICKUP, Cat 1/2	This magnetic pick-up speed probe is used to detect the gas producer ignition speed.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

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CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Electronic Aging		
010641	012, 034	54600 54600 65462	POWER SUPPLY, Cat 1/2	PS1 supply power to the governor controller. Since the AVR is shunt (generator) powered, substantial current overloads can affect the AVR operating power and lead to loss of excitation. The excitation control system compensates for this by providing the AVR with a constant power source. During overload conditions, the SBO1 obtains the additional power necessary for the AVR operation from the current transformers.	Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> EC# 114279 to be completed (includes replacement of power supply). <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
010643	012	54600 65462	PROBE, Cat 1/2	This magnetic pick-up speed probe is used to detect the power turbine speed.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010645	012, 034,0	54600 54600 54660	PUMP, Cat 1/2	These pumps provide fuel flow to SG combustion chambers and lube oil flow to Gas Producer and Power Turbine bearings.	Satisfactory	Corrosion / General Corrosion Corrosion / Pitting Corrosion Corrosion / Flow Assisted Corrosion (Erosion-Corrosion)	Inspection - Visual Predictive Maintenance - Lubrication Analysis Predictive Maintenance - Vibration Monitoring SRST - Functional Test	<u>Incremental recommendations for Plant EOL (2020):</u> 1. Ensure PdM for thermography is conducted routinely (at 12 week frequency) for fuel forwarding pumps only (i.e. 012/0-54660-P101/102/201/202/301/302). 2. Complete a one-time alignment inspection on all pumps except

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Corrosion / Fouling (accumulation of deposits) Fatigue / Mechanical Fatigue Fatigue / Environmentally-Assisted Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		fuel forwarding pumps (i.e. 012/034-54660-P101/102/201/202/301/302). <u>Incremental recommendations for CO EOL (2024):</u> 1. Ensure PdM for thermography is conducted routinely (at 12 week frequency for fuel forwarding pumps only (i.e. 012/034-54660-P101/102/201/202/301/302). 2. Complete a one-time alignment inspection on all pumps except fuel forwarding pumps (i.e. 012/034-54660-P101/102/201/202/301/302).
010647	012, 034	54600 54600	REGULATOR, Cat 1/2	The AVR controls the excitation current to the stationary exciter field winding and therefore the output of the exciter, and by that, the generator output voltage and the reactive power output.	Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> For 034-54600-SG2, project #13-49289 (014 SG AVR Upgrade Project) to be completed. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
010648	012, 034	54600 54600 54630 65460 65466	RELAY, Cat 1/2	Protective relays prevent unnecessary trips, isolate faults and protect motors and breakers. Time delay/bias relays are arranged for an intentional delay/additional bias in output.	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Project #13-41044 (Pickering A SG Protective Relay Upgrade) and EC# 121013 to be completed for replacement

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Thermal Aging Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue Obsolescence / Immediate Obsolescence Concern		of protection relays for 014 SGs with digital multi-functional relays. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010649	012, 034	54600 54600 65466	RELAY, Cat 3/4	These relays detect faults and cause an alarm state.	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Thermal Aging Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue Obsolescence / Immediate Obsolescence Concern		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Project #13-40972 (SG Reliability Project) to be completed (includes proactive replacement of these relays as part of 014 SG Control Logic Upgrade). Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010651	012, 034	54600 54600	SENSOR/THERMAL CONVERTER, Cat 1/2	These thermal sensors measure temperature in the generator/drive/control cubicle and send a signal to the louver controller.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Thermal Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging		
010652	012, 034,0	54600 54600 65466	SENSOR/THERMAL CONVERTER, Cat 3/4	The leak detector probe 0-65466-LSR2001 monitors the oil level in the waste fuel oil tank 0-54660-TK2000, which is used to determine if there is a leak in the tank.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010655	012, 034	54600 54600 54660	STRAINER, Cat 1/2	Strainers are used to filter out solid debris in the standby generators & fuel oil system.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion Corrosion / Fouling (accumulation of deposits) Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
010657	012, 034	54600 54600	SWITCH, Cat 1/2	These current sensing switches produce a signal based on the exhaust gas temperature.	Satisfactory	Corrosion / General Corrosion Corrosion / Abrasion, Erosion and Cavitation Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Electronic Aging Obsolescence / Immediate Obsolescence Concern		Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Resolve obsolescence issues.
010658	012, 034	54600 54600	SWITCH, Cat 1/2	The blocking switch is a manual switch that is used to block the SG from tripping.	Satisfactory	Corrosion / General Corrosion Corrosion / Abrasion, Erosion and Cavitation Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Electronic Aging Obsolescence / Immediate Obsolescence Concern		Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Resolve obsolescence issues.
010660	012, 034	54600 54600 54660	SWITCH, Cat 1/2	The disconnect switch interrupts or opens an electric circuit, isolating the downstream circuit for purposes of inspection and maintenance.	Good	Corrosion / General Corrosion Corrosion / Abrasion, Erosion and Cavitation		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue		
010661	012, 034	54600 54600	SWITCH, Cat 1/2	The control switches impact the field breaker control. The ignition switches are starter switches in the control system of the SGs.	Satisfactory	Corrosion / General Corrosion Corrosion / Abrasion, Erosion and Cavitation Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Electronic Aging Obsolescence / Immediate Obsolescence Concern		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Resolve obsolescence issues per Project #13-40972 (SG Reliability Project, which includes upgrade of control logic system). Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010662	012, 034,0	54600 54600 65466	SWITCH, Cat 3/4	The output contacts of these switches are tied to the control/protection/annunciation logic.	Satisfactory	Corrosion / General Corrosion Corrosion / Abrasion, Erosion and Cavitation Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Electronic Aging Obsolescence / Immediate Obsolescence Concern		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: 1. Obsolescence issues to be resolved per Project #13-40972 (SG Reliability Project), which includes upgrade of control/annunciation logic. 2. Complete WO# 3137333, 3152941, 3152945, 3152938, 3152930 and 3152927 for replacement of switches. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
010663	012, 034	54600 54600	SWITCH, Cat 3/4	Speed switches sense gas producer (GPL/GPM) speed and transmit signals for control logic, alarm and trip functions.	Good	Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010664	012, 034	54600 54600 65466	SWITCH, Cat 1/2	Temperature switches detect and transmit variations in process temperature to alarm, and control instruments.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Thermal Fatigue Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Electronic Aging	Calibration SRST - Functional Test	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete EC (e.g. 29774) that removes CO2 fire protection system from SGs. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Complete proactive replacement of 056/078-54600 SG louvres 11/12 and the actuator motor.
010667	012, 034,0	54600 54600 54660	TANK, Cat 3/4	Tanks for compressed air for louver actuators, lube oil storage, and fuel oil storage.	Good	Fatigue / Thermal Fatigue Fatigue / Environmentally-Assisted Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010668	0	54600 54660	TANK, Cat 3/4	Fuel oil is stored in storage tanks 0-54660-TK1 & 0-54660-TK2 for U012 & U034.	Good	Corrosion / General Corrosion Fatigue / Environmentally-Assisted Fatigue Fatigue / Thermal Fatigue		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> 1. Complete corrective action WOs to repair tanks. 2. Complete WOs to perform 10 year inspection of tanks. <u>Incremental recommendations for Plant EOL (2020):</u>

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Wear Mechanisms - Wear		No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010669	012, 034	54600 54600	THERMOCOUPLE, Cat 3/4	HTs are Thermocouples that sense SG exhaust temperature. SG controls activate alarms and SG trip based on sensed temperature profile.	Good	Mechanical and Thermal Degradation / Thermal Aging Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010670	012, 034	54600 54600 54660 65466	TRANSMITTER, Cat 1/2	Transformers perform various functions. Current Transformers (CTs) are used to produce a low current signal useful for measurement and control of SG speed and power output. Potential Transformers (PTs) reduce voltages to levels that can be used in measuring or control circuits. Power Transformers (T1s), reduce voltages to match those of supplied loads.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Thermal Aging	Inspection - Internal SRST - Functional Test	Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Perform one-time inspection/maintenance checks of all transformers in this CG, and if degraded repair/replace to ensure components reach CO EOL (2024).
010672	012 ,034	54600 54600 65462	TRANSMITTER, Cat 1/2	UT1s are frequency to voltage transducers, WT's are power to voltage transducers, PQM's convert SG power output to current signal for SG Governor controller feedback. PT2006 measures fuel supply pressure at outlet of fuel pumps	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Calibration SRST - Functional Test	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010673	012, 034	54600 54600 65462	TRANSMITTER, Cat 1/2	PT 2002 detects and transmits gas producer compressor pressure to control instrumentation. TX/EXH detects SG exhaust	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				temperature and transmits to control and recording instrumentation. PT2008 detects fuel oil supply pressure and transmits to indicator.		Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		
010674	012, 034	54600 54600	TURBINE, Cat 1/2	This CG includes the gas producer and power turbine of the SG. The turbines provide the torque to drive the load via reduction gearbox to generate class III power in the event of loss of class IV power.	Satisfactory	Corrosion / General Corrosion Corrosion / Fouling (accumulation of deposits) Corrosion / Oxidation Corrosion / Pitting Corrosion Corrosion / Stress Corrosion Cracking - Nickel-base Alloys Fatigue / Environmentally-Assisted Fatigue Fatigue / Mechanical Fatigue Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Thermal Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010676	012, 034	54600	VALVE, AIR RELEASE, Cat 1/2	These Air Release Valves are used to vent fuel from fuel oil heaters to overflow tank after maintenance, to	Good	Corrosion / General Corrosion		Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				allow air venting from overflow tank. These valves protect equipment that is sensitive to air.		Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Incremental recommendations for CO EOL (2024): Perform a one-time inspection of an ARV, repair/replace, and conduct further inspections if first ARV is degraded.
010678	012, 034	54600 54600 54660 65466	VALVE, CONTROL, Cat 1/2	These control valves are part of the lube flow control to the power turbine bearings, and recirculating flow for the fuel forwarding pumps.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion	Calibration Overhaul/Refurbishment SRST - Functional Test	Incremental recommendations for Plant EOL (2020): Perform a one-time inspection of all valves (including valve internals) that are not covered under PM's for periodic replacement (i.e. only CV3029 is regularly replaced). If inspection indicates aging degradation, then repair or replace to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Perform a further one-time inspection for all valves (including valve internals) that are not covered under PM's for periodic replacement (i.e. only CV3029 is regularly replaced). If inspection indicates aging degradation, then repair or replace to reach CO EOL (2024).
010679	034	54600 54600	VALVE, CONTROL, Cat 3/4	This control valve is part of the lube flow control to the power turbine bearings.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010680	012	54600 65462	VALVE, CONTROL, Cat 1/2	These metering valves are part of the fuel control, which regulates fuel supply to the Standby Generators.	Satisfactory	Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring,	Inspection - Visual SRST - Functional Test SRST - Stroke Test	Incremental recommendations for Plant EOL (2020): Perform a one-time inspection of valve internals, and repair / replace if degraded to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024):

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Perform a further one-time inspection of valve internals, and repair/replace if degraded to reach CO EOL (2024).
010688	012, 034	54600 54600	VALVE, PRESSURE REGULATING, Cat 1/2	These are pressure regulating valves for the gas producer lube oil flow.	Satisfactory	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Functional Test	Incremental recommendations for Plant EOL (2020): Complete a one-time inspection, and repair/replace if degraded. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach Plant EOL (2024).
010691	012, 034	54600 54600 54660	VALVE, PRESSURE RELIEF, Cat 1/2	These relief valves provide overpressure protection for the lube and fuel oil system.	Very Good	Fatigue / Mechanical Fatigue Fatigue / High Vibrations Corrosion / General Corrosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010693	012, 034	54600 54600	SOLENOID/SOLENOID OPERATED VAL, Cat 1/2	SVs dump fuel to storage tank on an SG fault.	Satisfactory	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Component Replacement SRST - Functional Test	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete NICR 125627 for 034 SG3 to replace SV3007. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
010694	012, 034	54600 54600	SOLENOID/SOLENOID OPERATED VAL, Cat 3/4	SG governor fuel stop valves control fuel level in day tank.	Satisfactory	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011215	056, 078	54600 54600	54600 Standby Generators-TRANSMITTER-ANALYTICAL-CAT 1&2	Temperature Transmitters (TT) convert RTD temperature signals to an electronic output current for transmission to local and station monitoring equipment.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Electronic Aging		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Ensure all BOM items verified for use in this application. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011260	056, 078	54600 54600	54600 Standby Generators - Lube Oil Cooler Dampers/Generator air cooling Dampers	PT Lube oil cooler:- Model SMC-204008S employs one cooling oil circuit and is complete with indoor and outdoor dampers with electric motors for control of damper operation. It is also possible to control the air supply and air exhaust to permit other useful functions such as: A) Combustion Air. B) Exhausting excess heat from the engine surfaces and from the generator. C) Maintaining reasonable temperature within the SG powerhouse. D) Preventing negative air pressure in the SG powerhouse.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion Corrosion / Fouling (accumulation of deposits)	SRST - Functional Test SRST - Logic Test	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> 1. Perform one-time replacement of Generator air cooling system Dampers (056/078-54600-SG1/2/3-MDP1/2) and spare parts through NICRs. 2. Perform one-time replacement of lube oil cooler dampers and spare parts through NICRs. 3. Perform one-time replacement of actuator motor for the PT lube oil cooler. 4. Procurement engineering to BOM and CAT ID components of new dampers (replaced via NICRs from above recommendations).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				Generator cooling air dampers:- Hot air is exhausted outside through Exhaust damper when SG is running. For controlling the generator so that it do not fall below (-) 10 Deg C during winter operation, hot air is mixed with inlet air by controlled opening of the Recirculation damper.				Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011261	056, 078	54600 54600	54600 Standby Generators - Gas Producer Assemblies	The compressors (CP) are part of the turbine. They use atmospheric air and bring it to higher pressure for later fuel addition and combustion. The combustion chambers (CHR) control the combustion of the fuel in a controlled manner, to drive the SG turbines. The Gear Boxes (GB) transfer power from SG to generator shaft. The Gas Produce Turbine (TU) drives the power turbine and Electric Generator.	Satisfactory	Corrosion / Environmental Degradation Fatigue / Fatigue due to Vibration Corrosion / General Corrosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Fretting Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete the zero time overhauls of the gas producers under W/O's: 02563229, 02563235 and 02563238 for 056-SG2, 078-SG1 and 078-SG2 respectively. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Adjust the current maintenance strategy based on as-found condition of gas producers during zero time overhauls.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Corrosion / Fouling (accumulation of deposits)		
011290	058	54600	54600, 54660 Standby Generators - Cables - 600V, 125V, 250V	Cables feed power / control signals from source to load.	Good	Fatigue / Mechanical Fatigue Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Inspection - Condition Monitoring Program	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011305	056, 078	54600 54600	54600 Standby Generators- Standby Generator	This CG covers several components that are part of the SG assembly. This includes: "Power Turbine" (PT) converts kinetic energy produced by the Gas Producer into rotational kinetic energy (torque). "Gear Box" is connected directly to the Power Turbine and reduces the P/T speed from 7500 RPM to 1800 RPM for the Generator. "Electric Generator" converts rotational kinetic energy into electrical energy. Each generator is capable of delivering Class III power to two units during a complete loss of class IV power.	Good	Fatigue / High Vibrations Fatigue / Thermal Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Overhaul/Refurbishment Predictive Maintenance - Electrical Testing Predictive Maintenance - Vibration Monitoring SRST - Functional Test	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete WO's for one-time inspections of power turbines (i.e. WO's 02696119, 02696120, 02696121, 02696116, 02696117 and 02696118). <u>Incremental recommendations for Plant EOL (2020):</u> Complete a one-time inspection of generator on 056 bank, similar to inspection performed on 078 bank (078-SG2, WO 2146890). Results of inspection to determine if corrective actions or further inspections on other SGs are required. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
011337	056, 078	54600	54660 Standby Generators Fuel Oil - Underground Piping	Fuel Oil System pipes supply fuel oil from the storage tanks to the Combustion Turbines that drive the Standby Generators	Satisfactory	Corrosion / Pitting Corrosion Corrosion / General Corrosion	Inspection - Buried Piping Program	Continue current practices. No additional practices are recommended to reach CO EOL (2024).

System 0466 – Structures

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
010818	014	21000	Calandria Vault, Vault Structure Cooling, Shield Tank - Structural Concrete - Concrete Walls and Slabs	The main purpose of the vault structure is to provide shielding against the radiation from an operating unit and to provide structural support to Calandria in its operating and non-operating modes. The Calandria Vault and its demineralized light water provides operational and shutdown shielding for the immediately surrounding areas. The water also provides cooling for the calandria assembly and the vault concrete.	Good	Corrosion / Corrosion of Embedded Steel Mechanical and Thermal Degradation / Thermal Aging Radiation Induced Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Leaching Calcium Hydroxide		<u>Incremental recommendations for Plant EOL (2020):</u> Conduct one-time visual inspection of accessible areas of Calandria Vault structural concrete for Pickering NGS P1,4 to inspect the severity of any potential damage and identify mitigating/remedial measures needed to be carried out. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
010819	014	21000	Calandria Vault/Vault Structure Cooling/Shield Tank - Embedded Parts and Supports	Reactivity Mechanism Deck Supports and Connections: The Reactivity Mechanism Deck is part of the reactor vault assembly. It closes the top of the calandria vault, thus providing a boundary between the vault and the boiler room atmospheres. The deck is supported by the calandria vault and it seals the vault atmosphere by seal plates welded to both the lower plate and the vault liner. End Shield Manhole: A manhole is located at the top of each End Shield, passing through both the support shell and the End Shield shell. It was used to provide access during fabrication and ball filing and was permanently sealed by welded cover plates.	Good	Radiation Induced Degradation / Radiation Embrittlement Corrosion / Corrosion of Embedded Steel		<u>Incremental recommendations for Plant EOL (2020):</u> Conduct one-time visual inspection of accessible areas to confirm the suitability of the components for Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
010820	014	21000	Calandria Vault/Vault Structure Cooling/Shield Tank - Seals & Sealants	*Atmospheric SS Seals & Rubber Seals The Atmospheric seals together with the silicone rubber seals at elevation 317'-6" (extending to elevation 324'-0"), separate the east and west F/M room atmospheres. The Elastomer Silicone rubber seals at elevation 274'-0" prevent liquid spills into the 25mm (1inch) and 76mm (3inch) gaps around the Calandria Vault. Integrity of the barrier/seal has a direct effect on environment inside the Calandria Vault. *Inconel 600 Manhole Seal The Inconel 600 Manhole expansion joint seal is necessary to allow free movement of the seal plate due to a temperature differential. *Shear Key Joints The Horizontal keys (elev. 312'-0") and vertical keys (elev. 312'-0") in north and south cross-walls are provided to resist forces due to earthquake and unbalanced header failure pressure on the east or west face of the vault.	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Cracking Due to Expansion or Contraction		<u>Incremental recommendations for Plant EOL (2020):</u> 1. Complete one-time inspection of seals. 2. Replace the elastomeric seals if required. 3. Replace seals if they show advanced degradation. <u>Incremental recommendations for CO EOL (2024):</u> 1. Replace the elastomeric seals if required. 2. Replace seals if they show advanced degradation.
010821	014	21000	Calandria Vault/Vault Structure Cooling/Shield Tank-Liners - Steel Liners	Steel liner is a crucial part of the Calandria shielding system and provides a leak-tight seal for containment of the vault demineralised light water.	Good	Corrosion / General Corrosion		<u>Incremental recommendations for Plant EOL (2020):</u> Conduct one-time inspection for Pickering NGS P1,4 to verify if there is no evidence of component failure or degradation discovered. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
010822	014	21000	Calandria Vault/Vault Structure Cooling/Shield Tank-Penetrations - Steel Sleeves Surrounding Concrete Vault Openings	With the exception of the large diameter opening in each end shield wall and the rectangular opening in the roof slab the only other openings and penetrations in the vault wall are to provide support for moderator and other systems piping. These penetrations and their associated embedded parts for piping and other systems are designed to contain the shield water within the confines of the Calandria Vault.	Good	Radiation Induced Degradation / Radiation Embrittlement Corrosion / General Corrosion		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Maintain demineralized light water shield at a pH of 10.0 to 10.5 and a conductivity of 10 to 100 micro mhos/cm. Chemical oxygen scavenging also takes place. Incremental recommendations for Plant EOL (2020): Perform one-time visual inspection of accessible areas and local leakage testing to confirm components' suitability. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010823	0	21000	Cooling Water Intake & Outfall - Foundations, Steel - Piles	The function of steel piles is to provide structural supports for the structures above by transferring the load to bedrock.	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Cracking Due to Cyclic Loading		Continue current practice. No additional practices are recommended to reach CO EOL (2024).
010824	0	21000	Cooling Water Intake & Outfall - Foundations, Concrete - Footings	The concrete foundations for the trash removal area are designed to support the trash removal structure. The CW supply pipe anchor blocks provide support for the pipes that carry cooling water to the condenser units.	Good	Corrosion / Chemical Attack Mechanical and Thermal Degradation / Freeze-Thaw Mechanical and Thermal Degradation / Leaching Calcium Hydroxide Mechanical and Thermal Degradation / Cracking Due to Expansion or Contraction Corrosion / Corrosion of Embedded Steel		Continue current practice. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
010825	0	21000	Cooling Water Intake & Outfall - Structural Concrete - Concrete Slabs, Piers, Walls	Provides lake water to the Cooling Water	Good	Mechanical and Thermal Degradation / Silt build-up and zebra mussel infestation Corrosion / Chemical Attack Mechanical and Thermal Degradation / Leaching Calcium Hydroxide Corrosion / Abrasion, Erosion and Cavitation Corrosion / Corrosion of Embedded Steel		Incremental recommendations for Plant EOL (2020): Perform one-time inspection on accessible concrete structures as per U.S.NRC Regulatory Guide 1.127 Inspection of Water-Control Structures Associated with Nuclear Power Plants. Incremental recommendations for CO EOL (2024): Conduct a detailed in-situ investigation during the next station outage to check the condition of the buried concrete and evaluate its structural safety and operational adequacy.
010826	018	21000	Relief Ducts - Foundations - Concrete Footings	The foundations are designed to support the PRD piers and frames which support the main relief duct structure.	Good	Corrosion / Chemical Attack Mechanical and Thermal Degradation / Freeze-Thaw Mechanical and Thermal Degradation / Leaching Calcium Hydroxide Mechanical and Thermal Degradation / Cracking Due to Expansion or Contraction Corrosion / Corrosion of Embedded Steel		Continue current practices. No additional practices are required to reach CO EOL (2024).
010827	018	21000	Relief Ducts - Foundations, Steel	The steel H-piles are designed to support the pile caps, which in turn support the PRD piers, frames and the main relief duct.	Good	Fatigue / Fatigue due to Vibration Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
010828	018	21000	PRD - Bearing Plates & Sliding Joints	The intended function of bearing plates/sliding joints is to allow for bearing support, as well as East-West movement of the Pressure Relief Duct.	Good	Corrosion / Fouling (accumulation of deposits) Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010829	018	21000	PRD Structure and Seals	Concrete Structure: The PRD structure connects the four Pickering “B” units and four Pickering “A” units to the common VB via ‘bulkheads’. In an accident event, a large RB pressure build-up occurs and contaminants such as radioactive fission products including tritium could be released into the PRD to be transferred and contained in the VB. Seals and Sealants: The joints in the PRD box section are designed to allow differential settlement, longitudinal thermal contraction and expansion, and facilitate construction. Steel Parts: The steel parts, including embedded structural steels, clamping plates, bolts, washers, and tendons, are a portion of the joint assembly. They are designed for clamping the rubber seals to be firmly held in place without leakage.	Good	Mechanical and Thermal Degradation / Freeze-Thaw Mechanical and Thermal Degradation / Self Loosening Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Cracking Due to Expansion or Contraction Corrosion / General Corrosion Corrosion / Corrosion of Embedded Steel		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010830	014	21000	Reactor Building - Foundations, Concrete	The foundation slab serves two main purposes. Firstly, to support the loads from the interior and exterior structure of the Reactor Building. Secondly, to form part of the containment structure and provide the	Good	Corrosion / Chemical Attack Mechanical and Thermal Degradation / Freeze-Thaw		Continue current practice. No additional practices are recommended for CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				bottom of the airtight enclosure. Therefore, the RB foundation slab forms part of the containment system and is designed to prevent leakage in the event of any postulated accident in the reactor or associated systems		Mechanical and Thermal Degradation / Leaching Calcium Hydroxide Mechanical and Thermal Degradation / Cracking Due to Expansion or Contraction Corrosion / Corrosion of Embedded Steel		
010831	014	21000	Reactor Building - Seals & Sealants	Elastomeric seals and sealants provide leak tightness for air and/or water penetrations.	Good	Radiation Induced Degradation / UV exposure Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010832	014	21000	Reactor Building - Steel Liners	(1) The steel liners are designed to provide additional shielding to the maintenance personnel in the F/M Service Rooms. Each shielding window allows remote viewing of the fuelling machine during fuel handling operation. (2) On the East and West Fuelling Machine Vault Roofs The steel liners on the East and West Fuelling Machine Vault roofs, through which the embedded re-circulating cooling water piping penetrate. The two steel liners also served as formwork for concrete at the time of construction.	Good	Radiation Induced Degradation / Radiation Embrittlement Corrosion / Corrosion of Embedded Steel		Continue current practice. No additional practices are recommended to reach CO EOL (2024).

TETRA TECH

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
010833	014	21000	Reactor Building-Foundations, Steel H-Piles	The steel H-piles support the pile cap, which in turn is designed to support the entire RB structure.	Good	Corrosion / Chemical Attack Fatigue / Fatigue due to Vibration Corrosion / General Corrosion Fatigue / Environmentally-Assisted Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010834	014	21000	Reactor Building-Structural Concrete - Concrete Dome, Exterior Walls, Pressure walls, Columns, Slabs	The reactor building encloses the reactor with its fuel loading and discharge equipment, and the entire primary heat transport and moderator systems. It shields staff from radiation, and together with the vacuum building and pressure relief duct, acts as containment following accidents.	Good	Corrosion / Corrosion of Embedded Steel Mechanical and Thermal Degradation / Elevated Temperature Mechanical and Thermal Degradation / Freeze-Thaw Fatigue / High Vibrations Fatigue / Thermal Fatigue Corrosion / Chemical Attack		Continue current practices. No additional practices are recommended for CO EOL (2024).
010835	018	21000	VB/Emergency Water Tank & Spray System - Foundations, Concrete	The pile cap is designed to support the entire VB structure and serve as the floor slab of VB basement	Good	Corrosion / Chemical Attack Mechanical and Thermal Degradation / Freeze-Thaw Mechanical and Thermal Degradation / Leaching Calcium Hydroxide Mechanical and Thermal Degradation / Cracking Due to		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Expansion or Contraction Corrosion / Corrosion of Embedded Steel		
010836	018	21000	VB/Emergency Water Tank & Spray System - Foundations, Steel	The steel H-piles support the pile cap, which in turn is designed to support the entire VB structure and serves as the floor slab of the VB basement.	Good	Mechanical and Thermal Degradation / Cracking Due to Cyclic Loading Corrosion / General Corrosion		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010837	018	21000	VB/Emergency Water Tank & Spray System - Pre-Stressed Steel	The horizontal pre-stressed cables keep the ring girder in longitudinal compression and ensure that no tensile stresses can exist in the perimeter under any design condition. Crack paths are therefore closed and corrosion and leakage minimized. The vertical post-tensioned rods add compressive stresses in addition to the self-weight of the ring girder onto the upper part of the VB wall. The joint between the ring girder and the perimeter wall is therefore closed and leakage minimized. The horizontal post-tensioned rods hold the buttress walls onto the VB wall by creating a large compressive stress and resultant friction force at the interface. This friction force counteracts the self-weight of the Buttress Walls, loads from the supported piping, and seismic loads.	Good	Mechanical and Thermal Degradation / Creep and Stress Relaxation Mechanical and Thermal Degradation / Shrinkage Corrosion / Corrosion of Embedded Steel		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010838	018	21000	VB/Emergency Water Tank & Spray System -	(1) Elastomeric Roof Seals provide continuation of the containment boundary between structurally	Good	Mechanical and Thermal Degradation / Self Loosening		Continue current practices No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
			Seals & Sealants	separated concrete elements –the VB perimeter wall and the VB roof slab, that contribute to the air tightness of the Vacuum Building containment. (2) Joint Sealants provide barrier from external atmospheric elements (water, ice, dirt) for the construction joints and other penetrations.		Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion		
010839	018	21000	VB/Emergency Water Tank & Spray System - Structural Concrete	The Vacuum Building is a Negative Pressure Containment System (NPCS) structure and is designed for 0 psi absolute pressure. It also houses the self-actuating Dousing System used for post-accident pressure suppression by steam condensation. The NPCS limits radioactivity release to the environment during accidents.	Good	Mechanical and Thermal Degradation / Cracking Due to Expansion or Contraction Mechanical and Thermal Degradation / Freeze-Thaw Corrosion / Corrosion of Embedded Steel		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010840	018	21000	VB/Emergency Water Tank & Spray System-Tank	The Emergency Water Storage Tank supplies water for the Vacuum Building dousing spray system that condenses steam and reduces the Vacuum Building pressure. The Tank also provides water to the Calandria (moderator makeup) and to the bleed cooler, as necessary.	Good	Mechanical and Thermal Degradation / Leaching Calcium Hydroxide Corrosion / Corrosion of Embedded Steel		Continue current practices No additional practices are recommended to reach CO EOL (2024).
010841	0	21000 25100	SEAL, Cat 1/2	(1) Elastomeric Roof Seals provide continuation of the containment boundary between structurally separated concrete elements –the VB perimeter wall and the VB roof slab, that contribute to the air tightness of the Vacuum Building containment.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Self Loosening		Continue current practices No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				(2) Joint Sealants provide barrier from external atmospheric elements (water, ice, dirt) for the construction joints and other.		Corrosion / General Corrosion		

System 0469 – VB Emergency Water Tank & Spray System

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
010720	018	25100 63420	AMPLIFIER, Cat 1/2	Monitor the radiation activity level in the Vacuum Building exhaust.	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010721	018	25100 63420	ANALYZER, Cat 1/2	Monitor the radiation activity level in the Vacuum Building exhaust.	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010725	018	25100 63420	CONNECTOR, Cat 3/4	Connect activity monitor 018-63420-R2N-RM2 to associated devices	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010729	1,4	25100 63420	DEVICE, Cat 1/2	Radiation detectors measure the radioactivity level of Reactor Building exhaust. Check sources provide radiation sources to test the monitoring systems.	Good	Radiation Induced Degradation / Radiation Depletion of Material Properties Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Mechanical Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
010730	018	25100 63420	DETECTOR, Cat 1/2	Measure the radiation activity level in the Vacuum Building exhaust and send signals to activity monitors.	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging Radiation Induced Degradation / Radiation Embrittlement		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010735	1,4	25100 63420	FUSE, Cat 1/2	Provide electrical protection to RB activity monitoring circuits.	Good	Mechanical and Thermal Degradation / Thermal Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010740	1,4	25100 63420	INDICATOR, Cat 1/2	Provide indication of radioactivity level of Reactor Building exhaust.	Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010743	1,4	25100 63420	MODULE, Cat 1/2	Amplify signals representing radioactivity level of Reactor Building exhaust.	Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010744	1,4	25100 63420	MONITOR, Cat 1/2	Monitor the radioactivity level of Reactor Building exhaust.	Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Thermal Aging		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010745	0	25100 34220	MOTOR, Cat 1/2	Pump motors used to drive main volume vacuum and upper chamber vacuum pumps.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation	Functional Test Predictive Maintenance - Vibration Monitoring	Incremental recommendations for Plant EOL (2020): Replace Main Vacuum Pump Motors. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						/ Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Lubricant Degradation		
010748	0	25100 34220	PUMP, Cat 1	Main volume vacuum pumps, maintain sub atmospheric pressure in vacuum building main chamber during normal operation. Following a DBA, pumps are required to initially stop, and later operate in the post-accident period for FADS operations to maintain containment at sub atmospheric conditions.	Good	Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Calibration Component Replacement Inspection - Non - Intrusive Lubrication Predictive Maintenance - Vibration Monitoring SRST - Functional Test SRST - Logic Test	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete the following: 1. Complete WOs 2810540 & 2956861 to do a post-mortem inspection of 0-34220-P3 (Sent to OEM, Waiting for Report). 2. Implement the applicable ACE recommendations written for SCR P-2014-30189, and implement the recommended changes resulting from the OEM inspection. 3. Complete WOs 2898045 and 2898043 to inspect the strainer/ separator on 0-34220-P1 and 0-34220-P3 respectively. 4. Complete WO 02956864 to overhaul 0-34220-P1. 5. Complete WO 04930103. 0-34220-P3 could not shutdown at target pressure. Repair / replace as necessary. 6. Complete WO 02953941 to rebuild the spare main volume pump and return to stores. 7. Complete WO 04808854 to investigate an oily mist observed at the pump inboard bearing. 8. Complete WO 02956863 0-34220-P2. Investigate and fix oil loss issue with P2. 9. Complete WO 04773326 and install In-pro Seals on P3. 10. Progress ECR 18400 to install permanent vibration transducers in pump bearing cartridges for improved

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
								transmission of high frequency signal. 11. Complete EC 88952 and install proper oil sampling ports on the pumps. 12. Complete EC 74633 to change the seal water distribution to each of the main volume vacuum pumps. 13. Complete EC 83997 to modify the breathers on the main volume vacuum pumps (0-34220-P1, P2, P3) to prevent pump oil leakage. 14. Complete EC 107060 for 0-34220-P1 and 0-34220-P2 to update the OEM flow diagram to as built conditions. 15. Complete EC 108025 for P2 0-71620-NICR Demin. Water piping change to support 0-34220-P2 Seal water float valve replacement. Incremental recommendations for Plant EOL (2020): Implement PdM activities to perform vibration testing, infrared readings and oil analysis. Incremental recommendations for CO EOL (2024): Review OEM inspection / recommendation from WOs 2810540, 2956861 and SCR P-2014-30189, and apply to assure condition of components to CO EOL (2024).
010749	0	25100 34220	PUMP, Cat 2	These are vacuum pumps which maintain differential pressure between vacuum building upper chambers and main chamber so that dousing is not initiated during normal operation.	Satisfactory	Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Incremental recommendations for Plant EOL (2020): Implement tasks for oil analysis, vibration analysis, and thermography. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010752	018,0	25100	REGULATOR, Cat 1/2,	These are pressure regulating valves for the air	Good	Mechanical and Thermal Degradation		Incremental recommendations for Plant EOL (2020):

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
			supplying program AOVs	supply for their corresponding pneumatically operated valves.		/ Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Obtain sufficient spares valves and components on site to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Complete a one-time replacement of 0-34220-MV9-PRV1, 0-34220-MV79-PRV1, 0-34220-MV81-PRV1 to reach CO EOL (2024).
010753	018,0	25100 63421 63422	RELAY (U0, 018), Cat 1/2	Relays, for example, 018-63422-R154 thru R173, used to control vacuum pumps lockout.	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Thermal Fatigue	Calibration SRST - Functional Test	Incremental recommendations for Plant EOL (2020): Complete a proactive replacement of relays in the system that are experiencing higher failure rates. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010754	1,4	25100 63420	RELAY (U1,4), Cat 1/2	Relays used for negative pressure containment box-up logic.	Good	Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue		Incremental recommendations for Plant EOL (2020): Complete a proactive replacement of relays in the system that are experiencing higher failure rates. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
010756	4	25100 63420	SENSOR/THERMAL CONVERTER, Cat 1/2	Check source is part of the NPC RB radiation detector (4-63420-R1N-RE1) which monitors radiation dose released from the RB.	Good	Radiation Induced Degradation / Radiation Depletion of Material Properties Mechanical and Thermal Degradation / Creep and Stress Relaxation Fatigue / Mechanical Fatigue		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010776	018	25100 34220	VALVE, PNEUMATIC/ PNEUMATIC	These MVs are pneumatic pressure control valves (normally closed) to prevent	Very Good	Mechanical and Thermal Degradation / Deterioration of		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
			AC, Cat 1/2, A31A with 1052	IPRV from opening too quickly. The valves are also required for manual controlled mode operation to permit venting of the RB atmosphere under all inside containment DBA conditions that will not raise the pressure to 4.48 kPa (g)		Material (Wiring, Seals, Gaskets, O-rings etc.)		
010777	0	25100 34220	VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2, A31A or A41 with 1035 actuator	These MVs are pneumatically operated valves on suction lines for main volume vacuum pumps and upper chamber vacuum pumps. Opens when a pump starts and closes when a pump stops.	Very Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Component Diagnostics Inspection - Visual Overhaul/Refurbishment	Continue current practices. No other additional practices are recommended to reach CO EOL (2024).
010778	0	25100 34220	VALVE, PNEUMATIC/ PNEUMATIC AC, Cat 1/2, Grinell valves	These pneumatically operated valves are the vacuum manifold/atmospheric line isolation valves. Normally, MV46, 47, 48 are open and MV29, 30, 31 are closed, which connects the top of the IPRV housing to atmospheric pressure, thus closing the IPRVs. To open the IPRVs, the valves MV46, 47, 48 are closed, and the valves MV29, 30, 31 are open, which connects the top of the IPRV housing to the vacuum manifold.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
010780	018	25100 34220	VALVE, PRESSURE RELIEF, Cat 1/2	These valves provide overpressure protection for instrument air supply for 018-34220-MV49 and 018-34220-MV2005.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Continue current practices. No other additional practices are recommended to reach CO EOL (2024).
010781	1,4, 018,0	25100 34220	SOLENOID/SO LENOID	SVs for MV29/30/31 are used to control the main	Good	Mechanical and Thermal Degradation	Calibration	<u>Current initiatives that need to be completed for Plant EOL (2020) that are</u>

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
		63420 63422	OPERATED VAL, Cat 1/2	pressure relief valves of the negative pressure containment system.		/ Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Electronic Aging	Component Replacement Overhaul/Refurbishment Predictive Maintenance - Electrical Testing SRST - Functional Test	incremental to current periodic maintenance practices: Execute PMs that have not been performed in the last 10 years (e.g. PM #88954). Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
011310	018	25100	25100 VB/Emergency Water Tank & Spray System - Foundations, Concrete	The pile cap is designed to support the entire VB structure and serve as the floor slab of VB basement	Good	Corrosion / Corrosion of Embedded Steel Corrosion / Chemical Attack Mechanical and Thermal Degradation / Cracking Due to Expansion or Contraction Mechanical and Thermal Degradation / Leaching Calcium Hydroxide Mechanical and Thermal Degradation / Freeze-Thaw		Continue current practices. No other additional practices are recommended to reach CO EOL (2024).
011311	018	25100	25100 VB/Emergency Water Tank & Spray System - Foundations, Steel	The steel H–piles support the pile cap, which in turn is designed to support the entire VB structure and serves as a floor slab of the VB basement.	Good	Corrosion / General Corrosion Fatigue / Fatigue due to Vibration		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011342	018	25100	25100 VB/Emergency Water Tank & Spray System -	The Vacuum Building is a Negative Pressure Containment System (NPCS) structure and is designed for 0 psi absolute	Satisfactory	Corrosion / Corrosion of Embedded Steel		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
			Structural Concrete	pressure. It also houses the self-actuating Dousing System used for post-accident pressure suppression by steam condensation. The NPCS limits radioactivity release to the environment during accidents.		Mechanical and Thermal Degradation / Freeze-Thaw Mechanical and Thermal Degradation / Cracking Due to Expansion or Contraction		
011343	018	25100	25100 VB/Emergency Water Tank & Spray System-Tank	The Emergency Water Storage Tank supplies water for the Vacuum Building dousing spray system that condenses steam and reduces the Vacuum Building pressure. The Tank also provides water to the Calandria (moderator makeup) and to the bleed cooler, as necessary.	Good	Corrosion / Corrosion of Embedded Steel Mechanical and Thermal Degradation / Leaching Calcium Hydroxide		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011344	018	25100	25100 VB/Emergency Water Tank & Spray System - Seals & Sealants	(1) Elastomeric Roof Seals provide continuation of the containment boundary between structurally separated concrete elements –the VB perimeter wall and the VB roof slab, that contribute to the air tightness of the Vacuum Building containment. (2) Joint Sealants provide barrier from external atmospheric elements (water, ice, dirt) for the construction joints and other penetrations.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion Mechanical and Thermal Degradation / Self-Loosening		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
011345	018	25100	25100 VB/Emergency Water Tank & Spray System - Pre-Stressed Steel	The horizontal pre-stressed cables keep the ring girder in longitudinal compression and ensure that no tensile stresses can exist in the perimeter under any design condition. Crack paths are therefore closed and corrosion and leakage minimized.	Good	Mechanical and Thermal Degradation / Creep and Stress Relaxation Corrosion / Corrosion of Embedded Steel Miscellaneous / Pre-Stress Relaxation		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

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				The vertical post-tensioned rods add compressive stresses in addition to the self-weight of the ring girder onto the upper part of the VB wall. The joint between the ring girder and the perimeter wall is therefore closed and leakage minimized. The horizontal post-tensioned rods hold the buttress walls onto the VB wall by creating a large compressive stress and resultant friction force at the interface. This friction force counteracts the self-weight of the Buttress Walls, loads from the supported piping, and seismic loads.				

System 0472 – Fuelling Machine

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
009118	1,4	35000 63535	ACTUATOR, Cat 1/2	These actuators are used to open or close the catenary isolation valves (63536-FMA-MV1/MV2) and the FM D2O snout cavity valves (63536-QLD-MV1/MV2/MV3).	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Radiation Induced Degradation / Radiation Embrittlement	Surveillance	Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Reactivate PMs (#121195-01, 12297-01, 121198-01, and 121199-01) to replace the catenary isolation valve and FM D2O valves every 5 years to each CO EOL (2024).
009119	1,4	35000 63535	ACTUATOR, 90 Degree Rotation	The actuator is used to provide Fueling Machine 90 degree head rotation.	Satisfactory	Mechanical and Thermal Degradation / Lubricant Degradation Radiation Induced Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Component Replacement	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009120	1,4	35000 63535	ACTUATOR, Snout Clamp	To actuate snout clamping or un-clamping.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Radiation Induced Degradation / Radiation Embrittlement		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009123	1,4	35000	BOX, Cat 1/2	These junction boxes provide local cable and wire connections for the Fuelling Machines.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete outstanding WOs to execute a one-time replacement of JB side Marotta / Enertech connectors (ref. SCR P-2014-09798 and -11783).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Corrosion / General Corrosion Radiation Induced Degradation / Radiation Embrittlement		Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
009125	1,4	35000 35300	BRAKE, Cat 1/2	Fueling machine bridge brakes on the FM Bridge Y-drive hold it stationary when required during normal operation and for emergency stopping.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Lubricant Degradation		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009127	1,4	35000 63534 63542	BRAKE, Cat 1/2	These brakes are used on the rolling shield to keep it motionless after the drive motor has stopped.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Lubricant Degradation Mechanical and Thermal Degradation / Thermal Aging		Incremental recommendations for Plant EOL (2020): Establish PMs for rolling shield brakes to perform function test on the brake circuits and/or inspection. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
009129	1,4	35000 63536	CATENARY ASSEMBLY, Cat 1/2	Hose assemblies for supply/return of pressurized D2O to/from various parts of the Fuelling Machine.	Good	Fatigue / Mechanical Fatigue Radiation Induced Degradation / Radiation Depletion of Material Properties Radiation Induced Degradation / Radiation Embrittlement	Component Replacement Functional Test	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Correct configuration management errors in current PMs. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
009130	1,4	35000 63536	CATENARY ASSEMBLY, Cat 3/4	Catenary assembly are hose assemblies for supply/return of pressurized D2O to/from various parts of the fuelling machine.	Good	Fatigue / Mechanical Fatigue Radiation Induced Degradation / Radiation Depletion of Material Properties Radiation Induced Degradation / Radiation Embrittlement		Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009134	1,4	35000 63534	CONTROL, Cat 1/2	These controllers are used to control the X-Drive of the Fuelling Machines.	Poor	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Electronic Aging Obsolescence / Immediate Obsolescence Concern	Component Replacement Overhaul/Refurbishment	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete one-time replacement per WOs 2636180, 2636179, 2731627, and 2636257. Incremental recommendations for Plant EOL (2020): Resolve obsolescence and spare part issues. Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
009136	1,4	35000 63536	CONVERTER, Cat 1/2	These converters are used to convert electrical signals to pneumatic signals for F/M magazine pressure control valves.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Radiation Induced Degradation / Radiation Embrittlement	Surveillance	<u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Perform calibrations at a frequency consistent with approved IQ Review Maintenance Template.
009138	1,4	35000 63535	CYLINDER, Cat 1/2	NLO is a piston-operated actuator that actuates FM head rotation locking pin. NZM is a hydraulic actuator that provides FM z-motion actuation.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Radiation Induced Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Component Replacement	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete one-time replacements of 1/4-63535-NZM-A1E & W and 1/4-63535-NZM-A2E & W via WO#3163574, 3163568, 3163587 and 3163654 (scheduled for 2019) to reach Plant EOL (2020). <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Complete one-time replacements of 1/4-63535-NZM-A1E & W and 1/4-63535-NZM-A2E & W to reach CO EOL (2024).
009143	1,4	35000 35300	FUELING MACHINE, Cat 1/2	The Fuelling Machine pressure vessel contains and maintains the pressure of D2O within the Fuelling Machine.	Satisfactory	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Creep and Stress Relaxation	Functional Test Inspection - Set point checks Inspection - Cable Supports	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Resolve outstanding spare parts issues for Fuelling Machines, sufficient spares should be on hand to complete outstanding WOs for replacement and support maintenance. <u>Incremental recommendations for Plant EOL (2020):</u>

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Cracking Due to Cyclic Loading Mechanical and Thermal Degradation / Lubricant Degradation Mechanical and Thermal Degradation / Thermal Aging Obsolescence / Immediate Obsolescence Concern		No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Update recent fatigue analysis for Fuelling Machine Pressure Boundary (i.e. NA44-CALC-35310-00002 R001), to account for life extension to CO EOL (2024).
009148	1,4	35000 63535 63536	HOSE, Cat 1/2	Catenary hoses supply oil / D2O for Fueling Machine. Line 1, 2, 32, 47, 95, 106, 224 contain oil, and Line 5, 11, 12, 37, 132 contain D2O.	Good	Fatigue / Mechanical Fatigue Radiation Induced Degradation / Radiation Depletion of Material Properties Radiation Induced Degradation / Radiation Embrittlement	Component Replacement Inspection - Visual	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009149	1,4	35000 63535	HOSE, Cat 3/4	Catenary hoses supply oil for Fuelling machine.	Good	Fatigue / Mechanical Fatigue Radiation Induced Degradation / Radiation Depletion of Material Properties Radiation Induced Degradation /		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Radiation Embrittlement		
009155	1,4	35000 63535	METER, Cat 1/2	The servo tachometers indicate bi-directional X-motion motor speed.	Good	Radiation Induced Degradation / Radiation Depletion of Material Properties Mechanical and Thermal Degradation / Electronic Aging Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Component Replacement Functional Test	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete one-time replacement of the x-drive train (WO # 2416645, 4940414, 4940416, 2676831) to restore condition to reach Plant EOL (2020). <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
009161	1,4	35000 63535	MOTOR, HYDRAULIC, Cat 1	Oil Hydraulic Motors for driving various parts of the Fuelling Machine: NBR - B Ram Motor NLA - Latch Ram Motor NMA - Magazine Rotation Drive Motor NXM - X-Drive Motor NYC - Y Correction Motor NYM - Y-Drive Motor	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Radiation Induced Degradation / Radiation Embrittlement Radiation Induced Degradation / Radiation Depletion of Material Properties	Functional Test Inspection - Visual	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete a one-time replacement of the x-drive motors, including WOs 4940414, 4940416, 2416645 and 2676831. <u>Incremental recommendations for Plant EOL (2020):</u> Complete a one-time replacement of Magazine motors (NMA) upon resolution of vendor quality issue with subject bent axis motors. <u>Incremental recommendations for CO EOL (2024):</u> 1. Complete a one-time replacement of the X-Drive Motors (NXM). 2. Complete a one-time replacement of the Magazine motors (NMA).
009167	1,4	35000 63534	POTENTIOMETER, Cat 1/2	These potentiometers are used to provide resistance signals representing the positions of the Y Drive of the Fuelling Machines.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation		<u>Incremental recommendations for Plant EOL (2020):</u> Resolve obsolescence and spares issues associated with 1/4-63534-NYM-POTE/POTW.

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						/ Wear Mechanisms - Wear Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Radiation Induced Degradation / Radiation Embrittlement		<u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
009176	1,4	35000 63536	REGULATOR, Cat 1/2	The valves in the CG are air pressure regulating valves that regulate air pressure supply to PMA-CV1 (the Magazine D2O Supply Pressure Control Valve).	Poor	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Radiation Induced Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / General Corrosion Fatigue / Fatigue due to Vibration	Valve Diagnostics	<u>Incremental recommendations for Plant EOL (2020):</u> Complete a one-time replacement of all PRVs in this CG to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Complete a one-time replacement of all PRVs in this CG to reach CO EOL (2024).
009181	1,4	35000 35335	SHIELD, Cat 1/2	The rolling shields are cantilevered concrete shields which can be moved over the vault floor bridge opening. They provide additional shielding in the service rooms. The rolling shields are not accessible during unit operation. The shield is required to retract away from the FM	Satisfactory	Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / General Corrosion		<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> 1. Complete one-time replacement of 4-63542-NRS-LS149E/LS149W/LS150E/LS50W per WO#2457639 and WO#2457650. 2. Replace rolling shield chains, mitre boxes and pillow blocks

TETRA TECH

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				vault floor opening, to allow the FM bridge to ascend into the vault. The rolling shield is driven by an electric motor via shafts and couplings. The motor is accessible and is located on the ceiling just above the FM service room door.		Mechanical and Thermal Degradation / Lubricant Degradation Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Thermal Aging		under the FM Bridges Reliability Improvement Project. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
009183	1,4	35000 63534	SOLENOID, Cat 1/2	These solenoid valves are used to actuate/de-actuate the safety latch mechanism for the F/M separators.	Satisfactory	Mechanical and Thermal Degradation / Thermal Aging Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Radiation Induced Degradation / Radiation Embrittlement	Functional Test Overhaul/Refurbishment	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009195	1,4	35000 63535	VALVE, CONTROL, Cat 1/2	The valves in the CG are flow control valves: SLA-CV2 - Latch Ram Speed Control, SRO-CV1/2 - Control speed of FM 90 degree head rotation, SYC-CV1/2 - Control speed of Y-Correction (fine Y movement), SZM-CV2/3 - Control speed of Z-motion.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Radiation Induced Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Calibration Functional Test Inspection - Visual	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> 1. Complete one-time replacements of Latch Ram and Z-Drive valves not completed via WO 2260381, 2260387, 2260389, 2392149, and 2392150. 2. Complete ECs #106679 to 106686 to replace the 90-degree rotation control valves. <u>Incremental recommendations for Plant EOL (2020):</u>

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
								1. Perform a one-time replacement of the Latch Ram Speed valves (SLA). 2. Complete a one-time replacement of the Z-Drive speed valves (SZM). 3. Complete a one-time replacement of the Y-Correction Speed valves (SYC). 4. Complete a one-time replacement of the FM 90 degree rotation valves (SRO). Incremental recommendations for CO EOL (2024): 1. Complete a one-time replacement of the Z-Drive speed valves (SZM). 2. Complete a one-time replacement of the Y-Correction Speed valves (SYC). 3. Complete a one-time replacement of the FM 90 valves (SRO).
009197	1,4	35000 63536	VALVE, MAG PRESSURE CONTROL, Crit 1	These control valves provide FM magazine D2O Supply and close following withdrawal of the sealing plugs and hydraulic integration of the fuelling machine with the reactor.	Satisfactory	Mechanical and Thermal Degradation / Electronic Aging Radiation Induced Degradation / Radiation Depletion of Material Properties Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Valve Diagnostics	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete WOs 04803924, 4948548 and 4832604 for 1/4-63536-PMA-CV1E/W, REPLACE/OVERHAUL VALVE. Incremental recommendations for Plant EOL (2020): Complete one-time replacement for all valves (except 1/4-63536-PMA-CV1E/W) to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): Complete a one-time replacement of all valves to reach CO EOL (2024).
009198	1,4	35000	VALVE, CONTROL, Cat 1/2	The valves in the CG are counterbalance valves for the Fuelling Machine Bridge. The valves provide bridge	Satisfactory	Mechanical and Thermal Degradation / Deterioration of Material (Wiring,		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				weight counterbalancing and also braking force to stop the bridge motion on descent.		Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Corrosion / Fouling (accumulation of deposits)		Complete WOs for replacing Y-drive counter balance valves (WO 2836421, 2836749, 2836755, 2836849, 2836762, 2836722, 4838684, and 4838685). Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
009200	1,4	35000 63535 63536	VALVE, HYDRAULIC, Cat 1/2	Fuelling Machine and FM Bridge directional and isolation valves. FMA-MVs close automatically to isolate the magazine D2O return line in the event of failure of the hose from the FM magazine. NBR-MVs are directional valves for B-ram. NYC-MVs (Y-Correction Directional valves) control the direction of the Y-correction drives. QLD-MVs (FM Snout valves) open to equalize pressure between the magazine and the snout cavity, and on both sides of the leak detector.	Poor	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Radiation Induced Degradation / Radiation Embrittlement Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.)	Surveillance	Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: Complete WOs for snout valve replacement (QLD-MVs): WO 3006097 (U1E) ACTIVE WO 2917819 (U1W) ACTIVE WO 4822459 (U4E) ACTIVE WO 4731348 (U4W) ACTIVE. Incremental recommendations for Plant EOL (2020): No additional practices are recommended to reach Plant EOL (2020). Incremental recommendations for CO EOL (2024): No additional practices are recommended to reach CO EOL (2024).
009203	1,4	35000 63535 63536	VALVE, MANUAL/HAND OPERATED, Cat 1/2	The valves are manual control valves. FFE-V14, 15 are needle valves to adjust operation speed of actuators. FRE-V11, 12 are retractor speed valves. FST-V11, 12 are fuel stop speed valves.	Poor	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Radiation Induced Degradation / Radiation Embrittlement		Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices: 1. Complete a one-time replacement of B-ram and Latch Ram cross connect valves. 2. Complete a one-time replacement of 1/4-63535-NBR-

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				NBR-V3 are needle valves to allow manual actuation of the B-ram. NLA-V3 are needle valves to allow manual actuation of the latch ram.		Corrosion / General Corrosion Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Fatigue / Fatigue due to Vibration		V3E/W and 1/4-63535-NLA-V3E/W. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Complete a one-time replacement of 1/4-63535-NBR-V3E/W and 1/4-63535-NLA-V3E/W.
009206	1,4	35000 63535 63536	VALVE, CHECK/NONRETURN/BACKFL, Cat 1/2	Check valves in this CG are 1/4-63536-FMA-NV1E/W, 1/4-63535-NBR-NV2E/W, 1/4-63535-PHS-NV10E/W, 1/4-63535-PHS-NV4E/W, 1/4-63535-PHS-NV5E/W and 1/4-63535-PHS-NV9E/W. Their functions are to prevent flow reversal for magazine oil supply, B-ram, carriage drive, head oil supply manifold, carriage oil supply.	Satisfactory	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Radiation Induced Degradation / Radiation Embrittlement Corrosion / General Corrosion Fatigue / Mechanical Fatigue Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Corrosion / Fouling (accumulation of deposits)	Component Replacement Overhaul/Refurbishment	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete WOs: 2106863, 1972285, 2362488, 2362478 to replace Magazine return check valves (FMA-NV2). <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
009208	1,4	35000 63536	VALVE, PNEUMATIC/PNEUMATIC AC, Cat 1/2	QLD-MV3 is a F/M snout bleed valve (oil actuated valve).	Satisfactory	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation	Surveillance	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Replace 1/4-63536-QLD-MV3E/MV3W via NICR 131568 and associated WOs (e.g. WOs 2451453, 2451451) and monitor performance to determine replacement

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						/ Wear Mechanisms - Wear Radiation Induced Degradation / Radiation Embrittlement		frequency based on vendor's expected design life and operating duty. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
009209	1,4	35000 63535	VALVE, PRESSURE REGULATING, Cat 1/2	The valves in the CG are pressure regulating valves for the Fuelling Machine oil hydraulic controls: PBR-PRVs controls supply pressure to the B Ram hydraulic motor, PHS-PRVs limits supply pressure for the hydraulic oil supply to the carriage, PLA-PRVs limits the Latch Ram oil hydraulic system pressure, PMA-PRVs limits magazine drive system pressure, PSC-PRVs limits snout clamp actuating system oil hydraulic pressure, PZM-PRVs limits pressure of the Z-motion system, 1/4-63536 -PSE-PRVE/W - regulates separator D2O pressure.	Satisfactory	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue	Component Replacement Functional Test Inspection - Internal Inspection - Visual	<u>Incremental recommendations for Plant EOL (2020):</u> Complete a one-time replacement of 1/4-63535-PHS-PRV4E/W, 1/4-63536-PSE-PRVE/W and 1/4-63535-PZM-PRVE/W and monitor performance to determine replacement frequency based on vendor's expected design life and operating duty. <u>Incremental recommendations for CO EOL (2024):</u> No additional practices are recommended to reach CO EOL (2024).
009211	1,2,4	35000 63535 63536	VALVE, PRESSURE RELIEF, Cat 1/2	The valves are differential pressure relief valves: PCR-RVs prevents build-up of excessive ram force, PHS-RVs protects hydraulic system for carriage actuators, PMA-RVs are for over-pressure protection for the FM head.	Satisfactory	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear	Component Replacement Inspection - Visual	<u>Incremental recommendations for Plant EOL (2020):</u> Complete a one-time replacement of 1/4-63536-PCR-RV2/3/E/W to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Complete a one-time replacement of 1/4-63536-PCR-RV2/3/E/W to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
						Corrosion / General Corrosion Fatigue / Mechanical Fatigue		
009213	1,4	35000 63535	VALVE, SPEED CONTROL, Cat 1/2	The valves in the CG (1/4-63535-SMA-CVE & W) are magazine drive rotation speed control valves. This valve is used to meter-out the hydraulic flow of the magazine motor.	Satisfactory	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Radiation Induced Degradation / Radiation Depletion of Material Properties	Functional Test Inspection - Visual Overhaul/Refurbishment	<u>Incremental recommendations for Plant EOL (2020):</u> Complete a one-time replacement of all valves in this CG to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Complete a one-time replacement of all valves in this CG to reach CO EOL (2024).
009215	1,4	35000 63535	SOLENOID/SOLENOID OPERATED VAL, Cat 1/2	These solenoid valves are used for various FM control functions such as locking the FM head rotational direction. 63535-NBR-SV3 Selects the desired direction of the B-ram. 63535-NLA-SV3 Selects the desired direction of the Latch ram. 63535-NRO-SV1 serves to actuate the rotation mechanism from stop to stop of the 90° Rotation Motion Drive. 63535-NLO-SV2 selects the position of the locking pin for the FM head rotation. 63535-NMA-SV Selects Magazine rotation direction. 63535-NZM-SV1 Selects Z-motion “advance” and “retract”. 63535-NYC-SV The valve actuates slave valve NYC-	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Radiation Induced Degradation / Radiation Embrittlement Radiation Induced Degradation / Radiation Depletion of Material Properties Corrosion / General Corrosion	Calibration Component Replacement Functional Test	<u>Current initiatives that need to be completed for Plant EOL (2020) that are incremental to current periodic maintenance practices:</u> Complete currently in-progress NICR (e.g. EC #116372) to address availability of spare parts. <u>Incremental recommendations for Plant EOL (2020):</u> No additional practices are recommended to reach Plant EOL (2020). <u>Incremental recommendations for CO EOL (2024):</u> Reactivate PMs to replace NMA-SV1, NSC-SV, NYC-SV2 and NZM-SV1 every 5 years to reach CO EOL (2024).

CG:	Units	USI#	CA Description	Function	Classification	Potential Aging Degradation Mechanisms	Current Practices	Incremental Work Recommended for Current and Extended Operating Life
				MV for the 'Y' correction drive 63535-NSC-SV control the direction of the supply flow to the Snout Clamp.				
009217	1,4	35000 63535	VALVE, SAFETY RELIEF/PRESSURE, Cat 1/2	1/4-63535-PSC-RVE & W are snout clamp safety relief valves on the east and west FM heads.	Good	Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) Mechanical and Thermal Degradation / Wear Mechanisms - Wear Fatigue / Mechanical Fatigue Radiation Induced Degradation / Radiation Depletion of Material Properties	Overhaul/Refurbishment	Continue current practices. No additional practices are recommended to reach CO EOL (2024).
009218	1,4	35000 63535	VALVE, SERVO, Cat 1/2	1/4-63535-NXM-SRV1E & W are X drive motion servo control valves for the east and west fueling machines.	Good	Mechanical and Thermal Degradation / Wear Mechanisms - Wear Mechanical and Thermal Degradation / Wear Mechanisms - Erosion Radiation Induced Degradation / Radiation Embrittlement Radiation Induced Degradation / Radiation Depletion of Material Properties		Continue current practices. No additional practices are recommended to reach CO EOL (2024).

APPENDIX C. OPG'S INTEGRATED AGING MANAGEMENT PROGRAM

This Appendix describes OPG's Integrated Aging Management (IAM) Program and how it interfaces with and supports other OPG programs including the Equipment Reliability (ER) program. It provides additional context on how the actual condition of SSCs is assessed and documented. OPG's IAM program is documented in N-PROG-MP-0008, "Integrated Aging Management" [34].

The objective of the Integrated Aging Management (IAM) Program is to ensure the condition of critical Nuclear Power Plant (NPP) equipment is understood and that required activities are in place to ensure the health of these components and systems while the plant ages. This is accomplished by establishing an integrated set of programs and activities which ensure performance requirements of all critical station equipment are met on an ongoing basis. The program also requires preparation of life cycle plans and *condition assessments* for critical plant equipment.

The main activities associated with the program are the preparation of major component Life Cycle Management Plans (LCMPs) for fuel channels, feeder piping, steam generators and reactor structures and components, and condition assessments for the balance of critical plant equipment. However, successful implementation of the program is also dependent on many other station programs to ensure plant equipment is maintained in the required condition. Some of the most important supporting programs are: Component Programs, the Maintenance Program and the Equipment Reliability (ER) program (which includes System Performance Monitoring).

The Aging Management Program also defines the method of condition assessment for plant SSCs. Condition assessments are not required for all SSCs. As per Appendix A in N-PROG-MP-0008 [34], condition assessment is managed by either LCMPs, Condition Assessments (CAs), and/or by System and Component Surveillance. Not all plant SSCs require an assessment of condition to be documented in a Detailed CA. Only those SSCs not screened out via the Aging Management process require a detailed condition assessment. The condition of the balance of SSCs is addressed by System and Component Surveillance and other programs. This is elaborated on further below.

OPG's Integrated Aging Management Program is aligned with International Standards, including IAEA Safety Guide No. NS-G-2.12, "Ageing Management for Nuclear Power Plants" [99]. This Reference identifies nine attributes of an effective aging management program including; Scoping of the AM program, and detection of aging effects and mitigating aging effects. As per Appendix C in OPG's program document, N-PROG-MP-008 [34], these nine attributes form an integral part of the program from which the condition assessment process is defined.

OPG's Equipment Reliability (ER) program, N-PROG-MA-0026, 'Equipment Reliability" [80], is based on INPO AP-913 standard [10] on preparation and implementation of an ER program. An effective ER program consists of the following six main areas:

- Scoping and Identification of Critical Components
- Performance Monitoring
- Corrective Action
- Continuing Equipment Reliability
- Long Term Planning and Life Cycle Management
- Preventative Maintenance Implementation

OPG's Integrated Aging Management addresses the Long Term Planning and Life Cycle Management element in AP913 [10]. A flowchart showing how the six AP913 elements are integrated in OPG's Equipment Reliability program is provided in Figure C.1. The figure also includes other programs and procedures which support ER, including Obsolescence Management, Safety and Regulatory, Modifications, and Periodic Safety Review. The inputs and outputs of the Aging Management program are also shown. One of the AM program outputs are AM actions which are input into System and Component Health Reports. These aging management actions are prioritized and tracked to completion via the System Health Reporting process.

The programs and processes contained in Figure C.1 are also listed in Table C.1 along with their numbers used in the figures.

The preparation of Condition Assessments (CAs) is one of the main activities in the IAM program. N-PROC-MP-0060, "Aging Management Process" [7] outlines the systematic process for conducting CAs of Aging Critical Plant Systems, Structures and Components (SSCs). The process consists of:

1. Scoping
2. Screening
3. Condition Assessment

Scoping:

N-PROC-MP-0060 describes the detailed steps followed to derive the scope of OPG's Aging Management Program. These steps define the boundary of System, Structures and Components (SSCs) that are considered within the program. The first step in the process is to define the systems to be addressed. Important station systems, both safety related and production critical are included and their boundaries are defined. Components within these systems are included in scope, using the Master Equipment List (MEL) as a basis. Components not addressed in the MEL, e.g. structures, some piping, etc., are then considered and those required to support a safety function or could negatively impact on a safety function if they fail are included in scope. Systems supporting these important systems are also included.

Components within these systems having similar attributes, e.g. criticality, type, subject to the same degradation mechanisms are then collected into Commodity Groups (CGs). The next step in the process is to screen the in-scope SSCs to determine if they should be subject to detailed condition assessment.

Screening:

The objective of the aging management screening process is to review the large number of SSCs (greater than 500,000 for Pickering) in the scope of the program and by employing a systematic process using defined criteria, determine which SSCs should be subject to detailed condition assessment. Per OPG Aging Management governance, the condition of screened-out SSCs does not require an in-depth condition assessment, but rather, their condition is assessed and managed by other processes in the AM program, e.g. Equipment Reliability program including system performance monitoring.
The screening process is illustrated in a flowchart shown in Figure C.2.

The first step in the screening process is to screen out non-critical components. Component Criticality is defined in N-PROC-MA-0077, “OPG Procedure ‘Critical Equipment Identification and Categorization” [9]. In this procedure, components are assigned critical categories CC1, 2, 3 or 4. Criticality sub-codes are also assigned in the areas of Reactor Safety (RS), Production (P) and Cost, Conventional Safety, Environmental (CCSE). CC1 and 2 components are “critical” components and CC3 and 4 are “non-critical”.

CC3 and 4 components are screened out from further assessment. For PSR2, CC3/RS3, i.e. having a lower level of importance in safety related function than RS1 or RS2, e.g. a component used in a safety related test, are not screened-out. Other non-critical components can also be added to the scope as requested by the system engineer. Per Reference [7], Critical Structures (e.g. Reactor Buildings, Pressure Relief Duct and Vacuum Building) are also addressed in the review.

A preliminary assessment of all components screened in to this step is performed, which collects pertinent information needed to conduct further screening. The information collected for each SSC is (i) Aging Related Degradation Mechanisms (ARDMs), i.e. modes or processes resulting in degradation of the component, e.g. corrosion and (ii) Aging Management Practices (AMPs), i.e. the methods in place to detect and manage the component’s aging.

Screening is continued based on this preliminary assessment. The next step in the processes screens out non-AM critical SSCs. Non-AM critical components are those which:

- There is assurance of design life over the length of station operation or extended operation, e.g. some structures.
- There is low risk associated with component degradation, i.e. low cost having limited and simple ARDMs with effective regular practices to detect degradation. Typical components falling into this category are hand switches, push-buttons used for testing, etc.

The next step in the process, reviews the practices used to mitigate age related degradation for a component. Practices can consist a set of programmatic requirements, e.g. the HX program, or can be defined by maintenance strategies or preventative maintenance or other tasks. A set of practices is considered effective if it addresses all component ARDMs and it has proven to be effective, is aligned with industry best practices, or other criteria documented in N-PROC-MP-0060 [7].

Upon completion of this last step in the screening process, the set of remaining non-screened out SSCs are subject to detailed condition assessment. All of the information used in the screening process is documented for each system in a System Screening Report and stored in Asset Suite as a controlled document.

Before describing the condition assessment process used for screened-in components, a description is provided of the method of managing aging used for screened-out components, i.e. not requiring a Detailed CA.

Treatment of SSCs not Requiring a Detailed CA:

The condition of screened out components is managed on an ongoing basis via OPG's Equipment Reliability (ER) program and other supporting programs. OPG's ER program is aligned with best industry standards and is comprised of a set of processes whose objective is to ensure that the reliability of systems and components is managed on an ongoing basis, including ensuring that all nuclear safety requirements are met.

Example Supporting Equipment Reliability processes are:

(i) The Corrective Action Program, "Corrective Action", N-PROG-RA-003 [92], executed to identify adverse trends in performance or component failures and put corrective actions in place to prevent re-occurrence of the adverse condition;
(ii) System Performance Monitoring, "System Performance Monitoring", N-PROC-MA-0024 [93] which requires surveillance, tracking, reporting on overall health and preparation of System Health Action Plans to improve system health and component condition;
(iii) The Component and Equipment Surveillance Program, N-PROG-MA-0017 [87], which addresses a number of different types of components, e.g. Power Operated Valves, Buried Piping, Cables, Heat Exchangers, etc. and;
(iv) The Preventative Maintenance Program, "Conduct of Maintenance", N-PROG-MA-0004 [94] which uses component operating history to optimize component performance and maintenance practices via PM feedback mechanisms and

conducts the required maintenance on components. Work reports document the observed condition of equipment subjected to maintenance.

Ongoing assessment, monitoring and the documenting information on the condition of station systems and components is conducted per the ER program.

With respect to the documentation of the condition of screened-out components, the objective of screening is not to assign a condition or classification for screened out components. As per N-PROC-MP-0060 [7], this is only required for components for which a detailed condition assessment is performed. However, during the preliminary assessment of component aging used in the screening process, the condition of AM-critical components is reviewed based on operating history, system health reports and other data sources. This information is documented in the System Screening Reports.

In addition, a number of comprehensive programs are in place which document component condition including: System and Component Health Reporting; The Maintenance Program which documents as-found condition of components; Predictive Maintenance, e.g. vibration monitoring; the Corrective Action Program, documenting adverse conditions on equipment in SCRs; Annual Reliability Reports; Design Assessments and many other station processes.

Detailed Condition Assessment:

Aging Critical components not screened out are subject to detailed condition assessment.

Detailed Condition Assessments (CAs) involve:

- Identifying and understanding component degradation mechanisms.
- Collecting data to evaluate the degree of degradation experienced to date, e.g. SHR data, OPEX, SCRs, etc.
- Evaluating component condition by comparing experienced degradation against established limits.
- Establishing actions required to minimize and control Aging Related Degradation Mechanisms (ARDMs) and improving condition.

All of the information above is documented in a Detailed CA. A sample Detailed CA is attached to this Appendix in Attachment C.1. This sample shows all of the required fields to be completed to derive overall condition. Once all of the required information is collected, an assessment of classification is performed. An overall Condition Classification (Very Good, Good, Satisfactory, Poor, and Very Poor) is defined in N-PROC-MP-0060 [7], which accounts for:

- The physical condition of the component at the time of assessment, and
- The adequacy of the practices in place to manage component aging.

Condition Classification is assigned by selecting the lesser of these two criteria. For example if the physical condition of a component is “Good”, but the adequacy of practices is “Satisfactory”, the component condition classification is “Satisfactory”. The definitions of the classifications are provided in Table C.2.

Improvement actions are required for components with a “Poor” or “Very Poor” classification. In many cases, actions are also provided for “Satisfactory” or “Good” components to maintain or improve the classification.

AM actions are prioritized and captured in system health reports (SHRs). These actions are assessed on an on-going basis and tracked to completion in SHR action plans. Detailed CAs are maintained and updated on a periodic basis to capture new information relevant to aging.

APPENDIX C.1 SUPPORTING TABLES AND FIGURES

Figure C.1 OPG-N Equipment Reliability Process Flow (Including Relationship to Periodic Safety Review) - P-REF-09710-0617392 (Modified)

OPG-N Equipment Reliability Process Flow (including Relationship to Periodic Safety Review) Modified – P-REF-09710-0617392

Note: This figure is not governance. It is for illustrative purposes only.

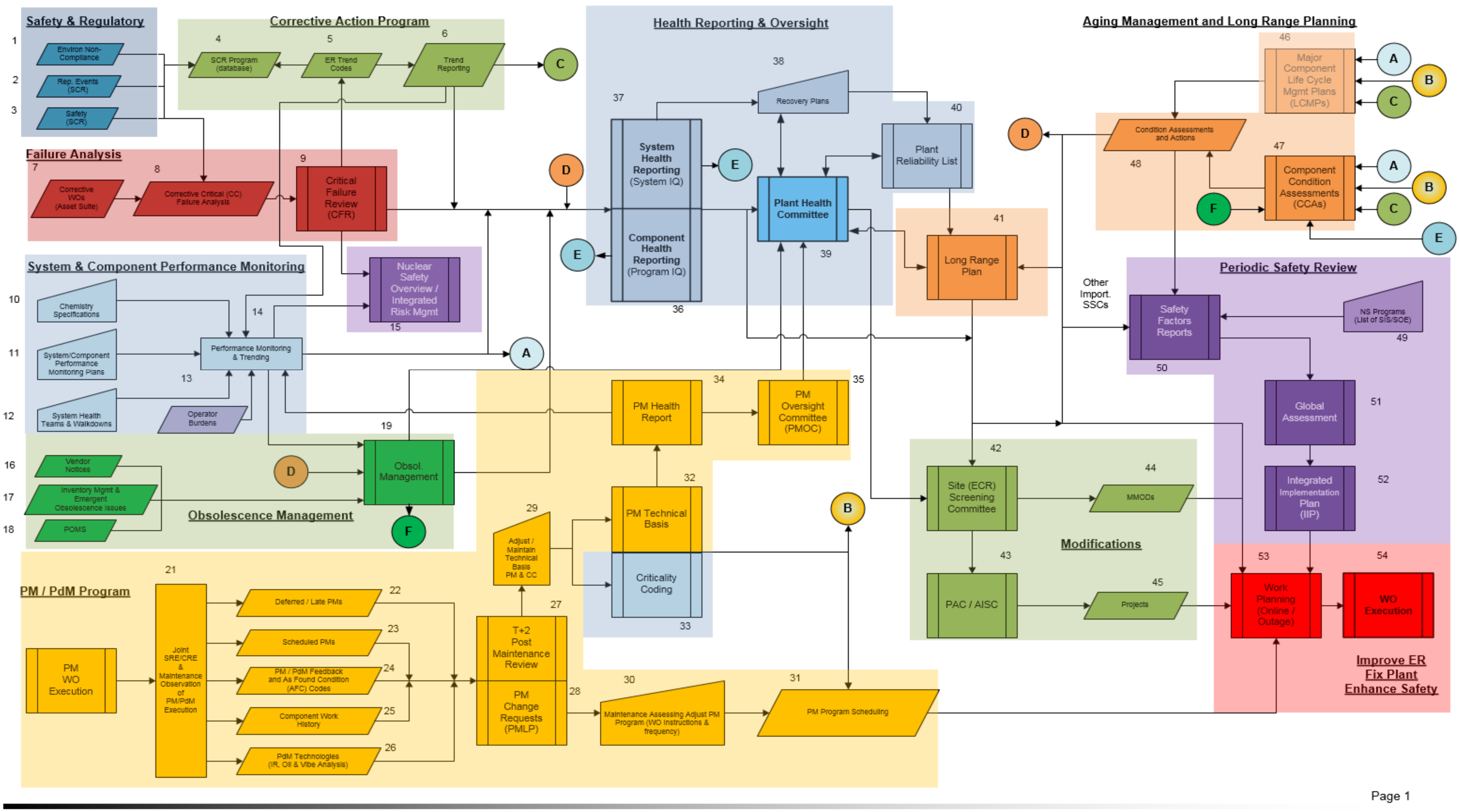

Table C.1 OPG-N Equipment Reliability Process Flow References – Associated with P-REF-09710-0617392

Box #	Box Title	Program/ Procedure Document Number	Program/ Procedure Document Name	Sample Documentation	Output
Safety & Regulatory					
1	Environmental Non-Compliance	N-PROG-RA-0002 N-PROG-OP-0006 OPG-PROG-0005 OPG-PROC-0042	Conduct Of Regulatory Affairs Environmental Management Environmental Management System Environmental Nonconformity, Corrective and Preventive Action	SCR No.: P-2015-05300 (Enviro. Non-Compliance SCR)	SCR Barrier analysis worksheet Interim actions
2	Reportable Events (SCR)	N-PROG-RA-0002 N-PROC-RA-0022 N-PROC-RA-0020	Conduct Of Regulatory Affairs Processing Station Condition Records Preliminary Event Notification	SCR No.: P-2014-25205 (Reportable Event SCR)	SCR Barrier analysis worksheet Interim actions
3	Safety (SCR)	N-PROG-RA-0002 N-PROC-RA-0022 N-INS-08965-10016	Conduct Of Regulatory Affairs Processing Station Condition Records Safety Hazard and Worker Safety Concern Resolution	SCR No.: P-2015-05169 (Safety SCR)	SCR Interim actions
Corrective Action Program					
4	SCR Program (database)	N-PROG-RA-0003 N-PROC-RA-0022	Corrective Action Processing Station Condition Records	See sample SCRs above.	Root Cause analysis Apparent cause analysis Corrective action plan Interim actions
5	ER Trend Codes	N-PROG-RA-0003 N-LIST-01966-10001	Corrective Action Trend Codes Applied To Station Condition Records	N-FORM-11318 N-PROC-MA-0097 Appendix C (ER Trend Codes)	Trend report

Box #	Box Title	Program/ Procedure Document Number	Program/ Procedure Document Name	Sample Documentation	Output
6	Trend Reporting	N-PROG-RA-0003 N-INS-01966.1-10000	Corrective Action Trending And Analysis Instruction And Performance Improvement Reporting	Bubble Chart Reports: P-REP-01966-0586702 ,2015 Q4 Equipment Failure Cognitive Trend Report P-REP-01966-0594738, 2016 Q1 Equipment Failure Cognitive Trend Report	Failure trends Recommendations
Failure Analysis					
7	Corrective WOs (Asset Suite)	N-PROG-RA-0003	Corrective Action	Excel List of CFR Items Related to P058 SG's	Scope of work
8	Corrective Critical (CC) Failure Analysis	N-PROC-MA-0097 N-PROG-MA-0026	Equipment Reliability Implementation Equipment Reliability Program	Excel List of CFR Items Related to P058 SG's	Apparent cause analysis
9	Critical Failure Review (CFR)			Excel List of CFR Items Related to P058 SG's	Corrective action plan Recurrence control action Trending
System and Component Performance Monitoring					
10	Chemistry Specifications	N-PROC-MA-0024 N-PROG-OP-0004 N-PROG-MA-0026	System Performance Monitoring Chemistry Program Equipment Reliability Program	N-REP-01824-00001 Section 1: Recommended Fuel Specification for Standby and Emergency Power Generators N-REP-01824-00001 Section 2: Recommended Chemistry for EPG and SG New Fuel Oil	Chemistry specifications System chemistry analysis Chemistry control recommendations

Box #	Box Title	Program/ Procedure Document Number	Program/ Procedure Document Name	Sample Documentation	Output
11	System/Component Performance Monitoring Plans	N-PROC-MA-0024 N-PROG-MA-0017 P-ESI-04610-00001-00001 P-ESI-05600-00001/00002/00003/00004 P-MAN-04940-00001 P-MAN-04940-00002 P-MAN-04946-00001 P-MAN-04660-10001 P-MAN-04916-00001	System Performance Monitoring Component and Equipment Surveillance Pump Strategy Motor Strategy AOV Strategy MOV Strategy NV Strategy HX Strategy Buried Piping	P-SPM-54600-0432570 (Standby Generator SPMP)	System Description Degradation mechanism Performance goals Functional failures
12	System Health Teams & Walkdowns	N-PROC-MA-0024	System Performance Monitoring	NK30-OP-54600-0027 (SG Daily Status Check) P-FORM-20196 , SG Performance Readings Field Walkdown Technical Surveillance – Standby Generators (See Appendix A of P-SPM-54600-0432570, SG SPMP)	Field walk down results OPS & Maint. feedback
13	Operator Burdens	N-PROC-MA-0024 N-STD-OP-0020 N-PROC-OP-0041 N-PROC-OP-0027	System Performance Monitoring Rounds And Routines Control of Operator Challenges Temporary Change Records	N-FORM-10540 TCR (Temporary Change Record) P-A-CMP-76100.01 (Routine Maintenance of the Boiler Room Cranes)	Operator burden description Operator Burden frequency Temporary alteration

Box #	Box Title	Program/ Procedure Document Number	Program/ Procedure Document Name	Sample Documentation	Output
14	Performance Monitoring & Trending	N-PROC-MA-0024 N-PROG-MA-0017 N-PROC-MA-0077	System Performance Monitoring Component and Equipment Surveillance Critical Equipment Identification And Categorization	Performance Monitoring Equipment List (Available in Section 4.0 of SG SPMP)	Updated PMEL Failure Trends and analysis Changes in criticality Reliability and maintainability assessments Performance reporting
15	Nuclear Safety Overview/ Integrated Risk Management	N-PROG-RA-0016	Risk and Reliability Program	Nuclear Safety Overview Presentation	Plant nuclear safety risk Probabilistic Safety Assessment (PSA)
Obsolescence Management					
16	Vendor Notices	N-STD-MA-0024	Obsolescence Management	POMS and RAPID data base	Vendor update Information (e.g. going out of business)
17	Inventory Management	N-STD-MA-0024 OPG-PROG-MM-0009	Obsolescence Management Items and Services Management	POMS and RAPID data base	Product availability in warehouse/store
18	POMS	N-STD-MA-0024	Obsolescence Management	POMS and RAPID data base	Obsolete equipment and their ranking
19	Obsolescence Management			POMS and RAPID data base	Product Obsolescence Information Corrective action plan
PM/ PdM Program					
20	PM WO Execution	N-PROC-MA-0026 N-PROC-MA-0034	Preventive Maintenance Technical Specifications Predictive Maintenance Program Requirements	Work Report Summary for Successful completion of PMID's: 18662-06 (PM) and 76368-04 (PdM)	Equipment information Scope of PM PM Frequency PM Technical basis Criticality code

Box #	Box Title	Program/ Procedure Document Number	Program/ Procedure Document Name	Sample Documentation	Output
21	Joint SRE/CRE & Maintenance Observation of PM/PdM Execution	N-PROC-MA-0026 N-PROC-MA-0034 N-PROG-MA-0026	Preventive Maintenance Technical Specifications Predictive Maintenance Program Requirements Equipment Reliability Program	N/A System Engineer Perspective: Any Preventative Maintenance Work that Work Management wants to defer is evaluated via the PMLP program by issuing a deferral.	
22	Deferred/Late PMs			PM Deferral for PM activity (18662-06)	
23	Scheduled PMs			N/A Done automatically by Asset Suite	
24	PM/PdM Feedback and As Found Condition (AFC) Codes			PM Feedback Summary (18662-06) As Found Condition Presentation	
25	Component Work History			Asset Suite Work Order History (18662-06)	
26	PdM Technologies (IR, Oil & Vibe Analysis)			Screenshot of PMLP Main Screen for SG Vibration Monitoring Predictive Maintenance Predefined (76368-04)	
27	T+2 Post Maintenance Review			N/A	

Box #	Box Title	Program/ Procedure Document Number	Program/ Procedure Document Name	Sample Documentation	Output
28	PM Change Requests (PMLP)			CR2014-00812: Sample Change Request filed against Standby Generator Major Outage PMID's (18662-06)	
29	Adjust/Maintain Technical Basis PM & CC			N/A	
30	Maintenance Assessing Adjust PM Program (WO Instructions & frequency)	N-PROC-MA-0026 N-PROC-MA-0034	Preventive Maintenance Technical Specifications Predictive Maintenance Program Requirements	N/A	
31	PM Program Scheduling			N/A	PM Scope PM Target date
32	PM Technical Basis			Screenshot of PMLP Main Screen for SG Vibration Monitoring Predictive Maintenance Predefined (76368-04)	Technical basis for PMs and PdMs
33	Criticality Coding	N-PROC-MA-0077	Critical Equipment Identification And Categorization	N-FORM-11294 (Criticality Coding Change Request Form)	Criticality code
34	PM Health Report	N-PROC-MA-0026 N-PROC-MA-0034 N-PROC-RA-0023	Preventive Maintenance Technical Specifications Predictive Maintenance Program Requirements Fleetview Program Health And Performance Reporting	PdM Program Health Report	PM Health Recommendations for health improvement

Box #	Box Title	Program/ Procedure Document Number	Program/ Procedure Document Name	Sample Documentation	Output
35	PM Oversight Committee (PMOC)	N-PROC-MA-0026 N-PROC-MA-0034	Preventive Maintenance Technical Specifications Predictive Maintenance Program Requirements	PM Oversight Committee Presentation	PM Health improvement plan PMOC endorsement of recommendations
Health Reporting & Oversight					
36	Component Health Reporting	N-PROG-MA-0017	Component And Equipment Surveillance	Component Health Report – Motors, Pumps (For SG-related Equipment)	Component condition assessment Component Health trending Component health improvement work
37	System Health Reporting	N-PROC-MA-0024 N-PROG-MA-0026 N-INS-01071-10000	System Performance Monitoring Equipment Reliability Program System Health Reporting	Standby Generators System Health Report	System condition assessment System Health trending System health improvement work
38	Recovery Plans	N-PROC-MA-0097 N-PROG-MA-0026	Equipment Reliability Implementation Equipment Reliability Program	P-PLAN-37000-0534519, Recovery Plan In Response To Pickering NGS Fuel Performance Problems.	Corrective actions Recommendations
39	Plant Health Committee			058 SG Critical Spares Status System Health Presentation System Advocacy	Monitoring of Plant Reliability List Monitoring of System Health
40	Plant Reliability List			P-CORR-01015-0520633 (Plant Reliability List 2015)	Work orders for plant reliability improvement

Box #	Box Title	Program/ Procedure Document Number	Program/ Procedure Document Name	Sample Documentation	Output
41	Long Range Plan	N-PROC-MA-0097 N-PROC-AS-0043 N-PROG-AS-0005 N-PROC-AS-0080	Equipment Reliability Implementation Nuclear Outage Generation Planning Nuclear Business Planning Program Nuclear Business Planning	System Health Presentation (See Plant Health Committee)	Equipment maintenance strategy Recommended actions
Modifications					
42	Site (ECR) Screening Committee	N-PROG-MP-0001 N-PROC-MP-0090 N-GUID-00700-10000	Engineering Change Control Modification Process Guide To Modification Process	ECR for Standby Generator Fire Alarm System Modifications and ECR Approval (Asset Suite)	Work approval Priority
43	PAC/AISC			N-FORM-10945 (Cost Estimate and Request for Conceptual Funding)	Project Approval Priority
44	MMODs			N/A (Modifications that meet certain cost requirements become Minor Modifications and do not require AISC Approval)	Scope of work Target Implementation date Work Priority
45	Projects	N-PROC-MP-0001 N-PROC-MP-0090 N-GUID-00700-10000 N-PROG-AS-0007	Engineering Change Control Modification Process Guide To Modification Process Project Management	NK30-DP-71400-00003 (SG Fire Alarm Upgrades Project Design Plan)	Scope of work Target Implementation date Work Priority Design Modification
Aging Management and Long Range Planning					

Box #	Box Title	Program/ Procedure Document Number	Program/ Procedure Document Name	Sample Documentation	Output
46	Major Component Life Cycle Management Plans (LCMPs)	N-PROG-MA-0025 N-PLAN-01060-10007 R02 NA44-PLAN-33110-10003 R005 NK30-PLAN-33110-10008 R007 N-REP-31100-10055 R001	Major Components Program Feeders Technical Basis Document Pickering Units 1 and 4 Steam Generator Life Cycle Management Plan Pickering Units 5-8 Steam Generator Life Cycle Management Technical Basis Report on Technical Basis for Fuel Channels Life Cycle Management Plan	Feeders Life Cycle Management Plan (N-PLAN-01060-10001)	Degradation mechanisms Current aging management practices Component assessment results and analysis Aging management strategy
47	Component Condition Assessment (CCAs)	N-PROG-MP-0008 N-PROC-MP-0060	Integrated Aging Management Aging Management Process	NK30-REP-54600-00111 (CCA – 54600 – Standby Generators – Gas Producer Assemblies)	Degradation mechanisms Current aging management practices Component assessment results and analysis
48	Condition Assessment and Actions	N-PROC-MP-0060	Aging Management Process	(See sample Condition Assessment above)	Recommended actions and aging management strategy
Periodic Safety Review					

Box #	Box Title	Program/ Procedure Document Number	Program/ Procedure Document Name	Sample Documentation	Output
49	NS Programs (List of SIS/SOE)	P-LIST-06937-00001 NA44-REP-03611-00004 NK30-REP-03611-00024 N-INS-03602-10001	Pickering A And B List Of Safety Related Systems Pickering A Systems Important To Safety Pickering B Systems Important To Safety Preparation of Safe Operating Envelope Compliance Tables	NA44-REP-03611-00004 Executive Summary (Pickering A SIS Systems) NK30-REP-03611-00024 Executive Summary (Pickering B SIS Systems) N-INS-03602-10001 Appendix A (Preparation of Safe Operating Envelope Compliance Tables)	List of systems important to safety Safe Operating Envelope
50	Safety Factors Reports	REGDOC-2.3.3	Periodic Safety Reviews	Safety Report (P-REP-03680-00001)	Equipment condition analysis Gaps
51	Global Assessment			Global Assessment Report (NK30-REP-03680-0400585)	Recommendations/ Actions Overall risk Methodology for gap resolution
52	Integrated Implementation Plan (IIP)			IIP (NK30-CORR-00531-06118)	Scope of work Target Implementation date Work Priority
Improve ER Fix Plant Enhance Safety					

Box #	Box Title	Program/ Procedure Document Number	Program/ Procedure Document Name	Sample Documentation	Output
53	Work Planning (Online/Outage)	N-PROG-MA-0019 N-PROC-MA-0049 N-PROC-MA-0013 N-PROC-MA-0022 N-PROC-MA-0008 N-PROG-MA-0004	Production Work Management Forced Outage Management Planned Outage Management Integrated On-Line Work Scheduling Work Initiation Approval and Prioritization Conduct Of Maintenance	Primary Equipment Group (PEG) Schedule NIMS	Scope of work Work plan Schedule Work Priority Resource requirement Material requirement
54	WO Execution	N-PROG-MA-0019 N-PROC-MA-0006 N-PROG-MA-0004	Production Work Management Work Performance Conduct Of Maintenance	Work Request (N-FORM-10048) approved	Work permits

TETRA TECH

Figure C.2 Aging Management Screening Process

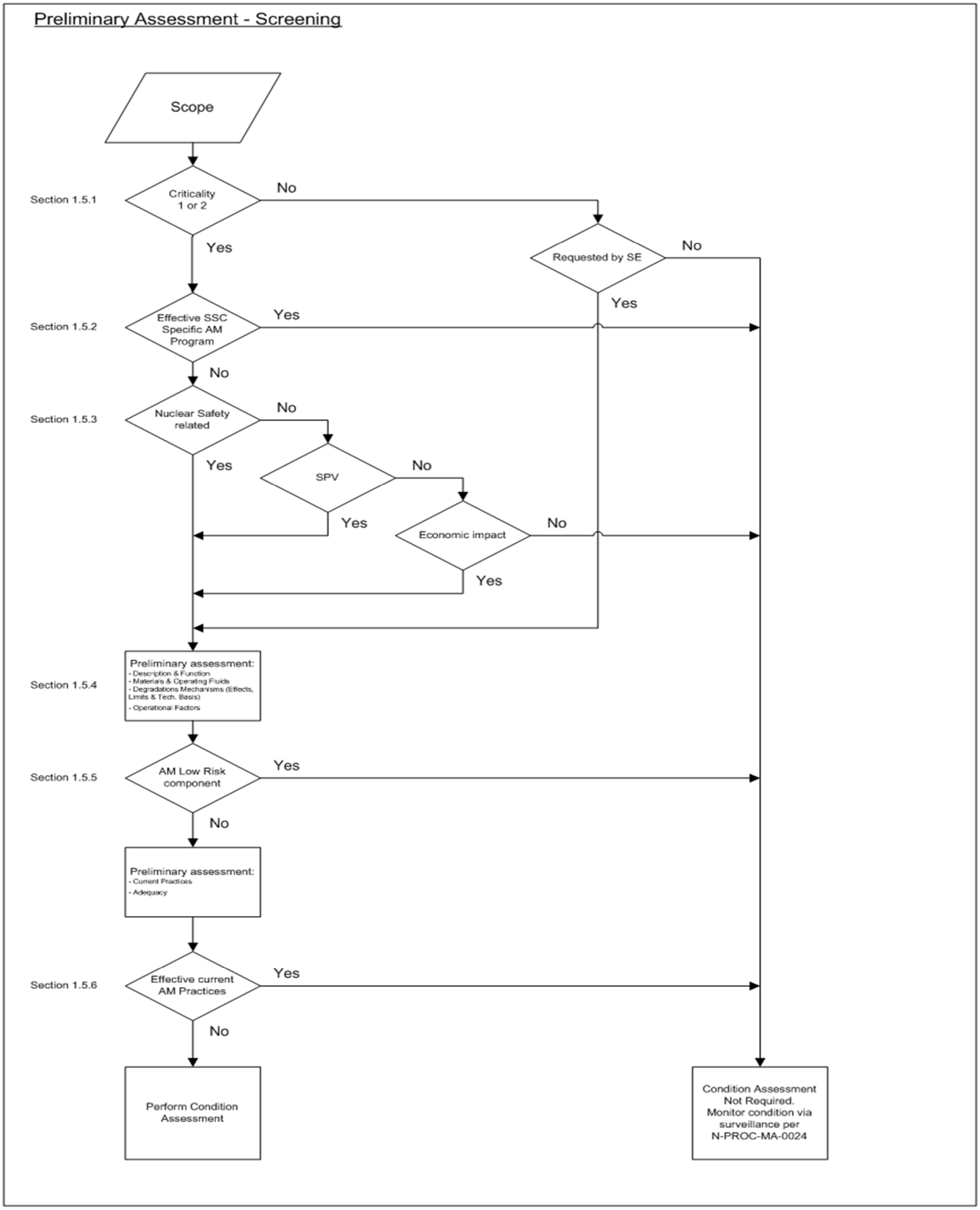

Table C.2 Classification of Component Condition and Aging Management Practices

Condition Classification	**Criteria**
Very Good	(a) The component meets all its functional design requirements, with no reduction in operating margin and exhibits no apparent degradation, (i.e., is in "like new" condition) <u>and</u>, (b) The *aging management* practices have been optimized to ensure the component remains in a "like new" condition.
Good	(a) The component meets all its functional design requirements, with only a slight reduction in operating margins. Some slight aging degradation is evident, <u>or</u>, (b) The *aging management* practices are adequate but has not been optimized to ensure that the component remains in a "like new" condition.
Satisfactory	(a) The component still meets all it functional design requirements, but operating margins are significantly eroded. This can be attributed to evidence of significant aging degradation, or, (b) The *aging management* practices are ineffective in only one area and should be reviewed and/or changed.
Poor	(a) The component can only marginally meet its functional design requirements and has lost all operating margin. Severe aging degradation is evident, <u>or</u>, (b) The *aging management* practices are ineffective in a number of areas and need to be revised.
Very Poor	(a) The component cannot meet one or more of its functional design requirements. The component needs immediate or near term maintenance, repair and/or replacement to restore its condition, <u>or</u>, (b) The current *aging management* practices are completely ineffective and need revision.

APPENDIX C.2 SAMPLE CA – SYSTEM 0412 CONTAINMENT

Note: This document is in the process of being issued in Asset Suite.

Document Number: NK30-REP-73110-00041-R000	
CCA: 011181 **System:** 0412 - Containment	
Pickering (amalgamated) – Aging Management Program Component Condition Assessment (CCA) **73110 RB Cooling - Boiler Room ACUs**	
Section 1	CCA Scope
USI	34200
USI(s) Included	73110,
CG Type	HVAC
CG Subtype	Air Conditioning Unit (ACU)
Equipment Tag(s)	6-73110-ACU1, 7-73110-ACU1, 8-73110-ACU1, 5-73110-ACU2, 6-73110-ACU2, 7-73110-ACU2, 8-73110-ACU2, 5-73110-ACU3, 6-73110-ACU3, 7-73110-ACU3, 8-73110-ACU3, 5-73110-ACU4, 6-73110-ACU4, 7-73110-ACU4, 8-73110-ACU4, 5-73110-ACU5, 6-73110-ACU5, 7-73110-ACU5, 8-73110-ACU5, 5-73110-ACU6, 6-73110-ACU6, 7-73110-ACU6, 8-73110-ACU6, 5-73110-ACU1
Equipment Cat ID(s)	0000694168, 0000694171
Unit(s)	5 (EOL 2020), 6 (EOL 2020), 7 (EOL 2020), 8 (EOL 2020)
Equipment Description	CAT ID 694168 (Status: BOMONLY ; On-Hand:0) E-Tags: 5/6/7/8-73110-ACU1/ ACU5/ ACU6 Description: ACU: C/W 286T LEESON 20 HP MOTOR EQ & R.H. COIL Manufacturer: SHELDONS ENGINEERING LTD Model: 10-1 MODULAR ACU RH COIL CAT ID: 694171 (Status: BOMONLY ; On-Hand:0) E-Tags: 5/6/7/8-73110-ACU2/ ACU3/ ACU4 Description: ACU: C/W 286T LEESON 20 HP MOTOR EQ & L.H. COIL Manufacturer: SHELDONS ENGINEERING LTD Model: 10-1 MODULAR ACU LH COIL This DCA covers the ACU units including the coils. It does not cover the motors, cables, or solenoid valves.
Equipment Function	E-Tags: 5/6/7/8-73110-ACU1/ ACU2/ ACU3/ ACU4 ACU5/ ACU6 Function: The Boiler Room Cooling system is comprised of six (6) Air Cooling Units (ACUs). The units are the draw-through fan-coil type which is supplied with cooling water from the process water system. These units are designed to limit the ambient temperature in the Boiler Room between 36°C (97°F) and 60°C (140°F). The Boiler Room ACUs are part of the safety system. The ACUs are Seismically Qualified to Category 'B' and have an alternative water supply from the EWS system. These ACUs are required to operate on demand in harsh environmental post-LOCA conditions. However, failure of the non-metallic materials (filters) would not prevent the ACUs from performing their function. Therefore the ACUs are not required to be environmentally qualified (NK30-EQL-73110-0001). Consequence of Failure: If the boiler room ACUs fail to operate, the boiler room temperature limit of 60°C (Reference P-SPM-73110-0465019 R003) could be exceeded. Prolonged operation above Safety Limit (65°C) would lead to local or global concrete deterioration. In addition, 3 ACUs are required to provide sufficient post-accident cooling. Therefore there is some redundancy with the ACUs. Reference NK30-OSR-08131.02-00003 Appendix A, Table A.1. Operating Limits: The ACUs operate to maintain the Boiler Room between 36°C (97°F) and 60°C (140°F). Duty Cycle: High Environment (Mild/Severe): Severe

	Operating Fluids: Tube side – HP service water, fin side – ambient air EQ: No Seismic: Yes (DBE 'B') Pressure Boundary: Yes STBR Phase: 3. Note that the system is required for phase 3. However, the cooling function is only required until phase 2. References: P-SPM-73110-0465019 R003, NK30-OSR-08131.02-00003, P-REP-01060-00002, NK30-SCL-73110-00001 Rev: 001, NK30-EQL-73110-0001
Section 2	Criticality and Consequences
AM Critical Components	Yes – based on: Nuclear Safety Related? Yes CG Criticality: Cat 1/2 CCA Requested by RSE? No Economic Impact? Yes Low Risk AM Component: No CG contains components with Criticality of 1 or 2 not classified as Non-AM Critical Components per Section 1.5.5 of N-PROC-MP-0060 Reactor Safety Ranking: 2 Overall CG Criticality: 2
Section 3	Condition Assessment and Life Cycle Issues
Summary	Scope
	ARDM Corrosion / Stress Corrosion Cracking - Carbon and low alloy steels **Effects:** Stress Corrosion cracking can lead to cracking, coil failures and leaks. Coil leaks are the major threat to ACUs availability and performance. **Limits:** It can be detected by observing brittle -appearing fracture and cracks at joints and high load bearing areas or by reduced performance of the ACU resulting in higher room temperatures. **Comments:** Stress corrosion cracking (SCC) is the unexpected sudden failure of normally ductile metals subjected to a tensile stress in a corrosive environment, especially at elevated temperature in the case of metals. Corrosion / Fouling (accumulation of deposits) **Effects:** Fouling of ACU components (including coils) **Limits:** Limited heat transfer – an increase in temperature of the 'cool' air being supplied to the control cubicle. **Comments:** Silting and Microbiological Influenced Corrosion (MIC) from HPSW impurities can contribute to accumulation of deposits and plugging drain lines on the tube side. Poor heat transfer will eventually prevent the component from meeting its performance requirements. Corrosion / General Corrosion **Effects:** General corrosion on coils, mounting brackets which the fan and motor are mounted causing failure at mountings. **Limits:** Component Failure **Comments:** Corrosion can cause reduced strength at mounts which can cause component failure. Mechanical and Thermal Degradation / Wear Mechanisms - Wear **Effects:** Bearings, and fan wear **Limits:** Component failure from unacceptable vibration levels **Comments:** Moving parts including bearings have a finite lifespan Mechanical and Thermal Degradation / Wear Mechanisms - Erosion **Effects:** Erosion of components from high pressure service water **Limits:** Reduction of coil thickness could eventually result in leakage.

	Comments: Silt, minerals, and MIC in HPSW could cause erosion on coil/inner tubes leading to leakage or loss of pressure boundary integrity. Fatigue / Mechanical Fatigue **Effects:** Vibration from the operation of the fan can lead to component malfunction. **Limits:** Component failure **Comments:** Mechanical fatigue is not expected to be a significant ARDM. Mechanical and Thermal Degradation / Deterioration of Material (Wiring, Seals, Gaskets, O-rings etc.) **Effects:** Deterioration of electrical components including insulation, windings, and connectors can cause the ACU to malfunction **Limits:** The ACU, including the motor and thermostat, fails to operate. **Comments:** While the ACU has material including insulation that will wear over time, the material degradation of electrical components is not expected to be significant compared to other ACU ARDMs
	Operational Factors N/A
	Condition: Satisfactory CHR REVIEW (Current Equipment Assessment Detail Report, dated 2/23/2016). A search was done using the equipment codes (5/6/7/8-73110-ACU1/ ACU2/ ACU3/ ACU4 ACU5/ ACU6). The following status was found: 6-73110-ACU2 – Watch List - No recent vibration data. All other equipment tags were classified as Acceptable (Green). SHR REVIEW (“Reactor Building Cooling”; Q4-2015). A search was done using the equipment codes (5/6/7/8-73110-ACU1/ ACU2/ ACU3/ ACU4 ACU5/ ACU6). The following information was found: WO# 02113023 (STATUS is ACTIVE). 7-73110-ACU4 END OF LIFE COIL PROACTIVE REPLACEMENT “There is one functional failure within 2 year rolling period. - WO 4767203, 6-73110-ACU4 and ACU5 isolated due to leak.” Level II Impairment due to ACU4 and ACU5 Isolated. P-2015-07286, WO# 4767703. Completion notes state “INVESTIGATED ACU5 FOR DRAIN BLOCKAGE AND FOUND NO ISSUES.” WO# 04767704 - 6-73110-ACU2 DRIP TRAY LINE LEAKING. Result: MM TIGHTEN FITTING ON DRIP TRAY LINE. REINSTALLED DRAIN LINE AS PER INSTRUCTIONS.” Task Status: FINISHED. WO# 04767705 - 6-73110-ACU1 DRIP TRAY LINE DISCONNECTED AND LEAKING. Result: “MM RECONNECT DRIP TRAY LINE. RECONNECTED DRAIN LINE AS PER INSTRUCTIONS.” Task Status: FINISHED. OPEX - SCR REVIEW (from 01Jan2010 to 15Jan2016). A search was done using the equipment codes (5/6/7/8-73110-ACU1/ ACU2/ ACU3/ ACU4 ACU5/ ACU6). The following status was found:

	SCR P-2015-07286 - 6-73110-ACU1, -ACU2 and -ACU5 isolated due to leakage (Level 3 Impairment, and ACUs leaking water to the reactivity mechanism deck which could affect its performance.) (March 2015). Apparent Cause Evaluation determined problem with -ACU4 and -ACU5 (recently replaced in P1361 outage) Based on the completion notes for AR 28177085 – 01 (dated 16-FEB-2016), which was generated from this SCR, "Based on the conclusion of SATPS, no additional actions are required." Therefore it is not necessary to "confirm the integrity of the condensate drain lines (leak tight and are not plugged for all ACUs)." P-2014-13350 - Containment Impairment on U4 Boiler Room ACU4 (tripped on motor overload). Broke- Fix (WO# 03163606 – motor replaced). P-2011-25045 CMU Trending: 6-73110-ACU1 coil leak. WO# 2696947 documents the coil replacement to correct the issue, P-2010-14935 - 1-73110-ACU4 & 1-73110-ACU6 Boiler Room ACU's Coupling bolts found sheared. RESULT – Predefined maintenance initiated to replace bolts every outage. WO REVIEW (from 01Jan2010 to 15Jan2016): There were work orders created for the proactive replacement of the coils in each Pickering B ACU. A review shows that only one WO is outstanding (WO# 2113023 - 7-73110-ACU4. The other 23 ACU coils were replaced between 2012 and 2014. Many work orders have been created and closed to manage the completion of PMIDs. There are 35 work orders for ACU Coil Leak, Investigate/Repair /Acu1 End Of Life Coil Proactive Replacement. There were seven (7) work orders related to Unit 6 drip tray or drain line being disconnected or plugged. This relates to SCR P-2015-07286 mentioned above. One work order (WO# 02279446) was created for 8-73110-ACU3 tripping. This WO was completed by replacing the overload relay. This is considered a one off and this was the only instance documented of any of the 24 ACUs experiencing a faulty overload relay. There are four completed work orders for motor bearing replacements. Bearing replacements were performed on 5-73110-ACU3/ACU4; 7-73110-ACU6; 8-73110-ACU3/ACU4; Condition Summary (from 01Jan2010 to 15Jan2016): Based on the above CHR, SHR, WO and OPEX reviews, there is some evidence of degradation related to ARDMs. These have been managed and fixed as part of ACU maintenance strategy (coil replacements) or as required (bearing, relay, drip tray issues). Therefore, the component condition as described above is rated as 'SATISFACTORY' Condition Classification: The current Aging Management Practices (AMP) as detailed below, are rated as Satisfactory. Rationale: The component still meets all it functional design requirements, but operating margins are significantly eroded. This can be attributed to evidence of significant aging degradation In summary, the overall Condition Classification is rated as 'Satisfactory'. There was a previous concern about blocked drain lines that resulted in SCR P-2015-07286. Completion Notes for the related AR (28177085 – 01) concluded, "Based on the

	conclusion of SATPs, no additional actions are required". The aging management practices are ineffective in only one area and should be reviewed and/or changed.
	Current Practices Detailed CA Equivalent Program? No Program Reference: N/A E-Tags: 5/6/7/8-73110-ACU1/ ACU2/ ACU3/ ACU4 ACU5/ ACU6 Programmatic Requirements: Not Applicable. PMID-Active Title: ACU1 VIBE CHECK / LUBE & INSPECT PMID #s: 12498, 12504, 12515, 12524, 12525, 12526, 14084, 14090, 14101, 14110, 14111, 14112, 15448, 15554, 15565, 15574, 15575, 15576, 17092, 17098, 17109, 17118, 17119, 17120 E-Tags: 5/6/7/8-73110-ACU1/ ACU2/ ACU3/ ACU4 ACU5/ ACU6 Scope: 1- Perform Pre-Maintenance Vibration Check 2- Lubricate, Inspect ACU And Replace Filters 3- Perform Post- Maintenance Vibration Check Frequency: 104 weeks (2 years). Supporting references (as applicable): Model Work Order 350258, Title: CHECK SEISMIC RESTRAINTS PMID #s: 1303, 1480, 1592, 1704 E-Tags: 5/6/7/8-73110-ACU1/ ACU2/ ACU3/ ACU4 ACU5/ ACU6 Scope: Perform the inspection of seismic restraints, on boiler room air conditioning unit Frequency: 312 weeks (6 years). Supporting references (as applicable): Model Work Order 01559914 PMID-Retired Title: REPLACE EQ INDUCTION FAN MOTOR/ EQ FAN MOTOR REPLACEMENT PMID #s: 12498, 12504, 12515, 12524, 12525, 12526, 14084, 14090, 14101, 14110, 14111, 14112, 15448, 15554, 15565, 15574, 15575, 15576, 17092, 17098, 17109, 17118, 17119, 17120, 96500, 96501, 96502, 96503 (task 2 for these PMIDs). E-tags: 5/6/7/8-73110-ACU1/ ACU2/ ACU3/ ACU4 ACU5/ ACU6 Scope: Replace This Fan Motor, With A New Environmentally Qualified Fan Motor per NK30-EQA-05600-00007, Rev 03. Frequency: 1300 weeks (25 years) – one time replacement for plant life. Note that the EQ qualified life of the motors is 27.9 years. These motors were replaced in late 2002 to 2004 which is acceptable to meet the operation requirements until 2029. Supporting references (as applicable):Model Work Order 01221698 Rationale for Retiring PMID: One time replacement required for plant life (2029). PMID-Retired Title: ACU'S 1 TO 6 Replace Filters PMID #s: 3339, 3475, 3598, 3708 E-tags: 5/6/7/8-73110-ACU1/ ACU2/ ACU3/ ACU4 ACU5/ ACU6 Scope: Replacement of inlet filters to BR ACU's to maintain proper air flow Frequency: 26 weeks (6 months) per Asset Suite notes. Supporting references (as applicable): Model Work Order 2166. Rationale for Retiring PMID: PM was retired "AS per engineering review, 22-OCT-1999). Note that filter changes are included in active PMIDs (see above). Title: VIBRATION READINGS PMID #s: 5474, 5670, 5786, 5874, 5973, 12498, 12504, 12515, 12524, 12525, 12526, 14084, 14090, 14101, 14110, 14111, 14112, 15448, 15554, 15565, 15574, 15575,

15576, 17092, 17098, 17109, 17118, 17119, 17120, 96500, 96501, 96502, 96503 (Task 3)
E-tags: 5/6/7/8-73110-ACU1/ ACU2/ ACU3/ ACU4 ACU5/ ACU6
Scope: Take vibration readings of ACU'S
Frequency: 104 weeks (2 years).
Supporting references (as applicable): Model Work Order 01210526
Rationale for Retiring PMID: PM was "RETIRED AS PER CR2005-01122 (01-Sept-2005). Note that vibration checks are included in active PMIDs (see above).

Life-cycle Replacement / Last Replacement Date:
The list below are the dates and work orders for when the coils were replaced in the Boiler Room ACUs.

E-tag	WO	WO Status	Replacement Date
5-73110-ACU1	02112158-02	CLOSED	29MAY2013
5-73110-ACU2	02112159-12	CLOSED	26MAY2013
5-73110-ACU3	02112161-04	CLOSED	23MAY2013
5-73110-ACU4	02112164-10	CLOSED	13MAY2013
5-73110-ACU5	02112167-03	CLOSED	16MAY2013
5-73110-ACU6	02112169-12	CLOSED	21MAY2013
6-73110-ACU1	02696947-13	CLOSED	11OCT2013
6-73110-ACU2	02112176-12	CLOSED	01OCT2013
6-73110-ACU3	02112178-01	CLOSED	22SEP2013
6-73110-ACU4	02112180-08	CLOSED	11OCT2013
6-73110-ACU5	02112182-12	CLOSED	30SEP2013
6-73110-ACU6	02112184-03	CLOSED	26SEP2013
7-73110-ACU1	02113016-13	CLOSED	13NOV2014
7-73110-ACU2	02113017-01	CLOSED	06NOV2014
7-73110-ACU3	02113018-05	CLOSED	27OCT2012
7-73110-ACU4	02113023-06	WORKING	N/A planned for P1671 outage.
	00303202-01	CLOSED	08JUL2000
7-73110-ACU5	02113024-07	CLOSED	06NOV2014
7-73110-ACU6	02113025-08	CLOSED	24OCT2012
8-73110-ACU1	02113044-01	CLOSED	30MAR2012
8-73110-ACU2	02113045-01	CLOSED	05APR2012
8-73110-ACU3	02113046-10	CLOSED	01MAY2014
8-73110-ACU4	02412300-28	CLOSED	28MAR2012
8-73110-ACU5	02113050-08	CLOSED	05APR2012
8-73110-ACU6	02113052-06	CLOSED	01MAY2014

Testing/Surveillance/Inspections:

Title: ACU Performance Checks
Document #: P-SPM-73110-0465019 R003
E-tags: 5/6/7/8-73110-ACU1/ ACU2/ ACU3/ ACU4 ACU5/ ACU6
Scope: Check Boiler Room temperature limit has not been exceeded (i.e. temperature is less than 60°C). Note that these ACUs are normally not accessible during unit operation.
Frequency: Daily
Supporting References (as applicable): N/A

Title: Visual Inspection
Document #: P-SPM-73110-0465019 R003
E-tags: 5/6/7/8-73110-ACU1/ ACU2/ ACU3/ ACU4 ACU5/ ACU6
Scope: Visual inspection and recording of any observed deficiencies.
Frequency: every outage.

Title: Routine Field Walkdown Plan (Operating)
Document #: P-SPM-73110-0465019 R003
E-tags: 5/6/7/8-73110-ACU1/ ACU2/ ACU3/ ACU4 ACU5/ ACU6

	Scope: The ACUs are normally inaccessible during unit operation. The scope of this surveillance is to 1) review ANO Log, Long Term Status Log and Equipment Status Log and then document status/any concerns; 2). Obtain temperatures of the Boiler Room from the Unit ANO Frequency: Once per month. Title: Field walkdown –Planned outage Document #: P-SPM-73110-0465019 R003 E-tags: 5/6/7/8-73110-ACU1/ ACU2/ ACU3/ ACU4 ACU5/ ACU6 Scope: The scope of this surveillance is to 1) review ANO Log, Long Term Status Log and Equipment Status Log and then document status/any concerns; 2). Obtain temperatures of the Boiler Room from the Unit ANO; 3) Meet with FSOS, SAT, SNO, Maintenance FLM and document any of their concerns; 4) Perform routine walkdown. In general, check housekeeping, water/oil leaks, Excessive pipe movement/vibrations (general state of pipe supports); Unusual noises from equipment; High operating temperatures on equipment; Material condition (e.g. Corrosion, state of insulation); Room/Area environment (lighting/ventilation/temperature); EQ Tags; Check for any non-conformances found e.g. unapproved operator field aids, Construction tags and marker, marker on equipment, documents posted in field without the approved field aid tag. Frequency: Every planned outage
	Adequacy E-Tags: 5/6/7/8-73110-ACU1/ ACU2/ ACU3/ ACU4 ACU5/ ACU6 Industry Best Practices: The IQ Review Template “PB-Fans and Coolers“ is used as the model for best practice for the Boiler Room ACUs. Per this application, (criticality 2, high duty cycle, severe service condition), the recommended practices are: PREDICTIVE: Operator Rounds – Per Shift Engineer/Performance Monitoring - Yearly Thermography – every 6 months Vibration Analysis – Every 2 months PERIODIC: Bearing Lubrication – Every 2 years Coupling Lube/Inspection – Every 2 years Filter Clean and Inspection – As Required Pulley Sheave/Belt Inspection – Every 2 years. PLANNED: Auxiliary Inspection – As Required Adequacy of Current AMP: Evidence of ARDM related degradation: There has been evidence of aging in coils and bearings. The maintenance strategy for coils is covered by proactive life cycle replacement WO’s based on internal operating experience which indicate a service life of approximately 15 years. Compliance with Programmatic Requirements: Not applicable Adequacy of PMID/Testing/Inspections/Surveillance: The scope of PMID/Testing/Inspections/Surveillance matches with Industry Best Practices except there are no PMIDs to perform thermography analysis. A review of work orders shows that between 2001 and 2004, the ‘perform thermography’ work orders were cancelled. Given the good performance of the ACUs, the motor and coil replacements, vibration analysis and inspections, it is considered acceptable from a condition assessment perspective to not perform thermography.

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	Status of Life-cycle Replacements: Coil replacement for 7-73110-ACU4 is in progress/WORKING status (Refer to Work Order 02113023). Coils in the other 23 ACUs have been completed in the previous 4 years. Once Work Order 02113023 has been completed, then the coils will last until the end of Phase 2 requirements (2029). As stated earlier, the system is required for phase 3 (2039) but the cooling function (i.e. cooling coil functionality) is only required for phase 2. Obsolescence and Spares: There are no unresolved obsolescence or spare issues. The amount of spares are sufficient to support planned and unplanned maintenance. Summary: The current aging management practices are adequate for the Boiler Room ACUs. As per SCR P-2015-07286, there was leaking of Unit 6 ACUs. The unit 6 ACUs were inspected and adequately fixed and addressed. Based on a review of the SCR and the AR completion notes, no further action is necessary to address the issue of blocked condensate drain lines.
	CP Adequate for EOL? No Recommendations to Reach Plant EOL (2020): Complete Work Order 02113023 to replace the coils in 7-73110-ACU4, by 2016.
	Additional AMP for LE? No
Obsolescence	**Obsolescence issues?** Yes **Issues resolved?** Yes Per Asset Suite:

E-tags	Item	Cat ID	Status
5/6/7/8-73110-ACU1/ ACU5/ ACU6	ACU	694168	BOMONLY
5/6/7/8-73110-ACU2/ ACU3/ ACU4	ACU	694171	BOMONLY

Per Proactive Obsolescence Management System (POMS):

E-tags	Existing Make/Model Per POMS	Obsolete per POMS? (Yes/No)	Replacement Make/Model per POMS
5/6/7/8-73110-ACU1/ ACU2/ ACU3/ ACU4 ACU5/ ACU6	SHELDONENG/ 10-1 MODULAR ACU LH COIL	YES	N/A

Summary:
The following E-tags are deemed obsolete: (5/6/7/8-73110-ACU1/ ACU2/ ACU3/ ACU4 ACU5/ ACU6). The rationale is that both of the CATIDs (694168 and 694174) are not at READY status and are listed as ‘BOMONLY’. However there are spares parts available for both CATIDs and these are set to READY status and are therefore the spare parts are considered to be available (not obsolete). It is not reasonable to expect that the entire ACU assembly would be replaced. Therefore obsolescence of the ACU assembly is not a significant issue.

Spares	**Spares applicable?** Yes **Issues resolved?** Yes **Replaceable/Repairable:** Repairable ACU Assembly spare status

E-tags	Item	Cat ID	Status	On-Hand
5/6/7/8-73110-ACU1/ ACU5/ ACU6	ACU	694168	BOMONLY	0
5/6/7/8-73110-ACU2/ ACU3/ ACU4	ACU	694171	BOMONLY	0

Piece Parts to support repairs (CATID 694168)

Applicable E-tags: 5/6/7/8-73110-ACU1/ ACU5/ ACU6

Item	Cat ID	Status	On-Hand
COIL, RIGHT HAND, PRESSURE BOUNDARY, 28" X 88" X 10", TYPE MR, 6 ROW.	168782	READY	0*
FILTER, AIR, -, PAD TYPE, 16" X 25" X 2", ADHESIVE COATED FIBERGLASS ELEMENT, NO FRAME	190484	READY	1,362
FILTER, AIR, DISPOSABLE, 16" X 25" X 2", USE CAT ID 190484	34333	NOPURCH	0
SPRING, ., ISOLATOR, 1" DEFLECTION, 5" OPERATING HEIGHT	686418	READY	0*
SPRING, ., ISOLATOR, USE CAT ID 686418 PER EC110101	87879	NOPURCH	0
MOTOR, ELECTRIC, INDUCTION, 20HP, 1200RPM, 575VAC, 60HZ, 3PH, 286T, CLASS H, TEFC, CONT, 1.15SF, ENVIRONMENTALLY & SEISMICALLY QUALIFIED	505260	READY	3
FAN, COOLING	257528	READY	0*
SHAFT	194168	NOPURCH	1
SHROUD, INLET	194167	NOPURCH	1
COUPLING, SHAFT, FLEXIBLE, 1.8735 X 1.9360 BORE	86779	READY	3
BEARING, PILLOW BLOCK, -, 1-15/16" ID	86409	READY	4
SPRING, ., ISOLATOR	87892	NOPURCH	0

* The Reorder point, and the current inventory quantity is not listed and is assumed to be 0.

The spare ACU coils are listed as READY and are not obsolete. There are no spares in stock. This is not an issue because all of the coils for 5/6/7/8-73110-ACU1/ ACU5/ ACU6 have been recently replaced and are not expected to require future replacement. There are no spare parts in stock for CATIDs 686418, 257528 (isolator spring and cooling fan). These are not considered to be critical spares. The status of these CATIDs is at 'READY' and therefore spares could be ordered if required. For other CATIDs, there are spare parts available to support planned and unplanned maintenance. Spare parts have CATIDs at READY status and quantities match or exceed the reorder points. To maintain environmental qualification of the ACU, care must be taken to ensure that seismic and EQ qualifications of fan motors are maintained.

Spares Conclusion: There are adequate spare parts for CATID 694168 to support future planned and unplanned maintenance.

Piece Parts to support repairs: (CATID 694171)

Applicable E-tags 5/6/7/8-73110-ACU2/ ACU3/ ACU4

Item	Cat ID	Status	On-Hand
COIL, LH, PRESSURE BOUNDARY, 28" X 88" X 10", TYPE MR, 6 ROW.	168783	READY	0*
FILTER, AIR, -, PAD TYPE, 16" X 25" X 2", ADHESIVE COATED FIBERGLASS ELEMENT, NO FRAME	190484	READY	1,362
FILTER, AIR, DISPOSABLE, 16" X 25" X 2", USE CAT ID 190484	34333	NOPURCH	0
SPRING, ., ISOLATOR, 1" DEFLECTION, 5" OPERATING HEIGHT	686418	READY	0*
SPRING, ., ISOLATOR, USE CAT ID 686418 PER EC110101	87879	NOPURCH	0
MOTOR, ELECTRIC, INDUCTION, 20HP, 1200RPM, 575VAC, 60HZ, 3PH,	505260	READY	3

286T, CLASS H, TEFC, CONT, 1.15SF, ENVIRONMENTALLY & SEISMICALLY QUALIFIED			
FAN, COOLING	257528	READY	0*
SHAFT	194168	NOPURCH	1
SHROUD, INLET	194167	NOPURCH	1
COUPLING, SHAFT, FLEXIBLE, 1.8735 X 1.9360 BORE	86779	READY	3
BEARING, PILLOW BLOCK, -, 1-15/16" ID	86409	READY	4
SPRING, ., ISOLATOR	87892	NOPURCH	0

* Reorder point are current inventory is not listed and assumed to be 0.

The spare ACU coils are listed as READY and are not obsolete. There are no spares in stock. This is not an issue because all of the coils for 5/6/7/8-73110-ACU2/ ACU3/ ACU4 have been recently replaced except for 7-73110-ACU4 and are not expected to require future replacement. Based on Asset Suite notes, the two coils required to complete the coil replacement for 7-73110-ACU4 have been delivered and are awaiting installation in the field. Therefore, there are no issues with obsolescence or spare quantities of the coil (CATID 168783)

There are no spare parts in stock for CATIDs 686418, 257528 (isolator spring and cooling fan). These are not considered to be critical spares. The status of these CATIDs is at 'READY' and therefore spares could be ordered if required in the future. For other CATIDs, there are spare parts available to support planned and unplanned maintenance. Spare parts have CATIDs at READY status and quantities match or exceed the reorder points. To maintain environmental qualification of the ACU, care must be taken to ensure that seismic and EQ qualifications of fan motors are maintained.

Spares Conclusion: There are adequate spare parts for CATID 694171 to support future planned and unplanned maintenance.

Time Limited Aging Analysis	**TLAA issues?** No **Issues resolved?** N/A **TLAA Reference:** N/A The components in this CG are non-passive, (i.e. Continuously operated, poised and exercised during outages, or on stand-by and function tested periodically to confirm availability) and hence TLAA is not applicable as the component life is assessed based on monitored performance.
Section 4	Aging Management Practices
CCA Equivalent Program	No **Program:** N/A
Current AM Practices	E-Tags: 5/6/7/8-73110-ACU1/ ACU2/ ACU3/ ACU4 ACU5/ ACU6 Programmatic Requirements: Not Applicable. PMID-Active Title: ACU1 VIBE CHECK / LUBE & INSPECT PMID #s: 12498, 12504, 12515, 12524, 12525, 12526, 14084, 14090, 14101, 14110, 14111, 14112, 15448, 15554, 15565, 15574, 15575, 15576, 17092, 17098, 17109, 17118, 17119, 17120 E-Tags: 5/6/7/8-73110-ACU1/ ACU2/ ACU3/ ACU4 ACU5/ ACU6 Scope: 1- Perform Pre-Maintenance Vibration Check 2- Lubricate, Inspect ACU And Replace Filters 3- Perform Post- Maintenance Vibration Check Frequency: 104 weeks (2 years). Supporting references (as applicable): Model Work Order 350258,

	Title: CHECK SEISMIC RESTRAINTS PMID #s: 1303, 1480, 1592, 1704 E-Tags: 5/6/7/8-73110-ACU1/ ACU2/ ACU3/ ACU4 ACU5/ ACU6 Scope: Perform the inspection of seismic restraints, on boiler room air conditioning unit Frequency: 312 weeks (6 years). Supporting references (as applicable): Model Work Order 01559914 PMID-Retired Title: REPLACE EQ INDUCTION FAN MOTOR/ EQ FAN MOTOR REPLACEMENT PMID #s: 12498, 12504, 12515, 12524, 12525, 12526, 14084, 14090, 14101, 14110, 14111, 14112, 15448, 15554, 15565, 15574, 15575, 15576, 17092, 17098, 17109, 17118, 17119, 17120, 96500, 96501, 96502, 96503 (task 2 for these PMIDs). E-tags: 5/6/7/8-73110-ACU1/ ACU2/ ACU3/ ACU4 ACU5/ ACU6 Scope: Replace This Fan Motor, With A New Environmentally Qualified Fan Motor per NK30-EQA-05600-00007, Rev 03. Frequency: 1300 weeks (25 years) – one time replacement for plant life. Note that the EQ qualified life of the motors is 27.9 years. These motors were replaced in late 2002 to 2004 which is acceptable to meet the operation requirements until 2029. Supporting references (as applicable):Model Work Order 01221698 Rationale for Retiring PMID: One time replacement required for plant life (2029). PMID-Retired Title: ACU'S 1 TO 6 Replace Filters PMID #s: 3339, 3475, 3598, 3708 E-tags: 5/6/7/8-73110-ACU1/ ACU2/ ACU3/ ACU4 ACU5/ ACU6 Scope: Replacement of inlet filters to BR ACU's to maintain proper air flow Frequency: 26 weeks (6 months) per Asset Suite notes. Supporting references (as applicable): Model Work Order 2166. Rationale for Retiring PMID: PM was retired “AS per engineering review, 22-OCT-1999). Note that filter changes are included in active PMIDs (see above). Title: VIBRATION READINGS PMID #s: 5474, 5670, 5786, 5874, 5973, 12498, 12504, 12515, 12524, 12525, 12526, 14084, 14090, 14101, 14110, 14111, 14112, 15448, 15554, 15565, 15574, 15575, 15576, 17092, 17098, 17109, 17118, 17119, 17120, 96500, 96501, 96502, 96503 (Task 3) E-tags: 5/6/7/8-73110-ACU1/ ACU2/ ACU3/ ACU4 ACU5/ ACU6 Scope: Take vibration readings of ACU'S Frequency: 104 weeks (2 years). Supporting references (as applicable): Model Work Order 01210526 Rationale for Retiring PMID: PM was “RETIRED AS PER CR2005-01122 (01-Sept-2005). Note that vibration checks are included in active PMIDs (see above). Life-cycle Replacement / Last Replacement Date: The list below are the dates and work orders for when the coils were replaced in the Boiler Room ACUs.

E-tag	WO	WO Status	Replacement Date
5-73110-ACU1	02112158-02	CLOSED	29MAY2013
5-73110-ACU2	02112159-12	CLOSED	26MAY2013
5-73110-ACU3	02112161-04	CLOSED	23MAY2013
5-73110-ACU4	02112164-10	CLOSED	13MAY2013
5-73110-ACU5	02112167-03	CLOSED	16MAY2013
5-73110-ACU6	02112169-12	CLOSED	21MAY2013
6-73110-ACU1	02696947-13	CLOSED	11OCT2013
6-73110-ACU2	02112176-12	CLOSED	01OCT2013
6-73110-ACU3	02112178-01	CLOSED	22SEP2013

6-73110-ACU4	02112180-08	CLOSED	11OCT2013
6-73110-ACU5	02112182-12	CLOSED	30SEP2013
6-73110-ACU6	02112184-03	CLOSED	26SEP2013
7-73110-ACU1	02113016-13	CLOSED	13NOV2014
7-73110-ACU2	02113017-01	CLOSED	06NOV2014
7-73110-ACU3	02113018-05	CLOSED	27OCT2012
7-73110-ACU4	02113023-06	WORKING	N/A planned for P1671 outage.
	00303202-01	CLOSED	08JUL2000
7-73110-ACU5	02113024-07	CLOSED	06NOV2014
7-73110-ACU6	02113025-08	CLOSED	24OCT2012
8-73110-ACU1	02113044-01	CLOSED	30MAR2012
8-73110-ACU2	02113045-01	CLOSED	05APR2012
8-73110-ACU3	02113046-10	CLOSED	01MAY2014
8-73110-ACU4	02412300-28	CLOSED	28MAR2012
8-73110-ACU5	02113050-08	CLOSED	05APR2012
8-73110-ACU6	02113052-06	CLOSED	01MAY2014

Testing/Surveillance/Inspections:

Title: ACU Performance Checks
Document #: P-SPM-73110-0465019 R003
E-tags: 5/6/7/8-73110-ACU1/ ACU2/ ACU3/ ACU4 ACU5/ ACU6
Scope: Check Boiler Room temperature limit has not been exceeded (i.e. temperature is less than 60°C). Note that these ACUs are normally not accessible during unit operation.
Frequency: Daily
Supporting References (as applicable): N/A

Title: Visual Inspection
Document #: P-SPM-73110-0465019 R003
E-tags: 5/6/7/8-73110-ACU1/ ACU2/ ACU3/ ACU4 ACU5/ ACU6
Scope: Visual inspection and recording of any observed deficiencies.
Frequency: every outage.

Title: Routine Field Walkdown Plan (Operating)
Document #: P-SPM-73110-0465019 R003
E-tags: 5/6/7/8-73110-ACU1/ ACU2/ ACU3/ ACU4 ACU5/ ACU6
Scope: The ACUs are normally inaccessible during unit operation. The scope of this surveillance is to 1) review ANO Log, Long Term Status Log and Equipment Status Log and then document status/any concerns; 2). Obtain temperatures of the Boiler Room from the Unit ANO
Frequency: Once per month.

Title: Field walkdown –Planned outage
Document #: P-SPM-73110-0465019 R003
E-tags: 5/6/7/8-73110-ACU1/ ACU2/ ACU3/ ACU4 ACU5/ ACU6

Scope: The scope of this surveillance is to 1) review ANO Log, Long Term Status Log and Equipment Status Log and then document status/any concerns; 2). Obtain temperatures of the Boiler Room from the Unit ANO; 3) Meet with FSOS, SAT, SNO, Maintenance FLM and document any of their concerns; 4) Perform routine walkdown. In general, check housekeeping, water/oil leaks, Excessive pipe movement/vibrations (general state of pipe supports); Unusual noises from equipment; High operating temperatures on equipment; Material condition (e.g. Corrosion, state of insulation); Room/Area environment (lighting/ventilation/temperature); EQ Tags; Check for any non-conformances found e.g. unapproved operator field aids, Construction tags and marker, marker on equipment, documents posted in field without the approved field aid tag.
Frequency: Every planned outage

Current AMP Adequacy	E-Tags: 5/6/7/8-73110-ACU1/ ACU2/ ACU3/ ACU4 ACU5/ ACU6

	Industry Best Practices: The IQ Review Template "PB-Fans and Coolers" is used as the model for best practice for the Boiler Room ACUs. Per this application, (criticality 2, high duty cycle, severe service condition), the recommended practices are: PREDICTIVE: Operator Rounds – Per Shift Engineer/Performance Monitoring - Yearly Thermography – every 6 months Vibration Analysis – Every 2 months PERIODIC: Bearing Lubrication – Every 2 years Coupling Lube/Inspection – Every 2 years Filter Clean and Inspection – As Required Pulley Sheave/Belt Inspection – Every 2 years. PLANNED: Auxiliary Inspection – As Required Adequacy of Current AMP: Evidence of ARDM related degradation: There has been evidence of aging in coils and bearings. The maintenance strategy for coils is covered by proactive life cycle replacement WO's based on internal operating experience which indicate a service life of approximately 15 years. Compliance with Programmatic Requirements: Not applicable Adequacy of PMID/Testing/Inspections/Surveillance: The scope of PMID/Testing/Inspections/Surveillance matches with Industry Best Practices except there are no PMIDs to perform thermography analysis. A review of work orders shows that between 2001 and 2004, the 'perform thermography' work orders were cancelled. Given the good performance of the ACUs, the motor and coil replacements, vibration analysis and inspections, it is considered acceptable from a condition assessment perspective to not perform thermography. Status of Life-cycle Replacements: Coil replacement for 7-73110-ACU4 is in progress/WORKING status (Refer to Work Order 02113023). Coils in the other 23 ACUs have been completed in the previous 4 years. Once Work Order 02113023 has been completed, then the coils will last until the end of Phase 2 requirements (2029). As stated earlier, the system is required for phase 3 (2039) but the cooling function (i.e. cooling coil functionality) is only required for phase 2. Obsolescence and Spares: There are no unresolved obsolescence or spare issues. The amount of spares are sufficient to support planned and unplanned maintenance. Summary: The current aging management practices are adequate for the Boiler Room ACUs. As per SCR P-2015-07286, there was leaking of Unit 6 ACUs. The unit 6 ACUs were inspected and adequately fixed and addressed. Based on a review of the SCR and the AR completion notes, no further action is necessary to address the issue of blocked condensate drain lines.
Current AMP Adequate to Reach Plant EOL?	No **Recommendations:** Recommendations to Reach Plant EOL (2020): Complete Work Order 02113023 to replace the coils in 7-73110-ACU4, by 2016.
Comp. End Mission Date	2039

Expected Degradation at CO EOL with CP	Assuming that the current AMP continue and the recommendations to reach Plant EOL (2020) are implemented (e.g. completion of WO#. 02113023), the expected condition at 2039 is SATISFACTORY. The rationale is as follows: - There will be effective AMP in-place to detect and mitigate the on-set of incipient age related degradation (vibration analysis and check seismic restraints inspections) - Life-cycle replacements will have effectively occurred (e.g. coil and EQ fan motor replacements) before the useful life of the component is exceeded. The aging management practices are adequate but has not been optimized to ensure that the component remains in a "like new" condition.
Additional AMP to Reach Plant CO EOL?	No **Recommendations:** N/A
Work Program Type	OM&A **Rationale:** WO#. 02113023-06 for replacement of 7-73110-ACU4 is already "Working".
Component Condition at CO EOL	Good **Comments and Basis:** Assuming that the current AMP continue, the expected condition at 2039 is 'Satisfactory'. The rationale is as follows: - There will be effective AMP in-place to detect and mitigate the on-set of incipient age related degradation (vibration analysis and check seismic restraints inspections) - Life-cycle replacements will have effectively occurred (e.g. coil and EQ fan motor replacements) before the useful life of the component is exceeded to restore the component to a like-new condition. - The cooling function of the ACUs will not be required post phase 2 (Units Being Defueled and Dewatered).
Comp. EOL Year	2039
Section 5	Cost and Schedule
Cost	$0.00 **Details:** Costs for Recommendations – Plant EOL (2020): $0 All estimates are considered Class III (+30% and -20% accuracy)". 1. Complete Work Order 02113023 to replace the coils in 7-73110-ACU4. Assumption – The work order is scheduled and the status is currently at 'WORKING'. This work is expected to be completed by 2016 and is assumed to be funded through the current work order process.
CCA Update	Next CCA Update Year: 2026 CCA Update Frequency: N/A

Action List	**Type:** **Tracking Number:** Plant EOL (2020) - Complete Work Order 02113023 to replace the coils in 7-73110-ACU4 by 2016.

References List	**Reference Number:** P-SPM-73110-0465019, R003 1. SPMP Reactor Building Cooling System **Reference Number:** NK30-OSR-08131.02-00003 2. Operational Safety Requirements - Pickering NGS 5-8 Negative Pressure Containment System **Reference Number:** P-REP-01060-00002 3. Pickering Component Condition Assessment Project - System Transition Boundary Report (STBR) **Reference Number:** NK30-SCL-73110-00001, Rev: 001 4. System Classification List Reactor Building Heating & Cooling **Reference Number:** NK30-EQL-73110-0001 5. Environmental Qualification List Development Package for Boiler Room ACUs 5/6/7/8-73110-ACU1 – ACU6 **Reference Number:** N/A 6. Current Equipment Assessment Detail Report Dated 2-23-2016. **Reference Number:** NK30-EQA-05600-00007, Rev 03 7. EQA PART 1 Rewound Continuous Duty Induction Type Electric Motors 575 Volts / 3 Phases / 60 HZ

Revisions Summary	**Revision:** 000 Initial Issue

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Document Number: NK30-REP-73110-00041-R000

CCA: 011181 **System:** 0412 - Containment

Approvals

Approval applies to CCA and CCA Cost Schedule

Preparer:	**Brent Achtymichuk** Mechanical Engineer	**Date:**	12-JAN-2017
Verifier: SANJAY KRISHNAN FOR	**Frank Meaney** Senior Mechanical Engineer	**Date:**	12 JAN 2017
Approver: SANJAY KRISHNAN FOR	**TJ Thomasson** Project Manager	**Date:**	12 JAN 2017

Accepted by:

ONTARIO POWER GENERATION

ACCEPTED	✓
ACCEPTED AS NOTED	
REVISE AND RESUBMIT	
	1/30/2017
Signature	Date
Name:	Stan Whatmough
Dept:	AMSI

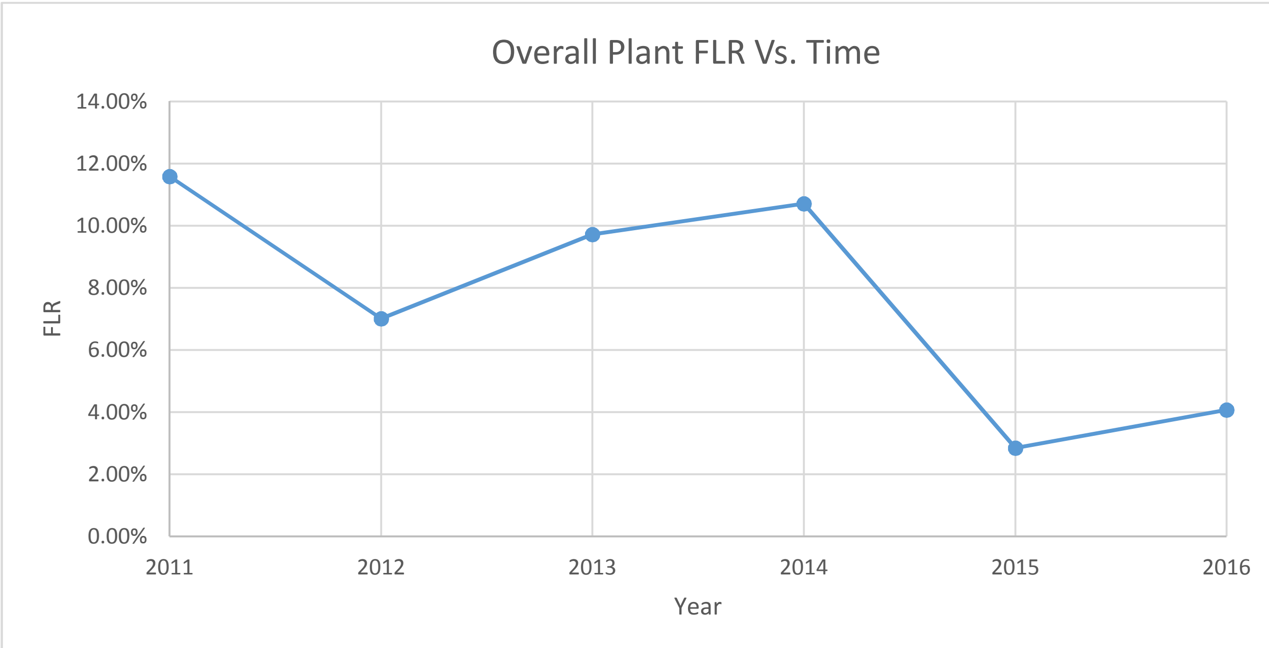
This acceptance does not relieve the contractor from responsibility for errors or omissions or from any obligations or liability under this contract.

Notes:

LICENSED PROFESSIONAL ENGINEER
M.B. Achtymichuk
100058241
12-JAN-2017
PROVINCE OF ONTARIO

TETRA TECH

APPENDIX D. FORCED LOSS RATE (FLR) DATA



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REVISE AND RESUBMIT	
Signature	**Date** 14 July 2016
Name: Mike Ruffolo, Manager	
Dept: Pickering Engineering - Aging Management & Strategic Initiatives	
OPG Proprietary	
Doc No.: P-REP-03680-00006 Rev: 000	
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Pickering NGS PSR2 Safety Factor 3 Report: Equipment Qualification (Seismic and Environmental)

PS112/RP/003 R01

July 13, 2016

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Revision Summary - For Amec Foster Wheeler Report PS112/RP/003

Rev	Date	Author	Comments
R00	May 6, 2016	R. Jayasundera, A. Johnstone	Initial issue for OPG review and comment.
R01	July 13, 2016	R. Jayasundera, A. Johnstone	Updated report addressing OPG comments on R00 Report.

EXECUTIVE SUMMARY

Ontario Power Generation (OPG) is performing a Periodic Safety Review (PSR) to support the possibility of continued operation of Pickering Nuclear Generating Station (NGS) beyond 2020. The PSR (referred to as "PSR2") is a subsequent PSR building on the review basis of earlier OPG Integrated Safety Reviews and other associated assessments. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

Part of PSR2 involves the preparation of Safety Factor reports for each of fifteen major topic areas. Safety Factor reports consist of:

- Assessing compliance against "Review Tasks" identified in the PSR2 Basis Document [1]. These Review Tasks are derived from Canadian Nuclear Safety Commission (CNSC) REGDOC-2.3.3, "Periodic Safety Reviews" [2] and International Atomic Energy Agency (IAEA) SSG-25, "Periodic Safety Review for Nuclear Power Plants" [3];

- Documenting assessments against applicable modern Laws, Regulations, Codes and Standards (L/R/C/Ss) as defined in Reference [1]; and

- Reviewing the effectiveness of applicable OPG Nuclear Programs that support the above assessments, through review of audit and self-assessment results.

The PSR2 review of Safety Factor 3, *Equipment Qualification (Seismic and Environmental)* (also referred to here as the "Equipment Qualification Safety Factor") is presented in this report. OPG Governance, Programs, Policies, Procedures, Instructions and Guidelines related to Pickering NGS equipment qualification (environmental and seismic) were reviewed for the ten PSR2 Review Tasks specified in Section 4.1 of this report. L/R/C/S and OPG Nuclear Program audit and self-assessment reviews for Safety Factor 3 were prepared per Sections 4.2 and 4.3, respectively. This report also includes a review of OPG commitments previously made to the CNSC, open CNSC action items, and exemptions granted by the CNSC since the current operating licence was issued (all related to Safety Factor 3), as well as identification and review of previously identified PSR1 gaps related to Safety Factor 3 (to ascertain the implications of extending Pickering NGS operation beyond 2020), per Section 4.4.

The results of the review of the Equipment Qualification Safety Factor are discussed in Section 5.0 of this report. The review has confirmed that the Pickering NGS equipment important to safety has been properly qualified and that this qualification is being maintained through an adequate programme of maintenance, inspection and testing. As discussed in Section 5.0, the review identified six gaps that will need to be addressed further as part of the PSR2 Global Assessment process.

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1.0 INTRODUCTION

Ontario Power Generation (OPG) is performing a Periodic Safety Review (PSR) to support the possibility of continued operation of Pickering Nuclear Generating Station (NGS) beyond 2020.[1] A comprehensive Integrated Safety Review (ISR) was completed for Pickering Units 5 through 8 in 2009 in support of refurbishment and continued operation. Pickering Units 1,4 integrated safety assessments were also performed for Pickering A Return to Service (PARTS) in support of approval to restart Units 1 and 4. In addition to these Pickering-specific studies, the 2013 Darlington ISR performed extensive code and standard reviews that were updated in relation to the versions that were assessed in the 2009 Pickering B ISR.[2] These previous ISRs are considered to constitute the first PSR completed for Pickering (referred to as "PSR1"). The current PSR (referred to as "PSR2") is a subsequent PSR building on the basis of earlier OPG integrated safety assessments through review of the various studies, assessments and licence renewals performed since PSR1. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

PSR2 will support and complement the licence renewal application for Pickering NGS going forward. Fifteen Safety Factors will be assessed as part of the PSR. The purpose of Safety Factor reviews is to confirm that the design, condition and operation of Pickering Units 1,4 and 5-8 (as well as common systems) will support continued safe operation for the period of PSR2 by:

- Assessing compliance against "Review Tasks" identified in the PSR2 Basis Document [1], which were derived from Canadian Nuclear Safety Commission REGDOC-2.3.3 [2] and IAEA SSG-25 [3];
- Documenting assessments against applicable modern Laws, Regulations, Codes and Standards (L/R/C/Ss) (as defined in Reference [1]); and
- Reviewing the effectiveness of applicable OPG Nuclear Programs that support the above assessments, through review of audit and self-assessment results.

As outlined in IAEA SSG-25 [3], the objective of the review of Equipment Qualification Safety Factor 3 is to: "determine whether plant equipment important to safety has been properly qualified (including for environmental conditions) and whether this qualification is being maintained through an adequate programme of maintenance, inspection and testing

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1 Currently, Pickering Units 5-8 are approved to operate to 247,000 Effective Full Power Hours. This operation limit is expected to be reached on some units in 2020. For the purposes of PSR2, OPG assumes operation of Pickering NGS for up to eight additional years, from 2020 until 2028. OPG will make a decision regarding the permanent shut down dates for the six reactors following the performance of a technical evaluation that will include PSR2, and will communicate it to the CNSC as required by the current Power Reactor Operating Licence.

2 Much of the compliance assessment and evaluation of Safety Factor health for the Darlington ISR is based on programs and practices that apply across OPG's nuclear operations. As a result, where Pickering is confirmed to follow the same nuclear programs and practices as were assessed for Darlington, the Darlington ISR programmatic conclusions are applicable to Pickering. As discussed in Section 1.0, an effectiveness review (effectiveness at Pickering NGS) of OPG programs used to demonstrate compliance with the PSR2 Assessment Basis will be conducted using recent audit and self-assessment results.

that provides confidence in the delivery of safety functions throughout the period of the PSR." REGDOC-2.3.3 [2] requires that: "The licensee shall conduct a PSR in accordance with this regulatory document for the period until the next PSR or, if applicable, until the end of commercial operation of the plant."

This report documents the results of the review of Safety Factor 3 for Pickering PSR2. The report is based on the OPG Governance, Programs, data, and material available up to January 15, 2016 which is the freeze date for PSR2.

2.0 SCOPE OF REVIEW

2.1 Review Task Assessments

The Pickering PSR2 Safety Factor 3 Review Tasks are defined in Reference [1]. Details of the derivation of these Review Tasks from CNSC REGDOC-2.3.3 and IAEA SSG-25 are shown in Reference [4]. The Safety Factor 3 Review Tasks are:

1) Confirm there exists a suite of engineering programs or processes to ensure equipment qualification requirements are met and documented.

2) Confirm equipment qualification has been adequately established for all service conditions expected during normal operation, anticipated operational occurrences and accident conditions. These service conditions are subdivided into environmental conditions and operational conditions. Environmental conditions include ambient temperature, pressure, humidity/steam, radiation, water/chemical sprays, fluid submergence, fire and seismic vibration. Operational conditions include process related conditions such as vibration, load cycling, electrical loading parameters, electromagnetic interference, mechanical loads and process fluid condition.

3) Perform an objective confirmation that the installed equipment is qualified to perform its Design Basis function for its operational life and that effective programs exist to monitor for timely maintenance or replacement, as required.

4) Confirm existence of a process for ensuring compliance with equipment qualification programs and of documented previous qualification measures taken to ensure qualification throughout the equipment's installed life (i.e., prescribed testing, calibration, maintenance, and parts replacement).

5) Confirm existence of a surveillance program and a feedback procedure to ensure aging degradation of qualified equipment remains insignificant.

6) Confirm existence of monitoring of actual environmental conditions and identification of 'hot spots' of high activity or temperature.

7) Confirm existence of an assessment that determines the effects of equipment failures on equipment qualification and appropriate corrective actions and/or safety improvements to maintain equipment qualification.

8) Confirm there is protection and adequate separation of qualified equipment from adverse environmental conditions.

9) Confirm physical condition and functionality capability of qualified equipment is being checked by walkdowns.

10) Confirm that changes to equipment classification have occurred, as required, as a result of major design modifications made since PSR1.

The methodology for the reviews is discussed in Section 3.1. Review Task findings are summarized in Section 4.1 of this Report.

2.2 L/R/C/S Reviews

The applicable Laws, Regulations, Codes and Standards relevant to the Equipment Qualification Safety Factor are identified in Reference [1] and are listed in Table 1 below. Table 1 also identifies the modern version and date of each L/R/C/S to be considered, the Safety Factor(s) to which each document is applicable, and the type of review that will be completed in PSR2.

All of the Safety Factor 3 L/R/C/S reviews are incremental in nature. The definition of an Incremental Review is as follows:

- Incremental Review: For L/R/C/Ss that have been reviewed in PSR1 but have had revisions since the last review, a topical review will be performed of the changes.

The methodology for the reviews is discussed in Section 3.2. A detailed compliance assessment for each L/R/C/S is provided in Appendix B of Reference [5]. Associated findings are summarized in Section 4.2 of this Report.

Table 1: L/R/C/Ss Reviewed for Equipment Qualification Safety Factor 3

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
L/R/C/Ss Referenced in Pickering NGS PROL 48.02/2018						
1	CSA N290.13	Environmental Qualification of Equipment for CANDU Nuclear Power Plants	N290.13-05	3, 4	Incremental	N290.13 addressed as part of Pickering B and Darlington ISRs
2	CSA N285.5	Periodic Inspection of CANDU Nuclear Power Plant Containment Components	N285.5-13	1, 2, 3, 4	Incremental	N285.5 addressed as part of Pickering B and Darlington ISRs
3	CSA N287.7	In-Service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants	N287.7-08	2, 3, 4	Incremental	N287.7 addressed as part of Pickering B and Darlington ISRs
4	CNSC RD/GD-210*	Maintenance Programs for Nuclear Power Plants	2012	3, 4	Incremental	S-210 and RD/GD-210 addressed as part of Darlington ISR
5	CNSC RD/GD-98	Reliability Programs for Nuclear Power Plants	2012	3, 4	Incremental	RD/GD-98 addressed as part of Darlington ISR and S-98 as part of Pickering B ISR

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
6	CNSC REGDOC-2.6.3*	Aging Management	2014	3, 4	Incremental	Transition plan in place and gap assessment between RD-334 and OPG Nuclear Integrated Aging Management governance performed by OPG
Additional L/R/C/Ss						
7	CSA N287.2	Material Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants	N287.2-08	1, 2, 3, 4	Incremental	N287.2 addressed as part of Pickering B and Darlington ISRs and PARTS
8	CSA N289.1	General Requirements for Seismic Design and Qualification of CANDU Nuclear Power Plants	N289.1-08	1, 3	Incremental	N289.1 addressed as part of Pickering B and Darlington ISRs
9	CSA N289.2	Ground Motion Determination for Seismic Qualification of Nuclear Power Plants	N289.2-10	1, 3	Incremental	N289.2 addressed as part of Pickering B and Darlington ISRs
10	CSA N289.3	Design Procedures for Seismic Qualification of Nuclear Power Plants	N289.3-10	1, 3	Incremental	N289.3 addressed as part of Pickering B and Darlington ISRs
11	CSA N289.4	Testing Procedures for Seismic Qualification of Nuclear Power Plants Structures, Systems, and Components	N289.4-12	1, 3	Incremental	N289.4 addressed as part of Pickering B and Darlington ISRs
12	CSA N289.5	Seismic Instrumentation Requirements for Nuclear Power Plants and Nuclear Facilities	N289.5-12	1, 3	Incremental	N289.5 addressed as part of Pickering B and Darlington ISRs

* Superseding documents to those currently in Pickering NGS PROL 48.02/2018.

2.3 Audit and Self-Assessment Reviews of OPG Programs

The OPG Nuclear Programs (N-PROGs) applicable to the Equipment Qualification Safety Factor are listed in Table 2 below. The methodology for the audit and self-assessment reviews is discussed in Section 3.3. The assessment results of each of the N-PROGs in Table 2 is provided in Appendix B, and findings are summarized in Section 4.3. It is noted that N-PROG-MP-0001, “Engineering Change Control”, N-PROG-MA-0004, “Conduct of Maintenance”, N-PROG-MP-0008, “Integrated Aging Management” and N-PROG-MP-0009, “Design Management” are examined in Appendix B in the context of any Pickering NGS seismic qualification related findings.

Table 2: OPG Programs Applicable to the Equipment Qualification Safety Factor

Document Number	Document Title
N-PROG-RA-0006 [6]	Environmental Qualification
N-PROG-MP-0001 [8]	Engineering Change Control
N-PROG-MA-0004 [9]	Conduct of Maintenance
N-PROG-MP-0008 [10]	Integrated Aging Management
N-PROG-MP-0009 [11]	Design Management

2.4 Additional Reviews

The PSR2 Safety Factor 3 Report includes a review of OPG commitments previously made to the CNSC, open CNSC action items, and exemptions granted by the CNSC since the current operating licence was issued (all related to Safety Factor 3). The Report also includes identification and review of previously identified PSR1 gaps related to Safety Factor 3 to ascertain the implications of extending Pickering NGS operation beyond 2020. The methodology for these reviews is described in Section 3.4.

In addition, any PSR2 gaps identified as a result of the Safety Factor 3 review which need to be addressed in other Safety Factor Reports are discussed in Section 4.4 of this report.

3.0 METHODOLOGY

The sub-sections below summarize the methodology used to assess Review Task and L/R/C/S compliance, and Nuclear Program effectiveness for the Equipment Qualification Safety Factor.

3.1 Review Tasks

As discussed earlier, the Safety Factor Review Tasks are derived from CNSC REGDOC-2.3.3 [2] and IAEA SSG-25 [3], taking into consideration the Review Tasks used in the Pickering B and Darlington ISRs (as derived in [4]).

For each Safety Factor 3 Review Task identified in Section 2.1, a confirmation of the existence of applicable OPG Governance, Programs, Policies and Procedures (as well as Instructions and Guidelines, as applicable) was performed. Compliance against Review Tasks is also assessed by reference to applicable Condition Assessments, safety analyses and operating experience, as required.

The Review Task assessments identify Compliances and Gaps as defined below:

- Compliance: Compliance indicates that either the safety requirement or the intent of the Review Task is met.

- Gap: A Gap indicates that the intent of the Review Task is not met.

3.2 L/R/C/S Reviews

The process to identify the modern L/R/C/Ss that are applicable to the PSR2 Assessment Basis involved first creating a broad list from multiple sources (potential candidate L/R/C/Ss) and then filtering it to identify those that are most significant and that are applicable to the PSR2 scope. The identification and selection criteria are detailed in Reference [1]. The result of the identification and selection process was a set of modern L/R/C/Ss that became part of the "PSR2 Assessment Basis".

PSR2 is focused on the extension of Pickering NGS operations beyond 2020, and will conduct reviews against a baseline of past PSR1 work. As a subsequent PSR, PSR2 focuses on changes in requirements, plant conditions, operating experience and new information. Since PSR2 is an update of previous ISRs, it incorporates reviews of L/R/C/Ss that have occurred as new versions have been issued. Since this assessment is a subsequent PSR, the focus is on identifying differences between what was previously assessed and what is now different within the current Pickering PSR2 Assessment Basis. In general, these differences relate to:

- More recent (new or revised) L/R/C/S versions than what was previously assessed as part of PSR1;[3]

- Safety significant differences between Pickering and Darlington, if the Darlington ISR is the basis for the earlier assessment;

- Implications of extending Pickering NGS operation beyond 2020; and

- Safety significant differences between Pickering Units 1,4 and Units 5-8.

L/R/C/Ss in the PSR2 Assessment Basis generally receive incremental reviews since PSR2 is an update of previous ISR assessments and clause-by-clause or high level reviews for the majority of the L/R/C/S in the PSR2 Assessment Basis have already been completed. Implementation plans (including gap analyses or code-over-code reviews) also exist for the latest editions of many L/R/C/Ss. As a result, incremental review is also used in circumstances where a L/R/C/S in the PSR2 Assessment Basis was not assessed in previous ISRs but an implementation plan currently exists for compliance.

The PSR2 incremental reviews in this Report include an assessment of the intent of recent changes to the L/R/C/Ss on a topic or subject-matter basis where there is potential to impact nuclear safety. Incremental reviews provide:

- A summary of the purpose of the L/R/C/S;

- Pertinent background information about the current revision of the L/R/C/S that is being considered;

- Identification of which Safety Factor(s) are applicable to the current revision of the L/R/C/S;

- A description of which version(s) of the L/R/C/S were assessed for PSR1 (i.e., Darlington ISR (for programmatic content), Pickering B ISR and PARTS code reviews);

- Identification of whether the current version of the L/R/C/S is an update of a previous version of the L/R/C/S that was assessed in PSR1 (and if so, a description of the major changes in the latest revision is provided as discussed below);

3 "New" refers to a code or standard that was not previously considered in the context of earlier assessments. "Revised" refers to an updated version of a code or standard that was previously considered in the context of earlier assessments. Where a document has a new number/type, but addresses the same topic from the same organization, it is a "revised", not "new", document (e.g., if a REGDOC replaces a CNSC G or RD document).

- An assessment of the applicability of PSR1 assessment findings (gaps and conclusions), including the implications of extending Pickering NGS operation beyond 2020 if any;
- An assessment of the applicability of assessment findings that address more recent (post-PSR1) editions of the L/R/C/S, including any implementation or transition plans that are already committed to by OPG; and
- Where PSR1 and post-PSR1 assessments are not sufficient to address changes in the latest edition of the L/R/C/S, an assessment of the changes from the previously assessed edition of the L/R/C/S (including identification of any safety significant PSR2 gaps which result).

The Safety Factor 3 L/R/C/S reviews identify Compliances and Gaps as defined below:[4]

- <u>Compliance</u>:
  - Where an Incremental review has been performed, Compliance indicates that the change in the safety requirement, per the topical review, is met.
  - Where a High Level review has been performed, Compliance indicates that the intent of the safety requirement is met. (Note: No High Level reviews were performed as part of Safety Factor 3.)
  - For Clause-by-Clause reviews of modern Laws, Regulations, Codes and Standards, Compliance indicates that the safety requirement is met. (Note: No Clause-by-Clause reviews were performed as part of Safety Factor 3.)
- <u>Gap</u>:
  - Where an Incremental review has been performed, a Gap indicates that the change in the safety requirement, per the topical review, is not met.
  - Where a High Level review has been performed, a Gap indicates that the intent of the safety requirement is not met. (Note: No High Level reviews were performed as part of Safety Factor 3.)
  - For Clause-by-Clause reviews of modern Laws, Regulations, Codes and Standards, a Gap indicates that the safety requirement is not met. (Note: No Clause-by-Clause reviews were performed as part of Safety Factor 3.)

[4] Safety Factor compliance assessments for Review Tasks and L/R/C/S reviews make use of: a) OPG Governance, Programs, Policies and Procedures which support the compliance arguments, b) Commitments previously made to the CNSC, open CNSC action items, and exemptions granted by the CNSC since the current operating licence was issued (Fukushima actions are included as appropriate as commitments or actions), c) Identification of previously identified ISR gaps related to each Safety Factor and the status of OPG's improvement plan(s) or other dispositions to address these, and d) Assessments and reviews performed since the PSR1 documents were completed.

The reviews assume that use of the word:

- “Shall" is used in an L/R/C/S to express a requirement, i.e., a provision that the licensee is obliged to satisfy in order to comply with the standard;
- "Should" is used to express a recommendation or that which is advised but not required;
- "May" is used to express an option or that which is permissible within the limits of the standard; and
- "Can" is used to express possibility or capability.

3.3 Audit and Self-Assessment Reviews

As discussed earlier, effectiveness reviews (effectiveness at Pickering NGS) of OPG programs used to demonstrate compliance with the PSR2 Assessment Basis were conducted, using recent audit and self-assessment results. Nuclear Program audit and self-assessment results were prepared for Safety Factor 3 by reviewing recent and applicable:

- OPG Program Health reports;
- OPG Nuclear Oversight independent performance based Program audits (typically performed in 1 to 5 year cycles) and self-assessments (typically performed on a yearly basis). This includes review of associated Station Condition Records and Action Requests to confirm that any findings have been completed; and
- CNSC “Type I” and “Type II” inspections of the effectiveness and performance of OPG programs, where called-up by OPG Program Health reports, audits or self-assessments.

The focus of these reviews was on effectiveness of the Programs at Pickering NGS, where specific information is available.

The list of Nuclear Programs to be assessed for each Safety Factor was derived from review of current OPG Governance, and has used the most recent version of these documents as of the PSR2 freeze date of January 15, 2016.

3.4 Additional Reviews

The PSR2 Safety Factor 3 assessment includes identification and review of previously identified PSR1 gaps related to Safety Factor 3 (as identified in the Pickering B and Darlington ISR Integrated Implementation Plans [12][13] and Pickering Units 5-8 Continued Operations Plan [14]) to ascertain the status of OPG's improvement plan(s) or other dispositions to address these and the implications of extending operation beyond 2020 (if any).[5]

A review was also performed of the following for Safety Factor 3 to determine if there are any impacts associated with operation of the Pickering Units past 2020:

- Commitments previously made to the CNSC in the R04 Pickering Licence Condition Handbook (LCH) [15];
- Open CNSC action items in the R04 Pickering LCH [15]; and
- Exemptions granted by the CNSC since the current operating licence was issued, per the R04 Pickering LCH [15] (Fukushima actions are included as appropriate as commitments or actions).

Any PSR2 gaps identified as a result of the Safety Factor 3 review which need to be addressed in other Safety Factor Reports are also discussed.

[5] PSR2 includes consideration and confirmation that the findings of PSR1 remain valid, as applicable, for the operation period. This includes assessment of PSR1 conclusions against implications resulting from extended operation. In particular, Pickering PSR1 results are applicable to PSR2 if there was a PSR1 gap that is still open, or if a closed PSR1 gap could be affected by extended operation. If so these gaps are carried forward into PSR2 for consideration in the Global Assessment. (When references to PSR1 are made, the source document is identified and the relevant text from that source document is summarized in the context of PSR2.) With respect to the Darlington ISR, much of the evaluation of Safety Factor health is based on programs and practices that apply across OPG's nuclear operations. As a result, Darlington ISR programmatic conclusions are applicable to Pickering PSR2 where Pickering is confirmed to follow the same nuclear programs and practices that were assessed for Darlington. Darlington PSR1 results are applicable to Pickering PSR2 if there are Darlington PSR1 gaps that are found to be relevant to Pickering PSR2.

4.0 REVIEW FINDINGS

4.1 Review Tasks

The sub-sections below provide an assessment of the adequacy of applicable OPG Governance, Programs, Policies and Procedures (as well as Instructions and Guidelines, as applicable) in demonstrating compliance against the Safety Factor 3 Review Tasks.

4.1.1 Review Task #1: Equipment Qualification Engineering Programs/Process

Confirm there exists a suite of engineering programs or processes to ensure equipment qualification requirements are met and documented.

4.1.1.1 Environmental Qualification

The Pickering NGS Environmental Qualification (EQ) Program has been established in accordance with the requirements of N-PROG-RA-0006, "Environmental Qualification" [6]. This program document provides direct authority to the following documents:

- N-PROC-RA-0051 "Environmental Qualification Lists" [16];
- N-PROC-RA-0044 "Environmental Qualification Assessment" [17]; and
- N-INS-03651-10023 "EQ Environment Monitoring and Equipment/Barrier Surveillance" [18].

The following text summarizes the contents of these program and procedure documents:

<u>N-PROG-RA-0006, "Environmental Qualification" [6]</u>

This program establishes the Environmental Qualification Program controls for OPG Nuclear and is applicable to all sites, including Pickering. The program establishes an integrated and comprehensive set of requirements that provide assurance that essential equipment can perform as required when exposed to harsh Design Basis Accident (DBA) conditions and that this capability is preserved over the life of the plant. Implementation of these program requirements provides the methodology, programmatic controls and interfaces for establishing and maintaining Environmental Qualification of equipment and components. N-PROG-RA-0006 Section 1.1.1 confirms that OPG's Environmental Qualification Program complies with CSA Standard N290.13-05 including Update 1, "Environmental Qualification of Equipment for CANDU Nuclear Power Plants", which is the most recent version of this Standard. Note: The L/R/C/S review of CSA N290.13 is addressed in Section 4.2 of this Report.

<u>N-PROC-RA-0051, "Environmental Qualification Lists" [16]</u>

This procedure defines work activities and instructions required to identify and document the equipment, components and qualification parameters applicable to the Environmental

Qualification Program. The process to identify equipment and components that must be environmentally qualified requires development of the following inputs:

- Environmental Qualification Design Guide;
- Harsh DBA List;
- Environmental Qualification Room Conditions Manual (RCM);
- Environmental Qualification Technical Basis Documents; and
- Environmental Qualification List Development Packages.

These inputs lead to the development of the Environmental Qualification List (EQL), which identifies safety-related equipment and components that may be subjected to harsh environment conditions resulting from a DBA in which they are required to function, or not fail, and have a potential failure mode caused by the environment.

<u>N-PROC-RA-0044, "Environmental Qualification Assessments" [17]</u>

The Environmental Qualification Assessments (EQAs) ensure that equipment subject to Environmental Qualification Program requirements are qualified for the environmental and operating service conditions and mission times for which they operate. Qualification is established through testing, analysis, Operating Experience (OPEX), ongoing qualification or a combination of these methods. EQAs identify equipment specific design, configuration, maintenance and procurement requirements necessary to establish and maintain the qualified status of equipment.

<u>N-INS-03651-10023, "EQ Environment Monitoring and Equipment/Barrier Surveillance" [18]</u>

This instruction outlines the requirements for Environmental Qualification environment monitoring, equipment and barrier surveillance as applicable to the station. For example:

- Environments where EQL equipment and components are installed are monitored to ensure values used in qualification are conservative or justification must be approved for monitoring a subset of rooms.
- Surveillance is completed to ensure installed configuration of electrical EQL equipment conforms to Environmentally Qualified related design and configuration requirements specified in EQAs.
- Periodic inspection and maintenance of Environmental Qualification barriers are completed to ensure their integrity throughout the life of the plant.

4.1.1.2 Seismic Qualification

Seismic qualification of Pickering A was established following the design and construction stages. Seismic qualification of Pickering B was established during the design and construction stages.[6]

There are a number of engineering programs and procedures used to maintain the Seismic Qualification of safety related SSCs. The programs and procedures that ensure Seismic Qualification is maintained include the following:

- N-PROG-MP-0009, "Design Management" [11];
- N-PROG-MP-0001, "Engineering Change Control" [8];
- N-PROG-MA-0004, "Conduct of Maintenance" [9];
- N-PROG-MP-0008, "Integrated Aging Management" [10];
- N-PROC-MA-0024, "System Performance Monitoring" [19]; and
- N-PROC-MA-0031, "Protection of Seismic Equipment and Routes" [20]

The specific applications for Seismic Qualification as captured in these programs are described below:

<u>N-PROG-MP-0009, "Design Management" [11]</u>

Engineering design changes maintain seismic design measures. Section 1.2.4 of this program specifies that seismic requirements are to be incorporated as part of the design basis and included in design inputs.

<u>N-PROG-MP-0001, "Engineering Change Control" [8]</u>

Section 1.9.1.2 (d) of this program requires that permanent design modifications to seismically qualified equipment be subjected to the stakeholder review process to ensure that the Seismic Qualification is not altered by a proposed design change. This program requires that during the Non-Identical Component Replacement (NICR) process, the Seismic Qualification of components must be maintained. Also, Section 1.6.2 of N-PROG-MP-0001 requires engineering to perform an engineering evaluation, in accordance with N-INS-08173-10048 "Item Equivalency Evaluation" [21], to determine if a prospective replacement meets all of the original design requirements (i.e., while performing an Item Equivalency Evaluation (IEE), qualified staff shall review the items design parameters,

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[6] As outlined further in Section 4.1.2.2 and 4.1.3.2 of this report, following the design and construction phase of Pickering A, the station was seismically assessed as part of the Seismic Margin Assessment and the necessary upgrades were made for those SSCs which belong to the seismic success path. For Pickering B however, Seismic Qualification was established during the design and construction phase. For the purposes of this report, when discussing the seismic status for both Pickering A and B, the generic term "Seismic Qualification" will be used despite the difference in qualification approach.

including requirements for seismic parameters during both normal operation and post-accident conditions).

N-FORM-10959, "Design Scoping Checklist", Item 3.19, "Seismic Requirements", is the mechanism within the Engineering Change Control (ECC) process which ensures that Seismic Qualification of SSCs is maintained. Item 3.19 ensures that the appropriate actions are taken when a modification will impact: a) Seismically qualified SSCs, b) SSCs where the seismic requirements in the National Building Code (NBC) or Ontario Building Code (OBC) apply, c) Non-seismically qualified systems or components in a station containing seismically qualified systems or components, d) Seismic routes, including entry or exit from doors on the route, or e) SSC for beyond design basis seismic effects.

<u>N-PROG-MA-0004, "Conduct of Maintenance" [9]</u>

This program establishes safe, uniform, and efficient maintenance practices at OPG Nuclear sites including Pickering. It ensures that effective implementation and control of maintenance activities are achieved by instituting high standards, providing a professional environment and sufficient resources, monitoring and assessing performance, and holding personnel accountable for their performance.

Section 1.2.4 of N-PROG-MA-0004 outlines precautionary measures to counter incidents that could impact the operation of seismically qualified equipment. It refers to N-PROC-MA-0031, "Protection of Seismic Equipment and Routes" [20] which sets criteria for controlling unsecured equipment and materials adjacent to seismically qualified SSCs in seismic areas and seismic routes (detailed below).

<u>N-PROG-MP-0008, "Integrated Aging Management" [10]</u>

The objective of the Integrated Aging Management (IAM) program is to ensure the condition of critical Nuclear Power Plant equipment is understood and that required activities are in place to ensure the health of these components and systems while the plant ages. The program also requires preparation of life cycle plans and condition assessments for critical plant equipment.

Section 1.3.11 of this program document states that the assessment of equipment condition shall consider established limits for the applicable degradation mechanisms as well as all applicable design requirements, including Seismic Qualification, in determining equipment condition.

<u>N-PROC-MA-0024, "System Performance Monitoring" [19]</u>

This procedure provides a consistent and comprehensive process for the designated system engineers to ensure effective monitoring, maintenance and enhancement of system performance and reliability.

Section 1.6.3 (h) requires that the acceptance band for system performance indicators or parameters have sufficient margin to allow for timely proactive intervention before component or system failures occur and are consistent with the requirements for Seismic Qualification.

N-PROC-MA-0031, "Protection of Seismic Equipment and Routes" [20]

This procedure establishes criteria to control unsecured equipment and material adjacent to seismically qualified SSCs and seismic routes, thereby ensuring operation, maintenance, modification or outage activities do not place seismically qualified SSCs at risk. The procedure includes guidelines for application of equipment restraints and separation distances to prevent interactions between unsecured equipment and material and seismically qualified SSCs during an earthquake.

4.1.1.3 Conclusion

The conclusion of this Review Task assessment is that there exists a suite of engineering programs or processes to ensure that environmental and seismic equipment qualification requirements are met and documented. The intent of Review Task #1 is met and therefore Pickering NGS is compliant.

4.1.2 Review Task #2: Establishment of Equipment Qualification Service Conditions

Confirm equipment qualification has been adequately established for all service conditions expected during normal operation, anticipated operational occurrences and accident conditions. These service conditions are subdivided into environmental conditions and operational conditions. Environmental conditions include ambient temperature, pressure, humidity/steam, radiation, water/chemical sprays, fluid submergence, fire and seismic vibration. Operational conditions include process related conditions such as vibration, load cycling, electrical loading parameters, electromagnetic interference, mechanical loads and process fluid condition.

4.1.2.1 Environmental Qualification

N-PROG-RA-0006, "Environmental Qualification" [6] defines the Environmental Qualification Program as a documented demonstration that equipment and components are capable of performing safety-related functions when subjected to environmentally harsh conditions resulting from DBAs. All equipment that is required to be Environmentally Qualified have EQAs prepared in accordance with N-PROC-RA-0044, "Environmental Qualification Assessment" [17]. The EQA is the design assurance document that demonstrates the capability of environmentally qualified equipment and components to perform their safety-related function(s) under the environmental stresses resulting from DBAs.

EQAs also dictate procurement, field configuration and maintenance activities necessary to maintain the equipment during its installed life to ensure operability. Operability is demonstrated through testing, analysis, OPEX or a combination of these methods.

Environmental Service Conditions

The normal and post-accident environmental service conditions for which equipment is qualified (temperature, pressure, radiation, humidity, flooding and chemical conditions) have been documented in the Pickering NGS Environmental Qualification RCMs (NA44-

MAN-03651-10001, "Environmental Qualification Room Conditions - Pickering A" [22] and NK30-MAN-03651-10001, "Environmental Qualification Room Conditions - Pickering B" [23]). The RCMs were prepared in compliance with N-INS-03651-10003, "Preparation of the Environmental Qualification Room Conditions Manual" [24], are based on safety analysis consistent with the Safety Report, and contain normal and accident service conditions for use in preparing EQAs. The RCMs contain the normal and limiting post-accident environmental conditions in the various rooms and areas inside and outside containment.

The accident transients and dose information inside containment are conservative, being derived from analytical simulations of a low probability large Loss of Coolant Accident (LOCA) event in conjunction with an impairment of the Emergency Coolant Injection (ECI) System. The accident conditions outside containment are bounded by a postulated Main Steam Line Break (MSLB) event.

Fire Protection

In terms of fire related environmental conditions (note, as per N-PROG-RA-0006 [6], the scope of the Environmental Qualification Program does not include fire protection) the Power Reactor Operating Licence requires compliance with the requirements of CSA N293, "Fire Protection for Nuclear Power Plants" which is defined in OPG program document N-PROG-RA-0012, "Fire Protection" [25]. The engineering evaluations which have assessed the station against the requirements of CSA N293 to ensure safe shutdown capability in the event of fire in any plant location include:

- Fire Protection Code Compliance Review: NA44-REP-71400-10001, "Pickering Nuclear Generation Station A Fire Protection Code Compliance Review" [26] and NK30-REP-71400-10001, "Fire Protection Code Compliance Review Pickering Nuclear Generating Station B" [27].

- Fire Hazard Assessment: NA44-REP-71400-10003, "Fire Hazard Assessment - Pickering A Nuclear Generating Station" [28] and NK30-REP-71400-10002, "Fire Hazards Assessment - Pickering B Nuclear Generating Station" [29].

- Fire Safe Shutdown Analysis: NA44-REP-71400-00023, "Fire Safe Shutdown Analysis - Pickering A Nuclear Generating Station" [30] and NK30-REP-71400-00001, "Fire Safe Shutdown Analysis - Pickering B Nuclear Generating Station" [31].

Operational Service Conditions

When environmentally qualifying equipment, operational service conditions that can have a demonstrably deleterious effect on the equipment are included in the testing and qualification of equipment as defined in the Environmental Qualification Program N-PROG-RA-0006, "Environmental Qualification Program" [6]. Operational service conditions that are reviewed include conditions such as temperature (this accounts for ambient temperature and heat rise due to the process fluid temperature as applicable), pressure, radiation, voltage, current frequency, load (e.g., mechanical loading), and frequency and magnitude of cycling.

Qualification requirements have been primarily specified in the design phase through imposition of technical specifications for equipment that have required demonstration of qualification by test. For example, the technical specification for electrical and electronic components and assemblies requires immunity (i.e., qualification) against electromagnetic interference, electrical transient interference, electrostatic discharge and external magnetic fields. As per N-INS-00700-10007, "Preparation of Modification Design Requirements" [32], N-DG-60407-10000 R000, "Guidelines for Electromagnetic Compatibility Test" [33] should be consulted and referenced during the preparation of Modification Design Requirements (MDR) for engineering changes to assist in identifying Electromagnetic Interference (EMI) and Radiofrequency Interference (RFI) design requirements. Design Guide N-DG-60407-10000 R000 [33] (which is applicable to all OPG Nuclear equipment that are either susceptible to and/or emits electromagnetic interferences) provides background and guidance for specifying requirements for electromagnetic compatibility tests to be applied to electrical/electronic equipment and systems for use within the plant environment. The recommended tests in N-DG-60407-10000 R000 are based primarily on TR-102323 Rev. 3, "EPRI Guidelines for Electromagnetic Interference Testing of Power Plant Equipment" together with operating experience from within OPG. By following the EPRI Guideline, the user fills out the template provided in the document which summarizes the applicable tests and associated information. The completed template is then included as part of an Engineering Specification.

N-PROG-MA-0004, "Conduct of Maintenance" [9] requires surveillance activities to be carried out, which include vibration monitoring. This assures early detection of deteriorating equipment conditions in order to ensure safe, reliable and economical operation of station equipment. The Environmental Qualification Surveillance Program is further detailed in Review Task 5.

Aging effects on equipment caused by operational stressors (e.g., vibration and mechanical loading) are addressed through application of Preventive Maintenance, Predictive Maintenance and Aging Management Programs which are discussed in detail in the Pickering PSR2 Safety Factor 4 (Aging) Report.

4.1.2.2 Seismic Qualification

Seismic Qualification for Pickering A was established by analysis. When the station was designed, no explicit seismic design standards for nuclear power plants were available and the seismic design provisions of the 1965 National Building Code of Canada were adopted for seismic design of the nuclear structures. Although the Pickering A systems were not originally required or designed to be Seismically Qualified, the equipment was subsequently evaluated and a Safe Shutdown Equipment List was defined by the 1998 Seismic Margin Assessment (SMA) [34]. A Probabilistic Risk Assessment (PRA) based SMA was then completed in 2013 [35] and a Seismic Equipment List was completed which identifies the SSCs required for the seismic success path.

Seismic qualification of Pickering B was established during the design and construction phases. Nuclear Safety Design Guide DG-30-68000-2, "Seismic Qualification of Safety Related Systems" [36] defines the fundamental seismic design requirements for safety related systems. It establishes the basis of Seismic Qualification and identifies those systems which are required to be qualified to permit execution of the basic safety functions following the occurrence of a low probability severe earthquake at the station site. This Design Guide also describes the two seismic categories (Design Basis Earthquake and Site Design Earthquake) that define functional requirements during or following an earthquake, as well as the levels of seismic excitation that structures or equipment must withstand. Acceptable methods for demonstrating qualification are also listed. The requirements of the Nuclear Safety Design Guides were implemented in design output documents such as Design Manuals, System Design Requirements, Technical Specifications and Specification Data Sheets for equipment, and System Flow Diagrams. More recently, the information related to the Pickering site seismic hazard has been included in the Pickering B PRA based SMA [37].

The NA44-SCL-USI/SCI and NK30-SCL-USI/SCI "System Classification List" are produced in accordance with the requirements of CSA N285.0, "General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants". These lists identify the seismic qualification (as defined by DG-30-68000-2 "Seismic Qualification of Safety Related Systems" [36]) for applicable pressure boundary systems.

4.1.2.3 Conclusion

The conclusion of this Review Task assessment is that environmental and seismic equipment qualification has been adequately established for all service conditions expected during normal operation, anticipated operational occurrences and accident conditions. The intent of Review Task #2 is met and therefore Pickering NGS is compliant.

4.1.3 Review Task #3: Qualification of Installed Equipment

Perform an objective confirmation that the installed equipment is qualified to perform its Design Basis function for its operational life and that effective programs exist to monitor for timely maintenance or replacement, as required.

4.1.3.1 Environmental Qualification

OPG Environmental Qualification Program N-PROG-RA-0006, "Environmental Qualification Program" [6] provides assurance that equipment and components will perform their safety related functions when exposed to harsh environmental conditions resulting from a DBA. The scope of the Environmental Qualification Program includes the following:

- Defining the list of equipment and components required to be environmentally qualified and maintaining the list current with plant licencing basis, design basis and service conditions;
- Ensuring auditable proof of performance under harsh DBA conditions is developed and maintained current with plant licencing basis, design basis, service conditions, and configuration; and
- Providing assurance that equipment and components within the Environmental Qualification Program are purchased, stored, installed, configured, maintained, monitored, and replaced to ensure qualified status is preserved.

Preparation of the EQL uses input from the following documents:

- Environmental Qualification Design Guide;
- Harsh DBA List;
- Environmental Qualification RCM;
- Environmental Qualification Technical Basis Documents; and
- EQL Development Packages.

Through these documents, the required safety function of the equipment requiring qualification is fully defined. The process for identifying equipment subject to Environmental Qualification is shown graphically in Figure 1 below (as documented in N-PROG-RA-0006 [6]).

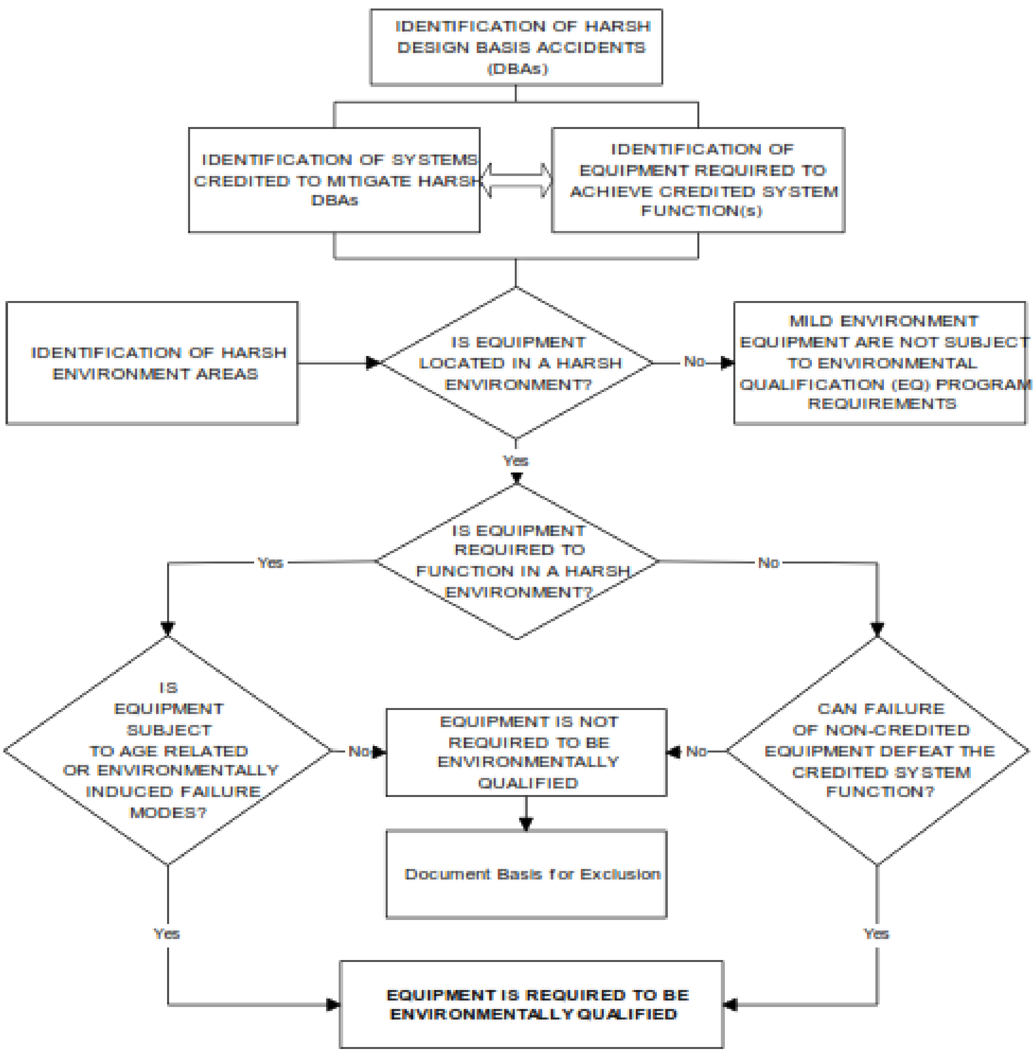

Figure 1: Criteria for Identifying Equipment Subject to Environmental Qualification Requirements

The EQA is a design assurance document that demonstrates the capability of environmentally qualified equipment and components to perform their safety-related function(s) under the environmental stresses resulting from DBAs. A typical EQA includes a summary of qualification levels by equipment tag, including required performance parameters. It also specifies configuration, maintenance, replacement, and purchasing requirements on a plant-specific equipment basis. EQAs are specifically designed to facilitate their use by plant engineering, maintenance and procurement personnel who plan, schedule, and maintain equipment and associated procedures and processes.

The EQA provides an evaluation of test and analysis documentation which establishes the basis for qualification, including a quantitative summary of any conditions of qualification. Additionally, it dictates Environmental Qualification configuration and maintenance requirements, and provides a quick overview of the complete list of potential conditions and limitations of qualification that have been identified which may be applied to plant-specific equipment. The evaluation is based either on plant and equipment-specific conditions, or a set of environmental conditions which envelop the worst case normal and DBA conditions postulated to occur at Pickering NGS.

Preventive maintenance requirements as well as any limitations on the life of the equipment are documented in the EQAs. In establishing maintenance intervals, margins are included to reduce the probability of exceeding the qualified life of components. These requirements are managed by N-PROG-MA-0004, "Conduct of Maintenance" [9] which provides the requirements for managing the execution of preventive maintenance. For components under the scope of the Environmental Qualification Program, extensions to maintenance intervals require acceptance from the Environmental Qualification Single Point of Contact (SPOC) to ensure compliance with the Environmental Qualification Program. In cases where the required maintenance date is missed, a Station Condition Record (SCR) is required to document the event, assess the impact and correct the situation. Individual equipment records for qualified equipment are recorded in the Asset Suite in accordance with N-GUID-03651-10002 "Environmental Qualification Asset Suite Data Conventions" [38]. This guideline establishes the convention used for storing equipment, component, and material data.

The Environmental Qualification Program N-PROG-RA-0006 "Environmental Qualification Program" [6] requires that whenever equipment on the EQL approaches the end of its qualified life, action must be taken to sustain the qualified status of that equipment regardless of what the station current life is taken to be. Hence, all EQAs will need to be re-assessed to ensure qualification is maintained in order to support continued operation of Pickering NGS beyond 2020. The current Environmentally Qualified life of all Pickering NGS SSCs may not necessarily extend to 2028 and a full review of Environmentally Qualified life-limited components impacted by operation past 2020 will need to be undertaken prior to life extension of Pickering NGS. This is therefore identified as a gap exists for Pickering PSR2 (**Pickering PSR2 Gap SF3-1**). It is noted that work to address this gap is currently underway as part of the update of Pickering NGS Condition Assessments for safety related systems and Life Cycle Management Plans for major components.

4.1.3.2 Seismic Qualification

Pickering B was designed to be seismically qualified through the implementation of the overall safety design requirements documented in Engineering Design Guide DG-30-68000-2 [36]. This formally established the design requirements for safety related systems to meet the seismic aspects of Canadian nuclear safety principles specified in CSA N289.1, "General Requirements for Seismic Design and Qualification of CANDU Nuclear Power Plants" (Table 2-2 of the Pickering B Safety Report, Part 2 [39] contains a list of seismically qualified systems). Note: The L/R/C/S review of CSA N289.1 is addressed in Section 4.2 of this Report.

Section 6.4 of DG-30-68000-2 [36] specifies the qualification method for civil, structural and mechanical equipment. The overall seismic design requirement is for the station to be resistant to the effects of a severe earthquake such that execution of the four critical safety functions is assured (i.e., safe shutdown of the reactors; decay heat removal; containment button up; and monitoring of critical safety parameters). The seismic design guide defines the scope, earthquake design levels, the extent to which systems must remain operational and the methodology of the seismic design of the station. It lists by Universal Subject Index (USI) the systems that are required to be qualified and the applicable earthquake level, and indicates the seismic classification category (i.e. Category A or B). Seismic Qualification of Safety System Instrumentation is covered in a separate specification [40].

The implemented seismic design requirements appear in various design documents (e.g., Design Manuals, Design Requirements, Drawings, Design Flow Diagrams, Design Reports, Qualification Test Reports, Technical Specification Data Sheets and Procurement Specifications). Correspondence between Ontario Hydro and Atomic Energy Control Board (AECB) in 1978 indicates that the regulator accepted the Pickering B seismic design [41]. Evidence that the overall design is in conformance to relevant codes, standards and regulations and in accordance with the Safety Report is indicated in correspondence with the AECB [42]. The Safety Design Matrix "Operation after an Earthquake" discusses the provisions made at Pickering Units 5-8 to cater to a Design Basis Earthquake. It concluded that: "The safety precautions described are more than adequate to ensure that the generating station poses negligible additional risk to the public after the worst credible seismic event".

As outlined in Section 2.3 of the Pickering A Safety Report – Part 2 [43], the Pickering Unit 1,4 SSCs required to perform the above mentioned critical safety functions during and following an earthquake were not originally required to be seismically qualified. However, the common containment structures (Reactor Building, Pressure Relief Duct and Vacuum Building) were designed to exceed the National Building Code 1965 seismic design provisions and were subsequently confirmed analytically to meet Pickering B Design Basis Earthquake seismic design requirements. The Pickering A Seismic Margin Assessment [34] evaluated the seismic capacity of the Pickering A SSCs required to perform the critical safety functions and identified necessary seismic upgrades. Seismic success path SSCs are identified in Reference [34], and are summarized in Table 32 of the Pickering A Safety Report Part 2 [43]. Adjacent or associated (including Unit 0) SSCs whose failure might impair functionality of seismic success path systems were also evaluated. NA44-DG-03650-00001, "Seismic Design Guide for Seismic Qualification of Pickering NGS A Success Path Structures, Systems and Components" [44], specifies the acceptable seismic design and qualification criteria for maintenance, replacements and modifications to Pickering A seismic success path SSCs.

The SMA methodology is based on documented performance of power and heavy industrial plant components in major earthquakes, on seismic fragility analysis and on seismic testing of components. Walkdown screening criteria evaluate seismic demand at the component location, anchorage adequacy, seismic interactions as well as equipment type-specific seismic susceptibilities. Where components do not satisfy screening criteria, seismic analysis methods consistent with the seismic margin assessment methodology have been utilized.

Low seismic capacity components were replaced; structure and component anchorage were upgraded and potential seismic interactions were dispositioned for the return to service of Units 1,4 [43]. A listing of these seismic upgrades includes:

- Boiler Room shield wall upgraded;
- Switchgear and panel anchorage upgraded;
- Anchoring of heat exchangers in Vacuum Building basement enhanced;
- Standby Generator oil pump house masonry block wall reinforced;
- Supports for Standby Generator batteries enhanced;
- Supports for Class I batteries enhanced;
- Main Control Room and Control Equipment Room panel anchorage upgraded;
- Control Equipment Room structural upgraded;
- Rerouted emergency air supply for airlocks;
- Anchored temporary breathing air system near Emergency Low Pressure Service Water pumps;
- Enhanced supports for bleed valves;
- Enhanced spring hangers;
- Restrained gas bottles and fire extinguishers;
- Improved supports for Reactor Building ACUs;
- Reviewed proximity issues for various valves;
- Improved Deaerator Storage Tank anchoring system;
- Provided lateral restraints for the High Pressure Feedwater Heaters;
- Reviewed the fuel channel positioning assembly rod;

- Reviewed anchor bolts used on the Fuelling Machine support column;
- Upgraded mercoid switches;
- Upgraded relays;
- A Seismic AIM was issued and staff was trained to respond to a seismic event;
- Periodic testing of systems credited in seismic analysis;
- Administrative controls added;
- Pressure Relief Duct analysis completed; and
- Piping supports on recirculating cooling water piping added.

Evaluation using the SMA methodology provides an equivalent level of assurance of seismic capacity to that provided by seismic qualification by conventional seismic design methods (per CSA N289.3, "Design Procedures for Seismic Qualification of Nuclear Power Plants) and seismic testing (per CSA N289.4, "Testing Procedures for Seismic Qualification of Nuclear Power Plant Structures, Systems and Components" and IEEE 344, "Standard for Seismic Qualification of Equipment for Nuclear Power Generating Stations"). Note: The L/R/C/S reviews of CSA N289.3 and N289.4 are addressed in Section 4.2 of this Report.

To ensure effective monitoring, maintenance and enhancement of system performance and reliability, N-PROC-MA-0024, "System Performance Monitoring" [19] provides a consistent and comprehensive process for System Engineers. Field walkdowns are the mechanism used for performing a field evaluation of system performance and forms an essential part of System Performance Monitoring. Walkdowns by System Engineers complement the routine inspections done by operations staff. Where adverse conditions are noted during walkdowns, appropriate corrective actions are initiated to restore equipment functionality. As per Section 1.8.1.4 of N-PROC-MA-0024 [19], the System Engineers are required to observe general conditions such as seismic route concerns (refer to N-PROC-MA-0031, "Protection of Seismic Equipment and Routes" [20]). This procedure is discussed in detail in Section 4.1.1.2 of this report. N-PROC-RA-0022, "Processing Station Condition Records" [45] provides a consistent reporting and evaluation process for identifying adverse conditions at OPG Nuclear. Upon identifying an adverse condition that directly impacts the ability of the station to operate safely, or one that represents an actual or potential operability concern, or one that represents a condition reportable under the operating licence, an SCR is initiated to document the condition. A corrective action plan is then developed to correct the adverse condition, as required. All actions are tracked to completion under the OPG Action Tracking system.

4.1.3.3 Conclusion

The installed equipment is seismically qualified to perform its Design Basis function for its operational life and effective programs exist to monitor for timely maintenance or replacement. However, the current Environmentally Qualified life of all Pickering NGS SSCs may not necessarily extend to 2028 and a full review of Environmentally Qualified life-limited components impacted by operation past 2020 will need to be undertaken prior to life extension of Pickering NGS. This is therefore identified as a gap for Pickering PSR2 (**Pickering PSR2 Gap SF3-1**). It is noted that work to address this gap is currently underway as part of the update of Pickering NGS Condition Assessments for safety related systems and Life Cycle Management Plans for major components.

4.1.4 Review Task #4: Compliance with Applicable Programs

Confirm existence of a process for ensuring compliance with equipment qualification programs and of documented previous qualification measures taken to ensure qualification throughout the equipment's installed life (i.e., prescribed testing, calibration, maintenance, and parts replacement).

4.1.4.1 Environmental Qualification

The processes for ensuring Environmental Qualification of Pickering NGS under the Environmental Qualification Program has been discussed in detail in Section 4.1.3. Measures are taken via interfacing programs to ensure qualification throughout the equipment's installed life. These interfacing programs are provided in N-PROG-RA-0006 "Environmental Qualification" [6] and include testing, calibration, maintenance and parts replacement.

OPG Training Program N-PROG-TR-0005, "Training" [46] provides the structure, processes and tools for defining, developing, implementing, documenting, assessing and improving required training. Engineering Support Personnel are required to take the Program Element Identification (PEL ID) 3875 "Introduction to EQ Engineering" and PEL ID 65799 "Environmental Qualification Refresher for Engineering" as part of the qualification "Engineering Support Personnel – Core Training" (Qualification Identification 6168) [47].

Maintenance staff receive training for PEL ID 3874 "EQ for Maintainers and Operators" and Supply Chain staff receive training for either PEL ID 3875 "Introduction to EQ Engineering" or PEL ID 3873 "EQ for Supply Chain". These training courses provide the high level EQ requirements expected from all staff. There are also more specialized training courses for staff performing the role of Environmental Qualification SPOC. N-PROG-AS-0002, "Human Performance" [48] provides initiatives under the Human Performance Program to ensure compliance with procedures and processes. Human performance tools such as pre-job and post-job briefings, self-check and peer verification are used to ensure that all required Environmental Qualification procedures and processes are followed.

The Training and Human Performance Programs are supplemented by regular Environmental Qualification Program self-assessments and audits as defined in N-PROG-RA-0010, "Independent Assessment" [49] and N-PROC-RA-0097, "Self-Assessment and Benchmarking" [50] which are conducted to ensure the Environmental Qualification Program is being sustained. The Environmental Qualification Program is audited with an interval of no greater than five years by Nuclear Oversight (NO) while self-assessments are conducted annually. Note: Recent audit and self-assessment results for N-PROGs applicable to the Equipment Qualification Safety Factor are addressed in Section 4.3 of this report.

Environmental Qualification issues identified by staff are documented through the SCR process and are reviewed on a daily basis. Relevant SCRs are also reviewed during the preparation of Environmental Qualification Program health reports and trends are identified. These activities ensure that problems are identified and corrected in a timely manner.

4.1.4.2 Seismic Qualification

The following engineering procedures describe processes for maintaining the original seismically qualified design configuration:

- N-PROC-MP-0090, "Modification Process" [51];
- N-PROC-MP-0047, "Design Verification" [52];
- N-PROC-MA-0031, "Protection of Seismic Equipment and Routes [20];
- N-PROC-MA-0024, "System Performance Monitoring" [19]; and
- N-PROC-TR-0008, "Systematic Approach to Training" [53].

These procedures are discussed below:

<u>N-PROC-MP-0090, "Modification Process" [51]</u>

This procedure provides guidelines for all engineering disciplines on maintaining and controlling the configuration of equipment when implementing design modifications in a system. For design modifications which involve a NICR process, the Seismic Qualification of components must be maintained. Also, while performing an IEE, qualified staff shall review the item's design parameters, including requirements for seismic parameters during both normal operation and post-accident conditions.

N-FORM-10959, "Design Scoping Checklist", Item 3.19 provides a list of issues which must be addressed to ensure Seismic Qualification is maintained for Non-Identical Component Replacement modifications. Appendix E of N-PROC-MP-0090 provides guidelines for the "Field Initiated Changes" and Section E.3.2 prohibits minor field-initiated changes from being applied to seismically qualified safety-related systems.

<u>N-PROC-MP-0047, "Design Verification" [52]</u>

This procedure provides direction for a systematic and uniform approach for design verification related to modification, licencing, operating and decommissioning activities at OPG. Section 1.2.1.2 of N-PROC-MP-0047 requires that design verification of seismically qualified equipment be performed only by qualified personnel. Section 1.2.4 also requires that if seismic testing is part of qualification testing that it demonstrates that the equipment or system meets the specified seismic requirements. It allows such testing to be performed in lieu of, or in addition to, analysis. Appendix C of N-PROC-MP-0047 provides guidelines for the design engineer to specify installation and operational requirements in the design of seismically qualified equipment.

N-PROC-MA-0031, "Protection of Seismic Equipment and Routes" [20]

This procedure receives its authority from N-PROG-MA-0004, "Conduct of Maintenance" [9], which has been discussed in detail in Section 4.1.1.2 of this report. N-PROC-MA-0031 is discussed in detail in Section 4.1.1.2 of this report.

N-PROC-MA-0024, "System Performance Monitoring" [19]

The system performance monitoring process associated with N-PROC-MA-0024 is discussed in detail in Section 4.1.1.2 of this report.

N-PROC-TR-0008, "Systematic Approach to Training" [53]

Training is provided to ensure that engineering, operations and maintenance staff are aware of station requirements (e.g. Seismic Qualification) while performing their respective duties. These duties may include, but are not limited to, ongoing testing, calibration, maintenance, and replacement of seismically qualified equipment. OPG Computer Assisted Learning Course 64196, "Seismic Qualification for Engineers", as well as ECC training, is part of the mandatory basic engineering qualification training for System Engineers, Design Engineers and Project Engineers.

Similar to Environmental Qualification, the Training and Human Performance Programs are supplemented by regular Program self-assessments and audits as defined in N-PROG-RA-0010, "Independent Assessment" [49] and N-PROC-RA-0097 "Self-Assessment and Benchmarking" [50] which are conducted to ensure the Seismic Qualification Program is being sustained. Note: Recent audit and self-assessment results for N-PROGs applicable to Seismic Qualification are addressed in Section 4.3 of this report.

4.1.4.3 Conclusion

The conclusion of this Review Task assessment is that a process exists for ensuring compliance with environmental and seismic equipment qualification programs and for documenting previous qualification measures taken to ensure qualification throughout the equipment's installed life. The intent of Review Task #4 is met and therefore Pickering NGS is compliant.

4.1.5 Review Task #5: Surveillance Program and Feedback Procedure

Confirm existence of a surveillance program and a feedback procedure to ensure aging degradation of qualified equipment remains insignificant.

4.1.5.1 Environmental Qualification

Equipment condition monitoring for age degradation is conducted to identify premature age related failures which negatively impact the Environmentally Qualified component's qualified life, and to ensure equipment failures are random and not common mode in nature.

Consistent with industry practice, and as per CANDU Owners Group (COG) Guideline GL 2008-02 "Environmental Qualification - Condition Monitoring of the Equipment" [54], equipment condition monitoring at Pickering NGS assesses the immediate and long-term operational readiness of Environmentally Qualified equipment through effective surveillance and feedback as outlined by the procedures and programs below.

N-PROG-RA-0006 "Environmental Qualification" [6] ensures that aging degradation is identified and mitigated through the following actions:

- The Environmental Qualification SPOC is required to ensure condition monitoring is conducted using the activities detailed in N-INS-03651-10023, "EQ Environment Monitoring and Equipment/Barrier Surveillance" [18] which includes periodic walkdowns of Environmentally Qualified equipment to confirm configuration requirements are met as specified in the applicable EQA.

- The Environmental Qualification SPOC reviews Environmental Qualification related SCRs, OPEX, and Environmental Qualification work reports to identify unanticipated age-related degradation which could affect the qualified life of Environmentally Qualified equipment/components. When failure trends or degradation is discovered, the Environmental Qualification SPOC initiates the actions to further evaluate and correct the situation, thereby providing the required feedback.

- The Manager Engineering Program Integration is responsible for generating and maintaining the EQA Part II[7] packages in accordance with N-PROC-RA-0044, "Environmental Qualification Assessment" [17] to support obsolescence, design changes and revised test reports.

N-PROG-MA-0026, "Equipment Reliability" [55] provides both surveillance and feedback to ensure aging degradation remains insignificant. There are various aspects of the System Engineer's role which support Environmental Qualification Condition Monitoring. Among these are:

- Functional failure evaluations, which are a surveillance and feedback process which include:
    - Evaluation of root cause;
    - Checking OPEX for similar failures and corrective actions;
    - Determination of the extent of condition; and
    - Setting up an action plan to prevent future repeat failures.

- System performance monitoring;

[7] EQA Part II is an evaluation of test and analysis documentation which establishes the basis for equipment Environmental Qualification. It is based either on plant and equipment specific environmental conditions, or a set of environmental conditions that envelop the worst-case conditions, normal and DBA, postulated to occur. EQA Part I evaluates qualification of individual equipment at station specific conditions based on generic equipment qualification evaluation completed in EQA Part II.

- Field walkdowns; and
- Preparation of System Health Reports which document the overall condition of the system, summarize the significant observations made during field walkdowns, and provide action plans listing the work activities which must be completed in order to improve system health.

Environmental Qualification monitoring is supported by N-PROG-MA-0004, "Conduct of Maintenance" [9] which is further detailed in the implementing procedure N-PROC-MA-0020, "Predefined Process" [56]. It requires as found conditions to be reported via work reports by maintenance personnel when equipment is disassembled, inspected or overhauled during preventive maintenance. In cases where unusual degradation is found, additional maintenance activities are initiated as required. Surveillance activities include infrared thermography, vibration monitoring, and lubricant analysis applied to equipment. These activities are performed to assure early detection of deteriorating equipment conditions in order to ensure safe, reliable and economical operation of station equipment. Predictive maintenance test and inspection data are trended using baseline or previous data for reference to determine extent of degradation. Data evaluation is performed by qualified maintenance personnel, reviewed and verified by the Predictive Maintenance Technology Owner and distributed to Component and Equipment Engineering and Performance Engineering.

N-PROG-MA-0017, "Component and Equipment Surveillance" [7] complements N-PROG-MA-0026, "Equipment Reliability" [55] by performing activities that evaluate, inspect, test and report on the health of specific component groups, including Environmentally Qualified equipment. These are monitored to provide assurance that the effects of aging are adequately managed during the operating life. Monitoring results are provided to the station System Engineers for incorporation into the system condition assessments. A specific example of component and equipment surveillance is the cable monitoring program implemented in accordance with N-PROC-MA-0099, "Cable Surveillance" [57].

N-PROG-OP-0001, "Nuclear Operations" [58] contributes to the overall surveillance of Environmentally Qualified equipment. Operator surveillance ensures that equipment and systems are operating within their design basis, that abnormalities are detected and that actions to resolve these abnormalities are completed in a timely fashion. Operator surveillance consists of operator rounds, readings, routines, predefines and tests. Operator rounds consist of checking accessible areas and equipment on a regular basis. All performance issues, deficiencies and problems are recorded and investigated to determine their cause. Follow up actions are performed or initiated to resolve issues.

N-PROG-RA-0003, "Corrective Action" [59] provides a consistent reporting and evaluation process for identified adverse conditions. Operations, Maintenance and Engineering personnel use the corrective action program to document any undesirable or questionable conditions. This process ensures the following:

- Adverse conditions are adequately documented;
- The cause of the adverse condition is determined;
- The extent of condition is determined; and

- Corrective actions are implemented to correct the adverse condition and where appropriate, to prevent the recurrence or reduce the risk of recurrence of similar adverse conditions.

The Corrective Action Program also provides a communication method of lessons learned (internal and external) to other facilities by providing a factual summary of the event or condition including the initial actions and observations.

4.1.5.2 Seismic Qualification

N-PROG-MA-0017, "Component and Equipment Surveillance" [7] outlines a set of activities that evaluate, inspect, test and report on the health of specific component groups, including seismically qualified equipment. The objective of these activities is to provide assurance that licensing, reactor safety, equipment reliability and conventional safety requirements are being met on an ongoing basis.

N-PROC-MA-0024, "System Performance Monitoring" [19] provides a consistent and comprehensive process for System Engineering to ensure effective monitoring, maintenance and enhancement of system performance and reliability. Field walkdowns are the mechanism used for performing a field evaluation of system performance and forms an essential part of System Performance Monitoring. Walkdowns by System Engineers complement the routine inspections completed by operations staff. Where adverse conditions are noted during walkdowns, appropriate corrective actions are initiated to restore equipment functionality.

N-PROG-MA-0004, "Conduct of Maintenance" [9], establishes safe, uniform and efficient maintenance practices. It ensures that effective implementation and control of maintenance activities are achieved by monitoring and assessing performance. The program includes work planning, work execution, and tool calibration and control, personnel and training, and performance indicators and assessment.

N-PROG-MP-0008, "Integrated Aging Management" [10] ensures that the condition of critical equipment (which includes seismically qualified equipment) is understood and that required activities are in place to ensure the health of the equipment while the plant ages. This is accomplished by establishing an integrated set of programs and activities which ensure performance requirements of all critical equipment are met on an ongoing basis. One of these activities are the Condition Assessments (CAs) covered under N-PROC-MP-0060, "Aging Management Process" [60] which are used to manage aging of critical plant equipment.

The practices that are being followed at Pickering NGS to identify and mitigate equipment aging include the following activities (as per Section 1.5.4.2 in N-PROC-MP-0060, "Aging Management Process" [60]):

- Operational tests and predefines;
- Engineering walkdowns;
- Component engineer review of maintenance results;

- Define component program activities (e.g., piping and valve periodic inspections);
- Activities to maintain Environmental Qualification, including preventative and predictive maintenance;
- In-plant inspections;
- Repair and replace strategies;
- Obsolescence strategies; and
- Research and development activities.

These practices supplement the aging information gathered under the IAM program. By managing the condition of the equipment, Seismic Qualification is also maintained.

Maintaining the integrity of system pressure boundaries of qualified systems during an earthquake is the fundamental requirement of seismic Category A.[8] Surveillance of pressure boundary components is accomplished through the Pickering NGS Periodic Inspection Program documents:

- NA44-PIP-03641.2-00001, "Pickering Nuclear Generating Station A Periodic Inspection Plan For Unit 1" [61];
- NA44-PIP-03641.2-00007, "Pickering Nuclear Generating Station A Periodic Inspection Plan For Unit 4" [62];
- NK30-PIP-03641.2-00001 "Pickering Nuclear Generating Station B Periodic Inspection Plan For Unit 5" [63];
- NK30-PIP-03641.2-00002 "Pickering Nuclear Generating Station B Periodic Inspection Plan For Unit 6" [64];
- NK30-PIP-03641.2-00003 "Pickering Nuclear Generating Station B Periodic Inspection Plan For Unit 7" [65]; and
- NK30-PIP-03641.2-00004 "Pickering Nuclear Generating Station B Periodic Inspection Plan For Unit 8" [66].

These plans provide in-service monitoring for degradation of fluid boundary portions of components, piping and supports of the Heat Transport System (HTS) and systems that are essential to reactor shutdown and safe cooling of fuel. The HTS is seismically qualified to maintain the pressure boundary integrity, and the Periodic Inspection Program (PIP) monitors aging degradation mechanisms such as pipe thinning due to Flow

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8 Category A systems must retain their structural and pressure boundary integrity during and following the earthquake to ensure and maintain the safety-related system function. Category B systems must retain their pressure boundary integrity and/or must function mechanically and/or electrically during and/or following an earthquake, to ensure and maintain the safety-related system function.

Accelerated Corrosion (FAC), formation and growth of cracks and flaws that are potential failure mechanisms under seismic excitation.

N-PLAN-01060-10001 "Feeders Life Cycle Management Plan" [67] requires monitoring of feeder piping wall thinning rate due to FAC. This has relevance to Seismic Qualification as the HTS is seismic Category A, and reduced wall thickness of feeder piping renders it susceptible to rupture during Design Basis Earthquake (DBE) induced loads.

In-service inspection programs for seismically qualified civil containment structures are also in place at Pickering NGS to support the surveillance program. The in-service inspections for the Reactor Buildings, Vacuum Building, and Pressure Relief Duct are performed in accordance with N-PROC-MA-0066, "Administrative Requirements for In Service Inspection And Testing For Concrete Containment Structures" [68]. This is performed in accordance with CSA N287.7, "In-Service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants". Note: The L/R/C/S review of CSA N287.7 is addressed in Section 4.2 of this Report.

4.1.5.3 Conclusion

The conclusion of this Review Task assessment is that a surveillance program and feedback procedure exist to ensure that aging degradation of qualified equipment remains insignificant. The intent of Review Task #5 is met and therefore Pickering NGS is compliant.

4.1.6 Review Task #6: Monitoring of Environmental Conditions

Confirm existence of monitoring of actual environmental conditions and identification of 'hot spots' of high activity or temperature.

4.1.6.1 Environmental Qualification

Pickering NGS temperature and radiation conditions are monitored on an ongoing basis. Regular monitoring of environmental parameters is the responsibility of the Station Environmental Qualification SPOC in accordance with Section 1.0 of N-INS-03651-10023, "EQ Environment Monitoring and Equipment/Barrier Surveillance" [18]. Pickering A is exempt from Radiation Surveillance based on the conservatism of the Pickering A RCM, as well as abnormal radiation conditions being reported through the SCR process. However, all other sections of N-INS-03651-10023 apply for Pickering Units 1,4.

Pickering NGS uses fixed on-line and portable batch environmental monitoring systems and devices to collect temperature and radiation data. The environmental room condition information is maintained in the Pickering NGS RCMs (NA44-MAN-03651-10001 "Environmental Qualification Room Conditions - Pickering A" [22] and NK30-MAN-03651-10001 "Environmental Qualification Room Conditions - Pickering B" [23]).

The Pickering NGS Environmental Qualification SPOC is responsible for the following environmental condition information:

- Maintaining records of environmental monitoring activities;

- Maintaining the Pickering NGS environmental conditions information in the RCM; and

- Preparing reports documenting any major deviations from the RCM and reviewing associated impacts to EQAs.

The process of collecting and analyzing data to review or revise the normal temperature profiles and normal radiation levels in the RCM is performed in accordance with NK30-INS-03651-00001, "Acquisition and Analysis of Temperature/Radiation Data for Environmental Monitoring to Preserve Environmental Qualification at Pickering NGS B" [69] (Note: As per N-INS-03651-10023, "EQ Environment Monitoring and Equipment/Barrier Surveillance" [18], with the Pickering amalgamation, identified Pickering Units 1,4 rooms are to be monitored in accordance with NK30-INS-03651-00001). NK30-INS-03651-00001 addresses:

Temperature Monitoring

The Plant Information (PI) system interface has been set up to monitor and trend reactor vault temperatures in four quadrants of each unit. The PI system provides temperature data on-line.

Radiation Monitoring

As per N-INS-03651-10023 [18], on-line monitoring provides trends of gamma radiation levels in different rooms using data from on-line radiation monitoring units. The resulting temperature and radiation information is used to update the RCM. Areas/rooms which are not instrumented, are periodically monitored and radiation surveys may be performed to acquire radiation measurements. The survey period should provide sufficient data for the establishment of a typical radiation dose level or for the verification of an established radiation dose level in the RCM for the Environmentally Qualified room of interest.

Addressing "Hot Spots" Of High Activity or Temperature

The first line of response in identifying abnormal room temperatures and radiation level is the station operators who respond to control room annunciations or local alarms during operator routines. When temperature/radiation levels exceed alarm set points and cannot be addressed by returning to normal levels within a reasonable amount of time, the alarm response manuals are followed.

The Radiation Protection department maintains a list of radiation "hot spots" within the station, which identifies the specific location and radiation level. Radiation "hot spots" are often highlighted in SCRs when station staff recognizes problems with radiation levels.

The qualified life of equipment located in the area of one time excursions is reviewed in accordance with Section 1.4.1.1 of N-INS-03651-10003, "Preparation of the Environmental Qualification Room Conditions Manual" [22]. Following an excursion, a check is made to ensure that the qualified life of qualified equipment in the area has not been significantly reduced. The excursion is initially documented through an SCR by station staff and later the review, cause of excursion and corrective actions are documented as part of the

Environmental Qualification history of the station. It is required that this documentation be retrievable and auditable.

In addition to temperature and radiation conditions, a variety of factors (including operation or maintenance activities) can impact the assessed qualified life. In order to maintain Environmental Qualification, a surveillance program monitors Environmentally Qualified equipment and components. The basis of the surveillance program relies on existing programs such as System Performance monitoring, Predictive Maintenance, Preventive Maintenance, Calibration Program and Corrective Action.

4.1.6.2 Seismic Qualification

This Review Task is not applicable to Seismic Qualification as monitoring of environmental conditions, and Radiation "hot spots" in particular, are not required to support seismic qualification.

4.1.6.3 Conclusion

The conclusion of this Review Task assessment is that actual environmental conditions are monitored and "hot spots" of high activity or temperature are identified and addressed. The intent of Review Task #6 is met and therefore Pickering NGS is compliant.

4.1.7 Review Task #7: Effects of Equipment Failures on Equipment Qualification

Confirm existence of an assessment that determines the effects of equipment failures on equipment qualification and appropriate corrective actions and/or safety improvements to maintain equipment qualification.

4.1.7.1 Environmental Qualification

This Review Task is closely associated with Review Task #5 (Section 4.1.5) which discusses the programs, processes and feedback procedures which ensure that equipment qualification is maintained.

In accordance with Section 1.4.2.2 of N-INS-03651-10023, "EQ Environment Monitoring and Equipment/Barrier Surveillance" [18] the Environmental Qualification SPOC is required to review Environmental Qualification related SCRs, OPEX and sample Environmental Qualification work reports to identify unanticipated age related degradation which could affect the qualified life of Environmentally Qualified equipment. When trends identify degradation, the Environmental Qualification SPOC initiates actions to further evaluate or correct the situation. Through this process, the failure histories of Environmentally Qualified equipment are reviewed and in-service degradation trends are identified. The Environmental Qualification SPOC recommends appropriate corrective actions to prevent equipment failures, including where required, a Technical Operability Evaluation.[9]

[9] A Technical Operability Evaluation (TOE) provides a substantiated engineering verification that a System, Structure or Component is capable of fulfilling its minimum credited safety function(s).

As per Section 1.4.2.2 of N-INS-03651-10023 [18], the Environmental Qualification SPOC performs, or assists in performing, the apparent or root cause analysis of Environmental Qualification related SCRs. This allows for the identification of potential common cause failure mechanisms and degraded conditions which could affect the qualified life of Environmentally Qualified equipment. This process is performed in accordance with guidelines specified in N-PROG-RA-0003, "Corrective Action" [59] which establishes the failure trending and analysis program and provides the program controls required for compliance to Environmental Qualification Industry Standards.

4.1.7.2 Seismic Qualification

N-PROC-MA-0024, "System Performance Monitoring" [19], which is discussed in Section 4.1.1.2 of this report, establishes the process for monitoring health of a system by trending performance and initiating proactive investigative and maintenance activities before failure can occur. This process is supported by N-PROG-MA-0017, "Component and Equipment Surveillance" [7] which lists a set of activities aimed at assuring the health of critical components by evaluating, inspecting, testing and reporting on the health of specific component groups.

N-PROG-MA-0004, "Conduct of Maintenance" [9], ensures that effective implementation and control of maintenance activities are achieved by monitoring and assessing performance. The program includes work planning, work execution, and tool calibration and control, personnel and training, and performance indicators and assessment. In accordance with N-PROC-MA-0020, "Predefined Process" [56] Preventative Maintenance tasks have been established for critical SSCs including Seismically Qualified equipment to assess the condition of equipment on a periodic basis and take appropriate corrective action to avoid functional failures. When failures occur, they are tracked, trended and appropriate actions (including a review of equipment design) are undertaken where required. Where modification of the plant is indicated to resolve the cause of failure, the ensuing activities are governed by N-PROC-MP-0090, "Modification Process" [51] which is discussed in Review Task #4 (Section 4.1.4.2). This process assesses the effects of equipment failures on Seismic Qualification and the appropriate corrective actions required to maintain Seismic Qualification.

4.1.7.3 Conclusion

The conclusion of this Review Task assessment is that programs and processes are in place to monitor the effects of equipment failures on equipment (environmental and seismic) qualification and to take appropriate corrective actions or safety improvements as required. The intent of Review Task #7 is met and therefore Pickering NGS is compliant.

4.1.8 Review Task #8: Equipment Protection from Adverse Environmental Conditions

Confirm there is protection and adequate separation of qualified equipment from adverse environmental conditions.

4.1.8.1 Environmental Qualification

Protection of equipment from adverse environmental conditions is addressed by OPG standard N-STQ-03651-10004, "Harsh Environment Protected Rooms" [70]. It defines the requirements for protected rooms located outside of the containment envelope which may be exposed to harsh environments during DBAs.

Environmental Qualification barriers are also controlled during modifications. In accordance with N-INS-03651-10023, "EQ Environment Monitoring and Equipment/Barrier Surveillance" [18] and N-GUID-03651-10000, "Guide to Environmental Qualification Completion Assurance" [71], the Environmental Qualification SPOC completes N-FORM-10649, "Environmental Qualification Completion Assurance". This form identifies barriers related to Environmental Qualification so that their effectiveness will be retained. In addition, the form verifies that required protective barriers and adequate separation of qualified equipment from adverse environmental conditions are in place.

To improve mitigation of powerhouse harsh environment events at Pickering NGS, the "H-line" concrete block wall and steam doors separating the powerhouse and reactor auxiliary bay, is designed to withstand steam pressure transients due to a postulated main steamline failure in the powerhouse. For Pickering A, the entire length of the block wall is supported with supplementary steel, erected along it at elevations 254ft, 274ft and 294ft [43]; while for Pickering B the block wall is reinforced with vertical beams and tiebacks at elevations 274ft and 294ft [39].

For Pickering B plant systems are separated into two groups (Group 1 and Group 2) for accident mitigation. This ensures that there is always a qualified safe shutdown success path for low-probability common mode events (e.g. local fires, turbine missiles), and events with widespread harsh environment conditions such as Main Steam Line Breaks. This ensures that sufficient systems remain available from at least one group to provide the required safety functions.

4.1.8.2 Seismic Qualification

An earthquake is a common mode event that is site wide and cannot be countered by two-group separation since both the groups are affected. The approach used has been to seismically qualify critical systems so that the capability to carry out essential safety functions is available following a DBE.

Pickering B was designed to be seismically qualified through the implementation of the overall safety design requirements documented in Engineering Design Guide DG-30-68000-2 [36]. This formally establishes the design requirements for safety related systems to meet the seismic aspects of Canadian nuclear safety principles specified in CSA N289.1, "General Requirements for Seismic Design and Qualification of CANDU Nuclear Power Plants" (Table 2-2 of the Pickering B Safety Report, Part 2 [39] contains a list of seismically qualified systems).

When Pickering A was designed, explicit seismic design standards for nuclear power plants were not available. Instead, the seismic design provisions of the 1965 National Building Code of Canada were adopted for seismic design of the nuclear structures. Although the Pickering A systems were not originally required to be seismically qualified, the equipment was evaluated and a Safe Shutdown Equipment List was defined by the 1998 Seismic Margin Assessment [34] which identifies the SSCs required for the seismic success path.

4.1.8.3 Conclusion

The conclusion of this Review Task assessment is that there is protection and adequate separation of qualified equipment from adverse environmental conditions. The intent of Review Task #8 is met and therefore Pickering NGS is compliant.

4.1.9 Review Task #9: Condition and Functionality of Qualified Equipment

Confirm physical condition and functionality capability of qualified equipment is being checked by walkdowns.

4.1.9.1 Environmental Qualification

The physical condition and functionality of environmentally qualified equipment is monitored by walkdowns on an ongoing basis. Walkdowns are part of the Condition Monitoring Program detailed in N-INS-03651-10023, "EQ Environment Monitoring and Equipment/Barrier Surveillance" [18].

The Environmental Qualification SPOC supports Performance Engineering for condition monitoring of Environmentally Qualified equipment. The Environmental Qualification SPOC works with the System Engineers to ensure that any adverse conditions found during the walkdowns are documented in the System Performance Monitoring Plans (SPMP) walkdown check sheets and integrated into the System Health Reports. In accordance with N-PROC-MA-0024, "System Performance Monitoring" [19], walkdowns are performed in accordance with an approved schedule (the normal expectation for routine walkdowns is weekly). SPMPs are used to detect deficiencies and anomalies associated with the Environmental Qualification equipment by checking equipment and system parameters. Any identified deficiencies are documented in SCRs and managed by N-PROG-RA-0003, "Corrective Action" [59].

N-GUID-03651-10001, "Field Guideline for Environmental Qualification Inspections and Walkdown" [72], provides guidance for Engineering personnel during field walkdowns of Environmentally Qualified equipment. It covers the most common Environmentally Qualified installations and concerns for all CANDU plants and is intended to augment site procedures, engineering drawings and other Environmental Qualification training programs.

In accordance with N-INS-03651-10023, "EQ Environment Monitoring and Equipment/Barrier Surveillance" [18], the Environmental Qualification SPOC ensures that field walkdowns are completed after installation of new or replaced Environmentally Qualified equipment to check that design and configuration requirements are met prior to the device being declared available for service. The Environmental Qualification SPOC also ensures that Environmental Qualification barriers (e.g., steam doors and radiation shielding) are checked periodically to ensure that environmentally qualified equipment, protected by these barriers, remains environmentally qualified.

4.1.9.2 Seismic Qualification

Field walkdowns to maintain physical condition and functional capability of Seismically Qualified SSCs are an essential part of the System Performance Monitoring process in accordance with N-PROC-MA-0024, "System Performance Monitoring" [19] and the results of these walkdowns are documented in System Health Reports. The purpose of these walkdowns is to identify any signs of degradation, functional failure or evidence of poor workarounds and complement the routine inspections completed by operations staff. Walkdowns of some equipment and components that are not accessible during the routine walkdowns are performed during Unit Planned Outages. Any deficiencies identified during walkdowns may generate an SCR or corrective work order, depending on the nature of the observation.

N-PROG-MA-0004, "Conduct of Maintenance" [9], ensures that effective implementation and control of maintenance activities are achieved by monitoring and assessing performance. The program includes work planning, work execution, and tool calibration and control, personnel and training, and performance indicators and assessment. N-PROG-MA-0004 provides authority to N-PROC-MA-0031, "Protection of Seismic Equipment and Routes" [20] establishes criteria to control unsecured equipment and material adjacent to seismically qualified SSCs and seismic routes, thereby ensuring operation, maintenance, modification or outage activities do not place seismically qualified SSCs at risk. The procedure includes guidelines for application of equipment restraints and separation distances to prevent interactions between unsecured equipment and material and seismically qualified SSCs during an earthquake. N-PROC-MA-0031 specifies good practices for maintaining seismic route integrity. It also includes a requirement for field walkdowns to confirm that designated internal and external seismic routes are maintained and accessible at all times.

4.1.9.3 Conclusion

The conclusion of this Review Task assessment is that field walkdown processes are in place to confirm the physical condition and functionality capability of qualified equipment is adequate. The intent of Review Task #9 is met and therefore Pickering NGS is compliant.

4.1.10 Review Task #10: Update to Equipment Classification

Confirm that changes to equipment classification have occurred, as required, as a result of major design modifications made since PSR1.

A listing of some of the major safety design modifications at Pickering NGS since PSR1, as outlined in Part 2 of the Pickering A and B Safety Reports ([39] and [43]) and the 2012 Power Reactor Operating Licence Renewal Application [73], are provided below:

- Fukushima Project related modifications, including:
    - Installation of Passive Autocatalytic Recombiners (PARs) on all units;
    - Addition of Emergency Mitigating Equipment including portable diesel pumps and diesel generators; and
    - Enhancements to water makeup/cooling capability for the Irradiated Fuel Bays; and

    - Additional flood barriers installed around the Pickering A Standby Generator Fuel Forwarding Pump house.

- Seismic Monitoring System Upgrades;
- ECI Strainer Capacity Margin increase;
- Airlock related Design improvements;
- Fuel Handling Equipment Reliability Improvement;
- Addition of Inter-Station Transfer Bus (ISTB); and
- Units 2 and 3 safe storage.

The bullets below provide examples of how some of the above mentioned design modifications have impacted equipment classification (i.e., Environmental/Seismic Qualification status) or where additional analysis demonstrated that equipment classification was not impacted:

- The PARS will mitigate hydrogen excursions, and are completely passive and do not rely on electrical power or operator intervention. The PARs are Seismically Qualified, but are not required to be Environmentally Qualified (i.e., do not have any known Environmentally Qualified degradation mechanisms);
- The airlocks in Pickering Units 1,4 were upgraded to be Environmentally and Seismically Qualified;
- Fuelling machine equipment seismic restraints have been added in order to avoid contributing to a LOCA;
- As part of Units 2 and 3 safe storage, the Unit 2 Class 1 batteries were replaced with new batteries in Seismically Qualified racks;
- In order to mitigate an MSLB in the Pickering A powerhouse, the ISTB was installed as a standby power source to transfer power from Pickering Units 5-8 to Units 1 and 4; and
- A review of the impact of Units 2 and 3 safe storage on Environmental Qualification conditions following MSLB or LOCA in the EQ Room Conditions Manual determined that the resulting impacts are acceptable [74].

The conclusion of this Review Task assessment is that equipment classification for major design modifications has occurred as required since PSR1 (Seismic and Environmental Qualification is being managed through the Engineering Change Control process). The intent of Review Task #10 is met and therefore Pickering NGS is compliant.

4.2 L/R/C/S Reviews

As per Section 2.2 of this report, detailed compliance assessments for twelve L/R/C/Ss with content applicable to Safety Factor 3 are provided in Reference [5]. Associated findings applicable to Safety Factor 3 are summarized in Table 3 below.

Table 3: PSR2 L/R/C/S Compliance Assessment Results for Safety Factor 3

L/C/R/S Reviewed	PSR2 Compliance Assessment for Safety Factor 3
N290.13-05, "Environmental Qualification of Equipment for CANDU Nuclear Power Plants"	There are no PSR2 gaps for N290.13-05. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with N290.13-05.
N285.5-13, "Periodic Inspection of CANDU Nuclear Power Plant Containment Components"	For Safety Factor 3 there are no PSR2 gaps for N285.5-13.
N287.7-08, "In-Service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants"	For Safety Factor 3 there are no PSR2 gaps for N287.7-08.
CNSC RD/GD-210 (2012), "Maintenance Programs for Nuclear Power Plants"	There are no PSR2 gaps for CNSC RD/GD-210. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with RD/GD-210.
CNSC RD/GD-98 (2012), "Reliability Programs for Nuclear Power Plants"	There are no PSR2 gaps for CNSC RD/GD-98. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with RD/GD-98.
CNSC REGDOC-2.6.3 (2014), "Aging Management"	For Safety Factor 3 there are no PSR2 gaps for CNSC REGDOC-2.6.3.
N287.2-08, "Material Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants"	There are no PSR2 gaps for CSA N287.2-08. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with N287.2-08.
N289.1-08, "General Requirements for Seismic Design and Qualification of CANDU Nuclear Power Plants"	There are no PSR2 gaps for CSA N289.1-08. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with N289.1-08.
N289.2-10, "Ground Motion Determination for Seismic Qualification of Nuclear Power Plants"	There are no PSR2 gaps for CSA N289.2-10. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with N289.2-10.

L/C/R/S Reviewed	PSR2 Compliance Assessment for Safety Factor 3
N289.3-10, "Design Procedures for Seismic Qualification of Nuclear Power Plants"	There is one PSR2 gap for Pickering NGS compliance with N289.3-10 which is applicable to Safety Factor 3 (**Pickering PSR2 Gap SF3-2**): 1. Clause 4.4.4.5 of CSA N289.3-10 states: "The power spectral density (PSD) function of each time-history shall be calculated and shown to not have any significant gaps in energy over the frequency intervals outlined in Table 2…." The calculation of PSD is not addressed in the Pickering A or B PRA Based SMAs. The Pickering NGS A PRA Seismic Guide and the OPG PRA Guide do not identify any requirements for PSD. Also, evidence in the form of a calculation for time histories which represent the design ground motion was not found (which is a precursor for the PSD calculation). The lack of evidence of calculated time histories was also identified as a gap in the Darlington ISR (ISR Issues #D352 and #D617 – Documented evidence in the form of a calculation to show that the generated time history correctly represents the design ground response spectrum within the prescribed requirements has not been provided). The closure reference for #D352 and #D617 makes use of the detailed assessment performed in NK38-REP-03680-10224 R000 which is specific to Darlington. A similar assessment for Pickering NGS could not be found. As a result, there is a gap for PSR2 to provide similar evidence to show that: a) the generated time history used within seismic analyses of safety-related systems correctly represents the design ground response spectrum for the Pickering site in compliance with N289.3-10, and b) the PSD function of each time-history has been calculated and shown to not have any significant gaps in energy over the frequency intervals.
N289.4-12, "Testing Procedures for Seismic Qualification of Nuclear Power Plants Structures, Systems, and Components"	There is one PSR2 gap for Pickering NGS compliance with N289.4-12 which is applicable to Safety Factor 3 (**Pickering PSR2 Gap SF3-3**): 1. Station-specific documents (including the Darlington seismic design guide, Darlington Reports and Darlington-specific technical specifications for seismic qualification) were used as the basis for compliance in the clause-by-clause Darlington code refresh review for clauses 4.2.1, 4.2.2.2, 4.2.3.1, 4.2.5, 4.3.2, 5.2.2.2.5, 5.7, 5.8.1, 5.8.1.2, 7.2.1, 7.7.1, 7.7.4 and 8.2. Pickering-specific seismic design guides, reports and technical specifications that are equivalent to those used to demonstrate Darlington compliance with the changes made in CSA N289.4-12 were identified. However, a detailed review to confirm that the Pickering-specific documents fully comply with the requirements of the clauses listed above is needed. As a result, this is a PSR2 gap.
N289.5-12, "Seismic Instrumentation Requirements for Nuclear Power Plants and Nuclear Facilities"	There is one PSR2 gap for Pickering NGS compliance with N289.5-12 which is applicable to Safety Factor 3 (**Pickering PSR2 Gap SF3-4**): 1. Darlington ISR Issues #D622, D623 and D624 require no further action for Darlington as they were either classified as Acceptable Deviations or were closed. However, the issues are identified as a PSR2 gap for the following reasons: (Note: These gaps are closely related and are therefore identified as a single PSR2 gap.) o Darlington ISR Issue #624 refers to specific Darlington instrumentation in order to classify the gaps as Acceptable Deviations. It must be demonstrated that Pickering seismic instruments have the same capabilities as the Darlington instruments (fleet-wide or Pickering-specific standards that would ensure that the Pickering seismic instruments have the same capabilities as the Darlington instruments could not be found). Therefore, this is identified as a gap for PSR2.

L/C/R/S Reviewed	PSR2 Compliance Assessment for Safety Factor 3
	◦ Darlington ISR Issue #D622 was deemed to be of low safety significance. The same rationale may apply at Pickering, but first it must be demonstrated that Pickering has the same set up of seismic instruments as Unit 0 at Darlington. Therefore, this is identified as a gap for PSR2. ◦ Darlington ISR Issue #D623 was deemed to be of low safety significance. The same rationale may apply at Pickering, but first it must be demonstrated that similar accelerometers are used at Pickering, and that their locations are not affected by strong ambient vibration. Therefore, this is identified as a gap for PSR2.

4.3 Audit and Self-Assessment Reviews

The OPG Nuclear Programs specifically applicable to the Equipment Qualification Safety Factor are identified in Table 2, and details of the associated audit and self-assessment results for each of the N-PROGs are provided in Appendix B. Based on the Appendix B.1 assessment of recent N-PROG-RA-0006 [6] audit and self-assessment results there are two gaps for Pickering PSR2 relating to the Environmental Qualification Program:

1) The Environmental Qualification documentation backlog (e.g., Document Change Requests for Environmental Qualification Assessments) increased from 9% in Q4 2013 to 14% in Q4 2014. As a result, CNSC staff requested that OPG assess and create a corrective action plan to ensure Environmental Qualification information remains current and more specifically to reduce and manage the document revision backlog. This is a gap (**Pickering PSR2 Gap SF3-5**) since the CNSC has identified this issue in an Action Notice following a regulatory inspection and the associated action (AR#28179009) is due to be completed by Q3 2016.

2) OPG is in non-compliance with N-PROC-RA-0044, "Environmental Qualification Assessment" for not correcting the documentation discrepancy for Unit 5-8 Vertical Flux Detector Tefzel cables with justification for a new qualified life value. As a result, CNSC staff requested that OPG revise the Environmental Qualification Assessment for Tefzel cables to reflect the change of qualified life of the Vertical Flux Detectors. This is a gap (**Pickering PSR2 Gap SF3-6**) since the CNSC has identified this issue in an Action Notice following a regulatory inspection and the associated action (AR#28170757) is due to be completed by Q4 2016.

The above-mentioned gaps are being tracked to completion and a Regulatory Management Action is in place to provide an update to the CNSC in August 2016 (per AR# 28179713).

In terms of the Component and Equipment Surveillance Program [7] and the Seismic Qualification Program (which consists of a series of engineering programs and procedures), no additional audit or self-assessment-related gaps were identified.

4.4 Additional Review Findings

As discussed in Section 3.4, the PSR2 Safety Factor 3 assessment also included a review of commitments previously made to the CNSC, open CNSC action items, and exemptions granted by the CNSC since the current operating licence was issued, per the R04 Pickering LCH [15], to determine if there are any associated impacts associated with operation of the Pickering Units past 2020. The review also included identification and review of previously identified PSR1 gaps related to Safety Factor 3 to determine impacts associated with operation of the Pickering Units past 2020. This assessment did not find any additional gaps to those already discussed in Section 4.2 (L/R/C/S reviews) or Section 4.3 (Audit and Self-Assessments Results) for Safety Factor 3.

There were no PSR2 gaps identified in this Safety Factor 3 Report that require discussion in other Safety Factor Reports.

5.0 RESULTS AND CONCLUSIONS

OPG Governance, Programs, Policies, Procedures, Instructions and Guidelines related to Pickering NGS equipment qualification (environmental and seismic) were reviewed for the ten PSR2 Review Tasks in Section 4.1 of this report and resulted in Pickering PSR2 Gap SF3-1 below. L/R/C/S and OPG Nuclear Program audit and self-assessment reviews for Safety Factor 3 were prepared per Sections 4.2 and 4.3, respectively, and resulted in PSR2 Gaps SF3-2 to SF3-6. This report also included a review of OPG commitments previously made to the CNSC, open CNSC action items, and exemptions granted by the CNSC since the current operating licence was issued (all related to Safety Factor 3), as well as identification and review of previously identified PSR1 gaps related to Safety Factor 3 (to ascertain the implications of extending Pickering NGS operation beyond 2020), per Section 4.4, which resulted in no additional PSR2 gaps.

The six PSR2 gaps that will need to be addressed as part of Pickering PSR2 are:

- **<u>Gap SF3-1</u>**: Per Review Task #3, the Environmental Qualification Program N-PROG-RA-0006, "Environmental Qualification Program" requires that whenever equipment on the EQL approaches the end of its qualified life, action must be taken to sustain the qualified status of that equipment regardless of what the station current life is taken to be. Hence, all EQAs will need to be re-assessed to ensure qualification is maintained in order to support continued operation of Pickering NGS beyond 2020. The current Environmentally Qualified life of all Pickering NGS SSCs may not necessarily extend to 2028 and a full review of Environmentally Qualified life-limited components impacted by operation past 2020 will need to be undertaken prior to life extension of Pickering NGS. This is therefore identified as a gap exists for Pickering PSR2. It is noted that work to address this gap is currently underway as part of the update of Pickering NGS Condition Assessments for safety related systems and Life Cycle Management Plans for major components.

- **<u>Gap SF3-2</u>**: For N289.3-10, "Design Procedures for Seismic Qualification of Nuclear Power Plants" there is a gap associated with Safety Factor 3. Clause 4.4.4.5 of CSA N289.3-10 states: "The power spectral density (PSD) function of each time-history shall be calculated and shown to not have any significant gaps in energy over the frequency intervals outlined in Table 2…." The calculation of PSD is not addressed in the Pickering A or B PRA Based SMAs. The Pickering NGS A PRA Seismic Guide and the OPG PRA Guide do not identify any requirements for PSD. Also, evidence in the form of a calculation for time histories which represent the design ground motion was not found (which is a precursor for the PSD calculation). The lack of evidence of calculated time histories was also identified as a gap in the Darlington ISR (ISR Issues #D352 and #D617 – Documented evidence in the form of a calculation to show that the generated time history correctly represents the design ground response spectrum within the prescribed requirements has not been provided). The closure reference for #D352 and #D617 makes use of the detailed assessment performed in NK38-REP-03680-10224 R000 which is specific to Darlington. A similar assessment for Pickering NGS could not be found. As a result, there is a gap for PSR2 to provide similar evidence to show that: a) the generated time history used within seismic analyses of safety-related systems correctly represents the design ground response spectrum for the Pickering site in compliance with N289.3-10, and b) the PSD function of each time-history has been calculated and shown to not have any significant gaps in energy over the frequency intervals.

- **Gap SF3-3**: For N289.4-12, "Testing Procedures for Seismic Qualification of Nuclear Power Plants Structures, Systems, and Components" there is a gap associated with Safety Factor 3. Station-specific documents (including the Darlington seismic design guide, Darlington Reports and Darlington-specific technical specifications for seismic qualification) were used as the basis for compliance in the clause-by-clause Darlington code refresh review for clauses 4.2.1, 4.2.2.2, 4.2.3.1, 4.2.5, 4.3.2, 5.2.2.2.5, 5.7, 5.8.1, 5.8.1.2, 7.2.1, 7.7.1, 7.7.4 and 8.2. Pickering-specific seismic design guides, reports and technical specifications that are equivalent to those used to demonstrate Darlington compliance with the changes made in CSA N289.4-12 were identified. However, a detailed review to confirm that the Pickering-specific documents fully comply with the requirements of the clauses listed above is needed. As a result, this is a PSR2 gap.

- **Gap SF3-4**: For N289.5-12, "Seismic Instrumentation Requirements for Nuclear Power Plants and Nuclear Facilities" there is a gap associated with Safety Factor 3. Darlington ISR Issues #D622, D623 and D624 require no further action for Darlington as they were either classified as Acceptable Deviations or were closed. However, the issues are identified as a PSR2 gap for the following reasons: (Note: These gaps are closely related and are therefore identified as a single PSR2 gap.)

    - Darlington ISR Issue #624 refers to specific Darlington instrumentation in order to classify the gaps as Acceptable Deviations. It must be demonstrated that Pickering seismic instruments have the same capabilities as the Darlington instruments (fleet-wide or Pickering-specific standards that would ensure that the Pickering seismic instruments have the same capabilities as the Darlington instruments could not be found). Therefore, this is identified as a gap for PSR2.
    - Darlington ISR Issue #D622 was deemed to be of low safety significance. The same rationale may apply at Pickering, but first it must be demonstrated that Pickering has the same set up of seismic instruments as Unit 0 at Darlington. Therefore, this is identified as a gap for PSR2.
    - Darlington ISR Issue #D623 was deemed to be of low safety significance. The same rationale may apply at Pickering, but first it must be demonstrated that similar accelerometers are used at Pickering, and that their locations are not affected by strong ambient vibration. Therefore, this is identified as a gap for PSR2.

- **Gap SF3-5**: The Environmental Qualification documentation backlog (e.g., Document Change Requests for Environmental Qualification Assessments) increased from 9% in Q4 2013 to 14% in Q4 2014. As a result, CNSC staff requested that OPG assess and create a corrective action plan to ensure that Environmental Qualification information remains current and more specifically to reduce and manage the document revision backlog. This is a gap for PSR2 since the CNSC identified this issue in an Action Notice following a regulatory inspection and the associated action (AR#28179009) is due to be completed by Q3 2016.

- **<u>Gap SF3-6</u>**: Pickering NGS is in non-compliance with N-PROC-RA-0044, "Environmental Qualification Assessment" for not correcting the documentation discrepancy for Unit 5-8 Vertical Flux Detector Tefzel cables with justification for a new qualified life value. As a result, CNSC staff requested that OPG revise the Environmental Qualification Assessment for Tefzel cables to reflect the change of qualified life of the Vertical Flux Detectors. This is a gap for PSR2 since the CNSC identified this issue in an Action Notice following a regulatory inspection and the associated action (AR#28170757) is due to be completed by Q4 2016.

The review of Safety Factor 3 has confirmed that the Pickering NGS equipment important to safety has been properly qualified and that this qualification is being maintained through an adequate programme of maintenance, inspection and testing.

6.0 REFERENCES

[1] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[2] CNSC REGDOC-2.3.3, *Periodic Safety Reviews*, April 2015.

[3] IAEA Safety Guide No. SSG-25, *Periodic Safety Review for Nuclear Power Plants*, 2013.

[4] OPG Report, P-REP-03680-00003 R000, *Pickering NGS PSR2: Definition of Safety Factor Review Tasks*, May 30, 2016.

[5] OPG Report, P-REP-03680-00004 R000, *Pickering NGS Periodic Safety Review 2: Code and Standard Reviews for Safety Factors 2 (Actual Condition of SSCs), 3 (Equipment Qualification) and 4 (Aging)*, July 2016.

[6] OPG Nuclear Program, N-PROG-RA-0006 R008, *Environmental Qualification*, May 28, 2015.

[7] OPG Nuclear Program, N-PROG-MA-0017 R008, *Component and Equipment Surveillance*, June 1, 2015.

[8] OPG Nuclear Program, N-PROG-MP-0001 R014, *Engineering Change Control,* January 7, 2015.

[9] OPG Nuclear Program, N-PROG-MA-0004 R011, *Conduct of Maintenance*, April 28, 2015.

[10] OPG Nuclear Program, N-PROG-MP-0008 R006A, *Integrated Aging Management,* October 2, 2015.

[11] OPG Nuclear Program, N-PROG-MP-0009 R011, *Design Management*, December 19, 2014.

[12] OPG Plan, NK30-PLAN-03680-00002 R000, *Pickering B - Integrated Implementation Plan*, December 15, 2011.

[13] OPG Report, NK38-REP-03680-10185 R002, *Darlington NGS Integrated Implementation Plan (IIP)*, April 30, 2015.

[14] OPG Plan, NK30-PLAN-00531-00001 R005, *Pickering 5-8 Continued Operations Plan*, December 2015.

[15] CNSC Report, LCH-PNGS-R004, *Pickering NGS: Licence Conditions Handbook*, December 23, 2015.

[16] OPG Nuclear Procedure, N-PROC-RA-0051 R008, *Environmental Qualification Lists*, September 3, 2014.

[17] OPG Nuclear Procedure, N-PROC-RA-0044 R008, *Environmental Qualification Assessment,* October 17, 2014.

[18] OPG Nuclear Instruction, N-INS-03651-10023 R003, *EQ Environment Monitoring and Equipment/Barrier Surveillance,* July 24, 2013.

[19] OPG Nuclear Procedure, N-PROC-MA-0024 R015, *System Performance Monitoring*, October 28, 2013.

[20] OPG Nuclear Procedure, N-PROC-MA-0031 R012, *Protection of Seismic Equipment and Routes,* March 19, 2013.

[21] OPG Nuclear Instruction, N-INS-08173-10048 R005, *Item Equivalency Evaluation,* July 11, 2014.

[22] OPG Manual, NA44-MAN-03651-10001 R002, *Environmental Qualification Room Conditions - Pickering A*, October 31, 2014.

[23] OPG Manual, NK30-MAN-03651-10001 R002, *Environmental Qualification Room Conditions - Pickering B*, November 06, 2014.

[24] OPG Nuclear Instruction, N-INS-03651-10003 R007, *Preparation of the Environmental Qualification Room Conditions Manual*, October 21, 2014.

[25] OPG Nuclear Program, N-PROG-RA-0012 R011, *Fire Protection*, July 28, 2015.

[26] OPG Report, NA44-REP-71400-10001 R001, *Pickering Nuclear Generating Station "A" Fire Protection Code Compliance Review*, March 11, 2011.

[27] OPG Report, NK30-REP-71400-10001 R001, *Fire Protection Code Compliance Review Pickering Nuclear Generating Station B,* November 23, 2010.

[28] OPG Report, NA44-REP-71400-10003 R001, *Fire Hazard Assessment – Pickering A Nuclear Generating Station*, April 30, 2012.

[29] OPG Report, NK30-REP-71400-10002 R002, *Fire Hazard Assessment - Pickering B Nuclear Generating Station,* November 23, 2011.

[30] OPG Report, NA44-REP-71400-00023 R000, *Fire Safe Shutdown Analysis - Pickering A Nuclear Generating Station*, May 8, 2012.

[31] OPG Report, NK30-REP-71400-00001 R002, *Fire Safe Shutdown Analysis - Pickering B Nuclear Generating Station*, October 05, 2011.

[32] OPG Nuclear Instruction, N-INS-00700-10007 R002, *Preparation of Modification Design Requirements,* March 12, 2015.

[33] OPG Nuclear Instruction, N-DG-60407-10000 R000, *Guidelines for Electromagnetic Compatibility Test,* October 21, 2011.

[34] OPG Report, NA44-REP-02004-0073 Volume 2, *Seismic Assessment of Pickering A NGS Summary Report*, February 25, 1998.

[35] OPG Report, NA44-REP-03611-00022 R00, *Pickering NGS A PRA Based Seismic Margin Assessment,* December 2013.

[36] OPG Design Guide, DG-30-68000-2 R02, *Seismic Qualification of Safety Related Systems,* December 3, 1979.

[37] OPG Report, NK30-REP-03611-00013 R01, *Pickering NGS B PRA Based Seismic Margin Assessment,* March 2014.

[38] OPG Nuclear Guide, N-GUID-03651-10002 R000, *Environmental Qualification Asset Suite Data Conventions*, October 31, 2014.

[39] OPG Report, NK30-SR-01320-00002 R004, *Pickering B Safety Report - Part 2,* October 10, 2012.

[40] OPG Specification, TS-30-68000-9 R00, *Seismic Qualifiaction of Safety System Instrumentation,* December 1976.

[41] OPG Letter, NK30-00531P (Tong to Mackie November 1978), *Pickering NGS B Seismic Design.*

[42] OPG Letter, NK30-00531P (Brown to Tong, February 1982), *Pickering B Design Completion Assurance.*

[43] OPG Report, NA44-SR-01320-00001 R015, *Pickering A Safety Report - Part 2,* July 24, 2012.

[44] OPG Design Guide, NA44-DG-03650-00001 R003, *Seismic Design Guide for Seismic Qualification of Pickering NGS A Success Path Structures, Systems and Components,* February 15, 2012.

[45] OPG Nuclear Procedure, N-PROC-RA-0022 R032 , *Processing Station Condition Records,* November 13, 2014.

[46] OPG Nuclear Program, N-PROG-TR-0005 R016, *Training,* January 5, 2016.

[47] OPG Nuclear Training and Qualification Description, N-TQD-403-00001 R011, *Nuclear Engineering Support Personnel Training and Qualification Description,* April 28, 2015.

[48] OPG Nuclear Program, N-PROG-AS-0002 R015, *Human Performance,* October 31, 2014.

[49] OPG Nuclear Program, N-PROG-RA-0010 R013, *Independent Assessment,* April 30, 2014.

[50] OPG Nuclear Procedure, N-PROC-RA-0097 R008, *Self- Assessment and Benchmarking, December 17, 2014.*

[51] OPG Nuclear Procedure, N-PROC-MP-0090 R012, *Modification Process,* April 20, 2015.

[52] OPG Nuclear Procedure, N-PROC-MP-0047 R006, *Design Verification, April 30, 2015.*

[53] OPG Nuclear Procedure, N-PROC-TR-0008 R019, *Systematic Approach to Training, January 24, 2014.*

[54] CANDU Owners Group (COG) Guideline, COG GL 2008-02, *Environmental Qualification - Condition Monitoring of the Equipment,* June 2008.

[55] OPG Nuclear Program, N-PROG-MA-0026 R02, *Equipment Reliability,* May 26, 2015.

[56] OPG Nuclear Procedure, N-PROC-MA-0020 R025, *Predefined Process, October 16, 2015.*

[57] OPG Nuclear Procedure, N-PROC-MA-0099 R001, *Cable Surveillance,* August 19, 2014.

[58] OPG Nuclear Program, N-PROG-OP-0001 R008, *Nuclear Operations,* November 13, 2015.

[59] OPG Nuclear Program, N-PROG-RA-0003 R010, *Corrective Action*, January 9, 2015.

[60] OPG Nuclear Procedure, N-PROC-MP-0060 R005B, *Aging Management Process,* October 1, 2015.

[61] OPG Plan, NA44-PIP-03641.2-00001 R006, *Pickering Nuclear Generating Station A Periodic Inspection for Unit 1,* September 8, 2006.

[62] OPG Plan, NA44-PIP-03641.2-00007 R005, *Pickering Nuclear Generating Station A Periodic Inspection Plan for Unit 4,* September 8, 2006.

[63] OPG Plan, NK30-PIP-03641.2-00001 R006, *Pickering Nuclear Generating Station B Periodic Inspection Plan for Unit 5,* August 29, 2006.

[64] OPG Plan, NK30-PIP-03641.2-00002 R004, *Pickering Nuclear Generating Station B Periodic Inspection Plan for Unit 6,* August 29, 2006.

[65] OPG Plan, NK30-PIP-03641.2-00003 R004, *Pickering Nuclear Generating Station B Periodic Inspection Plan for Unit 7,* August 29, 2006.

[66] OPG Plan, NK30-PIP-03641.2-00004 R004, *Pickering Nuclear Generating Station B Periodic Inspection Plan for Unit 8,* August 29, 2006.

[67] OPG Plan, N-PLAN-01060-10001 R17, *Feeders Life Cycle Management Plan, September 15, 2015.*

[68] OPG Nuclear Procedure, N-PROC-MA-0066 R005, *Administrative Requirements For In Service Inspection And Testing For Concrete Containment Structures,* April 24, 2014.

[69] OPG Nuclear Instruction, NK30-INS-03651-00001 R000, *Acquisition and Analysis of Temperature/Radiation Data for Environmental Monitoring to Preserve Environmental Qualification at Pickering NGS B,* February 15, 2005.

[70] OPG Nuclear Engineering Standard, N-STQ-03651-10004 R002, *Harsh Environment Protected Rooms,* July 20, 2010.

[71] OPG Nuclear Guide, N-GUID-03651-10000 R000, *Guide to Environmental Qualification Completion Assurance*, March 30, 2006.

[72] OPG Nuclear Guide, N-GUID-03651-10001 R001, *Field Guideline for Environmental Qualification Inspections and Walkdown,* May 28, 2013.

[73] OPG Letter, P-CORR-00531-03719, *Application for Renewal of Pickering Nuclear Generating Station Power Reactor Operator Licence,* July 04, 2012.

[74] OPG Report, NA44-SR-01320-00002 R004, *Pickering Nuclear 1-4 Safety Report: Part 3 – Accident Analysis,* October 31, 2013.

Appendix A: Nomenclature

AECB	Atomic Energy Control Board
CANDU	CANada Deuterium Uranium
CAP	Corrective Action Plan
CA	Condition Assessment
CMP	Control Maintenance Procedure
CNSC	Canadian Nuclear Safety Commission
COG	CANDU Owners Group
DBA	Design Basis Accident
DBE	Design Basis Earthquake
DCR	Document Change Request
ECC	Engineering Change Control
ECI	Emergency Coolant Injection
EMI	Electromagnetic Interference
EQA	Environmental Qualification Assessment
EQL	Environmental Qualification List
EQ	Environmental Qualification
FAC	Flow Accelerated Corrosion
GAR	Global Assessment Report
HTS	Heat Transport System
IAEA	International Atomic Energy Agency
IAM	Integrated Aging Management
IEE	Item Equivalency Evaluation
IIP	Integrated Implementation Plan
ISR	Integrated Safety Review
ISTB	Inter-Station Transfer Bus
LOCA	Loss of Coolant Accident
MDR	Modification Design Requirements
MSLB	Main Steam Line Break
NBC	National Building Code
NGS	Nuclear Generating Station
NICR	Non-Identical Component Replacement
OBC	Ontario Building Code

OPEX	Operating Experience
OPG	Ontario Power Generation
PARs	Passive Autocatalytic Recombiners
PARTS	Pickering A Return to Service
PEL ID	Program Element Identification
PI	Plant Information
PIP	Periodic Inspection Program
PRA	Probabilistic Risk Assessment
PSR	Periodic Safety Review
PSR1	Periodic Safety Review 1 (earlier OPG PSR work and other associated assessments)
PSR2	Periodic Safety Review 2 (subsequent PSR per CNSC REGDOC-2.3.3)
RCM	Room Conditions Manual
RFI	Radiofrequency Interference
SCR	Station Condition Record
SDE	Site Design Earthquake
SMA	Seismic Margin Assessment
SPOC	Single Point of Contact
SSC	Structures, Systems and Components
SPMP	System Performance Monitoring Plan
USI	Universal Subject Index

Appendix B: Audit and Self-Assessment Results

B.1 N-PROG-RA-0006, "Environmental Qualification"

The Environmental Qualification program provides assurance that essential equipment can perform as required when exposed to harsh DBA conditions and that this capability is preserved over the life of the plant. The OPG EQ Program complies with CSA Standard N290.13-05, "Environmental Qualification of Equipment for CANDU Nuclear Power Plants". Note: The L/R/C/S review of CSA N290.13 is addressed in Section 4.2 of this Report.

The EQ program consists of three distinct areas which are integrated to establish and maintain auditable proof of qualification throughout the life of station units:

- Documentation of EQ requirements: Defining list of equipment and components required to be environmentally qualified which is maintained current with the plant licensing basis, design basis and service conditions.

- Demonstration of qualification: Ensuring auditable proof of performance under harsh DBA conditions is developed and maintained current with plant licensing basis, design basis, service conditions and configuration.

- Implementation and preservation: Providing assurance that equipment and components within the EQ program are purchased, stored, configured, maintained, monitored and replaced to ensure qualified status is preserved.

In January 2014, Nuclear Oversight conducted a performance based audit at Pickering NGS, NO-2014-105 [B.1.1] to determine if the actions taken to address the issues identified in the EQ audit NO-2013-111 [B.1.2] are effective. NO-2013-111 [B.1.2] had identified deficiencies in EQ Room Condition Monitoring and Reporting, deficiencies in EQ Equipment surveillance, and untimely resolution of document change requests (DCRs) backlog on EQ documents. It was concluded in NO-2014-105 [B.1.1] that the previously identified deficiencies were resolved and the actions taken to address the issues were effective.

The Equipment Reliability department completed a self-assessment in March 2015 for Pickering NGS, P15-000842 [B.1.3] in order to provide a gap analysis of the corrective actions from the 2013 CNSC EQ Type II inspection [B.1.4] and the 2013 Nuclear Oversight Audit [B.1.2]. Overall, the corrective actions which were completed were considered to be completed with satisfactory results. However, a trend was identified that EQ DCRs on both Environmental Qualification Assessments and Environmental Qualification List Development Packages had a backlog increase. As part of the self-assessment, an action plan was put in place for managing the EQ document backlog.

A CNSC Type II inspection of the EQ program was conducted in March 2015 for Pickering NGS [B.1.5]. The audit was deemed successful and the CNSC recognized OPG for its effort in improving the EQ program from a "RED" program performance in 2013 to a "WHITE" by Q4-2014. The CNSC provided a final report in June 2015 which identified 4 Action Notices and 2

Recommendations (all Action Notices and Recommendations were known issues). As part of OPG's response to the CNSC, one Action Notice was closed out, while the remaining Action Notices had Corrective Action Plans put in place to enable tracking to completion. The Action Notices were related to:

- The Environmental Qualification documentation backlog (e.g., DCRs for Environmental Qualification Assessments) increased from 9% in Q4 2013 to 14% in Q4 2014. As a result, CNSC staff requested that OPG assess and create a corrective action plan to ensure Environmental Qualification information remains current and more specifically to reduce and manage the document revision backlog (tracked per Action Request (AR) #28179009, Due Q3 2016).

- OPG is in non-compliance with N-PROG-RA-0006, "Environmental Qualification" for not ensuring that all Environmental Qualification requirements are addressed in corresponding Control Maintenance Procedures (CMPs). As a result, CNSC staff requested that OPG ensure that all Environmental Qualification requirements are addressed in corresponding CMPs by performing a review between the maintenance requirements listed in all Pickering EQA Part 1 and those listed in the latest revision of applicable CMPs (tracked per AR#28179010, which is now complete, hence this is not a gap for PSR2).

- OPG is in non-compliance with N-PROC-RA-0044, "Environmental Qualification Assessment" for not correcting the documentation discrepancy for Unit 5-8 Vertical Flux Detector Tefzel cables with justification for a new qualified life value. As a result, CNSC staff requested that OPG revise the Environmental Qualification Assessment for Tefzel cables to reflect the change of qualified life of the Vertical Flux Detectors (tracked per AR#28170757, Due Q4 2016).

- The CNSC requested that OPG assess the cause of the low compliance (60%) of the Pickering NGS maintenance records and provide a corrective action plan to ensure the status of Environmentally Qualified equipment is validated and preserved. This has been tracked per AR#28174743, which is now complete, hence this is not a gap for PSR2.

There is high confidence that these actions will be completed successfully and are being tracked to completion since a Regulatory Management Action (AR#28179713) is in place to provide an update to the CNSC by August 2016.

Based on the above assessment of recent N-PROG-RA-0006 audit and self-assessment results there are two gaps for Pickering PSR2 relating to the Environmental Qualification Program:

1) The Environmental Qualification documentation backlog (e.g., DCRs for Environmental Qualification Assessments) increased from 9% in Q4 2013 to 14% in Q4 2014. As a result, CNSC staff requested that OPG assess and create a corrective action plan to ensure Environmental Qualification information remains current and more specifically to reduce and manage the document revision backlog. This is a gap (**Pickering PSR2 Gap SF3-5**) since the CNSC has identified this issue in an Action Notice following a

regulatory inspection and the associated action (AR#28179009) is due to be completed by Q3 2016.

2) Pickering NGS is in non-compliance with N-PROC-RA-0044, "Environmental Qualification Assessment" for not correcting the documentation discrepancy for Unit 5-8 Vertical Flux Detector Tefzel cables with justification for a new qualified life value. As a result, CNSC staff requested that OPG revise the Environmental Qualification Assessment for Tefzel cables to reflect the change of qualified life of the Vertical Flux Detectors. This is a gap (**Pickering PSR2 Gap SF3-6**) since the CNSC has identified this issue in an Action Notice following a regulatory inspection and the associated action (AR#28170757) is due to be completed by Q4 2016.

References

[B.1.1] N-REP-01070-0524925 (NO-2014-105), *Nuclear Oversight Report - Environmental Qualification Follow-up*, January 20, 2014.

[B.1.2] N-REP-01070-0435216 (NO-2013-111), *Nuclear Oversight Report –Environmental Qualification,* December 2, 2013.

[B.1.3] P15-000842-SA, *Self-Assessment Report - Snapshot Assessment on EQ Program as part of CNSC Type II Inspection Requirements,* March 13, 2015.

[B.1.4] NK30-CORR-00531-06595, *Pickering 5-8 Environmental Qualification (EQ) Program Implementation and Sustainability, CNSC Report#PRPD-PICKB-2012-157,* June 18, 2013.

[B.1.5] P-CORR-00531-04483, *Pickering NGS: CNSC Type II Compliance Inspection Report: PRPD-2015-005, Environmentally Qualified Equipment Inspection*, June 3, 2016.

B.2 Seismic Qualification (Per N-PROG-MP-0001 "Engineering Change Control", N-PROG-MP-0009 "Design Management", N-PROG-MA-0004 "Conduct of Maintenance" and N-PROG-MP-0008 "Integrated Aging Management")

There are a series of engineering programs and procedures used to maintain the Seismic Qualification of safety related SSCs, as per N-PROG-MP-0009, "Design Management", N-PROG-MP-0001, "Engineering Change Control", N-PROG-MA-0004, "Conduct of Maintenance", N-PROG-MP-0008, "Integrated Aging Management", and N-PROC-MA-0021, "System Performance Monitoring". Details of the audit and self-assessment results for compliance with the seismic elements of the applicable programmatic documents are described below.

Nuclear Oversight conducted a performance based audit of the ECC program for Pickering NGS in November 2014 per NO-2014-030 [B.2.1] to determine if ECC-related activities were being performed effectively and in compliance with program requirements for safe and reliable operations. In terms of Seismic related findings, it was determined that there was a misalignment between vendor documents associated with seismic classification/requirements. An SCR (N-2014-30369) was initiated which required corrective actions to be implemented. This SCR is now complete and the necessary corrective actions were finalized to address the underlying issues.

An Evaluating Organization Effectiveness review (EOER) was completed for Pickering NGS in May 2014 per P14-000541 [B.2.2] in which adverse conditions related to seismic equipment and routes were identified. The following 6 actions were generated (tracked under SCR P-2014-26922):

1) Assign a program owner to ensure compliance with N-PROC-MA-0031, "Protection of Seismic Equipment and Routes";

2) Develop a broad action plan which will address items such as management expectations, worker awareness and training, oversight of program compliance, etc;

3) Conduct a pre-effectiveness review of Corrective Action Plan (CAP) actions to determine the effectiveness of corrective actions initiated;

4) Conduct an EOER for SCR P-2014-26922;

5) Ensure oversight of the Seismic Program at the safety cornerstone meeting and review appropriate metrics as feasible; and

6) Identify training gaps within the Seismic Program and PELs to be updated accordingly.

EOER P16-000023 [B.2.3] was initiated in September 2015 (completed as per item 4 above) and determined that the corrective actions identified in SCR P-2014-26922 were effective at addressing the adverse conditions.

The above results for the audit and self-assessments therefore reveal that there are no gaps associated with implementation of the Seismic Program at Pickering NGS.

References

[B.2.1] N-REP-01070-0519973 T06, *Nuclear Oversight - Engineering Change Control Audit,* November 3, 2014*.*

[B.2.2] P14-000541-EOER, *P-2012-12185 - Field Issues around Seismic Equipment and Routes,* May 30, 2014*.*

[B.2.3] P16-000023-EOER*, EOER for P-2012-12185 CAP deemed ineffective,* March 29, 2016*.*

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ACCEPTED	✓
ACCEPTED AS NOTED	
REVISE AND RESUBMIT	
Signature 14 July 2016 **Date**	
Name: Mike Ruffolo, Manager	
Dept: Pickering Engineering - Aging Management & Strategic Initiatives	
OPG Proprietary	
Doc No.: P-REP-03680-00007 Rev: 000	
This acceptance does not relieve the contractor from responsibility for errors or omissions or from any obligations or liability under this contract.	

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Pickering NGS PSR2 Safety Factor 4 Report: Aging

PS112/RP/004 R01

July 13, 2016

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Revision Summary - For Amec Foster Wheeler Report PS112/RP/004

Rev	Date	Author	Comments
R00	May 6, 2016	R. Guhathakurta, A. Johnstone, R. Jayasundera	Initial issue for OPG review and comment.
R01	July 13, 2016	R. Guhathakurta, A. Johnstone, R. Jayasundera	Updated report addressing OPG comments on R00.

EXECUTIVE SUMMARY

OPG is performing a Periodic Safety Review (PSR) to support the possibility of continued operation of Pickering Nuclear Generating Station (NGS) beyond 2020. The PSR (referred to as "PSR2") is a subsequent PSR building on the review basis of earlier OPG Integrated Safety Reviews and other associated assessments. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

Part of PSR2 involves the preparation of Safety Factor reports for each of fifteen major topic areas. Safety Factor reports consist of:

- Assessing compliance against "Review Tasks" identified in the PSR2 Basis Document [1]. These Review Tasks are derived from Canadian Nuclear Safety Commission (CNSC) REGDOC-2.3.3, "Periodic Safety Reviews" [2] and International Atomic Energy Agency (IAEA) SSG-25, "Periodic Safety Review for Nuclear Power Plants" [3];

- Documenting assessments against applicable modern Laws, Regulations, Codes and Standards (L/R/C/Ss) as defined in Reference [1]; and

- Reviewing the effectiveness of applicable OPG Nuclear Programs that support the above assessments, through review of audit and self-assessment results.

The PSR2 review of Safety Factor 4, *Aging* is presented in this report. OPG Governance, Programs, Policies, Procedures, Instructions and Guidelines related to Safety Factor 4 were reviewed for the ten PSR2 Review Tasks specified in Section 4.1 of this report. L/R/C/S and OPG Nuclear Program audit and self-assessment reviews for both Safety Factors 2 and 4 were prepared per Sections 4.2 and 4.3, respectively. This report also includes a review of OPG commitments previously made to the CNSC, open CNSC action items, and exemptions granted by the CNSC since the current operating licence was issued (all related to Safety Factors 2 or 4), as well as identification and review of previously identified PSR1 gaps related to Safety Factor 4 (to ascertain the implications of extending Pickering NGS operation beyond 2020), per Section 4.4.

The results of the review of the Aging Safety Factor are discussed in Section 5.0 of this report. The review has confirmed that aging aspects affecting Structures, Systems and Components (SSCs) important to safety are being effectively managed and that an effective aging management program is in place at Pickering NGS. As discussed in Section 5.0, the review identified eighteen gaps that will need to be addressed further as part of the PSR2 Global Assessment process.

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LIST OF TABLES AND FIGURES

1.0 INTRODUCTION

OPG is performing a Periodic Safety Review (PSR) to support the possibility of continued operation of Pickering NGS beyond 2020.[1] A comprehensive Integrated Safety Review (ISR) was completed for Pickering Units 5 through 8 in 2009 in support of refurbishment and continued operation. Pickering 1,4 integrated safety assessments were also performed for Pickering A Return to Service (PARTS) in support of approval to restart Units 1 and 4. In addition to these Pickering-specific studies, the 2013 Darlington ISR performed extensive code and standard reviews that were updated in relation to the versions that were assessed in the 2009 Pickering B ISR.[2] These previous ISRs are considered to constitute the first PSR completed for Pickering (referred to as "PSR1"). The current PSR (referred to as "PSR2") is a subsequent PSR building on the basis of earlier OPG integrated safety assessments through review of the various studies, assessments and licence renewals performed since PSR1. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

PSR2 will support and complement the licence renewal application for Pickering NGS going forward. Fifteen Safety Factors will be assessed as part of the PSR. The purpose of Safety Factor reviews is to confirm that the design, condition and operation of Pickering Units 1,4 and 5-8 (as well as common systems) will support continued safe operation for the period of PSR2 by:

- Assessing compliance against "Review Tasks" identified in the PSR2 Basis Document [1], which were derived from CNSC REGDOC-2.3.3 [2] and IAEA SSG-25 [3];
- Documenting assessments against applicable modern Laws, Regulations, Codes and Standards (L/R/C/Ss) (as defined in Reference [1]); and
- Reviewing the effectiveness of applicable OPG Nuclear Programs that support the above assessments, through review of audit and self-assessment results.

As outlined in IAEA SSG-25 [3], the objective of the review of Aging Safety Factor 4 is to: "determine whether aging aspects affecting Systems, Structures and Components (SSCs) important to safety are being effectively managed and whether an effective aging management programme is in place so that all required safety functions will be delivered for the design lifetime of the plant." REGDOC-2.3.3 [2] requires that: "The licensee shall

[1] Currently, Pickering Units 5-8 are approved to operate to 247,000 Effective Full Power Hours. This operation limit is expected to be reached on some units in 2020. For the purposes of PSR2, OPG assumes operation of Pickering NGS for up to eight additional years, from 2020 until 2028. OPG will make a decision regarding the permanent shut down dates for the six reactors following the performance of a technical evaluation that will include PSR2, and will communicate it to the CNSC as required by the current Power Reactor Operating Licence.

[2] Much of the compliance assessment and evaluation of Safety Factor health for the Darlington ISR is based on programs and practices that apply across OPG's nuclear operations. As a result, where Pickering is confirmed to follow the same nuclear programs and practices as were assessed for Darlington, the Darlington ISR programmatic conclusions are applicable to Pickering. As discussed in Section 1.0, an effectiveness review (effectiveness at Pickering NGS) of OPG programs used to demonstrate compliance with the PSR2 Assessment Basis will be conducted using recent audit and self-assessment results.

conduct a PSR in accordance with this regulatory document for the period until the next PSR or, if applicable, until the end of commercial operation of the plant."

This report documents the results of the review of Safety Factor 4 for Pickering PSR2. The report is based on the OPG Governance, Programs, data, and material available up to January 15, 2016 which is the freeze date for PSR2.

2.0 SCOPE OF REVIEW

2.1 Review Task Assessment

The Pickering PSR2 Safety Factor 4 Review Tasks are defined in Reference [1]. Details of the derivation of these Review Tasks from CNSC REGDOC-2.3.3 and IAEA SSG-25 are shown in Reference [4]. The Safety Factor 4 Review Tasks are:

1) Confirm there is a documented method and criteria for identifying safety related SSCs covered by the Aging Management Program.

2) Ensure there is an effective Aging Management Program and dedicated organization with clearly defined roles and responsibilities with sufficient resources to continually assess aging effects in safety related SSCs.

3) Establish a list of SSCs covered by the aging management program and records that provide information in support of the management of aging.

4) Evaluate and document impact of potential aging degradation of safety related SSCs.

5) Confirm or develop understanding of dominant aging mechanisms of safety related SSCs.

6) Confirm existence of predictive maintenance program.

7) Ensure existence of programs for timely detection and mitigation of aging mechanisms and/or aging effects of any SSCs important to safety, including obsolescence of technology used in the plant or obsolescence of services or supplies external to the plant.

8) Establish acceptance criteria and required safety margin for safety related SSCs for the period of PSR2 through reliability and risk assessments.

9) Confirm adequacy of management of the effects of aging on those parts of the plant that will be required for safety when the reactor has ceased operation, for example the spent fuel storage facilities.

10) Confirm the models used to predict the evolution and advancement of aging degradation are properly supported in accordance with current accepted practices pertaining to aging degradation.

2.2 L/R/C/S Reviews

The applicable Laws, Regulations, Codes and Standards relevant to the Safety Factor for Aging are identified in Reference [1] and are listed in Table 1 below. Table 1 also identifies the modern version and date of each L/R/C/S to be considered, the Safety Factor(s) to which each document is applicable, and the type of review that will be completed in PSR2. Note that L/R/C/S reviews associated with Safety Factor 2, *Actual Condition of SSCs Important to Safety*, are also provided in this Safety Factor report.

All of the Safety Factor 2 and 4 L/R/C/S reviews are incremental in nature. The definition of an Incremental Review is as follows:

- Incremental Review: For L/R/C/Ss that have been reviewed in PSR1 but have had revisions since the last review, a topical review will be performed of the changes.

The methodology for the reviews is discussed in Section 3.2. A detailed compliance assessment for each L/R/C/S is provided in Appendix B of Reference [4]. Associated findings are summarized in Section 4.2 of this Report.

Table 1: L/R/C/Ss Reviewed for Safety Factors 2 and 4

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
L/R/C/Ss Referenced in Pickering NGS PROL 48.02/2018						
1	CSA N290.13	Environmental Qualification of Equipment for CANDU Nuclear Power Plants	N290.13-05	3, 4	Incremental	N290.13 addressed as part of Pickering B and Darlington ISRs
2	CSA N285.4	Periodic Inspection of CANDU Nuclear Power Plant Components	N285.4-14	1, 2, 4	Incremental	N285.4 addressed as part of Pickering B and Darlington ISRs
3	CSA N285.5	Periodic Inspection of CANDU Nuclear Power Plant Containment Components	N285.5-13	1, 2, 3, 4	Incremental	N285.5 addressed as part of Pickering B and Darlington ISRs
4	CSA N287.7	In-Service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants	N287.7-08	2, 3, 4	Incremental	N287.7 addressed as part of Pickering B and Darlington ISRs
5	CNSC RD/GD-210*	Maintenance Programs for Nuclear Power Plants	2012	3, 4	Incremental	S-210 and RD/GD-210 addressed as part of Darlington ISR
6	CNSC RD/GD-98	Reliability Programs for Nuclear Power Plants	2012	3, 4	Incremental	RD/GD-98 addressed as part of Darlington ISR and S-98 as part of Pickering B ISR
7	CNSC REGDOC-2.6.3*	Aging Management	2014	3, 4	Incremental	Transition plan in place and gap assessment between RD-334 and OPG Nuclear Integrated Aging Management governance performed by OPG
Additional L/R/C/Ss						
8	CSA N287.2	Material Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants	N287.2-08	1, 2, 3, 4	Incremental	N287.2 addressed as part of Pickering B and Darlington ISRs and PARTS

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
9	CSA N285.8	Technical Requirements for In-Service Evaluation of Zirconium Alloy Pressure Tubes in CANDU Reactors	N285.8-15	2, 4	Incremental	N285.8 addressed as part of Pickering B and Darlington ISRs

* Superseding documents to those currently in Pickering NGS PROL 48.02/2018.

2.3 Audit and Self-Assessment Reviews of OPG Programs

The OPG Nuclear Programs (N-PROGs) applicable to the Safety Factor for Aging are listed in Table 2 below. The methodology for the audit and self-assessment reviews is discussed in Section 3.3. The assessment results of each of the N-PROGs in Table 2 is provided in Appendix B, and findings are summarized in Section 4.3.

Table 2: OPG Programs Applicable to the Safety Factor for Aging

Document Number	Document Title
N-PROG-MP-0008 [6]	Integrated Aging Management
N-PROG-MA-0025 [8]	Major Components
N-PROG-MA-0026 [9]	Equipment Reliability
N-PROG-MA-0017 [10]	Component and Equipment Surveillance
N-PROG-MA-0004 [11]	Conduct of Maintenance
N-PROG-OP-0004 [13]	Chemistry
N-PROG-MA-0019 [15]	Production Work Management
N-PROG-MA-0016 [16]	Fuel

Audit and self-assessments results for N-PROGs associated with Safety Factor 2, *Actual Condition of SSCs Important to Safety*, are also provided in Appendix B, and findings are summarized in Section 4.3. The N-PROGs applicable to the Actual Condition of SSCs Safety Factor 2 are listed in Table 3 below.

Table 3: OPG Programs Applicable to the Actual Condition of SSCs Safety Factor

Document Number	Document Title
OPG-PROG-0009 [26]	Items and Services Management
N-PROG-MP-0001 [27]	Engineering Change Control
N-PROG-MP-0004 [28]	Pressure Boundary

2.4 Additional Reviews

The PSR2 Safety Factor 4 Report includes a review of OPG commitments previously made to the CNSC, open CNSC action items, and exemptions granted by the CNSC since the current operating licence was issued (all related to Safety Factors 2 or 4). The Report also includes identification and review of previously identified PSR1 gaps related to Safety Factor 4 to ascertain the implications of extending Pickering NGS operation beyond 2020. The methodology for these reviews is provided in Section 3.4.

In addition, any PSR2 gaps identified as a result of the Safety Factor 4 review which need to be addressed in other Safety Factor Reports are discussed in Section 4.4 of this report.

3.0 METHODOLOGY

The sub-sections below summarize the methodology used to assess Review Task and L/R/C/S compliance, and Nuclear Program effectiveness for the Aging Safety Factor.

3.1 Review Tasks

As discussed earlier, the Safety Factor Review Tasks are derived from CNSC REGDOC-2.3.3 [2] and IAEA SSG-25 [3], taking into consideration the Review Tasks used in the Pickering B and Darlington ISRs (as derived in [4]).

For each Safety Factor 4 Review Task identified in Section 2.1, confirmation of the existence of applicable OPG Governance, Programs, Policies and Procedures (as well as Instructions and Guidelines, as applicable) was performed. Compliance against Review Tasks is also assessed by reference to applicable Condition Assessments, safety analyses and operating experience, as required.

The Review Task assessments identify Compliances and Gaps as defined below:

- Compliance: Compliance indicates that either the safety requirement or the intent of the Review Task is met.
- Gap: A Gap indicates that the intent of the Review Task is not met.

3.2 L/R/C/S Reviews

The process to identify the modern L/R/C/Ss that are applicable to the PSR2 Assessment Basis involved first creating a broad list from multiple sources (potential candidate L/R/C/Ss) and then filtering it to identify those that are most significant and that are applicable to the PSR2 scope. The identification and selection criteria are detailed in Reference [1]. The result of the identification and selection process was a set of modern L/R/C/Ss that became part of the "PSR2 Assessment Basis".

PSR2 is focused on the extension of Pickering NGS operations beyond 2020, and will conduct reviews against a baseline of past PSR1 work. As a subsequent PSR, PSR2 focuses on changes in requirements, plant conditions, operating experience and new information. Since PSR2 is an update of previous ISRs, it incorporates reviews of L/R/C/Ss that have occurred as new versions have been issued. Since this assessment is a subsequent PSR, the focus is on identifying differences between what was previously assessed and what is now different within the current Pickering PSR2 Assessment Basis. In general, these differences relate to:

- More recent (new or revised) L/R/C/S versions than what was previously assessed as part of PSR1;[3]
- Safety significant differences between Pickering and Darlington, if the Darlington ISR is the basis for the earlier assessment;
- Implications of extending Pickering NGS operation beyond 2020; and
- Safety significant differences between Pickering Units 1,4 and Units 5-8.

L/R/C/Ss in the PSR2 Assessment Basis generally receive incremental reviews since PSR2 is an update of previous ISR assessments and clause-by-clause or high level reviews for the majority of the L/R/C/S in the PSR2 Assessment Basis have already been completed. Implementation plans (including gap analyses or code-over-code reviews) also exist for the latest editions of many L/R/C/Ss. As a result, incremental review is also used in circumstances where a L/R/C/S in the PSR2 Assessment Basis was not assessed in previous ISRs but an implementation plan currently exists for compliance.

The PSR2 incremental reviews in this Report include an assessment of the intent of recent changes to the L/R/C/Ss on a topic or subject-matter basis where there is potential to impact nuclear safety. Incremental reviews provide:

- A summary of the purpose of the L/R/C/S;
- Pertinent background information about the current revision of the L/R/C/S that is being considered;
- Identification of which Safety Factor(s) are applicable to the current revision of the L/R/C/S;
- A description of which version(s) of the L/R/C/S were assessed for PSR1 (i.e., Darlington ISR (for programmatic content), Pickering B ISR and PARTS code reviews);
- Identification of whether the current version of the L/R/C/S is an update of a previous version of the L/R/C/S that was assessed in PSR1 (and if so, a description of the major changes in the latest revision is provided as discussed below);

3 "New" refers to a code or standard that was not previously considered in the context of earlier assessments. "Revised" refers to an updated version of a code or standard that was previously considered in the context of earlier assessments. Where a document has a new number/type, but addresses the same topic from the same organization, it is a "revised", not "new", document (e.g., if a REGDOC replaces a CNSC G or RD document).

- An assessment of the applicability of PSR1 assessment findings (gaps and conclusions), including the implications of extending Pickering NGS operation beyond 2020 if any;
- An assessment of the applicability of assessment findings that address more recent (post-PSR1) editions of the L/R/C/S, including any implementation or transition plans that are already committed to by OPG; and
- Where PSR1 and post-PSR1 assessments are not sufficient to address changes in the latest edition of the L/R/C/S, an assessment of the changes from the previously assessed edition of the L/R/C/S (including identification of any safety significant PSR2 gaps which result).

The Safety Factor 4 L/R/C/S reviews identify Compliances and Gaps as defined below:[4]

- Compliance:
    - Where an Incremental review has been performed, Compliance indicates that the change in the safety requirement, per the topical review, is met.
    - Where a High Level review has been performed, Compliance indicates that the intent of the safety requirement is met. (Note: No High Level reviews were performed as part of Safety Factor 4.)
    - For Clause-by-Clause reviews of modern Laws, Regulations, Codes and Standards, Compliance indicates that the safety requirement is met. (Note: No Clause-by-Clause reviews were performed as part of Safety Factor 4.)
- Gap:
    - Where an Incremental review has been performed, a Gap indicates that the change in the safety requirement, per the topical review, is not met.
    - Where a High Level review has been performed, a Gap indicates that the intent of the safety requirement is not met. (Note: No High Level reviews were performed as part of Safety Factor 4.)
    - For Clause-by-Clause reviews of modern Laws, Regulations, Codes and Standards, a Gap indicates that the safety requirement is not met. (Note: No Clause-by-Clause reviews were performed as part of Safety Factor 4.)

[4] Safety Factor compliance assessments for Review Tasks and L/R/C/S reviews make use of: a) OPG Governance, Programs, Policies and Procedures which support the compliance arguments, b) Commitments previously made to the CNSC, open CNSC action items, and exemptions granted by the CNSC since the current operating licence was issued (Fukushima actions are included as appropriate as commitments or actions), c) Identification of previously identified ISR gaps related to each Safety Factor and the status of OPG's improvement plan(s) or other dispositions to address these, and d) Assessments and reviews performed since the PSR1 documents were completed.

The reviews assume that use of the word:

- "Shall" is used in an L/R/C/S to express a requirement, i.e., a provision that the licensee is obliged to satisfy in order to comply with the standard;
- "Should" is used to express a recommendation or that which is advised but not required;
- "May" is used to express an option or that which is permissible within the limits of the standard; and
- "Can" is used to express possibility or capability.

3.3 Audit and Self-Assessment Reviews

As discussed earlier, effectiveness reviews (effectiveness at Pickering NGS) of OPG programs used to demonstrate compliance with the PSR2 Assessment Basis were conducted, using recent audit and self-assessment results. Nuclear Program audit and self-assessment results were prepared for Safety Factor 4 by reviewing recent and applicable:

- OPG Program Health reports;
- OPG Nuclear Oversight independent performance based Program audits (typically performed in 1 to 5 year cycles) and self-assessments (typically performed on a yearly basis). This includes review of associated Station Condition Records and Action Requests to confirm that any findings have been completed; and
- CNSC "Type I" and "Type II" inspections of the effectiveness and performance of OPG programs, where called-up by OPG Program Health reports, audits or self-assessments.

The focus of these reviews was on effectiveness of the Programs at Pickering NGS, where specific information is available.

The list of Nuclear Programs to be assessed for each Safety Factor was derived from review of current OPG Governance, and has used the most recent version of these documents as of the PSR2 freeze date of January 15, 2016.

3.4 Additional Reviews

The PSR2 Safety Factor 4 assessment includes identification and review of previously identified PSR1 gaps related to Safety Factor 4 (as identified in the Pickering B and Darlington Integrated Implementation Plans (IIPs) [32][33] and Pickering Units 5-8 Continued Operations Plan (COP) [34]) to ascertain the status of OPG's improvement plan(s) or other dispositions to address these and the implications of extending operation beyond 2020 (if any).[5]

A review was also performed of the following for both Safety Factors 2 and 4 to determine if there are any impacts associated with operation of the Pickering Units past 2020:

- Commitments previously made to the CNSC in the R04 Pickering Licence Condition Handbook (LCH) [31];
- Open CNSC action items in the R04 Pickering LCH [31]; and
- Exemptions granted by the CNSC since the current operating licence was issued per the R04 Pickering LCH [31] (Fukushima actions are included as appropriate as commitments or actions).

Any PSR2 gaps identified as a result of the Safety Factor 4 review which need to be addressed in other Safety Factor Reports are also discussed.

[5] PSR2 includes consideration and confirmation that the findings of PSR1 remain valid, as applicable, for the operation period. This includes assessment of PSR1 conclusions against implications resulting from extended operation. In particular, Pickering PSR1 results are applicable to PSR2 if there was a PSR1 gap that is still open, or if a closed PSR1 gap could be affected by extended operation. If so these gaps are carried forward into PSR2 for consideration in the Global Assessment. (When references to PSR1 are made, the source document is identified and the relevant text from that source document is summarized in the context of PSR2.) With respect to the Darlington ISR, much of the evaluation of Safety Factor health is based on programs and practices that apply across OPG's nuclear operations. As a result, Darlington ISR programmatic conclusions are applicable to Pickering PSR2 where Pickering is confirmed to follow the same nuclear programs and practices that were assessed for Darlington. Darlington PSR1 results are applicable to Pickering PSR2 if there are Darlington PSR1 gaps that are found to be relevant to Pickering PSR2.

4.0 REVIEW FINDINGS

4.1 Review Tasks

The sub-sections below provide an assessment of the adequacy of applicable OPG Governance, Programs, Policies and Procedures (as well as Instructions and Guidelines, as applicable) in demonstrating compliance against the Safety Factor 4 Review Tasks.

4.1.1 Review Task #1: Criteria for Identifying Safety Related SSCs

Confirm there is a documented method and criteria for identifying safety related SSCs covered by the Aging Management Program.

The Integrated Aging Management (IAM) Program, N-PROG-MP-0008, "Integrated Aging Management" [6] ensures the condition of critical Nuclear Power Plant (NPP) equipment is understood and that required activities are in place to assure the health of these components and systems while the plant ages. The IAM program covers all critical SSCs having nuclear safety, production, cost, conventional safety and environmental significance.

Aging Management (AM) is managed through the implementation of several programs including:

- N-PROC-MP-0060, "Aging Management Process" [7];
- N-PROG-MA-0025, "Major Components" [8];
- N-PROG-MA-0026, "Equipment Reliability" [9];
- N-PROG-MA-0016, "Fuel" [16]; and
- N-PROG-MA-0017, "Component and Equipment Surveillance" [10].

N-PROC-MP-0060 "Aging Management Process" [7] is the predominant method for identifying SSCs to be included in the Aging Management Program. Selection of components to be included in the aging management scope is based on their criticality as determined through the process outlined in N-PROC-MA-0077, "Critical Equipment Identification and Categorization" [19]. Critical components that have effective maintenance practices (for example Periodic Inspection) that address Age Related Degradation Mechanisms (ARDMs) are excluded.

N-PROG-MA-0025, "Major Components" establishes an integrated set of processes and activities to justify fitness for service of Feeders (N-PLAN-01060-10001, "Feeders Life Cycle Management Plan" [35]), Fuel Channels (N-PLAN-01060-10002, "Fuel Channels Life Cycle Management Strategy and Plan" [36]), Reactor Components & Structures (N-PLAN-01060-10003, "Reactor Components and Structures Life Cycle Management Plan" [37]), and Steam Generators (N-PLAN-33110-10009, "Steam Generators Life Cycle Management

Plan" [38]). N-PROG-MA-0025, "Major Components" develops long-term Life Cycle Management strategies that support continued fitness for service.

It is noted that N-PROG-MA-0016, "Fuel" [16] establishes requirements to integrate and review Fuel-related data in order to ensure fuel performs safely and reliably over the life of the stations, consistent with design and licensing bases, while optimizing station reliability, production, and cost effectiveness. Fuel-related data includes any information that may impact fuel throughout its life-cycle including (but not limited to) manufacturing, inspections, research, station operating conditions, and fuel channel interactions. Also included is fuel channel data that may impact safety analysis, or the safety report, however, this program does not include responsibilities for fuel channel life-cycle management and fitness for service that are covered by N-PROG-MA-0025, "Major Components" [8].

Life Cycle Management Plans (LCMPs) outline the requirements to manage the aging of the major components and associated sub-systems which are developed based on the methodology identified in N-PROG-MP-0008, "Integrated Aging Management" [6]. These documents as well as the procedural methods used in preparing and revising LCMPs collectively ensure that critical NPP equipment is understood and that required activities are in place to support component and overall system health as the plant ages. LCMPs are to be updated within a timeframe that allows for the specified requirements (e.g., inspections) to be captured in the business planning cycle.

The Major Components program specifies the requirements for monitoring, integrating, and assessing information related to Feeders, Steam Generators, Reactor Components & Structures, and Fuel Channels. It also details the documentation requirements associated with demonstrating compliance with the requirements and limits applicable to the four major components areas. The program addresses all processes that impact or have the potential to impact the integrity and performance of the major components, including design, operation, chemistry, inspection, maintenance, and modifications.

N-PROG-MA-0026, "Equipment Reliability" [9] is the governing document that establishes the process for Equipment Reliability (ER) for critical components. The ER Program and its implementing procedures ensure that critical components meet their defined/desired level of reliability for the lifespan of the NPP.

N-PROG-MA-0017, "Component and Equipment Surveillance" [10] Program defines requirements for establishing programs to ensure the health of select nuclear power plant components and equipment. It also provides additional focus and program oversight which provides assurance that licensing, reactor safety, equipment reliability, environmental and conventional safety requirements are met on an ongoing basis. The program identifies component programs, ownership, roles, accountabilities, and component monitoring activity requirements driven by regulatory, safety (nuclear, conventional, environmental), and business (production) requirements. Not all component and equipment lines warrant the oversight of a unique component program, as critical components that have effective maintenance practices that address the ARDMs are excluded. Components not in the scope of N-PROG-MA-0017 are managed under N-PROG-MA-0025, "Major Components" [8] or N-PROG-MA-0026, "Equipment Reliability" [9].

The Component and Equipment Surveillance Program addresses both components and program health. Steam generators, fuel and fuel channels, feeder piping, and reactor components are excluded, as they are covered by formal Life Cycle Plans. Component programs within the scope of the Component and Equipment Surveillance Program include: Heat Exchangers, Power Operated Valves, Check Valves, Periodic Inspection of Pressure Retaining Nuclear Components, Periodic Inspection of Containment Components, In-service Examination and Testing of Concrete Containment Structures, Pressure Relief Devices, Buried Piping, Pipe Wall Thinning (Microbial Induced Corrosion, Flow Accelerated Corrosion), Lubrication, Pressure Vessel Certification, Calibration Process Engineering Requirements, Valve Stem Packing, Machine Safeguarding, Pump Testing, Critical Pipe Support Inspection and Cable Surveillance Program. This program defines the requirements for establishing component programs that manage component and equipment health including inspection, maintenance, certification and testing [10].

These specific component programs establish requirements relating to the following:

1. Performance criteria and monitoring: Parameters are in place to ensure that limiting conditions and equipment performance can be monitored to ensure that nuclear safety, reliability, and availability targets are met as follows:

    a. To identify specific alert levels and acceptance criteria for condition monitoring;

    b. To establish mission time testing requirements where required for specified poised and stand-by safety related equipment;

    c. To identify leading indicators that predict performance as well as indicators based on failures;

    d. To relate monitored parameters and acceptable levels of performance to measureable indications of component degradation.

2. Licensing and Regulatory requirements: Activities are in place to comply with Licensing and Regulatory requirements by ensuring special testing, design reviews and evaluations are identified and conducted;

3. Maintenance and Condition Monitoring: Activities are specified utilizing predictive and preventive maintenance processes;

4. Establishing action levels and to initiate action to correct problems or deviations from expected performance; and

5. Understanding, identifying and managing equipment aging issues by defining maintenance, inspection and other activities necessary to manage component aging.

N-PROC-MA-0077, "Critical Equipment Identification and Categorization" [19] is one of the implementing procedures under the ER umbrella. It identifies criticality coding and single point vulnerabilities. This procedure establishes a process for identification of critical components that is consistent with the guidance provided by INPO AP913, "Equipment Reliability Process Description". This includes:

- Identifying the basis and considerations for categorization and significance in supporting plant functions;
- Defining high, low, non-critical, run to maintenance and single point vulnerable components; and
- Application of component categorization methodology.

Part of the Aging Management Process (AMP) is the ongoing evaluation of the condition of critical SSCs. This is accomplished through the condition assessment process, which identifies actions required to assure the health of the SSCs as the plant ages. N-PROC-MP-0060, "Aging Management Process" [7] describes the process used to perform condition assessment of critical components.

The AMP outlines the process of scope definition and screening to identify the critical components requiring condition assessment. This includes:

- Identifying the SSCs subject to screening and defining the boundaries of each system;
- Grouping the components in order to manage the screening and condition assessment in an efficient manner; and
- Ensuring the effectiveness of the condition assessment process.

The AMP also provides guidance in preparing condition assessments to identify actions required to assure health of SSCs as the plant ages.

For other station components that do not have a component program or established periodic activities, Condition Assessments (CAs)[6] are prepared to determine and document the actual condition and to identify actions to maintain performance. CAs document the current condition of SSCs, relevant aging mechanisms and any actions required to mitigate aging related degradation. CAs are issued in Asset Suite and are also stored in a CA database which is reviewed and updated to incorporate new information (e.g. OPEX). System health reports summarize the results of Condition Assessments and identify action plans required to resolve aging related issues.

Conclusion:

The procedures and processes identified above provide a documented method and criteria for identifying safety related SSCs covered by the Aging Management program. The intent of Review Task #1 is met and therefore Pickering NGS is compliant.

[6] The terminology currently used is Condition Assessment (CA) instead of Component Condition Assessment (CCA). "Condition Assessment" will be used throughout this Report.

4.1.2 Review Task #2: Aging Management Program Effectiveness

Ensure there is an effective Aging Management Program and dedicated organization with clearly defined roles and responsibilities with sufficient resources to continually assess aging effects in safety related SSCs.

N-PROG-MP-0008, "Integrated Aging Management" [6], uses CNSC Regulatory Document RD-360, "Life Extension of Nuclear Power Plants", February 2008, CNSC Regulatory Document RD-310, "Safety Analysis for Nuclear Power Plants", February 2008 and CNSC Regulatory Document RD-334 , "Aging Management of Nuclear Power Plants", June 2011 as the bases for its program [6].

The IAM Program, N-PROG-MP-0008, "Integrated Aging Management" ensures the condition of critical NPP equipment is understood and that required activities are in place to assure the health of these components and systems while the plant ages [6].

This is accomplished by establishing an integrated set of programs and activities that ensure the performance requirements of all critical station equipment are met on an ongoing basis. These programs and activities all serve an integral function to ensure critical equipment degradation due to aging is managed such that operation of the NPP remains within the licensing basis of the facility and allows for station operational goals to be met.

The program also requires preparation of life cycle plans for critical plant equipment. The purpose of these plans is to determine and document actions required to ensure plant equipment will meet all design and operating objectives over the life of the plant in consideration of aging. The life cycle plans are established by a comprehensive condition assessment process. Condition assessments supplement the ongoing engineering surveillance activities in place which monitor and optimize system performance.

The IAM program consisting of this set of integrated programs and activities, provides for a sound technical basis for achievement of design life. The implementation of the integrated set of programs and activities to support the IAM program follows a "Plan-Do-Check-Act" feedback model, as shown in Figure 1, below, from N-PROG-MP-0008, "Integrated Aging Management" [6]. One of the objectives of the Integrated Aging Management Program is to ensure the actions required to meet equipment reliability objectives are identified and input into the system health reports.

To meet the requirements of the "Plan-Do-Check-Act" model, the Integrated Aging Management program interfaces with a number of programs. The programs listed below also have a role in managing equipment degradation and aging. These programs include:

- N-PROG-MA-0026, "Equipment Reliability" [9]
- N-PROG-MA-0019, "Production Work Management" [15]
- N-PROG-MA-0004, "Conduct of Maintenance" [11];
- N-PROG-OP-0001, "Nuclear Operations" [12];

- N-PROG-OP-0004, “Chemistry” [13]; and
- N-PROG-RA-0016, “Risk and Reliability Program” [14].

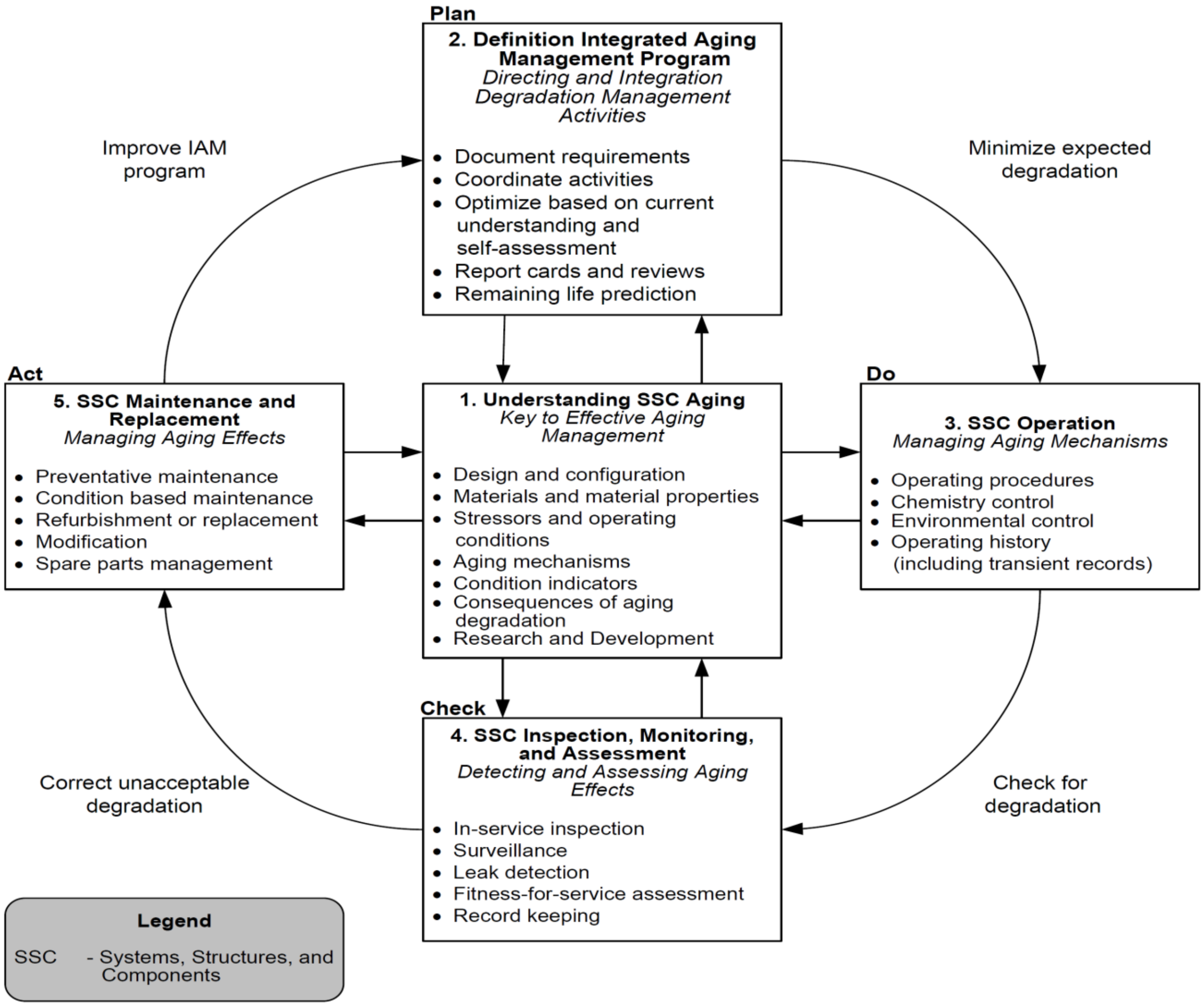

Figure 1: Aging Management Model

The Equipment Reliability program ensures that there are defined activities ensuring that equipment aging issues are identified, understood and effectively managed for equipment important to nuclear safety and equipment reliability. Any Aging Management station actions are captured in the system health reports and managed through the Production Work Management program.

The Production Work Management program specifies the requirements for identifying, prioritizing, planning, scheduling and performing work in support of plant operation, maintenance and modifications. As well, it establishes safe, uniform and efficient work control practices. Work management is the process by which maintenance, modifications, surveillances, testing, engineering support, and any work activities that require plant coordination or schedule integration are implemented. Accountabilities in this process are established from work initiation to work completion, and compliance is ensured through monitoring [15].

P-INS-06931-00003, "Accountability and Ownership for Pickering A and Pickering B Systems and Structures" [39] assigns ownership and responsibility for the SSCs common to both plants located at the Pickering site. Accountability for the scheduling of work pertaining to these SSCs rests with the organization defined in the Production Work Management program. Many of the documents contained in these programs are referred to or provide specific direction for the planning of activities and the performance of work. Common priority systems, work processes and methodologies are consistently applied in scheduling all work in support of the operation, maintenance, and modification of nuclear facilities across the fleet. The Production Work Management program facilitates the prioritization and scheduling of work for SSCs. It is through the work management system that the resources identified in the Production Work Management program [15] are assigned to perform the work.

The maintenance program includes activities required to ensure continued equipment reliability in accordance with defined equipment strategies, consisting of preventative and predictive elements. The implementation of this program is such that information from maintenance activities is fed back to engineering departments to enable effective completion of condition assessments. Results of condition assessments are implemented into maintenance programs as required to optimize equipment performance while the plant ages. These activities are highlighted in N-PROG-MA-0004, "Conduct of Maintenance" [11].

Equipment aging can be sensitive to the operating environment seen by the components. Systems are to be operated and tested consistent with approved operating procedures and chemistry specifications, to ensure aging degradation experienced remains as documented in the design basis and completed condition assessments. These activities are addressed in N-PROG-OP-0001, "Nuclear Operations" [12] and N-PROG-OP-0004, "Chemistry" [13].

N-PROG-RA-0016, "Risk and Reliability Program" [14] ensures that the reliability of systems important to nuclear safety meet requirements and that station public safety risk goals are met. In consideration of equipment aging, failure rates representative of actual plant condition are used in these analyses to ensure the conclusions are valid.

N-PROG-MP-0014, "Reactor Safety Program" [18] defines activities related to nuclear safety analysis including defining the Safe Operating Envelope (SOE). The aging management program establishes the activities required to maintain equipment performance, such that the requirements of the safety analysis and licensing basis are met. Safety limits for equipment performance used in condition assessments are defined in the SOE. Where component aging is not practical to mitigate, the rate of degradation must be determined and taken into account in the safety analysis (e.g., pressure tube diametrical creep) [6].

N-PLAN-01060-10009, "Integrated Aging Management Self-Assessment Plan" specifies the scope and the schedule of the IAM self-assessments in a three-year cycle [40]. The scope includes program execution, performance elements, and comparison against industry standards and best practices. Subsequent to this, per section 1.9.2 of N-PROC-MP-0060, "Aging Management Process" [7], the IAM program self-assessments are conducted at the site and corporate levels at a frequency determined by the Manager, Engineering Programs Integration department in accordance with N-PROC-RA-0097, "Self-Assessment and Benchmarking" [41] to measure the effectiveness of the AMP [7].

The Integrated Aging Management program, N-PROG-MP-0008, "Integrated Aging Management" [6] and its implementing document N-PROC-MP-0060, "Aging Management Process" [7] identify and define the roles of the owners and responsible persons involved in the AMP. The responsibility for the IAM program is split between corporate groups and the station:

- The Director, Components Engineering is the Integrated Aging Management program owner who provides oversight of the IAM program and engineering programs with aging management related activities to ensure program effectiveness;

- The Director of Station Engineering ensures program requirements in this document are implemented at the station and ensures adequate resources are available to effectively address aging management of critical components as required;

- The Manager, Engineering Program Integration Department monitors program implementation for consistency and compliance with program requirements, and monitors adequacy of the program through specific program performance, review of audit findings, benchmarking, and Station Condition Records [6], [7].

Ultimately, the responsibility of providing sufficient resources to continually assess aging effects in safety related SSCs is with the Directors and Managers stated above. In order to meet the requirements of the IAM, they provide the support to the staff so they can perform Aging Management engineering activities. The scope of IAM self-assessments confirms that sufficient resources are in place to sustain the program. Note that audit and self-assessments results for N-PROGs associated with Safety Factor 4 are summarized in Section 4.3.

Conclusion:

The procedures and processes identified above verify that there is an effective Aging Management Program and a dedicated organization with clearly defined roles and responsibilities with sufficient resources to continually assess aging effects in SSCs important to safety. The intent of Review Task #2 is met and therefore Pickering NGS is compliant.

4.1.3 Review Task #3: Aging Management Program SSCs and Records

Establish a list of SSCs covered by the aging management program and records that provide information in support of the management of aging.

N-PROC-MP-0060, "Aging Management Process" [7] describes the process used to perform condition assessments of critical components which uses a systematic and comprehensive approach consisting of the following two steps:

(a) Scope definition and screening to identify the critical components requiring condition assessment.

(b) Condition assessment to identify actions required to assure health of SSCs as the plant ages.

The purpose of scoping is to:

- Identify the SSCs subject to screening and define the boundaries of each system; and

- Group the components in order to manage the screening and condition assessment in an efficient manner.

The purpose of screening is to ensure the effectiveness of the condition assessment process and to apply the most effort to:

- Critical components;

- Components posing the highest risk both from a nuclear safety and economic perspective; and

- Components most susceptible to aging degradation.

The main steps in the screening process are:

(a) Screen-out components based on Criticality Code (CC).

(b) Screen-out major components in the scope of the SSC-specific aging management programs.

(c) Screen-out components that are non-safety related, unless they are Single Point Vulnerabilities (SPV) or have a cost-benefit impact.

(d) Screen-out non-AM critical components.

(e) Screen-out components with effective practices in place.

For the remaining components, a condition assessment report is prepared. The purpose of condition assessment is to evaluate the following:

- Possible and actual aging related degradation mechanisms (ARDM).

- Preventive actions to minimize and control ARDMs.

- Methods for detecting, monitoring and trending aging effects.

- Methods for mitigating aging effects.

The condition assessment evaluates the actual condition of the SSCs and recommends actions to maintain health as the plant ages.

The sustaining aspects of the IAM program are accomplished by periodically reviewing and assessing component condition based on new observations and information.

Also as described in N-PROC-MP-0060, "Aging Management Process" [7], a data collection and record keeping system is required to support aging management activities and to provide a basis for decisions on the type and timing of aging management actions. Management of data is in accordance with N-PROG-AS-0006, "Records and Document Control" [42]. This program governs management of records and documents throughout their life cycle.

Data and records relevant to aging management are divided into the following categories:

- Baseline information, consisting of data on the design and the condition at the beginning of the service life of SSCs;

- Operating history, covering service conditions and data on testing of availability and failure of SSCs;

- Maintenance history, including data on the monitoring of the condition and maintenance of SSCs; and

- Records of SSC screening and condition assessments.

All records are maintained in an approved records repository, in accordance with OPG-PROC-0019, "Records and Document Management" [43].

Documents that describe the current condition of components and identify actions required to maintain or restore performance, or ensure fitness for service, per the IAM program include the following:

- Life Cycle Management Plans (in N-PROG-MA-025, "Major Components" [8]);

- Condition Assessments (in N-PROC-MP-0060, "Aging Management Process" [7]);

- System Health Reports (in N-PROC-MA-0024, "System Performance Monitoring" [20]);

- Component Health Reports (in N-PROG-MA-0017, "Component and Equipment Surveillance" [10]);

- Inspection Program Reports (in N-PROG-MA-0017, "Component and Equipment Surveillance" [10]);

- Test Program Reports (in N-PROG-MA-0017, "Component and Equipment Surveillance" [10]); and

- Engineering Program Health Reports (in N-PROG-MA-0017, "Component and Equipment Surveillance" [10]).

Life Cycle Management Plans are developed and implemented for the major components (Feeders, Steam Generators, Fuel Channels and Reactor Components and Structures) in accordance with N-PROG-MA-0025, "Major Components" [8].

These documents outline the requirements to manage the aging of the major components and associated sub-systems which are developed based on the methodology identified in N-PROG-MP-0008, "Integrated Aging Management" [6]. These documents and the procedural methods used in preparing and revising LCMPs collectively ensure that critical Nuclear Power Plant (NPP) equipment is understood, and that the required activities are in place to support component and overall system health as the plant ages. LCMPs are updated within a timeframe that allows for the specified requirements (e.g., inspections) to be captured in the business planning cycle which is identified in the specific LCMP [6].

Per N-PROG-MA-0025, "Major Components" [8], the following elements are addressed in the Component Aging Management Strategies and Plans:

- Strategy and rationale for inspection scopes and schedules based on criteria that consider areas at the highest risk for the degradation mechanisms identified for each component or sub-system. This is considered within the context of the current understanding of the factors (environmental and otherwise) that affect the degradation mechanisms and corresponding rates;

- Acceptance and repair criteria for inspections;

- Identification of preventative and required maintenance activities;

- Identification of required modifications.

Aging Management practices identified are implemented through System Health Reports, which are created in accordance with N-PROC-MA-0024, "System Performance Monitoring" [20]. The actions from the Condition Assessments and Component Health Reports are monitored, tracked and documented in the System Health Report (or through other traceable means such as action tracking assignments, or in the work management system [7]) until completion. Other records used to support the System Health Reports include; Inspection Program Reports, Test Program Reports and Engineering Program Health Reports [20].

Component Health Reports are for components that are part of component programs. These component programs are covered under N-PROG-MA-0017, "Component and Equipment Surveillance" [10], which identifies a set of activities (component, inspection and test programs) to assure the health of a select group of components. Component Health Reports summarize component failures, inspection results, and performance trends. The reports also review current preventive and predictive maintenance practices to ensure that existing Aging Management measures are effective. The Component Health Reports also report on the health of the component program [10].

Inspection and test programs are implemented for components and equipment that have been identified as requiring a documented process for mandated inspection and test activities required by licensing, codes and standards, and regulatory commitments. Examples of inspection and test programs in N-PROG-MA-0017 include Periodic Inspection of Nuclear Pressure Retaining Components (CSA N285.4) and Containment Components (CSA N285.5), In-Service Examination and Testing of Concrete Containment (CSA N287.7), and Pressure Relief Devices (CSA N285.0 and B51). Inspection and test reports are issued following each inspection or test campaign. These reports document inspection or test results and any corrective actions taken to mitigate aging degradation. Additionally, Engineering Program Health Reports report on the status of program implementation execution for all the programs. Component Health Reports and Engineering Program Health Reports are issued at minimum once a year [10].

Conclusion:

The procedures and processes identified above confirm that a list of SSCs covered by the aging management program and records providing information in support of the management of aging have been established. The intent of Review Task #3 is met and therefore Pickering NGS is compliant.

4.1.4 Review Task #4: Aging Degradation of SSCs

Evaluate and document impact of potential aging degradation of safety related SSCs.

N-PROC-MA-0077, "Critical Equipment Identification and Categorization" [19] provides the methodology for categorizing structures and components into critical rankings based on their importance. Criticality rankings ensure reactor safety, production, cost, conventional safety, public safety and environmental requirements are satisfied. Criticality rankings are used to define suitable maintenance strategies, prioritize work activities and ensure appropriate levels of other programmatic activities are in place [19].

Considerations for categorizations include:

- Determination of component criticality is consequence-based. The higher the consequence associated with component failure, the higher the criticality ranking.
- Normal, accident and post-accident functions of a component should be considered.
- Consideration should be given to the role of a component in a system. In a critical system, not all components are critical and each should be examined separately.

The initiator ensures the plant components are categorized into the following categories:

1. Highly critical components - Criticality Code 1 (CC1): CC1 components are components defined as SPV where failure of a single component will result in an immediate (within 24 hrs) reactor trip, turbine trip, manual shutdown (including shutdowns within 24hrs due to risk to public safety) and/or a stepback of >10%. The highest level of preventive, predictive, testing surveillance and other activities would normally be in place for CC1 equipment to support the objective of "zero tolerance for their failure".

2. Low critical components - Criticality Code 2 (CC2): CC2 components exclude CC1 and are components whose individual failure would cause impairment of the reactor or the turbine, which would result in a shutdown of the station in 7 days if not rectified (including shutdowns within 7 days due to risk to public safety), de-rates the station by 5 to 10%, or reduces the redundancy of a System Important to Safety (SIS). For CC2 components a moderate amount of preventive maintenance (PM) or predictive maintenance (PdM) would normally be in place.

3. Non-critical components - Criticality Code 3 (CC3): CC3 components exclude CC1, CC2 and are components whose individual failure would cause a de-rate of the station less than 5%. CC3 components are also deemed to be more cost effective to maintain than replace providing a savings of greater than $500,000, or cause a shutdown of the station due to impairment after 7 days (including shutdowns after 7 days due to risk to public safety). For CC3 components basic PM or PdM activities would normally be in place. Risk may be acceptable if PM or PdM is not performed. An evaluation of the benefits of PM activities may be performed on a case by case basis to determine the value returned.

4. Run to maintenance components - Criticality Code 4 (CC4): CC4 components exclude CC1, CC2, and CC3 and are components that can tolerate equipment unavailability for extended periods of time without causing a trip, or de-rating of the station due to the components individual failure. These components are operated until they require maintenance or fail. Run to Maintenance (RTM) in this context means preventive or predictive maintenance is not performed on the component. Periodic testing may still be performed on some RTM components where no specific maintenance activities are performed. PM or PdM would normally not be in place for CC4 equipment.

N-PROC-MP-0060, "Aging Management Process" [7] outlines the process and provides guidance on scope definition and screening to identify the critical components requiring condition assessment. The AMP also provides guidance on preparing condition assessments to identify actions required to assure health of SSCs as the plant ages [7].

SSC screening and condition assessments are documented. The following are recorded and stored in such a way that they are secure and retrievable:

- Screening records are documented as Quality Assurance (QA) records and retained in Asset Suite. The screening records provide the disposition, i.e., technical justification, assessments, and rationale to screen out the SSCs from further evaluation.

- Condition assessments are documented as QA records and retained in Asset Suite as controlled documents. Condition assessments evaluate the condition of the SSC and recommend actions to minimize and control ARDMs.

- The recommended actions are traceable; for example, as action tracking assignments, in system health reports action plans, or in work management.

A condition assessment application (i.e., AM database), has been developed in order to facilitate all steps required for this process, i.e. scoping, component grouping, screening, and condition assessment. This application is not the official repository of the aging management information. The outputs of this application (i.e. Screening records, and Condition Assessment Reports) are reviewed and approved, and then issued in Asset Suite as official records.

Degradation mechanisms are documented in the System Performance Monitoring Plans (SPMP), Equipment Strategy Manuals (for components addressed in component programs) and external sources such as industry reports (external OPEX). Sources such as the Electric Power Research Institute (EPRI) Aging Assessment Field Guide and IAEA documents supplement these sources by providing extensive information on aging mechanisms. These sources are used in assessing and addressing degradation mechanisms that are likely to affect particular plant components.

Per Section 1.5.4.1 of N-PROC-MP-0060, available sources for degradation mechanism limits include:

- SPMPs;
- Operating Manuals;
- Impairments Manuals;
- Operational Safety Requirements (OSRs); and
- Design Manuals and reports for limits not addressed in the Safety Report or OSRs.

To assess the effects of degradation mechanisms, the as-found condition of a component is compared against baseline data and the effects of the degradation are determined by comparing current condition to the baseline. The baseline can be retrieved from the manufacturer's recommendations, operating limits established by applying appropriate margin to safety analysis limits, and OPEX where appropriate. For those components where no established baseline exist, engineering judgement based on past experience is used to assess degradation.

If a component group is not screened out, a condition assessment is prepared to evaluate the condition and recommend any additional actions required to maintain the health of the component. The report contains the following information:

- Component ID(s) indicating system application.
- Degradation mechanisms and limits, including whether obsolescence is applicable.
- Applicable environmental conditions (e.g., temperature, radiation).
- Operational factors including stressors.
- Applicable operating history including a review of pertinent operating transients.
- Aging management practices in place.

- Review of component aging information sources.
- Current physical condition of component.
- Actions required to maintain performance.

The condition evaluation technique depends on the nature of the component, its degradation mechanisms and available information. One of the following evaluation techniques is used to determine the component condition:

(a) In most cases, the condition of a component can be directly determined by a review of obtained results pertaining to its performance, such as from maintenance, testing, and inspections.

(b) In some cases, where condition cannot be determined based on observed results (e.g., piping not subject to periodic inspection), special testing or inspections may be required. Also, specific analyses or Research & Development (R&D) may be warranted (e.g., stress analysis for mechanical components).

(c) In special cases, component condition is determined by Time Limited Aging Assessment (TLAA). TLAA considers parameters either directly or typically associated with operating time. Examples of these parameters considered in TLAA are:

  - Number of allowable stress cycles, fatigue usage.
  - Corrosion rate.
  - Crack growth rate.

Where unanticipated aging is detected, it is reported in a Station Condition Record in accordance with N-PROC-RA-0022, "Processing Station Condition Records" [44], and evaluation and mitigation is managed through N-PROG-RA-0003, "Corrective Action" [17].

Conclusion:

The procedures and processes identified above indicate there is an evaluation and documentation of the impact of potential aging degradation of SSCs important to safety. The intent of Review Task #4 is met and therefore Pickering NGS is compliant.

4.1.5 Review Task #5: Dominant Aging Mechanisms

Confirm or develop understanding of dominant aging mechanisms of safety related SSCs.

N-PROG-MA-0017, "Component and Equipment Surveillance" [10] program defines requirements for establishing programs to ensure the health of unique critical nuclear power plant components and equipment. The program identifies component programs, ownership, roles, accountabilities, and component monitoring activity requirements driven by regulatory, safety (nuclear, conventional, environmental), and business (production) requirements. Not all component and equipment lines warrant the oversight of a unique component program, resulting in components or equipment excluded from this program [10]. Components not in the scope of N-PROG-MA-0017 are managed under N-PROG-MA-0026, "Equipment Reliability" [9]. In general, the scope of components and equipment in a component program are critical components as categorized in N-PROC-MA-0077, "Critical Equipment Identification and Categorization" [19].

The Component and Equipment Surveillance Program addresses both components and program health. Steam generators, fuel and fuel channels, feeder piping, and reactor components are excluded, as they are covered by formal Life Cycle Plans under N-PROG-MA-0025, "Major Components".

Component programs within the scope of the Component and Equipment Surveillance Program include: Heat Exchangers, Power Operated Valves, Check Valves, Periodic Inspection of Pressure Retaining Nuclear Components, Periodic Inspection of Containment Components, In-service Examination and Testing of Concrete Containment Structures, In-service Examination and Testing of Concrete Containment Structures, Pressure Relief Devices, Buried Piping, Pipe Wall Thinning (Microbial Induced Corrosion, Flow Accelerated Corrosion), Lubrication, Pressure Vessel Certification, Calibration Process Engineering Requirements, Valve Stem Packing, Machine Safeguarding, Pump Testing, Critical Pipe Support Inspection and the Cable Surveillance Program [10].

Program deliverables include component health reports, program health reports, and component program self-assessments.

System performance is assessed by collecting data from plant sources that is trended, analyzed, and reported as part of the system health report, as identified in N-PROC-MA-0024, "System Performance Monitoring" [20]. The SPMP includes functional failure evaluations, where the objective is to prevent system functional failures through the measurement, monitoring, trending, analysis, and correction of component or equipment functional failures for typical CC1 and CC2 components. The critical system functions and major components and equipment that provide those functions are identified. Failure mechanisms are then determined, and the parameters for measuring degradation mechanisms are monitored and trended with the intent of preventing any equipment functional failures.

The SPMP also includes the Performance Monitoring Equipment List (PMEL) which includes a column for degradation mechanisms. The degradation mechanisms and indicators are defined by taking the following into consideration:

1. Associated physical, electrical, mechanical, cyber, and chemical properties that can cause degradation, as well as short and long-term aging and operational wear processes.

2. Consultation with appropriate technical experts and specialists as required.

3. Sources of information on degradation mechanisms including but not limited to equipment/component strategy documents, maintenance histories, OPEX, vendor

manual and EPRI reports. (This information is also found in N-PROC-MP-0060, "Aging Management Process".)

Identifying and understanding component degradation mechanisms is critical in the condition assessment process. Component degradation mechanisms and degradation mechanism limits are one of the inputs in the screening. N-PROC-MP-0060, "Aging Management Process" specifies that, during the screening process, a preliminary assessment is to be performed for each Component Group based on degradation mechanisms and degradation mechanism limits, and identifies the practices in place to manage degradation. Understanding of dominant aging mechanisms is developed through required training [7].

Staff implementing aging management activities defined in this document are trained and qualified in accordance with N-TQD-403-00001, Nuclear Engineering Support Personnel Training and Qualification Description [66]. Specifically Program Element (PEL) 68601, "Aging Management" is a qualification which represents specific nuclear fundamentals training which are required across all duty areas within Engineering [66].

Conclusion:

The procedures and processes identified above confirm that the Aging Management Program results in understanding of dominant aging mechanisms of safety related SSCs. The intent of Review Task #5 is met and therefore Pickering NGS is compliant.

4.1.6 Review Task #6: Predictive Maintenance Program

Confirm existence of a predictive maintenance program.

N-PROC-MA-0034, "Predictive Maintenance Program Requirements" [24] is an implementing document of N-PROG-MA-0026, "Equipment Reliability Process" [9]. N-PROC-MA-0034, "Predictive Maintenance Program Requirements" establishes a process that manages the Predictive Maintenance (PdM) Program [24]. The PdM program applies to specific critical equipment (Criticality Code 1, 2 and possibly 3) identified through N-PROC-MA-0077, "Critical Equipment Identification and Categorization" [19] to ensure early detection of deteriorating equipment components, to ensure safe and reliable operation.

The Predictive Maintenance program examines and trends critical component data to assess immediate signs of premature aging. PdM test and inspection data are trended using baseline or previous data for reference to determine extent of degradation. Test and inspection data are trended to determine the extent of degradation. Abnormal inspection and test data is reviewed to determine the extent of equipment degradation, and to identify any additional inspection and testing requirements and mitigating actions required to prevent failure. The Predictive Maintenance program has the capability to facilitate identification of precursors for component and equipment degradation.

Briefly, the PdM process encompasses the following steps:

- Equipment selection, including documentation of PM bases;

- Develop/review alarm limits;
- Determine periodicity of PdM monitoring for equipment set by failure types while accounting for duty cycle, equipment history and design information. This information is included in the technical basis (co-ordinating with outages and/or existing safety system tests where applicable);
- Monitoring Intervals (Frequencies are defined through technical basis and documented in IQ Review[7] database. Any exceptions to the technical basis template is noted in IQ Review under individual equipment); and
- Establishing Predefineds.

Critical component monitoring is performed using a number of techniques including N-ED-09183-10001, "Predictive Maintenance Infrared Thermography Program" [45], N-ED-09183-10002, "Predictive Maintenance Vibration Monitoring Program" [46] and N-ED-09183-10000, "Predictive Maintenance Lubrication Screening Program" [47]. Following replacement or refurbishment, all equipment within the scope of the Predictive Maintenance program is subject to post maintenance baseline testing in accordance with N-STD-MA-0004, "Post Maintenance Testing" [21] and the results are documented for benchmarking purposes.

For anomalies that have a high potential of causing a unit outage or de-rating, the PdM Technology Owner, together with the appropriate Component and Equipment Engineer, and Performance Engineer, immediately assesses the problem and develop an action plan. N-PROG-RA-0003, "Corrective Action" [17] identifies the overall corrective action process.

In general, all maintenance personnel involved in PdM activities enter the acquired data into the appropriate PdM database, and summarize their analysis in OPG's reporting application (Plant IQ).[8]

Conclusion:

The procedures and processes identified above confirm the existence of a predictive maintenance program. The intent of Review Task #6 is met and therefore Pickering NGS is compliant.

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[7] IQ Review is a web based software application that is designed to document the technical basis for PdM program scope including equipment selection, performed tests, frequency and limits at a site and across OPG Nuclear [24].

[8] Plant IQ is a web based software application that is designed to automate the process of documenting, and reporting the condition of equipment (from PdM results) at a site and across OPG Nuclear. Inputs to the systems are generated from within the PdM group and outputs are directed towards Maintenance, Operations, Engineering, Work Control and any other group that is concerned with plant equipment health [24].

4.1.7 Review Task #7: Detection and Mitigation of Aging Mechanisms

> *Ensure existence of programs for timely detection and mitigation of aging mechanisms and/or aging effects of any SSCs important to safety, including obsolescence of technology used in the plant or obsolescence of services or supplies external to the plant.*

N-PROC-MP-0060, "Aging Management Process" outlines evaluation techniques to be used in assessing component condition as well as practices in place to mitigate equipment aging [7]. The recommended actions from the CAs are traceable and often tracked through system health report action plans.

When new information becomes available that introduces uncertainty with respect to the component's continued performance, the condition assessment is reviewed and revised, if necessary, to identify any additional required station activities. This is managed thorough N-PROC-MP-0060, Section 1.7.1.1 Revision of a Condition Assessment.

N-PROC-MA-0024, "System Performance Monitoring" [20], establishes a consistent and comprehensive process for effective monitoring, maintenance, and system performance and reliability. It establishes requirements for a structured, standardized reporting program (SystemIQ)[9] for system monitoring and performance to ensure Systems Important to Safety (SIS) and production will perform their intended functions under the design basis. In addition, system health is continuously monitored by trending performance. The SPMP is based on a comparison of performance indicators against established targets in order to improve system performance. These targets include direct measures of a system's health, such as the number of functional failures, as well as indirect measures, such as the operating corrective maintenance backlog. System performance is assessed by collecting data from plant sources that is trended, analyzed, and reported as part of the System Health Report.

N-PROG-MA-0017, "Components and Equipment Surveillance" [10], identifies a set of activities to assure the health of a select group of nuclear power plant components. The program consists of a number of program elements and managed processes. Program elements consist of Component, Inspection and Test Programs. These programs are implemented for component and equipment groups that have been identified [10].

N-PROC-MA-0020, "Predefined Process" [67], provides a process to manage preventive maintenance being performed on equipment of greater functional importance and related to the more stringent requirements applicable to nuclear safety, production or equipment reliability.

The primary objectives of the N-PROG-MA-0025, "Major Components" [8] program is to establish an integrated set of processes and activities to justify fitness for service of Feeders, Fuel Channels, Reactor Components & Structures, and Steam Generators and

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9 SystemIQ is a web-based software application that is designed to assist System Engineers and Managers in optimizing the operational performance and condition of Plant systems. SystemIQ centralizes, standardizes, and automates System Health Reporting, Performance and Condition Monitoring, and the generation and organization of System Notebooks [20].

develop long-term Life Cycle Management strategies that support preservation of these assets. Strategies and plans are developed and implemented to ensure sufficient monitoring is conducted and appropriate data is acquired to demonstrate that each of the major components is compliant with its respective design basis documents.

LCMPs address inspection requirements based on the criteria which consider areas at highest risk for the degradation mechanisms identified for each component or sub-system. The Life Cycle Management Plans document acceptance and repair criteria for inspections, preventive maintenance activities and any required modifications. The LCMPs incorporate inspection results, degradation mechanisms, various inspections and examinations (including non-destructive examinations in accordance with N-STD-MA-0021, "Non-Destructive Examination" [25]), updates to programs or processes and any new novel R&D information for the component.

Obsolescence Management is governed by N-STD-MA-0024, "Obsolescence Management" [22]. This standard defines and implements a sustaining program to manage the proactive and reactive obsolescence issues associated with critical equipment and components. The activities interface with equipment reliability and life-cycle management strategies designed to sustain continued safe and reliable plant operation.

The Obsolescence Management program provides direction on managing obsolescence issues pertaining to all critical equipment and components, as defined in N-PROC-MA-0077, "Critical Equipment Identification and Categorization" [19] related to the safe and reliable operation of the plant.

As identified in [22], the general approach to managing obsolescence is:

- Identifying obsolescence issues (proactively and reactively);
- Prioritizing and ranking obsolescence issues;
- Evaluating and developing cost effective resolution strategies;
- Executing and managing solutions to completion;
- Obsolescence reporting;
- Integrating with the existing reactive identification and resolution processes; and
- Maintaining information in applicable databases up to date.

Proactive Obsolescence Management includes the use of industry tools to implement the Obsolescence program. These tools are:

- Proactive Obsolescence Management System (POMS): A software database provided by Rolls Royce that proactively identifies when an item (equipment, component) is becoming obsolete. This service is performed by collecting equipment information from member utilities and contacting each manufacturer of installed equipment on a regular basis to determine if the model number is still supported. In addition the POMS priority mechanism looks at the manufacturer/model number, criticality of the equipment and stock availability. This allows for timely identification of obsolescence vulnerabilities by forecasting the impact in the station based on available stock.

- Obsolescence Manager (OM): A software application added to POMS, designed to facilitate the prioritization and resolution of obsolescence related issues. The goal of OM is to assist the Obsolescence Process Coordinator (OPC) in resolving a large amount of identified obsolescence issues in POMS.

- Obsolete Item Replacement Database (OIRD): A database that may be used for sharing information with other nuclear utilities that are subscribed to OIRD about components and spare parts that have been determined to be obsolete or manufactured in a different form.

Other OPG resources available to proactively identify obsolescence issues include the Equipment Reliability Plan, Aging Management Plan, Condition Assessments, Component Health Reports, and Station/System Health Reports. The Obsolescence Management Program facilitates an interactive approach with Engineering and other stakeholders.

The Reactive Obsolescence process is driven by the plant demand to support emergent and scheduled work. The identification of reactive issues is typically from:

- Maintenance Material Requests;

- Supply Chain Request for Quote or Purchase Orders;

- Maintenance assessment of Bill of Material;

- Maintenance emergent/discovery work;

- Performance/Components and Equipment Engineering.

Procurement Engineering and the OPC manage the prioritization and track the reactive obsolescence issues to completion.

Proactive and Reactive Obsolescence as described in N-STD-MA-0024, "Obsolescence Management" focuses on the obsolescence issues associated with critical equipment and components. The program activities interface with equipment reliability and life cycle management strategies designed to sustain continued safe and reliable plant operation. The standard does not explicitly address obsolescence of services or supplies external to the plant.

To effectively manage the Proactive and Reactive Obsolescence issues, the OPC uses the data in Asset Suite that has been prioritized along with the information from POMS to develop a Site Obsolescence List.

N-GUID-08173-10007, "Smart Ordering" [48] is a process for anticipated demands related to planned strategies for systematically replacing items in multiple end-uses. Proactive procurement of these types of materials improves the availability of materials, thus resulting in better work schedule adherence, outage scope stability, equipment reliability and economic benefit.

N-CHAR-AS-0002, "Nuclear Management System" [68], indicates that the Materials Management system owned and implemented by Supply Chain, ensures equipment, components, materials, and services meet appropriate and applicable design and quality requirements through review and approval of suppliers' quality programs, and audits or in-process surveillance of the suppliers' activities. Equipment, components, materials, and services are purchased to required specifications and codes.

Conclusion:

The procedures and processes identified above confirm that programs for timely detection and mitigation of aging mechanisms and/or aging effects, including obsolescence of technology, have been established. However, N-STD-MA-0024, "Obsolescence Management" does not explicitly address obsolescence of services or supplies external to the plant. This is therefore identified as a gap for Pickering PSR2 (**Pickering PSR2 Gap SF4-1**).

4.1.8 Review Task #8: Acceptance Criteria and Safety Margins for Safety Related SSCs

Establish acceptance criteria and required safety margin for safety related SSCs for the remaining life of the station through reliability and risk assessments.

N-STD-MP-0020, "Margin Management" [23] defines the expectations for the management of design and operating margins associated with systems, structures and components (SSCs) important to safe and reliable plant operation, including the processes and oversight mechanisms.

The IAM program is required to ensure that the impacts of age related degradation potentially affecting nuclear safety are identified, understood, and conservatively managed. This objective is achieved through execution of a number of programs and procedures that assess the condition of the SSCs and establish the activities required to maintain equipment performance such that the requirements of the safety analysis and licensing basis are continuously met.

The potential for plant aging to adversely impact safety margins and reactor operation (e.g., de-rating) is a recognized concern. Of particular concern are the degradation mechanisms that affect major reactor components, that are not easily replaced during the planned life of the reactor, and for which mitigation is not available without a refurbishment outage and/or extremely costly expenditure (e.g., Pressure Tubes and Steam Generators).

N-PROG-RA-0016, "Risk and Reliability Program" [14] establishes the framework for the development and use of Probabilistic Risk Assessment (PRA) as a means to manage radiological risk and to contribute to safe operation of nuclear reactors. Program elements have been developed to provide a high level framework for both risk assessment activities and reliability program requirements to meet the intent of the OPG Nuclear Safety Policy and CNSC standards on risk and reliability [14].

N-STD-RA-0033, "Reliability Monitoring and Reporting of Systems Important to Safety" [49] provides direction on carrying out reliability program activities at the station consistent with S-98, "Reliability Programs for Nuclear Power Plants" (now RD/GD-98, "Reliability Programs for Nuclear Power Plants")[10] and S-99 "Reporting Requirements for Operating Nuclear Power Plants" (now REGDOC-3.1.1, "Reporting Requirements for Nuclear Power Plants").[11]

N-PROG-MP-0014, "Reactor Safety Program", defines the organizational responsibilities and key program elements for the management of issues related to Nuclear Safety Analysis, in particular Generic Action Items, and the following major components of safe operation: Safety Analysis Basis (Safety Report and Analysis of Record), Safe Operating Envelope and Severe Accident Management.

N-STD-RA-0030, "Risk Management for Outage and On-Line Maintenance" [50] and N-STD-RA-0034, "Preparation, Maintenance and Application of Probabilistic Risk Assessment" [51] provide direction on the preparation of OPG PRAs and on carrying out risk assessments for units in a planned outage, as well as non-standard configurations in units at high power [52].

The reliability program collects data and uses this information to derive an annual result for system predicted future unavailability. Generic industry component failure data is used as the starting point and station-specific actual failure data is added to obtain updated failure rates. As part of the implementation of the reliability program, Annual Reliability Reports are submitted to the CNSC consistent with S-99 and REGDOC-3.1.1.[11] The results from this report allow OPG to assess system performance against the predicted future unavailability (PFU) targets, as well as to identify and take corrective actions in case the PFU results are below target [52].

At OPG, Design and Operating margins are managed through the implementation of N-STD-MP-0020, "Margin Management" [23]. The Margin Management standard applies to all SSCs with an important role in safe and reliable plant operation. N-INS-03600-10001 "Margin Management Implementation" [53] provides guidance on how to identify low margin issues, assess low margin conditions and develop a resolution plan. Determination of whether a margin issue is in the scope of the Margin Management program requires the assistance of the Margin Management Coordinator to complete N-FORM-11371, "Low Margin Assessment Form" [54]. The completion of N-FORM-11371 requires the combined effort of the Margin Management Coordinator and station staff [53]. N-FORM-11371 documents the low margin condition. Part 1 contains the System Operating and Design Margin Identification, and Part 2 contains the Risk Score Determination which also identifies a number of Risk Scenarios, such as "Major Reduction in margin of Safety (significant risk to public or station personnel or major impact on environment)", "Total loss or serious degradation of a special safety system, system for reactor power control or S-98 system", "Unplanned production loss of more than 30 Equivalent Full Power days"

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10 CNSC Regulatory Document RD/GD-98, "Reliability Programs for Nuclear Power Plants", has been issued and replaces S-98 in the regulatory framework. Per the Pickering License Conditions Handbook [31], the requirements set out in the newly issued document remain unchanged from those established in S-98.

11 Per the Pickering License Conditions Handbook [31], the most recent Pickering PROL (48.02/2018) has an amendment to replace S-99 with REGDOC-3.1.1.

and "Radiation Exposure to personnel in excess of regulatory limits" to provide a few examples [54].

Margin management addresses low margin issues arising from equipment degradation, plant configuration and operating procedure changes, engineering modifications, and reanalysis. The intent of the margin management standard is to provide assurance of SSC availability and operability, at least to its next planned maintenance or inspection cycle [53].

To resolve a low margin issue, the Issue Owner (typically the System Engineer), develops a resolution plan and priorities based on the risk posed by the low margin issues. The issue owner tracks the resolution of the low margin issue and communicates updates to the Margin Management Coordinator to ensure timely oversight scheduling is provided. The Margin Management Coordinator performs quarterly reviews of the list of margin issues to check for possible aggregate impact of margin concerns on an SSC. A combined impact of more than one margin issue on an SSC may require increasing the priority of the involved margin concerns. Resolution plans, as contained in the System and/or Component Health Reports are executed using existing station processes (i.e. On-line or Outage Work Management, Engineering Change Control, and Action Tracking). A low margin issue may be deemed resolved when the resolution plan has been executed, all corrective actions completed, and adequate margin has been restored [53].

In order to determine the available margins, the acceptance criteria must be known and understood. The margin is the gap between the acceptance criteria and the actual (or predicted) performance for a SSC. The acceptance criteria are the values of operational and design constraints found in operating and design documents.

The following documents may be consulted to determine the acceptance criteria for the SSC:

- Operating Manuals;
- Abnormal Incidence Manuals;
- Annunciation / Alarm Response Manuals;
- Operational Safety Requirements;
- Environmental Qualification Technical Basis Documents;
- System Performance Monitoring Plans;
- Condition Assessments;
- Design Manuals and Design Descriptions;
- Safety Report.

In 2000, OPG initiated a study to address the effects of aging of the Heat Transport System (HTS) on the Safety Analysis margins at Pickering B and Darlington NGS. This study, documented in N-REP-33000-10000 [55] concluded that in terms of impact on nuclear safety, Heat Transport System aging would have the largest impact on safety margin. The report acknowledged that the potential reduction in the safety margins might require operational and/or design changes to maintain adequate margins [55]. This report did not address Pickering A (Units 1,4); however, the strategy developed for Pickering Units 5-8 has been applied to Pickering Units 1,4 [74].

N-CORR-00531-06781 [56] provides a progress report on the Heat Transport System Aging safety analysis for Pickering Units 1, 4, Pickering Units 5-8 and Darlington which was provided to the CNSC in February 2015.

N-CORR-00531-06781 includes a status of the assessments completed to date for Pickering A and B:

- Completed Loss of Flow (LOF), Small Break Loss of Coolant Accident (SBLOCA) and neutron overpower protection (NOP) analysis for Darlington, Pickering Units 1,4 and Pickering Units 5-8 for future aged HTS conditions to demonstrate continued safe operation of each station with aged HTS;

- Improvement of safety margins through application of revised channel power limits, bundle power limits and improvement in trip coverage through modification to Shutdown System trip setpoints;

- Independent Technical Panel on trip acceptance criteria and completion of final report with recommendations;

- Interfacing with Pickering Continued Operations Plan to demonstrate continued safe operation to end of commercial operation; and

- Planning of REGDOC 2.4.1 "Deterministic Safety Analysis" compliance activities including initiation of the first Safety Analysis associated with the implementation plan for REGDOC-2.4.1.

The following analysis has been completed for Pickering Units 1,4 and Pickering Units 5-8:

- Pickering Units 1,4: SBLOCA, LOF, and NOP Analysis for 6010 Effective Full Power Days (EFPD) [56].

- Pickering Units 5-8: SBLOCA, and LOF for 10300 EFPD [75].

- Pickering Units 5-8: NOP Analysis for 10300 EFPD [56].

With respect to the NOP analysis, the CNSC has accepted that the enhanced NOP (E-NOP) Extreme Value Statistics (EVS) methodology can be used to determine NOP Trip Setpoints (TSPs) and stated it is acceptable to use previously submitted E-NOP EVS-based analyses results for station compliance purposes [69].

N-CORR-00531-06781 concluded that the future aged conditions demonstrate continued safe operation of the plants until reactor operation reaches the specified target EFPD [56]. The target aged HTS condition projections are limited by the availability and quality of the station data. This typically means projecting to a future aged condition three to four years from the time analysis is completed. Therefore currently the assessment does not address the plans for extended operation.

As Pickering pursues continued operation, the HTS aging issues and remaining life will be addressed. N-CORR-00531-06781 identifies the strategy for the management of HTS Aging Impact on Safety Margins which essentially includes two areas:

- Safety analysis methodology improvements to demonstrate larger margins; and
- Design and operational changes.

The CNSC has recently concurred with the industry proposed Derived Acceptance Criteria [70].

Conclusion:

The procedures and the study identified above confirms the known existence of the acceptance criteria and required safety margin for SSCs important to safety for the remaining life of the plant through reliability and risk assessments. The intent of Review Task #8 is met and therefore Pickering NGS is compliant.

4.1.9 Review Task #9: Management of Aging for Spent Fuel Storage Facilities

Confirm adequacy of management of the effects of aging on those parts of the plant that will be required for safety when the reactor has ceased operation, for example the spent fuel storage facilities.

Safe Storage of the Pickering Units will be addressed outside of the scope of PSR2, which is focussed on safe plant operation. The Pickering NGS Stabilization Activity Plan (SAP) [58] outlines OPG's plan for managing the arrangements and activities that will be conducted in support of the Safe Storage Project. As the operational footprint of the station is reduced, all unnecessary SSCs will be placed into an Inactive Safe State, that is, they will be removed from the design basis, de-energized, drained of gas or fluids and isolated from operational systems. SSCs which remain necessary to support continued operations will be reclassified and reconfigured, as required, to meet Storage with Surveillance operational demands. The resulting Safe Storage configuration will include a station with the following systems remaining available [58]:

- Irradiated fuel bays (IFBs), including cooling, purification and monitoring equipment. IFBs will remain in service until all fuel has been transferred to dry storage containers and relocated to designated dry storage areas on site;
- Select heavy water storage facilities located at various locations across the facility, including helium storage tanks and various storage, inventory and feed tanks;
- Spent resin storage and handling systems;
- The Pickering Waste Management Facility, including the means to receive, package and store dry storage containers containing spent fuel;

- Select fire protection equipment;
- Environmental monitoring equipment for intermittent or continuous monitoring of select atmospheric and liquid emission streams, as required;
- Radiation monitoring equipment;
- Active and inactive drainage systems, including the means to collect, store, treat and discharge liquid waste streams, as required;
- Low and intermediate level waste management systems, including the means to collect, store, package, and ship low and intermediate level waste generated on site;
- Heating and ventilation systems required to maintain minimum temperatures in all in-service (or partially in-service) areas;
- Security systems; and
- Auxiliary systems which will be required to support the above noted operational systems including, but not limited to, power supplies, lighting, air supplies, service water, domestic water and demineralized water supplies.

The Safe Storage Project will plan and execute the transition of the Pickering NGS from its final shutdown state to its Safe Storage State. OPG will continue to manage the effects of aging as required using the current Aging Management Program.

With respect to the above-mentioned Safe Storage SSCs, the primary consideration for management of aging relates to the Pickering IFBs which are discussed further below.

The safety related functions of the Pickering IFBs are to [57]:

- Maintain the containment boundary; and
- Provide cooling and storage of irradiated fuel.

There are three IFBs at Pickering NGS. These are the Pickering 1-4 IFB, the Pickering 5-8 IFB and the Pickering Auxiliary IFB. The IFBs at Pickering NGS will be required to operate safely for at least 15-20 years after all reactors have ceased operation to allow for decay of radioactivity of the fuel and so it can be moved to dry storage [59].

As discussed in Review Task #1, N-PROC-MP-0060, "Aging Management Process" [7] describes the process used to perform condition assessment of critical components. NK30-REP-21500-00001 R002, "PB Nuclear - Aging Management Program Component Condition Assessment 21500 - IFB (Irradiated Fuel Bay)" is a Condition Assessment written for the Pickering 5-8 IFB last revised in April 2015 [60].

From the CA it was identified that the IFB is a passive component which requires periodic maintenance and repair. The IFB is subject to surveillance and its life is determined based on performance.

A program to clean up the bottom of the fuel bay is identified as a current practice [60]. Other initiatives for the IFBs include [60]:

1. Maintaining the temperature of the bay water within limits: 28°C - 32 °C (with the minimum and maximum limits of 15°C and 50°C determined by design), so that the integrity of the bay structure and its ability to hold water is not challenged;

2. Maintaining the IFB-B Sumps 2A, 2B, 3A and 3B operation in AUTO;

3. Regularly monitoring sump level to ensure Auto operation control as part of operator rounds and so that the sump level is maintained at low level as per design requirements (below the groundwater levels). This is so that the bay interspace is maintained dry and the bay water does not migrate to the groundwater (environment);

4. Permanently monitoring the leak rate from IFB-B into the sumps. As past trending has shown, the amount of the leakage from IFB-B is seasonal with a very low leak rate during fall/winter months and higher leak rate during the spring/summer months;

5. Inspecting all four IFB-B sumps; and

6. Ensuring that the IFB-B purification system is available to control bay leakage.

The CA determined that the Pickering 5-8 IFB will reach its component End of Life (2035 assumed in the CA) provided the current practices are permanently in place with current Aging Management practices. No other CAs have been performed for the Pickering 1-4 IFB or Pickering Auxiliary IFB to date. System Condition Assessments were performed for the Irradiated Fuel Bays [71] and Irradiated Fuel Bay Auxiliaries [72]. These documents do not meet the requirements of the current Aging Management governance N-PROG-MP-00060. These documents do not provide sufficient detail to determine if the current AM practices are adequate to meet the required end of life (EOL) for the IFBs.

NK30-REP-34410-00002 R001 "PB Nuclear - Aging Management Program Component Condition Assessment 34410 IFB Cooling Heat Exchanger" [61] is a condition assessment written for Pickering 5-8 IFB Cooling Heat Exchanger (HX). The CA indicates that the current practice of HX tube cleaning, eddy current inspection and leak testing every four years is adequate to reach component End of Life (which is stated to be 2035 in the CA). However, according to P-CHAR-04660-00001 "Heat Exchangers Replacement and Procurement of Critical Spares" [73], in 2017 058-34410-HX1, 2, 3 tube bundle will be replaced with the same tube material. The 2018 replacement plan is to replace 0-34410-HX3 tube bundle with Titanium tubes similar to HX1, 2.

NK30-REP-344100-00003 R002, "PB Nuclear - Aging Management Program Component Condition Assessment 34410, 71310 Irradiated Fuel Bay Auxiliaries – Valves-Manual- CAT 3&4" [62] is a CA written for Pickering 5-8 IFB Auxiliary isolation valves to the heat exchangers. The CA indicates that the current practice of engineering surveillance, operator rounds, and repairing deficiencies under corrective work orders, plus an additional action to determine testing and maintenance strategies for the IFB HX isolation valves, is adequate to reach component EOL (which is stated to be 2040 in the CA).

Conclusion:

The Pickering NGS Safe Storage Project will plan and execute the transition of the Pickering NGS from its final shutdown state to its Safe Storage State. OPG will continue to manage the effects of aging as required using the current Aging Management Program.

As discussed above, CAs for the Pickering 5-8 IFB, IFB Cooling Heat Exchanger (HX) and HX isolation valves have been performed and it was concluded that the Pickering 5-8 IFB and HX will reach their End of Life in 2035 and HX isolation valves will reach their End of Life in 2040 with current Aging Management practices. However, no CAs have been performed for Pickering 1-4 IFB, IFB Cooling HX or HX isolation valves or the Pickering Auxiliary IFB, IFB Cooling HX or HX isolation valves to date. To meet the intent of N-PROC-MP-0060, "Aging Management Process", a CA is required for Pickering 1-4 IFB SSCs and the Pickering Auxiliary IFB SSCs.

The conclusion of this Review Task is that there is a gap with respect to the Aging Management practices of IFB facilities at Pickering NGS; that is to specifically produce a CA for the Pickering 1-4 IFB SSCs and the Pickering Auxiliary IFB SSCs. It is noted that work to address this gap is currently underway as part of the Pickering NGS Condition Assessments for the Pickering IFBs that will be addressed under Safety Factor Report 2, "Actual Condition of Structures, Systems and Components." Since this work is not yet complete, this is identified as a gap for Pickering PSR2 (**Pickering PSR2 Gap SF4-2**).

4.1.10 Review Task #10: Aging Degradation Models

Confirm the models used to predict the evolution and advancement of aging degradation are properly supported in accordance with current accepted practices pertaining to aging degradation.

Aging Management of SSCs is performed in accordance with the Integrated Aging Management Program, N-PROG-MP-0008 [6]. This program establishes a framework for ensuring that activities and conditions required to optimize equipment condition are in place. In addition, the program defines the condition assessment process used to evaluate the health of critical components and establish actions necessary to maintain health. The "Plan-Do-Check-Act" model illustrates the systematic, proactive life cycle Aging Management approach defined by the Integrated Aging Management Program. The "Plan-Do-Check-Act" model introduced in Review Task #2 addresses how the Integrated Aging Management program implements the integrated set of programs and activities and closely adheres to the "Plan-Do-Check Act" model developed in the IAEA document, "Implementation and Review of a Nuclear Power Plant Ageing Management Programme", Safety Reports Series No.15, (1999). This is applicable to all SSCs.

To meet the requirements of the "Plan-Do-Check-Act" model, the Integrated Aging Management program interfaces with a number of programs. The programs listed below also have a role in managing equipment degradation and aging. These programs include:

- N-PROG-MA-0026, "Equipment Reliability" [9]
- N-PROG-MA-0019, "Production Work Management" [15]

- N-PROG-MA-0004, "Conduct of Maintenance" [11];
- N-PROG-OP-0001, "Nuclear Operations" [12];
- N-PROG-OP-0004, "Chemistry" [13]; and
- N-PROG-RA-0016, "Risk and Reliability Program" [14].

Details of how these programs integrate into the Aging Management process are discussed in Review Task #2.

The IAM program is implemented by fulfilling the requirements of interfacing programs having a role in managing equipment degradation and aging. Specifically, N-PROG-MA-0025, "Major Components" [8], and N-PROC-MA-0100, "Major Components Life Cycle Management Plan" [63] call for all major components LCMPs to define, schedule, and provide justification for all long term actions that need to be performed over at least the next business planning cycle to ensure long-term safety and reliability of major components.

In order to achieve this, the following Life Cycle Management Strategies, outage and non-outage activities must be identified and performed:

1. Life Cycle Management Strategies
   - Identify current practices for managing various active and plausible degradation mechanisms to ensure safe and reliable operation of the system according to its design basis.
   - Define the required inspection and maintenance activities over short-term (1-2 years), mid-term (3-5 years), and long term (10 years or end of life) to meet the objective above.
   - Delineate proactive activities that are the subject of on-going Research and Development (R&D) programs, and provide recommendations for their implementation.
   - Address major life limits (to the end of design life) and key strategic issues.
2. Summary of outage actions
   - Identify life cycle management actions (summarized in tabular form) required during station outages.
   - Summary tables are unit specific and include all actions to be performed on the component or structure in the particular outage. The actions represent the results of all the assessments carried out and cover a specified time interval (10 years) to satisfy generation planning requirements of N-PROC-AS-0043, "Nuclear Outage and Generation Planning" [64].
   - Required actions are categorized by the following criteria:

    - Legal requirements;
    - Fitness-for-service requirements;
    - Asset preservation related activities;
    - The definition of each of the above criteria are clearly stated in the respective program LCMP.
  - Outage actions identified in the LCMP undergo stakeholder review to ensure they are practical and feasible.

3. Non-Outage and support activities :
   - LCMP may also include a summary table for non-outage and/or generic activities that support the LCMP. If included, this table may list R&D activities, engineering assessments or evaluations, management initiatives, and other actions as required. It is also acceptable to provide references to other applicable documents or plans for these non-outage activities.
   - The table provides action, owner and target completion dates for each activity.
   - It is important that activities be identified as pre-requisites to any outage related activities, where applicable.

In complying with these requirements, the Major Components LCMP:

- Identifies component degradation mechanisms, potential degradation sites, and consequences of aging degradation and failures.
- Evaluates component condition by comparing observed degradation against established limits.
- Implements all relevant codes, standards, regulatory commitments, Periodic Inspection Program (PIP) requirements, and OPG Nuclear program requirements into plans to mitigate the consequences of identified degradation mechanisms from both a safety and an economic perspective.
- Identifies risks, cost range estimates, tooling requirements, and other inputs as required for business planning.
- Identifies the schedule associated with inspection, repair, or modification activities.
- Serves as a basis for OPG strategic and long-range investment planning.

Through the LCMP process, activities are in place to monitor the degradation of the major components. The inputs from the inspections provide a means/model for evaluation of the fitness for service of the components.

In addition to the foregoing strategies and programs, the discussion of compliance with modern codes and standards in Section 4.2 demonstrates that current accepted practices are followed. Also the Report for Safety Factor 9, *Use of Experience from other NPPs and Research Findings*, will document the processes used to identify lessons and improvements across all disciplines.

Conclusion:

The procedures identified above confirm the models used to predict the evolution and advancement of aging degradation are properly supported in accordance with current accepted practices pertaining to aging degradation. The intent of Review Task #10 is met, and therefore, Pickering NGS is compliant.

4.2 L/R/C/S Reviews

As per Section 2.2 of this report, detailed compliance assessments for nine L/R/C/Ss with content applicable to Safety Factors 2 and 4 are provided in Appendix B of Reference [4]. Associated findings applicable to Safety Factor 4 are summarized in Table 4 below. (Note: There were no PSR2 L/R/C/S gaps identified for Safety Factor 2).

Table 4: **PSR2 L/R/C/S Compliance Assessment Results for Safety Factor 4**

L/C/R/S Reviewed	PSR2 Compliance Assessment for Safety Factor 4
N290.13-05, "Environmental Qualification of Equipment for CANDU Nuclear Power Plants"	There are no PSR2 gaps for N290.13-05. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with N290.13-05.
N285.4-14, "Periodic Inspection of CANDU Nuclear Power Plant Components"	There are six PSR2 CSA N285.4 gaps which all relate to Safety Factor 4. The first five of these N285.4 gaps are applicable to compliance with N285.4-09 including Updates 1 and 2. The sixth N285.4 gap is applicable to compliance with N285.4-14. 1. N285.4 PIP Governance references N285.4-05, not N285.4-09 including Updates 1 and 2. This (programmatic) Darlington gap is a PSR2 gap against N285.4-09 including Updates 1 and 2 (<u>**Pickering PSR2 Gap SF4-3**</u>). 2. There has been a significant change in the wording of clause 4.2.7 in CSA N285.4-09 including Updates 1 and 2. I-PROC-AS-0009, "Inspection Qualification of Non-Destructive Examination Processes" does not identify the authorized inspector as a qualifying authority as directed by clause 4.2.7. Instead it establishes the CANDU Inspection Qualification Bureau (CIQB) as the organization that would approve procedures and personnel. This (programmatic) Darlington gap is a PSR2 gap against N285.4-09 including Updates 1 and 2 (<u>**Pickering PSR2 Gap SF4-4**</u>). 3. New erosion and corrosion inspection requirements in N285.4-09 including Updates 1 and 2 are not reflected in current PIP governance. NK38-REP-03680-10137 R000 states that: "It should be noted specifically that [this ISR Issue] is likely to have a major impact on piping PIPs because sub-clauses 7.4.7.X in CSA N285.4-09 including UPD1 and UPD2 include substantive changes. Under the new standard erosion and corrosion inspection exemptions can no longer be justified

L/C/R/S Reviewed	PSR2 Compliance Assessment for Safety Factor 4
	on the basis of [sic] that conditions are determined to be non-erosive and non-corrosive." This Darlington PIP gap will also need to be addressed in the Pickering PIPs. Therefore, this is a PSR2 gap against N285.4-09 including Updates 1 and 2 (**Pickering PSR2 Gap SF4-5**). 4. Extended life inspection schedules in N285.4-09 including Updates 1 and 2 are not reflected in PIP governance. This (programmatic) Darlington gap is a PSR2 gap against N285.4-09 including Updates 1 and 2 (**Pickering PSR2 Gap SF4-6**). 5. An assessment of the prior operating non-conforming state, as required by N285.4-09 including Updates 1 and 2, is required when dispositioning inspection results. This requirement has not been included in the feeder PIP plan. This Darlington PIP gap will also need to be addressed in the Pickering PIPs. Therefore, this is a PSR2 gap against N285.4-09 including Updates 1 and 2 (**Pickering PSR2 Gap SF4-7**). 6. There is a PSR2 gap for Pickering NGS against N285.4-14 (**Pickering PSR2 Gap SF4-8**) to address: o Revised requirements for pressure tube volumetric and dimensional inspection (Clause 12.2), pressure tube hydrogen equivalent determination (Clause 12.3) and pressure tube material property testing (Clause 12.4); o Clause 12.5 which specifies minimum annulus spacer surveillance examination and testing requirements; o Selection criteria for identifying candidate tube for pressure tube surveillance examination and testing (Annex E) to include selection criteria for annulus spacer surveillance examination and testing; and o Clause 7.4.8 which specifies requirements for inspection of Environmentally Assisted Cracking, and Clauses 7.5.1/7.5.2 which specify requirements for inspection of identical components.
N285.5-13, "Periodic Inspection of CANDU Nuclear Power Plant Containment Components"	There are two PSR2 CSA N285.5 gaps which both relate to Safety Factor 4: 1. There were a number of concessions granted from the CNSC for compliance with N285.5-M90 that will need to be reconciled for Pickering for the period of PSR2: (Since these gaps are all concession-related and associated with N285.5-M90, they are tracked under a single PSR2 gap (**Pickering PSR2 Gap SF4-9**).) o The Pickering B ISR gap associated with N285.5-M90 clause 4.5.1 is closed. However, the disposition of the gap refers to OPG receiving a concession from the CNSC on the inspection of components deemed to be inaccessible. A similar (updated) concession may be required for Pickering operation past 2020. Therefore, this is a gap for PSR2. o The Darlington ISR disposition of the gaps for N285.5-M90 clauses 8.4.2.1 and 8.4.2.2 refer to OPG receiving a concession from the CNSC that insulation will not be removed in the absence of visible damage to a component, and only "light weight" access covers will be removed. The Darlington ISR states: "This is a concession from the regulator which is not assured in the case of a refurbished plant. As such, this represents a gap". By the same logic it will need to be

L/C/R/S Reviewed	PSR2 Compliance Assessment for Safety Factor 4
	reconciled for Pickering for the period of PSR2 (life extension past 2020). ○ The Darlington ISR disposition of the gap for N285.5-M90 for clause 8.5.2.2 refers to an exception of the numerical rules of this clause for reasons of practicality, and that a concession was received from the CNSC. The Darlington ISR stated "… it is categorized as a Gap, because a concession from the CNSC is not assured for a refurbished plant.". By the same logic it will need to be reconciled for Pickering for the period of PSR2. ○ Per the Darlington ISR disposition of the gap for N285.5-M90 clause 8.6.3, although CNSC acceptance was obtained, there is still a non-compliance with a portion of the clause related to the timing of inspections which is noted as needing to be reconciled for a refurbished station. The Darlington ISR stated "This represents a gap that will need to be reconciled with the regulator for a refurbished station." By the same logic it will need to be reconciled for Pickering for the period of PSR2. 2. The changes in N285.5-13 relative to N285.5-08 that are applicable to Fiberglass Reinforced Plastic material that is used at Pickering NGS have only been assessed for fitness for service to 2024 in the Pickering Continued Operations Plan. These changes related to aging management (monitoring and test programs) for FRP materials. As a result, additional assessment is required for Pickering to address FRP aging management at Pickering for operation to 2028, and to confirm the current program aligns with N285.5-13 clauses 8.2, 8.3.3, 8.3.4 and A.6.1.2 (**Pickering PSR2 Gap SF4-10**). (Note: This gap only exists if Pickering NGS intends to operate past 2024.)
N287.7-08, "In-Service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants"	There are three PSR2 CSA N287.7 gaps which all relate to Safety Factor 4: 1. N287.7-08 clause 7.11.2 Table 1 involving non-compliance with accuracy and repeatability requirements for dewpoint temperature was a gap for Darlington. No evidence can be found that this has been addressed for Pickering NGS. This is therefore a gap for Pickering PSR2 (**Pickering PSR2 Gap SF4-11**). 2. OPG initiated a Regulatory Management action to provide the CNSC with the latest Dow Corning 995 material test report in response to an Action Notice raised in the CNSC Type II Inspection. The work is currently in progress. Therefore, this is a gap for Pickering PSR2 (**Pickering PSR2 Gap SF4-12**). 3. Actions #31, #32, and #33 from the Pickering Units 5-8 Continued Operations Plan are related to N287.7 and although complete, need to be re-assessed for Pickering operation past 2020 (**Pickering PSR2 Gap SF4-13**). (IIP Action #31 involved submission of Periodic Inspection Plans and Life Cycle Management Plans for a number of safety-significant civil structures. IIP Action #32 involved submission of Aging Management Plans for concrete containment structures to the CNSC for acceptance. IIP Action #33 involved revising the Reactor Building Periodic Inspection Plan and submitting to the CNSC for acceptance.)
CNSC RD/GD-210 (2012), "Maintenance Programs for Nuclear Power Plants"	There are no PSR2 gaps for CNSC RD/GD-210. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with RD/GD-210.

L/C/R/S Reviewed	PSR2 Compliance Assessment for Safety Factor 4
CNSC RD/GD-98 (2012), "Reliability Programs for Nuclear Power Plants"	There are no PSR2 gaps for CNSC RD/GD-98. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with RD/GD-98.
CNSC REGDOC-2.6.3 (2014), "Aging Management"	There are two PSR2 REGDOC-2.6.3 gaps which both relate to Safety Factor 4: 1. OPG is not compliant with N-PROC-MP-0060 Aging Management Process, Section 1.7 for "not reviewing and updating the Component Condition Assessments[12] within the review cycle of the component, and when new information or feedback from the program was received." OPG has since revised these CAs, which are now valid until 2020. OPG has stated they will develop an implementation plan to prevent reoccurrence of: a) not reviewing and revising the CAs within the review cycle, and b) not updating the CAs when pertinent new information becomes available. OPG stated they will provide an update and a target implementation date on this action to the CNSC by October 30, 2016. This is a gap for Pickering PSR2 (**<u>Pickering PSR2 Gap SF4-14</u>**). 2. OPG is not compliant with N-PROC-MA-0077, "Critical Equipment Identification and Categorization", Section 1.2 because "the Reactor Safety (RS) category code and rationale for critical components was not always accurate or consistently applied in the CCAs[12]." OPG has stated they have since completed a review and update of the RS category code and rationale for a portion of the components to become fully compliant with N-PROC-MA-0077. However, OPG has stated that a review of the CAs will be conducted to ensure consistency with the revised Reactor Safety codes and that an update will be provided to the CNSC by October 30, 2016. This is a gap for Pickering PSR2 (**<u>Pickering PSR2 Gap SF4-15</u>**).
N287.2-08, "Material Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants"	There are no PSR2 gaps for CSA N287.2-08. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with N287.2-08.
N285.8-15, "Technical Requirements for In-Service Evaluation of Zirconium Alloy Pressure Tubes in CANDU Reactors"	There is one PSR2 gap for Pickering NGS compliance with N285.8-15 which is applicable to Safety Factor 4: 1. For the Pickering B ISR, no clause-by-clause review of the Standard was conducted on the basis that the pressure tubes will be replaced during the refurbishment outage for Pickering Units 5-8, and the condition of these components is well understood and managed through their own specific, detailed life cycle plans and fitness-for-service criteria. However, in November 2015, OPG issued Plan N-REP-31100-10061 R002 for Pickering NGS compliance with pressure tube in-service evaluation requirements in CSA N285.8-15. OPG had submitted a previous compliance plan for the long term use of the 2010 edition of CSA N285.8 and this compliance plan was accepted by the CNSC. The compliance plan was revised to document OPG's compliance to the 2015 edition of CSA N285.8. Since OPG has committed to fulfillment of the commitments in N-REP-31100-10061 R002, successful fulfillment by OPG of the commitments in the compliance plan is required for Pickering operation past 2020. This is

[12] The terminology currently used is Condition Assessment (CA) instead of Component Condition Assessment (CCA).

L/C/R/S Reviewed	PSR2 Compliance Assessment for Safety Factor 4
	therefore a gap for Pickering PSR2 (**<u>Pickering PSR2 Gap SF4-16</u>**). In particular, the significant changes to CSA N285.8-15 per the CSA Impact Statement will need to be reflected in Pickering procedures, including: o Implementation of statistically based fatigue crack initiation evaluation curves for axial flaws (Clauses D.4.2, D.4.3, and D.3.6); o Implementation of closed-form engineering relation for threshold peak stress for Delayed Hydride Cracking (DHC) initiation (Clauses D.5 and 5.4.3.4); o Implementation of statistically based threshold relation for peak stress for crack initiation due to hydrided region overloads (Clause D.5); o Implementation of new fracture toughness models for axial through-wall flaws (Clause D.13.2); and o Implementation of Methods 1 and 2 Probabilistic Leak-Before-Break (Clauses 3.1, 7.3 and 7.4).

4.3 Audit and Self-Assessment Reviews

The OPG Nuclear Programs specifically applicable to Safety Factors 2 and 4 are identified in Table 2 and Table 3 respectively, and details of the associated audit and self-assessment results for each of the N-PROGs are provided in Appendix B. Based on the Appendix B audit and self-assessment results for Safety Factors 2 and 4 related N-PROGs, there is one gap for Pickering PSR2:

- Per Section B.1, Nuclear Oversight conducted a performance based audit (NO-2016-027) of the IAM Program in March 2016. The purpose of the audit was to determine whether IAM program requirements are being met and are effectively implemented to support safe and reliable operation. The audit concluded that the managed system controls are not fully effective and identified the following two open findings applicable to Pickering NGS which result in a PSR2 gap (**<u>Pickering PSR2 Gap SF4-17</u>**) (Note: These gaps are closely related and are therefore identified as a single PSR2 gap.):
  - o The IAM program requires that the interfacing programs affecting critical component condition should be comprehensive and sufficiently integrated to ensure critical information and assumptions used in completing condition assessments and Aging Management activities are valid and effective. However, a lack of integrated life cycle initiatives has been identified, which has the potential to impact equipment health. In addition, the program defines the requirements for program oversight and implementation. However, issues were identified in the completion of Condition Assessments and the execution of related recommendations due to ineffective oversight and implementation of the IAM program. SCR N-2016-08041 (AR# 28189056) has been raised to address this issue and is expected to be completed by Q4 2017. This is a gap for PSR2 since the SCR is not yet closed, and missing information in Condition Assessments and incomplete

actions may lead to ineffective management of the aging equipment and impact the reliability of SSCs.

- The IAM implementing procedure identifies the requirement for qualified individuals to perform Aging Management engineering activities such as preparing and reviewing Condition Assessments and screening reports. The audit identified that some Engineering Support Personnel performed engineering work independently while they were not qualified in the Training Information Management System. SCR P-2016-08008 (AR# 28189028) has been raised to address this issue and is expected to be completed by Q3 2016. This is a gap for PSR2 since the SCR is not yet closed, and unqualified staff performing work independently could impact the quality of Engineering work including Aging Management work activities.

4.4 Additional Review Findings

As discussed in Section 3.4, the PSR2 Safety Factor 4 Report includes a review of OPG commitments previously made to the CNSC, open CNSC action items, and exemptions granted by the CNSC since the current operating licence was issued (all related to Safety Factors 2 or 4). The Report also includes identification and review of previously identified PSR1 gaps related to Safety Factor 4 to ascertain the implications of extending Pickering NGS operation beyond 2020.

Review of the Pickering Units 5-8 Continued Operations Plan [34] identified the following closed gaps from the Pickering B ISR that will need to be revisited in the context of continued operation past 2020 for PSR2 Safety Factor 4 (**Pickering PSR2 Gap SF4-18**):

COP [34] Appendix A Item #	IIP [32] Reference Number	Issue*
4	G01-04	Demonstrate adequate safety margins to operate Pickering B units from a Heat Transport System aging perspective to Jan 31, 2021. The 2015 strategy update to CNSC staff provided a progress report on HTS Aging Safety Analysis and related activities, and an updated revision of the HTS Aging Management Strategy for the period 2015-2020. This needs to be expanded to cover operation past 2020 for Pickering Units 1,4 and Units 5-8. Therefore, this is a gap for Pickering PSR2.
10, 11, 12, 13	F01 (including F01-1, F01-2, F01-3, F01-4)	Develop a strategy to provide evidence that the Calandria Tube (CT) - Liquid Injection Shutdown System (LISS) nozzle gap will be maintained beyond 240,000 Effective Full Power Hours (EFPH) for all Pickering B units. The strategy for CT - LISS nozzle gap preservation may apply beyond 2025, but this needs to be confirmed. Therefore, this is a gap for Pickering PSR2.
14	F02	Develop R&D justification for extending Fuel Channel design life beyond 240,000 EFPH in the areas of hydrogen ingress, fracture toughness, spacer mobility and integrity.

COP [34] Appendix A Item #	IIP [32] Reference Number	Issue*
		Actions were constrained by the shutdown date of 2020 assumed in the 2011 business plan. This needs to be expanded to cover operation past 2020 and is therefore a gap for Pickering PSR2. Note: An interim LCMP update for major components is documented in P-CORR-01060-0587604 R000 [76], which describes life cycle management strategies for major components to achieve extended operations to 2024. Strategies in this document may apply beyond 2024, but this needs to be confirmed as part of the resolution of this COP-related PSR2 gap.
19	F13	Update the NOP analysis for Pickering B. Actions were constrained by the shutdown date of 2020 assumed in the 2011 business plan. This is primarily relevant to Safety Factor 5 but is also of relevance to Safety Factor 4. This needs to be expanded to cover operation past 2020 for Pickering Units 1,4 and Units 5-8. Therefore, this is a gap for Pickering PSR2.
21	F14-1a	With respect to the Feeder LCMPs, clarify the impact of fuel channel axial elongation during operation beyond the fuel channel assumed design life of 210,000 EFPH on feeder stress analysis and acceptable feeder thickness. This was only addressed to 2025. This needs to be expanded to cover operation to 2028. Therefore, this is a gap for Pickering PSR2. Note: An interim LCMP update for major components is documented in P-CORR-01060-0587604 R000 [76], which describes life cycle management strategies for major components to achieve extended operations to 2024. Strategies in this document may apply beyond 2024, but this needs to be confirmed as part of the resolution of this COP-related PSR2 gap.
30	F14-4.1	Include the periodic inspection programs and LCMPs for the secondary side pressure retaining components and submit them for CNSC review. Although the action to submit PIPs and LCMPs for the secondary side pressure retaining components is complete, these documents will need to be extended to cover operation past 2020 for Pickering Units 1,4 and Units 5-8. Therefore, this is a gap for Pickering PSR2.
52	I15-1a	Perform Time Limiting Aging Analysis (TLAAs) and include such TLAAs in the LCMPs and in the CAs. OPG to provide commitment that TLAAs necessary to determine the actual conditions of components will be completed. Although the action is complete, this will need to be updated to cover operation past 2020 for Pickering Units 1,4 and Units 5-8. Therefore, this is a gap for Pickering PSR2.

COP [34] Appendix A Item #	IIP [32] Reference Number	Issue*
69	I15-7a	Include relevant information from COG JP 4271 Calandria and internals. Fitness for Life Extension Guidelines in N-PLAN-01060-10003 "Reactor Components and Structures Life Cycle Management Plan (LCMP)" and submit the LCMP to the CNSC in accordance with Pickering B PROL 08.20/2013 LC 1.2. Although the action is complete, this will need to be updated to cover operation past 2020 for Pickering Units 1,4 and Units 5-8. Therefore, this is a gap for Pickering PSR2. Note: An interim LCMP update for major components is documented in P-CORR-01060-0587604 R000 [76], which describes life cycle management strategies for major components to achieve extended operations to 2024. Strategies in this document may apply beyond 2024, but this needs to be confirmed as part of the resolution of this COP-related PSR2 gap.
Appendix C, Item 5	F11	Update the Pickering B HTS aging model. This action is complete but needs to be reviewed to assess impact of operation past 2020 for Pickering Units 1,4 and Units 5-8. Therefore, this is a gap for Pickering PSR2.
Appendix C, Item 6	F12	Update the Pickering B HTS aging management strategy. This action is complete but needs to be reviewed to assess impact of operation past 2020 for Pickering Units 1,4 and Units 5-8. Therefore, this is a gap for Pickering PSR2.

* Closed Pickering Units 5-8 COP actions were reviewed to determine whether they need to be reassessed (PSR2 gaps identified) to address operation past 2020. Where applicable, equivalent Pickering Units 1,4 PSR2 gaps are also identified where reassessment will be required for operation past 2020.

Review of the Darlington IIP [33] for gaps that may need to be reassessed in the context of Pickering PSR2 for operation past 2020 did not identify any additional gaps for Safety Factor 4.

The following concession in the R04 Pickering Licence Conditions Handbook [31] is applicable to Safety Factor 4 in the context of operation past 2020:

- "OPG shall carry out periodic inspections in accordance with the accepted PIP documents. If a deviation from the accepted PIP program is anticipated during inspection planning activities, OPG shall obtain CNSC acceptance prior to conducting the affected inspections. However, for any findings, discoveries or deviations from the accepted PIP that are identified during an inspection, OPG shall provide justification to CNSC in the inspection report submission following OPG governance, OPEX and Best Industry Practices. For permanently required exemptions to the requirements of CSA PIP standards, OPG shall revise the affected PIP document accordingly prior to issuing the next scheduled revision of the PIP document." The LCH wording provides a way for handling a deviation from the accepted PIP program. If a permanent concession is needed as a result of future updates to CSA Standards associated with PIPs, this concession will need to be revisited. CNSC acceptance will be obtained at that time if it is needed. This is therefore not a PSR2 gap.

The following concession in the Pickering LCH is applicable to Safety Factor 2 in the context of operation past 2020:

- "With respect to N285.4-05 clause 14.2.5.1.3, CNSC staff have accepted OPG's request to use COG Report 07-4089 R1 "Fitness-for-Service Guidelines for Steam Generator and Preheater Tubes", with one exception pertaining to the use of the Level D safety factors stipulated in ID 2.3.2.2 Paragraph (a), *Deterministic Leak-Before-Break,* to other load levels (levels A, B, C) and any other portions of the Fitness for Service Guidelines (FFSG) that invoke the use of ID-2.3.2.2 (per CNSC letter e-Doc 4298097). Instead, OPG is required to continue using the safety factors defined in ID 2.3.2.2 Paragraph (a) of Revision 0 of the fitness-for-service guidelines for the appropriate load levels." This exception relates to a pressure limit for which tubes pulled from an SG need to be tested for showing either Leak Before Break or Maximum Tolerable Flaw Size calculations. This value does not change as the SG tubes age and does not need to be revisited for operation past 2020. This is therefore not a PSR2 gap.

Per Appendix B.5 of this report, there were three PSR2 gaps identified that are associated with the Conduct of Maintenance Program and related to Foreign Material Exclusion, Work Planning and organizational learning. These gaps are not directly related to Aging and will be captured under Safety Factor Report 11, "Procedures".

5.0 RESULTS AND CONCLUSIONS

OPG Governance, Programs, Policies, Procedures, Instructions and Guidelines related to Aging Safety Factor 4 were reviewed for the ten PSR2 Review Tasks in Section 4.1 of this report and resulted in Pickering PSR2 Gaps SF4-1 and SF4-2 below. L/R/C/S and OPG Nuclear Program audit and self-assessment reviews for both Safety Factors 2 and 4 were prepared per Sections 4.2 and 4.3, respectively, and resulted in PSR2 Gaps SF4-3 to SF4-17. This report also includes a review of OPG commitments previously made to the CNSC, open CNSC action items, and exemptions granted by the CNSC since the current operating licence was issued (all related to Safety Factors 2 or 4), as well as identification and review of previously identified PSR1 gaps related to Safety Factor 4 (to ascertain the implications of extending Pickering NGS operation beyond 2020), per Section 4.4, which resulted in PSR2 Gap SF4-18.

The eighteen PSR2 gaps that will need to be addressed as part of Pickering PSR2 are:

- **<u>Gap SF4-1</u>**: The conclusion of Safety Factor 4 Review Task #7 is that programs for timely detection and mitigation of aging mechanisms and/or aging effects, including obsolescence of technology, have been established. However, N-STD-MA-0024, "Obsolescence Management" does not explicitly address obsolescence of services or supplies external to the plant. This is therefore identified as a gap for Pickering PSR2.

- **<u>Gap SF4-2</u>**: Per Safety Factor 4 Review Task #9, there is a gap with respect to the Aging Management practices of IFB facilities at Pickering NGS that is to specifically produce a CA for the Pickering 1-4 IFB SSCs and the Pickering Auxiliary IFB SSCs. It is noted that work to address this gap is currently underway as part of the Pickering NGS Condition Assessments for the Pickering IFBs that will be addressed under Safety Factor Report 2, "Actual Condition of Structures, Systems and Components." Since this work is not yet complete, this is identified as a gap for Pickering PSR2.

- **<u>Gap SF4-3</u>**: N285.4 PIP Governance references N285.4-05, not N285.4-09 including Updates 1 and 2. This (programmatic) Darlington gap is a PSR2 gap against N285.4-09 including Updates 1 and 2.

- **<u>Gap SF4-4</u>**: There has been a significant change in the wording of clause 4.2.7 in CSA N285.4-09 including Updates 1 and 2. I-PROC-AS-0009, "Inspection Qualification of Non-Destructive Examination Processes" does not identify the authorized inspector as a qualifying authority as directed by clause 4.2.7. Instead it establishes the CANDU Inspection Qualification Bureau as the organization that would approve procedures and personnel. This (programmatic) Darlington gap is a PSR2 gap against N285.4-09 including Updates 1 and 2.

- **<u>Gap SF4-5</u>**: New erosion and corrosion inspection requirements in N285.4-09 including Updates 1 and 2 are not reflected in current PIP governance. NK38-REP-03680-10137 R000 states that: "It should be noted specifically that [this ISR Issue] is likely to have a major impact on piping PIPs because sub-clauses 7.4.7.X in CSA N285.4-09 including UPD1 and UPD2 include substantive changes. Under the new standard erosion and corrosion inspection exemptions can no longer be justified on the basis of [sic] that conditions are determined to be non-erosive and non-corrosive." This Darlington PIP gap will also need to be addressed in the Pickering PIPs. Therefore, this is a PSR2 gap against N285.4-09 including Updates 1 and 2.

- **<u>Gap SF4-6</u>**: Extended life inspection schedules in N285.4-09 including Updates 1 and 2 are not reflected in PIP governance. This (programmatic) Darlington gap is a PSR2 gap against N285.4-09 including Updates 1 and 2.

- **<u>Gap SF4-7</u>**: An assessment of the prior operating non-conforming state, as required by N285.4-09 including Updates 1 and 2, is required when dispositioning inspection results. This requirement has not been included in the feeder PIP plan. This Darlington PIP gap will also need to be addressed in the Pickering PIPs. Therefore, this is a PSR2 gap against N285.4-09 including Updates 1 and 2.

- **<u>Gap SF4-8</u>**: There is a PSR2 gap for Pickering NGS against N285.4-14 to address:

    - Revised requirements for pressure tube volumetric and dimensional inspection (Clause 12.2), pressure tube hydrogen equivalent determination (Clause 12.3) and pressure tube material property testing (Clause 12.4);

    - Clause 12.5 which specifies minimum annulus spacer surveillance examination and testing requirements;

    - Selection criteria for identifying candidate tube for pressure tube surveillance examination and testing (Annex E) to include selection criteria for annulus spacer surveillance examination and testing; and

    - Clause 7.4.8 which specifies requirements for inspection of Environmentally Assisted Cracking, and Clauses 7.5.1/7.5.2 which specify requirements for inspection of identical components.

- **<u>Gap SF4-9</u>**: There were a number of concessions granted from the CNSC for compliance with N285.5-M90 that will need to be reconciled for Pickering for the period of PSR2: (Since these gaps are all concession-related and associated with N285.5-M90, they are tracked under a single PSR2 gap.

    - The Pickering B ISR gap associated with N285.5-M90 clause 4.5.1 is closed. However, the disposition of the gap refers to OPG receiving a concession from the CNSC on the inspection of components deemed to be inaccessible. A similar (updated) concession may be required for Pickering operation past 2020. Therefore, this is a gap for PSR2.

    - The Darlington ISR disposition of the gaps for N285.5-M90 clauses 8.4.2.1 and 8.4.2.2 refer to OPG receiving a concession from the CNSC that insulation will not be removed in the absence of visible damage to a component, and only "light weight" access covers will be removed. The Darlington ISR states: "This is a concession from the regulator which is not assured in the case of a refurbished plant. As such, this represents a gap". By the same logic it will need to be reconciled for Pickering for the period of PSR2 (life extension past 2020).

  - The Darlington ISR disposition of the gap for N285.5-M90 for clause 8.5.2.2 refers to an exception of the numerical rules of this clause for reasons of practicality, and that a concession was received from the CNSC. The Darlington ISR stated "… it is categorized as a Gap, because a concession from the CNSC is not assured for a refurbished plant.". By the same logic it will need to be reconciled for Pickering for the period of PSR2.

  - Per the Darlington ISR disposition of the gap for N285.5-M90 clause 8.6.3, although CNSC acceptance was obtained, there is still a non-compliance with a portion of the clause related to the timing of inspections which is noted as needing to be reconciled for a refurbished station. The Darlington ISR stated "This represents a gap that will need to be reconciled with the regulator for a refurbished station." By the same logic it will need to be reconciled for Pickering for the period of PSR2.

- **Gap SF4-10**: The changes in N285.5-13 relative to N285.5-08 that are applicable to Fiberglass Reinforced Plastic material that is used at Pickering NGS have only been assessed for fitness for service to 2024 in the Pickering Continued Operations Plan. These changes related to aging management (monitoring and test programs) for FRP materials. As a result, additional assessment is required for Pickering to address FRP aging management at Pickering for operation to 2028, and to confirm the current program aligns with N285.5-13 clauses 8.2, 8.3.3, 8.3.4 and A.6.1.2 (Note: This gap only exists if Pickering NGS intends to operate past 2024.).

- **Gap SF4-11**: N287.7-08 clause 7.11.2 Table 1 involving non-compliance with accuracy and repeatability requirements for dewpoint temperature was a gap for Darlington. No evidence can be found that this has been addressed for Pickering NGS. This is therefore a gap for Pickering PSR2.

- **Gap SF4-12**: OPG initiated a Regulatory Management action to provide the CNSC with the latest Dow Corning 995 material test report in response to an Action Notice raised in the CNSC Type II Inspection. The work is currently in progress. Therefore, this is a gap for Pickering PSR2.

- **Gap SF4-13**: Actions #31, #32, and #33 from the Pickering Units 5-8 Continued Operations Plan are related to N287.7 and although complete, need to be re-assessed for Pickering operation past 2020. (IIP Action #31 involved submission of Periodic Inspection Plans and Life Cycle Management Plans for a number of safety-significant civil structures. IIP Action #32 involved submission of Aging Management Plans for concrete containment structures to the CNSC for acceptance. IIP Action #33 involved revising the Reactor Building Periodic Inspection Plan and submitting to the CNSC for acceptance.).

- **Gap SF4-14**: OPG is not compliant with N-PROC-MP-0060 Aging Management Process, Section 1.7 for "not reviewing and updating the Component Condition

Assessments[13] within the review cycle of the component, and when new information or feedback from the program was received." OPG has since revised these CAs, which are now valid until 2020. OPG has stated they will develop an implementation plan to prevent reoccurrence of: a) not reviewing and revising the CAs within the review cycle, and b) not updating the CAs when pertinent new information becomes available. OPG stated they will provide an update and a target implementation date on this action to the CNSC by October 30, 2016. This is a gap for Pickering PSR2.

- **<u>Gap SF4-15</u>**: OPG is not compliant with N-PROC-MA-0077, "Critical Equipment Identification and Categorization", Section 1.2 because "the Reactor Safety (RS) category code and rationale for critical components was not always accurate or consistently applied in the CCAs[13]." OPG has stated they have since completed a review and update of the RS category code and rationale for a portion of the components to become fully compliant with N-PROC-MA-0077. However, OPG has stated that a review of the CAs will be conducted to ensure consistency with the revised Reactor Safety codes and that an update will be provided to the CNSC by October 30, 2016. This is a gap for Pickering PSR2.

- **<u>Gap SF4-16</u>**: For the Pickering B ISR, no clause-by-clause review of CSA N285.8 was conducted on the basis that the pressure tubes will be replaced during the refurbishment outage for Pickering Units 5-8, and the condition of these components is well understood and managed through their own specific, detailed life cycle plans and fitness-for-service criteria. However, in November 2015, OPG issued Plan N-REP-31100-10061 R002 for Pickering NGS compliance with pressure tube in-service evaluation requirements in CSA N285.8-15. OPG had submitted a previous compliance plan for the long term use of the 2010 edition of CSA N285.8 and this compliance plan was accepted by the CNSC. The compliance plan was revised to document OPG's compliance to the 2015 edition of CSA N285.8. Since OPG has committed to fulfillment of the commitments in N-REP-31100-10061 R002, successful fulfillment by OPG of the commitments in the compliance plan is required for Pickering operation past 2020. This is therefore a gap for Pickering PSR2. In particular, the significant changes to CSA N285.8-15 per the CSA Impact Statement will need to be reflected in Pickering procedures, including:

    - Implementation of statistically based fatigue crack initiation evaluation curves for axial flaws (Clauses D.4.2, D.4.3, and D.3.6);

    - Implementation of closed-form engineering relation for threshold peak stress for Delayed Hydride Cracking (DHC) initiation (Clauses D.5 and 5.4.3.4);

    - Implementation of statistically based threshold relation for peak stress for crack initiation due to hydrided region overloads (Clause D.5);

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[13] The terminology currently used is Condition Assessment (CA) instead of Component Condition Assessment (CCA).

    - Implementation of new fracture toughness models for axial through-wall flaws (Clause D.13.2); and

    - Implementation of Methods 1 and 2 Probabilistic Leak-Before-Break (Clauses 3.1, 7.3 and 7.4).

- **<u>Gap SF4-17</u>**: Per Section B.1, Nuclear Oversight conducted a performance based audit (NO-2016-027) of the IAM Program in March 2016. The purpose of the audit was to determine whether IAM program requirements are being met and are effectively implemented to support safe and reliable operation. The audit concluded that the managed system controls are not fully effective and identified the following two open findings applicable to Pickering NGS which result in a PSR2 gap (Note: These gaps are closely related and are therefore identified as a single PSR2 gap.):

    - The IAM program requires that the interfacing programs affecting critical component condition should be comprehensive and sufficiently integrated to ensure critical information and assumptions used in completing condition assessments and Aging Management activities are valid and effective. However, a lack of integrated life cycle initiatives has been identified, which has the potential to impact equipment health. In addition, the program defines the requirements for program oversight and implementation. However, issues were identified in the completion of Condition Assessments and the execution of related recommendations due to ineffective oversight and implementation of the IAM program. SCR N-2016-08041 (AR# 28189056) has been raised to address this issue and is expected to be completed by Q4 2017. This is a gap for PSR2 since the SCR is not yet closed, and missing information in Condition Assessments and incomplete actions may lead to ineffective management of the aging equipment and impact the reliability of SSCs.

    - The IAM implementing procedure identifies the requirement for qualified individuals to perform Aging Management engineering activities such as preparing and reviewing Condition Assessments and screening reports. The audit identified that some Engineering Support Personnel performed engineering work independently while they were not qualified in the Training Information Management System. SCR P-2016-08008 (AR# 28189028) has been raised to address this issue and is expected to be completed by Q3 2016. This is a gap for PSR2 since the SCR is not yet closed, and unqualified staff performing work independently could impact the quality of Engineering work including Aging Management work activities.

- **<u>Gap SF4-18</u>**: Per Section 4.4, review of the Pickering Units 5-8 Continued Operations Plan [34] identified 11 closed gaps from the Pickering B ISR that will need to be revisited in the context of continued operation past 2020 for PSR2. These relate to COP Appendix A Actions #4, 10, 11, 12, 13, 14, 19, 21, 30, 52, 69, and Appendix C Items 5 and 6. A number of these gaps will also need to be addressed for Pickering Units 1,4.

The review of Safety Factor 4 has confirmed that aging aspects affecting SSCs important to safety are being effectively managed and that an effective aging management program is in place at Pickering NGS.

6.0 REFERENCES

[1] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[2] CNSC REGDOC-2.3.3, *Periodic Safety Reviews*, April 2015.

[3] IAEA Safety Guide No. SSG-25, *Periodic Safety Review for Nuclear Power Plants*, March 2013.

[4] OPG Report, P-REP-03680-00003 R000, *Pickering NGS PSR2: Definition of Safety Factor Review Tasks,* May 30, 2016.

[5] OPG Report, P-REP-03680-00004 R000, *Pickering NGS Periodic Safety Review 2: Code and Standard Reviews for Safety Factors 2 (Actual Condition of SSCs), 3 (Equipment Qualification) and 4 (Aging)*, July 2016.

[6] OPG Nuclear Program, N-PROG-MP-0008 R006A, *Integrated Aging Management*, October 2, 2015.

[7] OPG Nuclear Procedure, N-PROC-MP-0060 R005B, *Aging Management Process*, October 1, 2015.

[8] OPG Nuclear Program, N-PROG-MA-0025 R002, *Major Components*, March 25, 2015.

[9] OPG Nuclear Program, N-PROG-MA-0026 R002, *Equipment Reliability*, May 26, 2015.

[10] OPG Nuclear Program, N-PROG-MA-0017 R008, *Component and Equipment Surveillance*, June 1, 2015.

[11] OPG Nuclear Program, N-PROG-MA-0004 R011, *Conduct of Maintenance*, April 28, 2015.

[12] OPG Nuclear Program, N-PROG-OP-0001 R008, *Nuclear Operations*, November 13, 2015.

[13] OPG Nuclear Program, N-PROG-OP-0004 R008, *Chemistry*, January 22, 2016.

[14] OPG Nuclear Program, N-PROG-RA-0016 R008, *Risk and Reliability Program*, October 16, 2015.

[15] OPG Nuclear Program, N-PROG-MA-0019 R009, *Production Work Management*, December 4, 2014.

[16] OPG Nuclear Program, N-PROG-MA-0016 R009, *Fuel*, October 26, 2015.

[17] OPG Nuclear Program, N-PROG-RA-0003 R010, *Corrective Action*, January 9, 2015.

[18] OPG Nuclear Program, N-PROG-MP-0014 R005, *Reactor Safety Program*, September 16, 2015.

[19] OPG Nuclear Procedure, N-PROC-MA-0077 R006, *Critical Equipment Identification and Categorization*, February 20, 2015.

[20] OPG Nuclear Procedure, N-PROC-MA-0024 R015, *System Performance Monitoring*, October 28, 2013.

[21] OPG Nuclear Standard, N-STD-MA-0004 R005, *Post Maintenance Testing*, April 17, 2015.

[22] OPG Nuclear Standard, N-STD-MA-0024 R000, *Obsolescence Management,* August 10, 2015.

[23] OPG Nuclear Standard, N-STD-MP-0020 R003, *Margin Management*, November 27, 2015.

[24] OPG Nuclear Procedure, N-PROC-MA-0034 R008, *Predictive Maintenance Program Requirements*, March 6, 2014.

[25] OPG Nuclear Standard, N-STD-MA-0021 R001, *Non-Destructive Examination*, July 30, 2015.

[26] OPG Program, OPG-PROG-0009 R002, *Items and Services Management*, May 8, 2015.

[27] OPG Nuclear Program, N-PROG-MP-0001 R014, *Engineering Change Control*, January 7, 2015.

[28] OPG Nuclear Program, N-PROG-MP-0004 R016, *Pressure Boundary*, November 27, 2015.

[29] OPG Nuclear Program, N-PROG-MP-0005 R005, *Configuration Management*, June 28, 2012.

[30] OPG Nuclear Program, N-PROG-MP-0009 R011, *Design Management*, December 19, 2014.

[31] CNSC Report, LCH-PNGS-R004, *Pickering NGS: Licence Conditions Handbook*, December 23, 2015.

[32] OPG Plan, NK30-PLAN-03680-00002 R000, *Pickering B - Integrated Implementation Plan*, December 15, 2011.

[33] OPG Report, NK38-REP-03680-10185 R002, *Darlington NGS Integrated Implementation Plan (IIP)*, April 30, 2015.

[34] OPG Plan, NK30-PLAN-00531-00001 R005, *Pickering 5-8 Continued Operations Plan*, December 2015.

[35] OPG Plan, N-PLAN-01060-10001 R017, *Feeders Life Cycle Management Plan*, November 2015.

[36] OPG Plan, N-PLAN-01060-10002 R016, *Fuel Channels Life Cycle Management Strategy and Plan*, November 2015.

[37] OPG Plan, N-PLAN-01060-10003 R013, *Reactor Components and Structures Life Cycle Management Plan*, November 2015.

[38] OPG Plan, N-PLAN-33110-10009 R006, *Steam Generators Life Cycle Management Plan*, November 2015.

[39] OPG Pickering Instruction, P-INS-06931-00003 R004, *Accountability and Ownership for Pickering A and Pickering B Systems and Structures*, October 2015.

[40] OPG Plan, N-PLAN-01060-10009 R000, *Integrated Aging Management Self Assessment Plan*, May 13, 2013.

[41] OPG Nuclear Procedure, N-PROC-RA-0097 R008, *Self-Assessment and Benchmarking*, December 17, 2014.

[42] OPG Nuclear Program, N-PROG-AS-0006 R011, *Records and Document Control*, October 23, 2014.

[43] OPG Procedure, OPG-PROC-0019 R007, *Records and Document Management*, March 14, 2016.

[44] OPG Nuclear Procedure, N-PROC-RA-0022 R032, *Processing Station Condition Records*, November 13, 2014.

[45] OPG Engineering Standard/Equipment-Related Standard, N-ED-09183-10001 R004, *Predictive Maintenance Infrared Thermography Program*, April 25, 2014.

[46] OPG Engineering Standard/Equipment-Related Standard, N-ED-09183-10002 R004, *Predictive Maintenance Vibration Monitoring Program*, August 2010.

[47] OPG Engineering Standard/Equipment-Related Standard, N-ED-09183-10000 R006, *Predictive Maintenance Lubrication Screening Program*. August 2010.

[48] OPG Guideline, N-GUID-08173-10007 R000, *Smart Ordering*, October 31, 2012.

[49] OPG Nuclear Standard, N-STD-RA-0033 R002, *Reliability Monitoring and Reporting of Systems Important to Safety,* October 24, 2013.

[50] OPG Nuclear Standard, N-STD-RA-0030 R003, *Risk Management for Outage Planning and On-line Maintenance*, November 13, 2013.

[51] OPG Nuclear Standard, N-STD-RA-0034 R003, *Preparation, Maintenance and Application of Probabilistic Risk Assessment*, October 1, 2013.

[52] OPG Letter, P-CORR-00531-03719, G. Jager to M.A. LeBlanc, *Application for Renewal of Pickering Nuclear Generating Station Power Reactor Operating Licence,* July 4, 2012.

[53] OPG Nuclear Instruction, N-INS-03600-10001 R002, *Margin Management Implementation*, November 27, 2015.

[54] OPG Form, N-FORM-11371 R001, *Low Margin Assessment Form*, October 2014.

[55] OPG Report, N-REP-33000-10000 R00, *Heat Transport System Aging – Safety Margin Management Phase 1 Report*, December 21, 2001.

[56] OPG Letter, N-CORR-00531-06781 R000, W.M. Elliot to M. Santini and F. Rinfret, *Progress Report on OPG Heat Transport System Aging Safety Analysis*, February 24, 2015.

[57] OPG List, P-LIST-06937-00001 R000, *Pickering A and B List of Safety Related Systems*, April 2011.

[58] OPG Plan, P-PLAN-00990-00007 R00, *Pickering NGS Stabilization Activity Plan (SAP)*, December 4, 2015.

[59] AECL Assessment Document, 30-21500-ASD-001 Revision 0 (OPG Record, NK30-REP-21500-0557128, August 2015), *Pickering B Irradiated Fuel Bay (IFB-B) - Leakage Assessment*, September 2010.

[60] OPG Report, NK30-REP-21500-00001 R002, *PB Nuclear - Aging Management Program Component Condition Assessment (CCA) 21500 - IFB (Irradiated Fuel Bay)*, April 24, 2015.

[61] OPG Report, NK30-REP-34410-00002 R001, *PB Nuclear - Aging Management Program Component Condition Assessment (CCA) 34410 - IFB Cooling Heat Exchanger*, August 12, 2010.

[62] OPG Report, NK30-REP-34410-00003 R002, *PB Nuclear - Aging Management Program Component Condition Assessment (CCA) 34410, 71310 - Irradiated Fuel Bay Auxiliaries-Valves-Manual – CAT 3&4*, April 24, 2015.

[63] OPG Nuclear Procedure, N-PROC-MA-0100 R002, *Major Components Life Cycle Management Plan*, March 27, 2013.

[64] OPG Nuclear Procedure, N-PROC-AS-0043 R011, *Nuclear Outage and Generation Planning*, December 11, 2014.

[65] CNSC Letter, e-Doc 4759048, File No. 4.01.02 (OPG File No. P-CORR-00531-04474 R000), M. Santini to B. McGee, *Pickering Planned Outages P1351, P1441 and P1481, Final Periodic Inspection Reports for CSA N285.4, N285.5 & N287.7 (Action Items 2014-48-5288, 2014-48-5576 and 2014-48-5588)*, May 20, 2015.

[66] OPG Nuclear Training and Qualification Description, N-TQD-403-00001 R011, *Nuclear Engineering Support Personnel Training and Qualification Description,* April 28,2015

[67] OPG Nuclear Procedure, N-PROC-MA-0020 R027, *Predefined Process*, June 2, 2016

[68] OPG Nuclear Charter, N-CHAR-AS-0002 R018, *Nuclear Management System*, March 6, 2015

[69] CNSC Letter, e-Doc 4907195, File No. 4.01.03 N-CORR-00531-17979 Rev: 000, B.D Howden to W. S Woods, *DNGS and PNGS Enhanced Neutron Overpower Protection (E-NOP) Extreme Value Statistics (EVS) Methodology, Closure of Action Item 20090OPG-06 – New Action Item 2016-OPG-7349,* January 21, 2016.

[70] CNSC Letter, e-Doc 4981431, File No. 4.01.0 N-CORR-00531-18045 Rev: 000, B.D Howden to W. S Woods, *Darlington and Pickering NGS: Derived Acceptance Criteria for Deterministic Safety Analysis of Slow Events and Re-categorization of CANDU Safety Issue PF18 Fuel Bundle/Element Behaviour under Post Dryout*, April 21, 2016.

[71] OPG Report, NA44-REP-34400-00002 R00, *Screening Record for Pickering A Ageing Management Compliance – SCA 41 Irradiated Fuel Bays,* June 29, 2012.

[72] OPG Report, NA44-REP-35710-00006, R00, *Screening Record for Pickering A Ageing Management Compliance – SCA 41 Irradiated Fuel Bays Auxiliaries*, June 29, 2012.

[73] OPG Project Charter, P-CHAR-04660-00001 R000, *Heat Exchangers Replacement and Procurement of Critical Spares*, January 28, 2016.

[74] OPG Letter, N-CORR-00531-04435 R000, *Progress Report on OPG Safety Report Improvement Activities, Part 3: Accident Analysis, Action Items: Pickering A 2007-4-17, Pickering B 2007-8-13 and Darlington 20071317*, January 28, 2009.

[75] CNSC Letter, N-CORR-00531-07408 Rev: 00, W.S. Woods to M. Santini, *Heat Transport System Aging Safety Analysis for Pickering Units 5 to 8*, October 27, 2015.

[76] OPG Memorandum, P-CORR-01060-0587604 R000, *Fitness for Service of Major Components*, March 29, 2016.

Appendix A: Nomenclature

AECL	Atomic Energy of Canada Limited
AFI	Area For Improvement
AM	Aging Management
AMP	Aging Management Plan
AR	Action Request
ARDM	Age Related Degradation Mechanism
ASME	American Society of Mechanical Engineers
CANDU	CANada Deuterium Uranium
CC	Criticality Code
CA	Condition Assessment
CCA	Component Condition Assessment
CIQB	CANDU Inspection Qualification Bureau
CM	Configuration Management
CNSC	Canadian Nuclear Safety Commission
COG	CANDU Owners Group
COP	Continued Operations Plan
CSA	Canadian Standards Association
CT	Calandria Tube
E-NOP	Enhanced Neutron Overpower Protection
ECI	Emergency Coolant Injection
EFPD	Effective Full Power Day
EFPH	Effective Full Power Hours
EFPY	Effective Full Power Year
EPRI	Electric Power Research Institute
EOL	End of Life
ER	Equipment Reliability
FFSG	Fitness for Service Guidelines
FME	Foreign Material Exclusion
HPECI	High Pressure Emergency Coolant Injection
HTS	Heat Transport System

HX	Heat Exchanger
IAEA	International Atomic Energy Agency
IAM	Integrated Aging Management
IFB	Irradiated Fuel Bay
IIP	Integrated Implementation Plan
INPO	Institute of Nuclear Power Operators
ISR	Integrated Safety Review
LCH	Licence Conditions Handbook
LCMP	Life Cycle Management Plan
LISS	Liquid Injection Shutdown System
LOF	Loss of Flow
L/R/C/Ss	Laws, Regulations, Codes and Standards
NGS	Nuclear Generating Station
NOP	Neutron Overpower Protection
NPP	Nuclear Power Plant
N-PROGs	Nuclear Programs
OIRD	Obsolete Item Replacement Database
OM	Obsolescence Manager
OPC	Obsolescence Process Coordinator
OPG	Ontario Power Generation
OPEX	Operational Experience
OSR	Operational Safety Requirements
PARTS	Pickering A Return to Service
PdM	Predictive Maintenance
PEL	Program Element
PFU	Predicted Future Unavailability
PIP	Periodic Inspection Program
PMEL	Performance Monitoring Equipment List
POMS	Proactive Obsolescence Management System
PRA	Probabilistic Risk Assessment
PROL	Power Reactor Operation Licence
PSR	Periodic Safety Review
PSR1	Periodic Safety Review 1 (earlier OPG PSR work and other associated assessments)

PSR2	Periodic Safety Review 2 (subsequent PSR per REGDOC-2.3.3)
RS	Reactor Safety
RTM	Run to Maintenance
SAP	Stabilization Activity Plan
SBLOCA	Small Break LOCA
SCR	Station Condition Record
SG	Steam Generator
SIS	Systems Important to Safety
SOE	Safe Operating Envelope
SSC	Structures, Systems and Components
SOE	Safe Operating Envelope
SPMP	System Performance Monitoring Plan
TIMS	Training Information Management System
TLAA	Time Limiting Aging Analysis
TSP	Trip Setpoint
WANO	World Association of Nuclear Operators

Appendix B: Audit and Self-Assessment Results

B.1 N-PROG-MP-0008, "Integrated Aging Management"

The objective of the Integrated Aging Management Program is to ensure that the condition of critical OPG Nuclear equipment is understood and that required activities are in place to ensure the health of these components and systems while the plant ages. This is accomplished by establishing an integrated set of programs and activities which ensure performance requirements of all critical station equipment are met on an ongoing basis. These programs and activities all serve an integral function to ensure critical equipment degradation due to aging is managed, such that operation of the station remains within the licensing basis and allows for station operational goals to be met.

N-PROG-MP-0008 also requires preparation of life cycle plans for critical plant equipment. The purpose of these plans is to determine and document actions required to ensure plant equipment will meet all design and operating objectives over the life of the plant in consideration of aging. The life cycle plans are established by a comprehensive condition assessment process. Condition assessments supplement the ongoing engineering surveillance activities in place to monitor and optimize system performance.

Nuclear Oversight conducted a performance based audit of IAM Program in March 2016, NO-2016-027 [B.1.1], for both Pickering and Darlington NGS. The purpose of the audit was to determine whether IAM program requirements are being met and are effectively implemented to support safe and reliable operation. The audit concluded that the managed system controls are not fully effective and identified the following two findings applicable to Pickering NGS:

- The IAM program requires that the interfacing programs affecting critical component condition should be comprehensive and sufficiently integrated to ensure critical information and assumptions used in completing Condition Assessments and aging management activities are valid and effective. However, a lack of integrated life cycle initiatives has been identified, which has the potential to impact equipment health.

  In addition, the program defines the requirements for program oversight and implementation. However, gaps were identified in the completion of Condition Assessments and the execution of related recommendations due to ineffective oversight and implementation of the IAM program.

- The IAM implementing procedure identifies the requirement for qualified individuals to perform Aging Management engineering activities such as preparing and reviewing Condition Assessments and screening reports. The audit identified that some Engineering Support Personnel performed engineering work independently while they were not shown as qualified in the Training Information Management System (TIMS).

Two SCRs (N-2016-08041 and P-2016-08008), were initiated to address the above findings, which required corrective actions to be implemented and are expected to be completed by Q4 2017 (N-2016-08041) and Q3 2016 (P-2016-08008).

Engineering Programs completed a self-assessment in February 2016, NO16-000190-SA [B.1.2] in order to assess the effectiveness of the IAM program and to identify areas for improvement for both Pickering and Darlington NGS as well as the Nuclear Waste Management Division. It was concluded that the primary configuration of the IAM Program is aligned with industry practices as well as CNSC expectations. However, areas for improvement were identified as follows:

- Various measures to improve IAM program effectiveness were identified. For example, revision of the 3 year cycle for the IAM self-assessment plan (N-PLAN-01060-10009) from 2016 to 2018; a roll out or briefing card to operations, maintenance and management staff to create an adequate awareness on IAM; and a review the applicability of DCR 126201 (AR 28185478);

- Revision to the governance document;

- Update to the Program Health Report Indicators; and

- IAM training requirements.

Two SCRs (N-2016-01710 and N-2016-02451) and an Action Request (AR 28185478) were initiated as a result of the self-assessment, which required corrective actions to be implemented. Although the Action Request is expected to be completed by Q2 2016, a check has confirmed that the two SCRs have already been closed and the necessary corrective actions were completed to address the underlying issues.

The above results for the audits and self-assessments revealed the following PSR2 gap related to the IAM Program (**Pickering PSR2 Gap SF4-17**):

- The IAM program requires that the interfacing programs affecting critical component condition should be comprehensive and sufficiently integrated to ensure critical information and assumptions used in completing Condition Assessments and Aging Management activities are valid and effective. However, a lack of integrated life cycle initiatives has been identified, which has the potential to impact equipment health.

  In addition, the program defines the requirements for program oversight and implementation. However, issues were identified in the completion of Condition Assessments and the execution of related recommendations due to ineffective oversight and implementation of the IAM program. SCR N-2016-08041 (AR# 28189056) has been raised to address this issue and is expected to be completed by Q4 2017. This is a gap for PSR2 since the issue is outstanding and missing information in Condition Assessments and incomplete actions may lead to ineffective management of the aging equipment and impact the reliability of SSCs.

- The IAM implementing procedure identifies the requirement for qualified individuals to perform Aging Management engineering activities such as preparing and reviewing Condition Assessments and screening reports. The audit identified that some Engineering Support Personnel performed engineering work independently while they were not shown as qualified in the Training Information Management System. SCR P-

2016-08008 (AR# 28189028) has been raised to address this issue and is expected to be completed by Q3 2016. This is a gap for PSR2 since the issue is outstanding and unqualified staff performing work independently could impact the quality of Engineering work including Aging Management work activities.

References

[B.1.1] OPG Nuclear Oversight Report N-REP-01070-0588401 T06, *NO-2016-027: Integrated Aging Management*, March 2016.

[B.1.2] OPG Self-Assessment Report, NO16-000190-SA, *Integrated Aging Management (IAM) Program Self-Assessment prior to Nuclear Oversight Audit NO2016-027*, February 2016.

B.2 N-PROG-MA-0025, "Major Components"

The major components program establishes an integrated set of processes and activities to justify fitness for service of Feeders, Fuel Channels, Reactor Components and Structures, and Steam Generators (SGs) and develops long-term Life Cycle Management strategies that support preservation of these assets. The program incorporates the reporting requirements associated with demonstrating compliance with design basis documentation relevant to each of the Major Components Program areas. Also described are processes for conducting regular cross-discipline reviews of information to ensure each of the major components cited above contributes to the safe operation of Pickering NGS.

The program calls for an effective, formal, and systematic process for integrating and reviewing data reported from each of the major components, which includes, but is not necessarily limited to, the following:

- Component design and manufacturing;
- Component inspection;
- Surveillance of sub-components (e.g. SG tubing and internal tube supports);
- Station operating conditions (e.g. physics, temperature, chemistry, etc.); and
- Research described in the technical basis documentation of the Life Cycle Management Plans (LCMP) applicable to the respective major component.

The Governance and Services Section completed a self-assessment in December 2015, BAS15-001756-SA [B.2.1] in order to assess program compliance with N-PROG-MA-0025 for both Pickering and Darlington NGS. The self-assessment concluded that the program is performing well with no overdue documents. No findings/SCRs were generated as a result of the self-assessment. Audit reports specifically for the Major Components Program have not been prepared and hence there are no pertinent findings.

The above results for the self-assessment reveal that there are no gaps for PSR2 related to the Major Components Program.

References

[B.2.1] OPG Self-Assessment Report, BAS15-001756-SA, *Program Assessment of N-PROG-MA-0025 Major Components*, December 2015.

B.3 N-PROG-MA-0026, "Equipment Reliability"

The purpose of the Equipment Reliability (ER) Program is to ensure ongoing high levels of reliable performance of components important to nuclear safety, production, and environmental protection. Reliable performance of components means very low numbers of component failures, degraded equipment condition is minimized, and redundancy is maintained on key Pickering NGS systems.

The ER Program contains the following elements which ensure ongoing high levels of reliable performance of critical components:

- Identifying critical components that require focused attention;
- Specifying the required maintenance strategies to maintain high levels of reliability, and continuously improving the maintenance strategies based on corrective actions and maintenance feedback;
- Executing Predictive Maintenance, and Preventive Maintenance programs;
- Monitoring system and component condition and implementing plans to restore and maintain system and component health;
- Taking prompt and effective action, when critical equipment fails, to understand the technical and organizational causes and to prevent a recurrence; and
- Identifying and predicting aging and obsolescence issues on important components and embedding mitigating strategies and actions into the business plan.

Nuclear Oversight conducted a performance based audit of the ER Program in May 2013, NO-2013-002 [B.3.1] for both Pickering and Darlington. The objective of the audit was to assess the implementation of the ER Program and to ensure on-going high levels of reliable performance of components important to safety, production and environmental protection. The audit found performance deficiencies in the following three areas:

- Preventive Maintenance Implementation;
- System Surveillance Activities; and
- Predictive Maintenance Implementation and Health Reporting.

Three Pickering NGS SCRs (P-2013-07134, P-2013-07136 and P-2013-07138) were initiated to address the above findings which required corrective actions to be implemented. These SCRs have since been closed and the necessary corrective actions were completed to address the underlying issues.

Operations and Maintenance Support completed a self-assessment in February 2014, NO14-000381-SA [B.3.2] in order to assess the health of the Equipment Reliability governance framework, which is applicable for both Darlington and Pickering. This involved a review of related SCRs, governance framework, revision records and previous program assessment reports. No actions were generated as a result of the self-assessment.

The above results for audits and self-assessments reveal that there are no gaps for PSR2 related to the Equipment Reliability Program.

References

[B.3.1] OPG Nuclear Oversight Audit Report, N-REP-01070-0435138 T06, *Audit OPGN NO-2013-002: Equipment Reliability*, May 2013.

[B.3.2] OPG Self-Assessment Report, NO14-000381-SA, *Program Assessment: N-PROG-MA-0026, Equipment Reliability*, February 2014.

B.4 N-PROG-MA-0017, "Component and Equipment Surveillance"

The Component and Equipment Surveillance program is a set of activities to assure the health of a select group of OPG Nuclear components. A series of program elements (component, inspection, and test) and managed processes have been developed under the authorization of this program. Program oversight provides assurance that licensing, reactor safety, equipment reliability, environmental and conventional safety requirements are met on an ongoing basis. For example, the periodic inspection program is in place to ensure compliance with applicable CSA Standards and the Power Reactor Operating Licence. The power operated valve program complements the in-service inspection test program by demonstrating the functionality of critical valves following design basis accidents.

OPG's comprehensive component and equipment monitoring is accomplished through the implementation of this program and the integration of interfacing activities that are managed under the Equipment Reliability program consisting of system performance monitoring, as well as component monitoring and response to equipment failures and degradation.

Operations and Maintenance Support completed a self-assessment in January 2014, NO13-000857-SA [B.4.1] in order to assess the health of the Component and Equipment Surveillance governance framework, which is applicable for both Darlington and Pickering NGS. This involved a review of the Governance Framework, SCR database, Asset Suite, and revision records. DCR #0000125104 was initiated against N-PROG-MA-0017 to capture the following observations of the self-assessment:

- Revisions/Addition to several implementing references;
- Reference to CSA N286-12, "Management System Requirements for Nuclear Facilities"; and
- Removal of non-governance documents which were identified as an implementing program document.

DCR #0000125104 has since been completed and the necessary corrective actions were finalized to address the underlying issues. Audit reports specifically for the Component and Equipment Surveillance Program have not been prepared and hence there are no pertinent findings.

The above results for the self-assessment reveal that there are no gaps for PSR2 related to the Component and Equipment Surveillance Program.

References

[B.4.1] OPG Self-Assessment Report, NO13-000857-SA, *Program Assessment: N-PROG-MA-0017, Component and Equipment Surveillance*, January 2014.

B.5 N-PROG-MA-0004, "Conduct of Maintenance"

The purpose of the Maintenance program is to ensure personnel and public safety, protection of the environment and reliable operation. The program includes work planning, work execution, calibration and tool control, personnel and training, and performance indicators and assessment.

Nuclear Oversight conducted a performance based audit of the Maintenance program in December 2015 for Pickering NGS, NO-2015-030 [B.5.1]. The purpose of the audit was to determine whether the Maintenance program requirements have been effectively implemented to support safe and reliable operation. The audit concluded that the managed system controls are not fully effective and identified the following findings:

- Foreign Material Exclusion (FME) is the process used to prevent the intrusion of Foreign Material into systems, equipment, or components. The goal of FME is to ensure that all staff plan and execute their activities to include precautions to prevent introducing foreign material into plant equipment. Significant gaps were found in the FME health reporting and event characterization and field issues were also noted (SCR P-2015-28880).

- OPG Governance requires that learning behaviours such as observation and coaching, self-assessments and benchmarking are used to continuously improve operations. Pickering Maintenance is not effectively utilizing performance improvement tools. Actions from the observation and coaching, self-assessment, and corrective action programs are lacking continuity, content and depth. This has resulted in gaps not being identified, actioned or effectively corrected (SCR P-2015-28884).

- N-PROC-MA-0002, "Work Planning", describes the requirements to establish the process for planning work to ensure common base requirements are uniformly supported. The process starts with a Work Request in which work requirements and constraints have not been identified and ends with fully planned Work Order with associated tasks ready to be scheduled. However, inconsistent quality in Work Instruction and non-compliances were noted (SCR P-2015-28887).

- Non-compliances with governance or management expectation were observed in work reporting, walk down feedback and Pre Job Briefing form use (SCR P-2015-28890).

Four SCRs were initiated to address the above findings which required corrective actions to be implemented. One SCR has been completed (P-2015-28890), while the remaining SCRs (P-2015-28880, P-2015-28884 and P-2015-28887) have actions in place and are all expected to be completed by Q2 2016 (P-2015-28884 and P-2015-28887) or Q3 2017 (P-2015-28880).

The Maintenance department completed a self-assessment in February 2014, P14-000101-SA [B.5.2], in order to validate the effectiveness of current actions, identify potential gaps and recommend additional actions required to address the Areas For Improvement (AFIs) identified in the 2013 WANO Peer Evaluation related to Conduct of Maintenance for Pickering NGS. The review identified the following AFIs:

- Maintenance Workers are sometimes not applying fundamental maintenance practices and following standards such as procedural use and adherence. This has resulted in a forced outage, a reactor trip, returning a unit to a guaranteed shutdown state, and rework. Managers and first line managers not enforcing standards is a contributing factor.

- Late, deferred, and late-in-grace preventive and predictive maintenance tasks have led to several critical component failures, loss of redundancy, and degraded system health. Contributors are ineffective corrective actions and an acceptance of ineffective execution of preventive maintenance and predictive maintenance tasks.

The self-assessment concluded that the current actions were not effective and provided recommendations to address the corrective actions related the AFIs, which were captured under two ARs (28159866 and 28160182). These ARs have since been closed and the necessary corrective actions were completed to address the underlying issues.

The above results for the audits and self-assessments, revealed that the Conduct of Maintenance Program possesses the following gaps:

1) Foreign Material Exclusion is the process used to prevent the intrusion of Foreign Material into systems, equipment, or components. The goal of FME is to ensure that all staff plan and execute their activities to include precautions to prevent introducing foreign material into plant equipment. Significant gaps were found in the FME health reporting and event characterization and field issues were also noted. SCR P-2015-28880 (AR# 28186936) has been raised to address this issue and is expected to be completed by Q3 2017. This is a gap for PSR2 since the SCR is not yet closed and inaccurate reporting of FME program health allows conditions to remain latent and increases the risk of a consequential event (Note: Since FME is not directly related to Aging, this gap will be captured under Safety Factor Report 11, "Procedures").

2) OPG Governance requires that learning behaviours such as observation and coaching, self-assessments and benchmarking are used to continuously improve operations. Pickering Maintenance is not effectively utilizing performance improvement tools. Actions from the observation and coaching, self-assessment, and corrective action programs are lacking continuity, content and depth. This has resulted in gaps not being identified, actioned or effectively corrected. SCR P-2015-28884 (AR# 28186012) has been raised to address this issue and is expected to be completed by Q2 2016. This is a gap for PSR2 since the SCR is not yet closed and organizational learning and progress cannot effectively occur without input from observation and coaching. Also, industry OPEX prevents duplication of errors and potential vulnerabilities can continue to exist without the use of effectiveness checks. (Note: Since organizational learning is not directly related to Aging, this gap will be captured under Safety Factor Report 11, "Procedures").

3) N-PROC-MA-0002, "Work Planning", describes the requirements to establish the process for planning work to ensure common base requirements are uniformly supported. The process starts with a Work Request in which work requirements and

constraints have not been identified and ends with fully planned Work Order with associated tasks ready to be scheduled. However, inconsistent quality in Work Instruction and non-compliances with N-PROC-MA-0002 were noted. SCR P-2015-28887 (AR# 28186347) has been raised to address this issue and is expected to be completed by Q2 2016. This is a gap for PSR2 since the SCR is not yet closed and this issue could increase the risk of errors by maintenance staff in the field. (Note: Since Work Planning is not directly related to Aging, this gap will be captured under Safety Factor Report 11, "Procedures").

References

[B.5.1] OPG Nuclear Oversight Report, N-REP-01070-0576012 T06, *NO-2015-030 Conduct of Maintenance - Pickering*, December 2015.

[B.5.2] OPG Self-Assessment Report, P14-000101-SA, *Technical Support Mission: Conduct of Maintenance – Supervisory Effectiveness; Equipment Performance and Condition*, February 2014.

B.6 N-PROG-OP-0004, "Chemistry"

The purpose of the Chemistry program is specify processes, requirements and staff accountabilities to ensure effective control of Pickering NGS chemistry, including provision of analytical services. The Chemistry program covers activities associated with overall objectives of controlling plant chemistry, including the following:

- Identification of issues or conditions that may impact on chemistry control performance;
- Maintenance of specifications for chemistry control;
- Control of laboratory methods;
- Sampling and analysis;
- Data management;
- Application of actions to maintain or restore chemistry control;
- Performance monitoring, including data review; and
- Control of process chemical quality.

Nuclear Oversight conducted a performance based audit of the Chemistry program for Pickering NGS in May 2014, NO-2014-024 [B.6.1] in order to determine if chemistry activities are being performed effectively and in compliance with the program requirements for safe and reliable operations. The audit noted positive insights in the areas of intra-laboratory testing and good work practices during both sampling collection and chemistry lab analysis. However, the audit determined that the performance of the managed system controls for the Chemistry Program is not fully effective and identified three findings:

- Deficiencies with inspection and maintenance of chemistry lab fume hoods;
- Untimely resolution of long standing issues; and
- Misalignment between chemistry specifications, chemistry control procedures and Chemistry and Environmental Monitoring (CEM) database.

Three SCRs (P-2014-16543, P-2014-16546 and P-2014-16550) were initiated to address the above findings, which required corrective actions to be implemented. These SCRs have since been completed (although the three SCRs have an "Approved" status, all the associated assignments have been completed) and the necessary corrective actions were completed to address the underlying issues.

Operations and Maintenance Support completed a self-assessment in April 2013, NO13-000227-SA [B.6.2] in order to assess the health of the Chemistry governance framework, which is applicable for both Darlington and Pickering NGS. This involved a review of related SCRs, governance framework, revision records and previous program assessment reports. No further actions or recommendations were generated as a result of the self-assessment.

The above results for the audit and self-assessment reveal that there are no gaps for PSR2 related to the Chemistry Program.

References

[B.6.1] OPG Nuclear Oversight Audit Report, N-REP-01070-0500280 T06, *Audit OPGN NO-2014-024: Chemistry Program - Nuclear Engineering and Pickering*, May 2014.

[B.6.2] OPG Self-Assessment Report, NO13-000227-SA, *Program Management Assessment: N-PROG-OP-0004, Chemistry*, April 2013.

B.7 N-PROG-MA-0019, "Production Work Management"

The Production Work Management program specifies the requirements for identifying, prioritizing, planning, scheduling, and performing work in support of the operation, maintenance, and modification of OPG Nuclear stations. The program also establishes safe, uniform, and efficient work control practices for nuclear sites.

The following objectives are satisfied through the implementation of the Production Work Management Program:

- Standardization: Common priority systems, work processes and methodologies are consistently applied in scheduling all work in support of the operation, maintenance, and modification of nuclear facilities across the fleet;
- Possible safety consequences of concurrent or sequential maintenance, testing, or operations are considered;
- Operational readiness of required equipment is ensured and the plant is not placed in a high-risk configuration;
- Accountabilities in this process are established from work initiation to work completion, and compliance is ensured through monitoring;
- Value for money: Resource utilization is maximized to ensure health of the systems and to satisfy internal and external commitments of the business plan; and
- Schedule compliance is continually reinforced.

Nuclear Oversight conducted a performance based audit of the Online Work Management Process (which is an element of the Work Management Program) in March 2013, N-NO-2013-012 [B.7.1], for both Pickering and Darlington. The purpose of the audit was to verify that the online work management process is utilized to identify, select, plan schedule and execute work in a manner that ensures high levels of safe and reliable plant operation. The audit concluded that the managed system controls were not effective and identified the following two findings applicable to Pickering:

- Work integral to maintaining safe and reliable plant operation at Pickering is not surviving through the stability window; and
- The current practice for mitigating emergent work at Pickering is not fully effective in meeting its objectives.

Two SCRs (P-2013-03748 and P-2013-03751) were initiated to address the above findings, which required corrective actions to be implemented. These SCRs have since been closed (although P-2013-03748 has an "Approved" status, all the associated assignments have been completed) and the necessary corrective actions were completed to address the underlying issues.

The Governance and Services Section completed a self-assessment in November 2015, BAS15-001596-SA [B.7.2] in order to assess compliance with N-PROG-MA-0019 for both Pickering and Darlington. It was concluded that the Production Work Management Program performed well with only minor recommendations identified (e.g., incorporate DCRs on superseded references at next revision; consideration for the addition of best practices text to the program). No findings/SCRs were initiated as result of this self-assessment.

The above results for the audit and self-assessment reveal that there are no gaps for PSR2 related to the Production Work Management Program.

References

[B.7.1] OPG Nuclear Oversight Audit Report, N-REP-01070-0435148 T06, *Audit OPGN NO-2013-012*: *Work Management*, March 2013.

[B.7.2] OPG Self-Assessment, BAS15-001596-SA, *Assessment of N-PROG-MA-0019 Production Work Management Program compliance to OPG-PROC-0001/OPG-STD-0001*, November 2015.

B.8 N-PROG-MA-0016, "Fuel"

The Fuel Program establishes a formal and systematic process for integrating and reviewing information related to fuel, and reporting its performance, condition, and compliance with associated design basis documents. N-PROG-MA-0016 specifies the requirements for monitoring, integrating and assessing fuel-related information and details the documentation requirements for issues identified by this program. N-PROG-MA-0016 also incorporates the reporting requirements associated with demonstrating fuel compliance with fuel design basis documentation. These activities are performed to ensure fuel performs safely and reliably over the life of the stations, maintaining fuel design basis, license bases, and Operational Safety Requirements, while optimizing station reliability, production, and cost effectiveness.

The Fuel Program also takes responsibility for integrating and reviewing fuel channel data which may impact safety analysis or the safety report. It is compliant with the requirements specified in applicable subsections (Section 7.3 through to Section 7.9) of N286-12, "Management System Requirements for Nuclear Facilities".

Operations and Maintenance Support completed a self-assessment in February 2013, NO13-000174-SA [B.8.1], in order to assess the health of the N-PROG-MA-0016 governance framework which is applicable for both Pickering and Darlington NGS. The self-assessment concluded that there were no program compliance issues and no findings/SCRs were generated.

Nuclear Oversight conducted a performance based audit of the Fuel Program in December 2014, NO-2014-025 [B.8.2] in order to determine if the requirements defined in governance have been met and effectively implemented for both Pickering and Darlington NGS. The audit determined that performance of the managed system controls for the Fuel program and related activities is not fully effective and identified the following four findings:

- Weaknesses in Fuel Program Oversight;
- Challenges to Timely Resolution of In-Core Fuel Defects;
- Weaknesses in Fuel Program Implementation; and
- Deficiencies in Fuel Program Governance.

Four SCRs (SCRs N-2014-34717, P-2014-34722, N-2014-34724 and N-2014-34726) were initiated to address the above findings, which required corrective actions to be implemented. These SCRs have since been completed and the necessary corrective actions were completed to address the underlying issues.

The above results for the audit and self-assessment reveal that there are no gaps for PSR2 related to the Fuel Program.

References

[B.8.1] OPG Self-Assessment Report, NO13-000174-SA, *Program Management Assessment: N-PROG-MA-0016, Fuel*, February 2013.

[B.8.2] OPG Nuclear Oversight Audit Report, N-REP-01070-0525012 T06, *Fuel Program: OPGN NO-2014-025 T6,* December 2014.

B.9 OPG-PROG-0009, "Items and Services Management"

The Items and Services Management program supports Supply Chain activities across all of OPG and is summarized by the following points:

- The processes identified in this program ensure:
    - Procurement is planned, and that purchased, stored, and issued items and purchased services meet appropriate and applicable design and quality requirements.
    - Requirements for a managed process are in place to ensure that the procurement of nuclear fuel is planned, and that purchased fuel materials and services meet appropriate design and quality requirements.
    - Items, services, and nuclear fuel materials and bundles are purchased in accordance with stated requirements and controlled through proper identification, handling, storage, issuance, and shipping to ensure the quality of equipment and components is preserved.
- The execution of requisitioning and procurement processes shall be fair and transparent and shall be conducted in accordance with OPG-STD-0017, Organizational Authority Register, and OPG's Business Code of Conduct.
- This program supports Quality Assurance requirements stated in the CSA N285 and N286 series of standards.
- This program is applicable to the purchase, storage and handling of items or services used to support construction, operation, maintenance and decommissioning activities within OPG nuclear power facilities, waste management and nuclear fuel materials and services for use in OPG reactors and support programs, and aligns with requirements of N-MAN-01913.11-10000, "Pressure Boundary Program Manual".
- This program is not applicable to the handling, storage, or control of nuclear fuel at OPG nuclear facilities, and items or services specifically described in other approved governing programs and supporting documents.

Nuclear Oversight conducted a performance based audit of the Items and Services Management program in October 2015 for Pickering NGS, NO-2015-024 [B.9.1]. The purpose of the audit was to assess the implementation effectiveness of the Items and Services Management Program in: meeting performance expectations; pressure boundary QA requirements; and safety and reliability concerns. The audit concluded that the managed system controls are effective, however the following two findings were generated:

- Requirements for maintaining the approved supplier list and warehousing activities are not always met; and

- Training records and qualifications for some Supply Chain staff were not maintained as per requirements.

Two SCRs (N-2015-24495 and N-2015-24497) were initiated to address the above findings, which required corrective actions be implemented. These SCRs have since been completed and the necessary corrective actions were completed to address the underlying issues. Audit reports specifically for the Items and Services Management Program have not been prepared and hence there are no pertinent findings

The above results for the self-assessment reveal that there are no gaps for PSR2 related to the Items and Services Management Program.

References

[B.9.1] OPG Nuclear Oversight Report, N-REP-01070-0566991 T10, *NO-2015-024, Items and Services Management Program (including Pressure Boundary)*, October 2015.

B.10 N-PROG-MP-0001, "Engineering Change Control"

The Engineering Change Control (ECC) program ensures all modifications to SSCs, including software and engineered tooling, are planned, designed, installed, commissioned, placed into service or removed from service within the Safe Operating Envelope or Safety and Design Envelope, design basis and licensing conditions. This program ensures all problems or betterment ideas requiring a modification are reviewed prior to approval to ensure they improve or maintain operability, maintainability, radiological and conventional safety, regulatory or licence compliance, and production at an acceptable cost.

Nuclear Oversight conducted a performance based audit of the ECC program for Pickering NGS in November 2014, NO-2014-030 [B.10.1] and the Nuclear Waste Management Facilities. The purpose of the audit was to determine if ECC related activities are being performed effectively and in compliance with program requirements. The audit concluded that the managed system controls are not fully effective and identified the following three findings:

- Quality issues in engineering vendor products affecting the ECC process (N-2014-30369);

- Examples of non-compliance and gaps in engineering administrative activities were identified in several modification packages across Pickering Plant Design and Pickering Projects Design. Ineffective oversight, the lack of attention to detail and insufficient knowledge of the ECC process by OPG and the vendors may be contributing to these gaps. Although, the impact of these gaps individually may not be significant, the cumulative effect of these gaps in the modification packages are challenging the product quality and configuration management due to long delays in close outs (SCR P-2014-30444); and

- Staff signing ECC related documents do not have the required qualifications or training as per governance requirements (SCR N-2014-30373).

Three SCRs were initiated to address the above findings, which required corrective actions to be implemented. Two SCRs have been completed (N-2014-30369 and N-2014-30373), while the remaining SCR (P-2014-30444) has actions in place and is expected to be completed by Q2 2016.

Operations and Maintenance Support completed a self-assessment in March 2013, NO2013-000200-SA [B.10.2] in order to assess the health of the ECC governance framework, which is applicable for both Darlington and Pickering NGS. This involved a review of related SCRs, governance framework, revision records and previous program assessment reports. No findings were identified or SCRs initiated as a result of this self-assessment.

The above results for the audit and self-assessment reveal that there are no gaps for PSR2 related to the Engineering Change Control Program (note, although SCR P-2014-30444 is still open, the identified gap is administrative in nature and is not considered safety significant).

References

[B.10.1] OPG Nuclear Oversight Report, N-REP-01070-0519973 T06, *Audit OPGN NO-2014-030: Engineering Change Control (ECC) Audit*, November 2014.

[B.10.2] OPG Self-Assessment Report, N013-000200-SA, *Program Management Assessment: N-PROG-MP-0001, Engineering Change Control*, March 2013.

B.11 N-PROG-MP-0004, "Pressure Boundary"

The Pressure Boundary (PB) Program defines the managed process to control the quality of PB activities at Ontario Power Generation's nuclear facilities. It provides the requirements and defines the responsibilities for compliance with and maintenance of the PB Quality Assurance Program and provides the governance framework for the execution of PB field work activities. This is to ensure nuclear facilities retain the PB Certificates of Authorization necessary to perform PB activities and, remain compliant with the Nuclear Station Power Reactor Operating Licences, Waste Facility Operating Licences and applicable CSA standards.

The PB Program covers activities related to quality at Pickering NGS and is applicable to the following:

- Pressure-retaining systems, components, storage tanks and supports that are registered or eligible for registration with an Authorized Inspection Agency as per CSA N285.0 and CSA B51;
- Component supports (American Society of Mechanical Engineers (ASME) Class NF) mandated by Codes and Standards applicable to this PB Program; and
- Propane systems.

The program complies with CSA N285.0, N286, B51 and ASME Boiler and Pressure Vessel Code versions as referenced in the applicable facility's Operating Licence.

Nuclear Oversight conducted a performance based audit of the Pressure Boundary Program for Pickering NGS in October 2013, N-NO-2013-018 [B.11.1]. The purpose of the audit was to verify compliance with applicable sections of the Pressure Boundary Program Manual (N-MAN-01913.11-10000), review areas of risk identified in previous PB audit findings and identify any potential opportunities for improvement at the station or its support facilities, with a focus on safety and reliability. The audit concluded that the managed system controls associated with the Pressure Boundary program are effective and no findings were identified or SCRs issued as a result of the audit.

The Governance and Services Section completed a self-assessment in November 2015, BAS15-001616-SA [B.11.2] in order to assess the governance structure of N-PROG-MP-0004, "Pressure Boundary" which is applicable for both Pickering and Darlington NGS. No findings/SCRs were generated as a result of the self-assessments. However, two recommendations were generated which were dispositioned as part of the technical review process for N-PROG-MP-0004 (e.g., simplification of roles to those driven by the PB Program alone and the update of referenced forms).

The above results for the audit and self-assessment reveal that there are no gaps for PSR2 related to the Pressure Boundary Program.

References

[B.11.1] OPG Nuclear Oversight Audit Report, N-REP-01070-0435154 T10, *Audit OPGN NO-2013-018: Pressure Boundary QAM - Pickering and Support Facilities*, October 2013.

[B.11.2] OPG Self-Assessment Report, BAS15-001616-SA, *Program Assessment N-PROG-MP-0004 Pressure Boundary*, November 2015.

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ONTARIO POWER GENERATION	
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REVISE AND RESUBMIT	
Signature	03 MAR 2017 **Date**
Name: Mike Ruffolo, Manager	
Dept: Pickering Engineering - Aging Management & Strategic Initiatives	
OPG Proprietary	
Doc No.: P-REP-03680-00009	**Rev: 000**
This acceptance does not relieve the contractor from responsibility for errors or omissions or from any obligations or liability under this contract.	

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Pickering NGS PSR2 Safety Factor 5 Report: Deterministic Safety Analysis

PS112/RP/006 R01

March 2, 2017

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Revision Summary – For Amec Foster Wheeler Report PS112/RP/006

Rev	Date	Author	Comments
R00	July 4, 2016	R. Jayasundera	Initial issue for OPG review and comment.
R01	March 2, 2017	R. Jayasundera, A. Johnstone	Updated report addressing OPG comments on R00 Report.

EXECUTIVE SUMMARY

Ontario Power Generation (OPG) is performing a Periodic Safety Review (PSR) to support the possibility of continued operation of Pickering Nuclear Generating Station (NGS) beyond 2020. The PSR (referred to as "PSR2") is a subsequent PSR building on the review basis of earlier OPG Integrated Safety Reviews and other associated assessments. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

Part of PSR2 involves the preparation of Safety Factor reports for each of fifteen major topic areas. Safety Factor reports consist of:

- Assessing compliance against "Review Tasks" identified in the PSR2 Basis Document [1]. These Review Tasks are derived from Canadian Nuclear Safety Commission (CNSC) REGDOC-2.3.3, "Periodic Safety Reviews" [2] and International Atomic Energy Agency (IAEA) SSG-25, "Periodic Safety Review for Nuclear Power Plants" [3];

- Documenting assessments against applicable modern Laws, Regulations, Codes and Standards (L/R/C/Ss) as defined in Reference [1]; and

- Reviewing the effectiveness of applicable OPG Nuclear Programs that support the above assessments, through review of audit and self-assessment results.

The PSR2 review of Safety Factor 5, *Deterministic Safety Analysis*, is presented in this report. OPG Governance, Programs, Policies, Procedures, Instructions and Guidelines related to Safety Factor 5 were reviewed for the seven PSR2 Review Tasks specified in Section 4.1 of this report. L/R/C/S and OPG Nuclear Program effectiveness reviews for Safety Factor 5 were prepared per Sections 4.2 and 4.3, respectively. Per Section 4.4, the PSR2 assessment includes a review of previously identified PSR1 gaps related to Safety Factor 5 (to ascertain the implications of extending Pickering NGS operation beyond 2020), as well as a review of the R04 Pickering Licence Conditions Handbook [4] for any impacts of Pickering NGS operation beyond 2020 on: a) OPG commitments previously made to the CNSC, b) open CNSC action items, and c) exemptions granted by the CNSC (all related to Safety Factor 5).

The results of the review of Safety Factor 5 are discussed in Section 5.0. The review of Safety Factor 5 has confirmed that the deterministic safety analysis programs and procedures at OPG are comprehensive, resulting in a systematic and disciplined approach to identifying, prioritizing and addressing any safety analysis related issues. As discussed in Section 5.0, the review identified seven gaps that will need to be addressed further as part of the PSR2 Global Assessment process.

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1.0 INTRODUCTION

Ontario Power Generation (OPG) is performing a Periodic Safety Review (PSR) to support the possibility of continued operation of Pickering Nuclear Generating Station (NGS) beyond 2020.[1] A comprehensive Integrated Safety Review (ISR) was completed for Pickering Units 5 through 8 in 2009 in support of refurbishment and continued operation. Pickering Units 1,4 integrated safety assessments were also performed for Pickering A Return to Service (PARTS) in support of approval to restart Units 1 and 4. In addition to these Pickering-specific studies, the 2013 Darlington ISR performed extensive code and standard reviews that were updated in relation to the versions that were assessed in the 2009 Pickering B ISR.[2] These previous ISRs are considered to constitute the first PSR completed for Pickering (referred to as "PSR1"). The current PSR (referred to as "PSR2") is a subsequent PSR building on the basis of earlier OPG integrated safety assessments through review of the various studies, assessments and licence renewals performed since PSR1. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

PSR2 will support and complement the licence renewal application for Pickering NGS going forward. Fifteen Safety Factors will be assessed as part of the PSR. The purpose of Safety Factor reviews is to confirm that the design, condition and operation of Pickering Units 1,4 and 5-8 (as well as common systems) will support continued safe operation for the period of PSR2 by:

- Assessing compliance against "Review Tasks" identified in the PSR2 Basis Document [1], which were derived from Canadian Nuclear Safety Commission (CNSC) REGDOC-2.3.3 [2] and International Atomic Energy Agency (IAEA) SSG-25 [3];

- Documenting assessments against applicable modern Laws, Regulations, Codes and Standards (L/R/C/Ss) (as defined in Reference [1]); and

- Reviewing the effectiveness of applicable OPG Nuclear Programs that support the above assessments, through review of audit and self-assessment results.

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1 Currently, Pickering Units 5-8 are approved to operate to 247,000 Effective Full Power Hours. This operation limit is expected to be reached on some units in 2020. For the purposes of PSR2, OPG assumes operation of Pickering NGS for up to eight additional years, from 2020 until 2028. OPG will make a decision regarding the permanent shut down dates for the six reactors following the performance of a technical evaluation that will include PSR2, and will communicate it to the CNSC as required by the current Power Reactor Operating Licence (PROL).

2 Much of the compliance assessment and evaluation of Safety Factor health for the Darlington ISR is based on programs and practices that apply across OPG's Nuclear operations. As a result, where Pickering is confirmed to follow the same Nuclear programs and practices as were assessed for Darlington, the Darlington ISR programmatic conclusions are applicable to Pickering. As discussed in Section 1.0, an effectiveness review (effectiveness at Pickering NGS) of OPG programs used to demonstrate compliance with the PSR2 Assessment Basis will be conducted using recent audit and self-assessment results.

As outlined in IAEA SSG-25 [3], the objective of the review of Deterministic Safety Analysis Safety Factor 5 is to: "determine to what extent the existing deterministic safety analysis is complete and remains valid when the following aspects have been taken into account:

- The actual plant design, including all modifications of Structures, Systems and Components (SSCs) since the last update of the safety analysis report or the last PSR;
- Current operating modes and fuel management;
- The actual condition of SSCs important to safety and their predicted state at the end of the period covered by the PSR;
- The use of modern, validated computer codes;
- Current deterministic methods;
- Current safety standards and knowledge (including research and development outcomes); and
- The existence and adequacy of safety margins".

REGDOC-2.3.3 [2] requires that: "The licensee shall conduct a PSR in accordance with this regulatory document for the period until the next PSR or, if applicable, until the end of commercial operation of the plant."

This report documents the results of the review of Safety Factor 5 for Pickering PSR2. The report is based on the OPG Governance, Programs, data, and material available up to January 15, 2016 which is the freeze date for PSR2.

2.0 SCOPE OF REVIEW

2.1 Review Task Assessments

The Pickering PSR2 Safety Factor 5 Review Tasks are defined in Reference [1]. Details of the derivation of these Review Tasks from CNSC REGDOC-2.3.3 [2] and IAEA SSG-25 [3] are shown in Reference [5]. The Safety Factor 5 Review Tasks are:

1) Confirm the existence of current deterministic safety analyses and the assumptions used to perform these analyses.

2) Evaluate the documentation and processes for defining, implementing, and maintaining the Safe Operating Envelope.

3) Perform assessment of OPG's Deterministic Safety Analysis to determine if the postulated events, event sequences and event combinations covered by the existing analysis are sufficient when compared against those for a modern nuclear power plant in accordance with the methodology in CNSC REGDOC-2.4.1, "Deterministic Safety Analysis".

4) Review adequacy of the documented guidelines for Deterministic Safety Analysis.

5) Evaluate the supporting analyses for design extension conditions to confirm that the arrangements aimed at preventing or mitigating severe core damage meet regulatory requirements.

6) Confirm that the impact of equipment failures and human errors, as well as the adequacy of engineering and administrative measures to prevent and mitigate accidents, have been analyzed and documented.

7) Confirm that the capabilities of the plant in its current state, and where relevant with account taken of planned safety improvements, have been demonstrated to be within regulatory requirements and expectations for both normal operation and accident conditions.

   In addition, confirm that plans are in place to ensure that forecast operational conditions of the plant will meet acceptance criteria for the design basis, including adequacy of safety margins, throughout the period of PSR2.

The methodology for the reviews is discussed in Section 3.1. Review Task findings are summarized in Section 4.1 of this report.

2.2 L/R/C/S Reviews

The applicable Laws, Regulations, Codes and Standards relevant to the Deterministic Safety Analysis Safety Factor are identified in Reference [1] and are listed in Table 1 below. Table 1 also identifies the modern version and date of each L/R/C/S to be considered, the Safety Factor(s) to which each document is applicable, and the type of review that will be completed in PSR2.

All of the Safety Factor 5 L/R/C/S reviews are incremental in nature. The definition of an Incremental Review is as follows:

- Incremental Review: For L/R/C/Ss that have been reviewed in PSR1 but have had revisions since the last review, a topical review will be performed of the changes.

The methodology for the reviews is discussed in Section 3.2. A detailed assessment for each L/R/C/S is provided in References [6], [7] and [8]. Associated findings are summarized in Section 4.2 of this report.

Table 1: L/R/C/Ss Reviewed for Deterministic Safety Analysis Safety Factor 5

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
L/R/C/Ss Referenced in Pickering NGS PROL 48.02/2018						
1	CSA N286	Management System Requirements for Nuclear Facilities	N286-12	5, 6, 9, 10, 11	Incremental	N286 addressed as part of Pickering B and Darlington ISRs.
2	CSA N290.15	Requirements for the Safe Operating Envelope of Nuclear Power Plants	N290.15-10	5, 8	Incremental	N290.15 not addressed as part of Pickering B or Darlington ISRs, but gap analysis has been performed against OPG Governance and N290.15.
3	CSA N286.7	Quality Assurance of Analytical, Scientific and Design Computer Programs	N286.7-16	1, 5, 6, 7, 10	Incremental	N286.7 addressed as part of Pickering B and Darlington ISRs.
4	CNSC REGDOC-2.4.1	Deterministic Safety Analysis	2014	5, 7	Incremental	C-6 addressed as part of the Pickering B ISR and PARTS. S-310 and RD-310 addressed as part of the Pickering B and Darlington ISRs, respectively. Implementation plan in place and gap

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
						assessment between REGDOC-2.4.1 and OPG Safety Analysis Program already performed.
Additional L/R/C/Ss						
5	CNSC REGDOC-2.5.2	Design of Reactor Facilities: Nuclear Power Plants	2014	1, 5, 6, 7	Incremental	RD-337 and NS-R-1 (precursors to REGDOC-2.5.2) addressed as part of Darlington ISR. NS-R-1 also addressed as part of Pickering B ISR.
6	CNSC REGDOC-2.3.2	Accident Management, Version 2	2015	1, 5, 6, 7, 8, 10	Incremental	REGDOC-2.3.2 addressed as part of Darlington ISR.
7	CSA N286.7.1	Guideline for the Application of N286.7-99, Quality Assurance of Analytical, Scientific, and Design Computer Programs for Nuclear Power Plants	N286.7.1-09	1, 5, 6, 7, 10	N/A [3]	N286.7.1 not addressed as part of Pickering B or Darlington ISRs.
8	CNSC G-144	Trip Parameter Acceptance Criteria for the Safety Analysis of CANDU Nuclear Power Plants	2006	5	Incremental	G-144 addressed as part of Pickering B and Darlington ISRs
9	CNSC G-149	Computer Programs Used in Design and Safety Analyses of Nuclear Power Plants and Research Reactors	2000	1, 5, 6, 7	Incremental	G-149 addressed as part of Pickering B and Darlington ISRs
10	CSA N288.2	Guidelines for Calculating Radiological Consequences to the Public from a Release of Airborne Radioactive Material for Nuclear Reactor Accidents	N288.2-14	5	Incremental	N288.2 addressed as part of Pickering B and Darlington ISRs

[3] The N286.7.1 guide has been amalgamated into the new (-16) edition of the N286.7 Standard. The N286.7 CSA Impact Statement [9] states: "The CSA N286.7.1 guide will no longer be maintained after this new edition of N286.7 is issued. Any relevant guidance has been put into the new edition of N286.7." As a result, only the review of N286.7-16 has been prepared for PSR2.

2.3 OPG Program Effectiveness Reviews

The OPG Nuclear Program (N-PROG) reviewed for Safety Factor 5 is listed in Table 2 below.[4] The methodology for the effectiveness reviews is discussed in Section 3.3. The assessment results of the N-PROG in Table 2 are provided in Appendix B, and findings are summarized in Section 4.3.

Table 2: OPG Program Reviewed for Safety Factor 5

Document Number	Document Title
N-PROG-MP-0014 [10]	Reactor Safety Program

2.4 Additional Reviews

The PSR2 Safety Factor 5 report includes a review of the R04 Pickering Licence Conditions Handbook (LCH) [4] for any impacts of Pickering NGS operation beyond 2020 on the following (all related to Safety Factor 5):

- OPG commitments previously made to the CNSC;
- Open CNSC action items; and
- Exemptions granted by the CNSC.

The PSR2 assessment includes identification and review of previously identified PSR1 gaps related to Safety Factor 5 to ascertain the implications of extending Pickering NGS operation beyond 2020. The methodology for these reviews is described in Section 3.4. Any PSR2 gaps identified as a result of programmatic Darlington PSR1 gaps related to Safety Factor 5 are discussed in Section 4.4 of this report. The review of Pickering PSR1 gaps previously identified in the Pickering Units 5-8 Continued Operations Plan (COP) [11] is provided in a separate PSR2 COP Review Report.

In addition, Fukushima Action Items (FAIs) were reviewed to identify implications of extending operation beyond 2020 (if any). This review is presented in a separate PSR2 FAI Review Report.

Any PSR2 gaps identified as a result of the Safety Factor 5 review which are relevant to other Safety Factors are discussed in Section 4.4 of this report.

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[4] The list of Nuclear Programs to be assessed for effectiveness for PSR2 was derived from review of current OPG Governance. Although there may be content in Nuclear Programs that is applicable to multiple Safety Factors, N-PROG reviews are only provided in one Safety Factor report and are not duplicated.

3.0 METHODOLOGY

The sub-sections below summarize the methodology used to assess Review Tasks, L/R/C/Ss, and Nuclear Program effectiveness for the Deterministic Safety Analysis Safety Factor.

3.1 Review Tasks

As discussed earlier, the Safety Factor Review Tasks are derived from CNSC REGDOC-2.3.3 [2] and IAEA SSG-25 [3], taking into consideration the Review Tasks used in the Pickering B and Darlington ISRs (as derived in [5]).

For each Safety Factor 5 Review Task identified in Section 2.1, a confirmation of the existence of applicable OPG Governance, Programs, Policies and Procedures (as well as Instructions and Guidelines, as applicable) was performed. Compliance against Review Tasks is also assessed by reference to applicable Condition Assessments, safety analyses and operating experience, as required.

The Review Task assessments identify Compliances and Gaps as defined below:

- Compliance: Compliance indicates that either the safety requirement or the intent of the Review Task is met.
- Gap: A Gap indicates that the intent of the Review Task is not met.

3.2 L/R/C/S Reviews

The process to identify the modern L/R/C/Ss that are applicable to the PSR2 Assessment Basis involved first creating a broad list from multiple sources (potential candidate L/R/C/Ss) and then filtering it to identify those that are most significant and that are applicable to the PSR2 scope. The identification and selection criteria are detailed in Reference [1]. The result of the identification and selection process was a set of modern L/R/C/Ss that became part of the "PSR2 Assessment Basis".

PSR2 is focused on the extension of Pickering NGS operations beyond 2020, and will conduct reviews against a baseline of past PSR1 work. As a subsequent PSR, PSR2 focuses on changes in requirements, plant conditions, operating experience and new information. Since PSR2 is an update of previous ISRs, it incorporates reviews of L/R/C/Ss that have occurred as new versions have been issued. Since this assessment is a subsequent PSR, the focus is on identifying differences between what was previously assessed and what is now different within the current Pickering PSR2 Assessment Basis. In general, these differences relate to:

- More recent (new or revised) L/R/C/S versions than what was previously assessed as part of PSR1;[5]

- Safety significant differences between Pickering and Darlington, if the Darlington ISR is the basis for the earlier assessment;

- Implications of extending Pickering NGS operation beyond 2020; and

- Safety significant differences between Pickering Units 1,4 and Units 5-8.

As described in Reference [1], L/R/C/S review types are clause-by-clause, high level or incremental. Most of the L/R/C/Ss in the PSR2 Assessment Basis receive incremental reviews since PSR2 is an update of previous PSR1 assessments and clause-by-clause or high level reviews for the majority of the L/R/C/Ss in the PSR2 Assessment Basis have already been completed. Implementation plans (including gap analyses or code-over-code reviews) also exist for the latest editions of many L/R/C/Ss. As a result, incremental review is also used in circumstances where a L/R/C/S in the PSR2 Assessment Basis was not assessed in previous PSR1 reviews but an implementation plan currently exists for compliance.

The PSR2 incremental reviews in this report include an assessment of the intent of recent changes to the L/R/C/Ss on a topic or subject-matter basis where there is potential to impact nuclear safety. Incremental reviews provide:

- A summary of the purpose of the L/R/C/S;

- Pertinent background information about the current revision of the L/R/C/S that is being considered;

- Identification of which Safety Factor(s) are applicable to the current revision of the L/R/C/S;

- A description of which version(s) of the L/R/C/S were assessed for PSR1 (i.e., Darlington ISR (for programmatic content), Pickering B ISR and PARTS code reviews);

- Identification of whether the current version of the L/R/C/S is an update of a previous version of the L/R/C/S that was assessed in PSR1 (and if so, a description of the major changes in the latest revision is provided as discussed below);

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[5] "New" refers to a code or standard that was not previously considered in the context of earlier assessments. "Revised" refers to an updated version of a code or standard that was previously considered in the context of earlier assessments. Where a document has a new number/type, but addresses the same topic from the same organization, it is a "revised", not "new", document (e.g., if a REGDOC replaces a CNSC G or RD document).

- An assessment of the applicability of PSR1 assessment findings (gaps and conclusions), including the implications of extending Pickering NGS operation beyond 2020 if any;

- An assessment of the applicability of assessment findings that address more recent (post-PSR1) editions of the L/R/C/S, including any implementation or transition plans that are already committed to by OPG; and

- Where PSR1 and post-PSR1 assessments are not sufficient to address changes in the latest edition of the L/R/C/S, an assessment of the changes from the previously assessed edition of the L/R/C/S (including identification of any safety significant PSR2 gaps which result).

High Level reviews provide the same information as above, where applicable, in a similar format. However, given that High Level L/R/C/Ss generally have not received past assessment during PSR1, the Incremental review content is augmented by a high level, section-by-section assessment of the degree of conformance of Pickering NGS with the L/R/C/S (demonstrating, with supporting evidence, whether the intent of the requirements stipulated in the document are met).

There are currently no L/R/C/S clause-by-clause reviews identified in the PSR2 Assessment Basis.

The Safety Factor 5 L/R/C/S reviews identify Compliances and Gaps as defined below:[6]

- <u>Compliance</u>:

    - Where an Incremental review has been performed, Compliance indicates that the change in the safety requirement, per the topical review, is met.

    - Where a High Level review has been performed, Compliance indicates that the intent of the safety requirement is met. (Note: No High Level reviews were performed as part of Safety Factor 5.)

    - For Clause-by-Clause reviews of modern L/R/C/Ss, Compliance indicates that the safety requirement is met. (Note: No Clause-by-Clause reviews were performed as part of Safety Factor 5.)

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[6] Safety Factor assessments for Review Tasks and L/R/C/S reviews make use of: a) OPG Governance, Programs, Policies and Procedures which support the assessment arguments, b) Commitments previously made to the CNSC, open CNSC action items, and exemptions granted by the CNSC (all related to the Safety Factor under review), as identified in the R04 Pickering LCH [4], c) Identification of previously identified Pickering-specific or programmatic PSR1 gaps related to the Safety Factor under review and the status of OPG's improvement plan(s) or other dispositions to address these, and d) Assessments and reviews performed since the PSR1 documents were completed.

- Gap:
    - Where an Incremental review has been performed, a Gap indicates that the change in the safety requirement, per the topical review, is not met.
    - Where a High Level review has been performed, a Gap indicates that the intent of the safety requirement is not met. (Note: No High Level reviews were performed as part of Safety Factor 5.)
    - For Clause-by-Clause reviews of modern L/R/C/Ss, a Gap indicates that the safety requirement is not met. (Note: No Clause-by-Clause reviews were performed as part of Safety Factor 5.)

The reviews assume that use of the word:

- "Shall" is used in an L/R/C/S to express a requirement, i.e., a provision that the licensee is obliged to satisfy in order to comply with the standard;
- "Should" is used to express a recommendation or that which is advised but not required;
- "May" is used to express an option or that which is permissible within the limits of the standard; and
- "Can" is used to express possibility or capability.

3.3 OPG Program Effectiveness Reviews

As discussed earlier, effectiveness reviews (effectiveness at Pickering NGS) of OPG programs used to demonstrate compliance with the PSR2 Assessment Basis were conducted, using recent applicable audit and self-assessment results:

- OPG Nuclear Oversight independent performance-based Program audits (typically performed in 1 to 5 year cycles) and self-assessments. This includes review of associated Station Condition Records and Action Requests to determine the status of any resulting corrective actions; and
- CNSC "Type I" and "Type II" inspections of the effectiveness and performance of OPG programs, where discussed in OPG audits or self-assessments.

There are many audits and self-assessments that are performed to assess the effectiveness of important aspects of each program. A sample of audits and self-assessments has been summarized for each program in order to demonstrate that program effectiveness is being assessed on an ongoing basis. The focus of these reviews was on effectiveness of the programs at Pickering NGS, where specific information is available. Results from these audits and self-assessments will be considered in the Global Assessment process. It is noted that audits and self-assessments are, by their nature, self-critical and are used to drive excellence in

performance. As a result, the broad review scope of program audits focuses on identifying improvement opportunities rather than presenting a balanced picture of program performance.

Program effectiveness is also monitored and addressed through the Fleetview Program Health and Performance Reporting process [12]. This process involves direct oversight by the Chief Nuclear Officer, and includes participation by the Nuclear Executive Committee members. Programs are reviewed, senior oversight is provided, and improvement plans are generated.

The list of Nuclear Programs to be assessed for each Safety Factor was derived from review of current OPG Governance, and has used the most recent version of these documents as of the PSR2 freeze date of January 15, 2016.

3.4 Additional Reviews

A review of the R04 Pickering LCH [4] was performed to determine if there are any impacts associated with operation of the Pickering Units past 2020 on the following (all related to Safety Factor 5):

- Commitments previously made to the CNSC;
- Open CNSC action items; and
- Exemptions granted by the CNSC.

The PSR2 assessment includes identification and review of previously identified Pickering-specific or programmatic PSR1 gaps related to Safety Factor 5 (as identified in the Darlington ISR Integrated Implementation Plan (IIP) [13] and Pickering Units 5-8 Continued Operations Plan [11]) to ascertain the status of OPG's improvement plan(s) or other dispositions to address these and the implications of extending operation beyond 2020 (if any).[7]

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[7] PSR2 includes consideration and confirmation that the findings of PSR1 remain valid, as applicable, for the operation period. This includes assessment of PSR1 conclusions against implications resulting from extended operation. In particular, Pickering PSR1 results are applicable to PSR2 if there was a PSR1 gap that is still open, or if a closed PSR1 gap could be affected by extended operation. If so these gaps are carried forward into PSR2 for consideration in the Global Assessment. (When references to PSR1 are made, the source document is identified and the relevant text from that source document is summarized in the context of PSR2.) With respect to the Darlington ISR, much of the evaluation of Safety Factor health is based on programs and practices that apply across OPG's Nuclear operations. As a result, Darlington ISR programmatic conclusions are applicable to Pickering PSR2 where Pickering is confirmed to follow the same Nuclear programs and practices that were assessed for Darlington. Darlington PSR1 results are applicable to Pickering PSR2 if there are Darlington PSR1 gaps that are found to be relevant to Pickering PSR2.

Fukushima Action Items were reviewed to identify implications of extending operation beyond 2020 (if any). The methodology for this review is provided in a separate PSR2 FAI Review Report.

Any PSR2 gaps identified as a result of the Safety Factor 5 review which are relevant to other Safety Factors are also discussed.

4.0 REVIEW FINDINGS

4.1 Review Tasks

The sub-sections below provide an assessment of the adequacy of applicable OPG Governance, Programs, Policies and Procedures (as well as Instructions and Guidelines, as applicable) in demonstrating compliance against the Safety Factor 5 Review Tasks.

4.1.1 Review Task #1: Current Deterministic Safety Analyses and Assumptions

Confirm the existence of current deterministic safety analyses and the assumptions used to perform these analyses.

As defined in CNSC REGDOC-2.4.1, "Deterministic Safety Analysis" [14], deterministic safety analysis is an analysis of a facility's response to an event, which is performed by using predetermined rules and assumptions (e.g., those concerning the initial facility operational state, availability and performance of facility systems and operator actions). Analysis of such events is retained in the Pickering 1,4 and 5-8 Safety Reports, which are made up of three parts:

- Part 1 – Plant/Site Description [15], [16];
- Part 2 – Design Description [15], [17]; and
- Part 3 – Accident Analysis [18], [19].

The licensing basis for deterministic safety analysis is contained in Part 3 of the Safety Report [18], [19], which was last updated in October 2013 for Pickering 1,4 and October 2014 for Pickering 5-8.

As outlined in N-PROC-MP-0086, "Safety Analysis Basis and Safety Report Update" [20], the Analysis of Record (AoR) is a collection of documents which consists of the latest revision of the Safety Report as well as CNSC submissions that update or supersede analysis reported in the latest revision of the Safety Report. Guidelines for the maintenance and control of the AoR are provided in N-INS-09000-10004, "Guidelines for the Control of the Analysis of Record" [21]. The latest revisions of the Pickering 1,4 AoR (NA44-REP-00531.7-10001, "Pickering A Analysis of Record" [22]) and Pickering 5-8 AoR (NK30-REP-00531.7-00001, "Pickering B Analysis of Record" [23]) were issued in November 2015.

<u>Safety Report Update</u>

Per Section 4.1 of CNSC Regulatory Document REGDOC-3.1.1, "Reporting Requirements for Nuclear Power Plants" [24], the Safety Report is required to be updated and submitted to the CNSC within five years of the date of the previous submission, to incorporate new findings and analyses. N-PROC-RA-0094, "Discovery Issue Resolution Process" [25], also requires that upon discovery of an issue (or potential issue) with deterministic safety analysis (termed Discovery Issue Resolution Process (DIRP)), a Station Condition Record (SCR) must be raised and the Safety

Report Update process must be initiated. N-PROC-MP-0086, "Safety Analysis Basis and Safety Report Update" [20], outlines the Safety Report Update (SRU) process, which is further detailed in the following instructions:

- N-INS-09000-10001, "Processing of Safety Report Analysis Issues: Overview" [26]
  This instruction provides an overview of the steps involved in updating the Safety Report, including the process for managing safety analysis issues.

- N-INS-09000-10002, "Guidelines for Evaluating and Prioritizing Safety Report Analysis Issues" [27]
  This instruction describes guidelines for evaluating and prioritizing Safety Report analysis issues. Potential safety report analysis issues of varying complexity and safety significance are transmitted to the Nuclear Safety and Technology Department for dispositions and the issues are assessed to determine whether they impact the current Safety Report analysis sections and whether new analysis is required.

- N-INS-09000-10004, "Guidelines for the Control of the Analysis of Record [21]
  This instruction provides guidance to determine when a submission needs to be included in the AoR and how to maintain the AoR for each station. This instruction is intended to apply to portions of the Safety Report which present safety analysis results (i.e. Part 3 – Accident Analysis).

- N-INS-09000-10005, "Safety Report Issue Database Management" [28]
  This instruction provides guidance for the management of the Safety Report Analysis Issue Database, which is a Microsoft Access Database which has been developed for recording Safety Report issue information.

As outlined in N-CORR-00531-07409, "OPG Safety Analysis Improvement and REGDOC-2.4.1 Implementation – Action Item 2014OPG-5461" [29], OPG has developed an action plan to comply with CNSC Regulatory Document REGDOC-2.4.1, "Deterministic Safety Analysis" [14] for both Pickering and Darlington NGS. A systematic process was utilized to prioritize Safety Report Appendices which are to undergo Safety Analysis Improvements based on their associated safety significance. For Pickering NGS, the analyses identified for development are the Common Mode Events Appendices. Note: The L/R/C/S review of REGDOC-2.4.1 [14] is addressed in Section 4.2 of this Report.

Safety Analysis Assumptions

Each initiating event analyzed in the Appendices of Part 3 of the Safety Report [18], [19], is characterized by a description of its conditions, including:

- Definition of initiating events;
- Initial conditions of the system and applicable boundary conditions;
- Control system conditions and logic;

- Availability/credits for systems and components;
- Operator actions;
- Key phenomenon and process;
- Method of analysis;
- Deterministic assumptions;
- Demonstration of computer code(s) applicability; and
- Acceptance criteria.

Some of the key generic assumptions included in deterministic safety analysis are as follows:

- Plant State and Configuration - Plant states are divided into normal operation and accident conditions. This includes specifying both system and component configurations and corresponding initial condition values for plant parameters. Plant models used in deterministic safety analysis (Table 1-12 and Table S.1-11 in Part 3 of the Pickering 1,4 and 5-8 Safety Report respectively [18], [19]) reflect the design of the plant, consistent with its current state, and the operating period to be covered. Depending on the analysis methodology, simplified conservative representation and conservative boundary conditions are assumed, which results in a conservative estimate of the safety margin.

- Operator Actions - Tables 1-2 to 1-11 and Tables S.1-1 to S.1-10 in Part 3 of the Pickering 1,4 and 5-8 Safety Report respectively [18], [19], summarize all required operator action credits. Typically, the earliest time of required operator action following a signal is 15 minutes for control room action and 30 minutes for field action.

- Impact of Aging –Plant aging has the potential to adversely impact safety margins and reactor operation. As identified in Section 4.4 of the Application for Renewal of Pickering NGS Power Reactor Operating Licence [30], an OPG Heat Transport System (HTS) Aging Management Program was initiated in 2000 to evaluate the impact of the HTS component aging on safety margins. The objective was to provide an integrated assessment of the collective effects of the identified aging mechanisms, and to develop effective safety margin management strategies based on the results of the assessment. Appendix 11 of the Pickering 1,4 and 5-8 Safety Reports [18],[19], provide details of the integration of HTS aging effects in Design Basis Accident (DBA) analysis (Note, Section 4.1.7, Review Task #7 provides further details on the status of HTS aging at Pickering 1,4 and 5-8).

Conclusion:

The conclusion of this Review Task assessment is that there exist current deterministic safety analyses and assumptions used to perform these analyses. The intent of Review Task #1 is met and therefore Pickering NGS is compliant.

4.1.2 Review Task #2: Documentation for Safe Operating Envelope

Evaluate the documentation and processes for defining, implementing and maintaining the Safe Operating Envelope.

The Safe Operating Envelope (SOE) is the set of limits and conditions within which the plant shall be operated to ensure conformance with the Safety Report and that the Safety Report conclusions remain valid. N-STD-MP-0016, "Safe Operating Envelope" [31], provides requirements for defining, implementing and maintaining the SOE. The specific objectives of the SOE are to establish the following:

- Thorough and current record of safety credits and operating limits in the form of Operational Safety Requirements (OSRs) and associated Instrument Uncertainty Calculation (IUC) reports. Safe Operating Limits and Conditions of Operability (SOE Limits) are captured in station operating documentation, which provide plant operators with the information required to ensure safe operation of the plant in conformance with the requirements of the Safety Analysis.
- A compliance framework whereby plant operation within the requirements established as part of the SOE is verified on a regular basis and appropriate corrective actions are initiated upon discovery of plant operation outside of the SOE.
- Infrastructure by which the SOE is integrated with other relevant business processes and maintained current over the life of the station.

The methodology for the preparation and revision of OSR reports is described in N-ST-08131.02-10000, "Preparation of Operational Safety Requirements" [32]. The OSR reports contain the following information (as outlined in Section 1.2.4 of N-STD-MP-0016 [31]):

- A brief overview of the safety functions of the system in relation to the Design Basis Accidents for which it is credited;
- Safety limits defining the minimum acceptable standards with respect to component or parameter performance;
- Surveillance requirements identifying specific objectives of tests or checks to ensure plant operation within the defined Safety Limits; and
- Conditions of Operability defining the impact on overall availability of the system or safety function in the event of operation outside of the defined Safety Limits.

N-STI-03602-10000, "SOE Instrument Uncertainty and Allowable Value Calculations" [33], describes the methodology for the preparation and revision of IUC reports. The IUC reports contain the following calculations for applicable instrumentation loops (as outlined in Section 1.2.7 of N-STD-MP-0016 [31]):

- Total uncertainty for instrument loops having a credited post-accident function, for comparison against the allowance for uncertainty made in the safety analysis or design, to confirm the existence of an adequate safety margin;
- Allowable values to confirm instrument loop availability during calibration for instrument loops required for the post-accident actuation of credited safety functions;
- Surveillance limits to define the limiting acceptable indicated value of parameters for instrument loops required to perform indication functions (either post-accident or during surveillance); and
- As-found and as-left tolerances for use during calibration to confirm instrument operation within the normal expected range for all instrument loops associated with a numerical Safety Limit.

Implementation of SOE

As per Section 1.3.2 of N-STD-MP-0016, "Safe Operating Envelope" [31], information contained in the OSR and IUC reports is incorporated into affected station operating documentation (e.g., Operating Manuals, Abnormal Incident Manuals, Safety Related System Tests and the preventative maintenance program) in accordance with N-INS-03602-10001, "Preparation of Safe Operating Envelope Compliance Tables" [34]. This requires that a gap analysis be performed comparing the content of the OSR and IUC reports to the corresponding station operating documents. The SOE Compliance Table lists all the SOE parameters for a given system and provides references showing where each of the SOE Limits and surveillance requirements are captured within the applicable station operating documentation.

Section 4.1 of the Pickering Licence Conditions Handbook [4], states that OPG has had an implemented SOE program since 2012 and a complete set of SOE documentation (i.e. OSRs, IUCs and Compliance Tables) exists for the following Pickering 1,4 and 5-8 systems important to safety:

- Shutdown Systems;
- Emergency Coolant Injection System;
- Negative Pressure Containment System;
- Fuel and Reactor Physics;
- Reactor Regulating System;
- Heat Transport System;
- Moderator System;
- Shutdown Cooling System;
- Main Steam Supply System;
- Feedwater System;
- Emergency Boiler Water Supply System (Pickering 1,4) / Emergency Water Supply System (Pickering 5-8);
- Boiler Emergency Cooling System;
- Powerhouse Emergency Venting System;
- Service Water Systems;
- Electrical Power System (Pickering 1,4) / Group 1 Electrical Power Supplies;

- Emergency Power Supply System (Pickering 5-8);
- Fuel Handling System and Irradiated Fuel Bay;
- Shield Cooling System;
- Annulus Gas System;
- Critical Safety Parameter Monitoring Instrumentation;
- Inter-Station Transfer Bus (Pickering 1,4);
- High Pressure Emergency Coolant Injection Power Supplies; and
- Powerhouse Environmental Protection.

<u>SOE Maintenance</u>

As per Section 1.3.4 of N-STD-MP-0016, "Safe Operating Envelope" [31], SOE is maintained by ensuring the following:

- Revisions of OSR and IUC reports are kept current in relation to changes in plant design, operation, Safety Analysis (e.g., SRU) or license requirements.
- The need for revision to the OSR report, SOE Compliance Table or IUC reports may be identified by any of the following processes:
    - N-PROC-MP-0086, "Safety Analysis Basis and Safety Report Updates" [20];
    - N-PROC-AS-0028, "Development, Review and Approval of Technical Procedures" [35];
    - N-PROG-MP-0001, "Engineering Change Control" [36]; and
    - N-PROG-RA-0003, "Corrective Action" [37].
- Changes to SOE reports are managed using the Engineering Change Control process as follows:
    - Engineering Change Request initiated;
    - Once the required actions are completed that identify the scope of changes, a Document Change Request (DCR) is created summarizing the required SOE report revisions; and
    - The approved DCR becomes the basis for identifying and implementing any required changes to station operating documentation. This ensures operation in compliance with the revised SOE requirements including the basis for revising the SOE reports and any other documentation identified as part of the DCR process impacted by the change (e.g., the Safety Report).
- Revisions of the SOE Compliance Tables are required to reflect the resolution of previously identified gaps over time and changes made to station operating documentation not related to OSR and IUC report revisions, but which affect the referencing of SOE requirements.

Also, when uncertainty arises with respect to the operability of equipment to meet the functional requirements of the defined SOE, a Technical Operability Evaluation (TOE) is performed. N-PROC-MP-0045, "Technical Operability Evaluation" [38], provides a process for identifying and evaluating degraded station conditions when the ability of SSCs to carry out their defined safety-related functions come into question. A formal

TOE provides a substantiated engineering verification that a SSC is capable of fulfilling its minimum credited safety function(s).

Note: The L/R/C/S review of CSA N290.15, "Requirements for the Safe Operating Envelope of Nuclear Power Plants" is addressed in Section 4.2 of this Report.

Conclusion:

The conclusion of this Review Task assessment confirms that documentation and processes for defining, implementing and maintaining the SOE exists. The intent of Review Task #2 is met and therefore Pickering NGS is compliant.

4.1.3 Review Task #3: Postulated Events

Perform assessment of OPG's Deterministic Safety Analysis to determine if the postulated events, event sequences and event combinations covered by the existing analysis are sufficient when compared against those for a modern nuclear power plant in accordance with the methodology in CNSC REGDOC-2.4.1, "Deterministic Safety Analysis".

An event is comprised of a discrete initiating failure (also known as a Postulated Initiating Event (PIE) or initiating failure) and the subsequent response of the plant, including human interaction as per procedures. An event can be initiated by an equipment failure, a human error or an external failure/event. The scope of initiating events addressed by the existing deterministic safety analysis for Pickering 1,4 and 5-8 (Part 3 of the Safety Report) [18], [19], consists of single/dual failure events[8] based on the requirements of the Siting Guide [39].

As outlined in Section 2.1 of the Safety Report [18], [19], all systems and components are reviewed to identify those containing significant quantities of radioactive materials. For each source of radioactive material, it is possible to determine ways in which unplanned releases of this material can occur, based on knowledge of the plant processes and past experience in selecting initiating events. This process leads to a comprehensive list of internal initiating events. To complete the list, all combinations of internal initiating events and compounding failures in the special safety systems are identified. Table 2-1 of the Pickering 1,4 Safety Report [18] and Table S.2-1 of the Pickering 5-8 Safety Report [19], identify these events, where in the Safety Report the analysis details are located and the dose category for each event. The initiating event categories include:

- Fuel Handling System Failures;
- Electrical Failures;
- Control Failures;

[8] Single failures include failures in process systems, while dual failures are single failures combined with a failure of a special safety system.

- Small Break Loss of Coolant Accidents (SBLOCA);
- Large Break Loss of Coolant Accidents(LBLOCA);
- Heat Transport System Breaks Outside Containment, Steam Generator Tube Failure and Bleed Cooler Failure;
- Feedwater and Steam Supply System Failures;
- Shutdown Cooling System Failures;
- Main Moderator and Moderator Auxiliary System Failures; and
- Shield Cooling System Failures.

Section 4.2 of CNSC REGDOC-2.4.1 [14], requires a systematic process to identify events, event sequences and event combinations that can potentially challenge the safety or control functions for the station. As outlined in N-CORR-00531-07409 [29], OPG has developed an Implementation Plan to comply with the requirements of REGDOC-2.4.1 [14]. **Pickering PSR2 Gap SF5-4**, identified as part of the L/R/C/S review of REGDOC-2.4.1 [14] and described in Section 4.2 of this Report, relates to consideration of analysis updates to comply with REGDOC-2.4.1 for operation beyond 2020. Any changes to the REGDOC-2.4.1 Implementation Plan will be addressed under this gap, and therefore there is no incremental gap identified in the context of this Review Task.

Conclusion:

The conclusion of this Review Task assessment is that the postulated events, event sequences and event combinations covered by the existing analysis are identified in the existing deterministic safety analysis, as summarized and documented in the Pickering NGS Safety Reports. OPG has developed an Implementation Plan to comply with the requirements of REGDOC-2.4.1 [14]. **Pickering PSR2 Gap SF5-4**, identified as part of the L/R/C/S review of REGDOC-2.4.1 [14] and described in Section 4.2 of this Report, relates to consideration of analysis updates to comply with REGDOC-2.4.1 for operation beyond 2020. Any changes to the REGDOC-2.4.1 Implementation Plan will be addressed under this gap, and therefore there is no incremental gap identified in the context of this Review Task.

4.1.4 Review Task #4: Guidelines for Deterministic Safety Analysis

Review the adequacy of the documented guidelines for Deterministic Safety Analysis.

As outlined in the CANDU Owners Group (COG) "Principles and Guidelines for Deterministic Safety Analysis" [40], for currently operating CANDU stations the Safety Report contains analysis based on the Siting Guide [39] (single/dual failure events) or Consultative Document C-6 "Requirements for the Safety Analysis of CANDU Nuclear

Power Plants" (5 event classes)[9]. However, REGDOC-2.4.1 [14] supersedes both the Siting Guide and C-6. Under REGDOC-2.4.1 [14], a Safety Report may contain Anticipated Operational Occurrences (AOO)/DBA[10] analysis; analysis of legacy single failure/dual failure/class 5 events with beyond design basis accident (BDBA) frequency retained as part of the original design basis; and summaries of BDBA analysis. As outlined in Reference [29], OPG has developed an implementation plan, which defines the REGDOC-2.4.1 compliant analyses to be undertaken in the 2014-2017 timeframe.

The governing programmatic document for deterministic safety analysis is N-PROG-MP-0014, "Reactor Safety Program" [10], which defines the organizational responsibilities and key program elements for management of issues related to deterministic safety analysis and the following major components for safe operation:

1) The Safety Analysis Basis
   The Safety Analysis Basis is formed from the safety analysis, which has been performed to characterize and quantify the consequences of various design basis accident events and to demonstrate that regulatory requirements have been met.

2) The Safe Operating Envelope
   The SOE identifies and implements the operating limits required by the Safety Analysis Basis.

3) Severe Accident Management
   Severe Accident Management (SAM) examines BDBAs, which are low frequency event sequences that are not included in the plant design basis (due to the low frequency of occurrence) and is not bounded by analyses of the station design basis. If the consequences of such events are significant core degradation, these BDBAs are referred to as Severe Accidents (SA). Severe Accident Management Guidelines (SAMGs) provide a framework for responding to BDBAs, in order to manage residual risk.

N-PROC-MP-0086, "Safety Analysis Basis and Safety Report Update" [20] (which is an implementing procedure of N-PROG-MP-0014, "Reactor Safety Program" [10]), provides the basis for the establishment of the Safety Analysis Basis and the SRU process (Section 4.1.1 of this report provides details of the SRU process). N-MAN-03600-10002, "Nuclear Safety Analysis Planning" [41], provides staff with instructions for the preparation and application of safety analysis plans and describes the roles and

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9 C-6 was applied to the Safety Analysis of the Darlington NGS only, while the Siting Guide was applied to Pickering NGS, Bruce A, Bruce B and Point Lepreau.

10 REGDOC-2.4.1 has subdivided the design basis events to include AOOs, which are events that are more complex than the normal operation manoeuvres, with the potential to challenge the safety of the reactor, and which might be reasonably expected to occur during the lifetime of a plant (previously analyzed as single failures under the Siting Guide). DBAs are not expected to occur during the lifetime of a plant, but in accordance with the principle of defence in depth, are considered in the design of a plant.

responsibilities of the staff involved. The preparation of the safety analysis plan allows the analyst to determine all aspects of the problem, including the plant systems, and items of existing analysis which may be affected by any work that is committed. N-MAN-03600-10003, "Nuclear Safety Analysis Execution" [42], provides instructions for the execution of the analysis plans.

N-STD-MP-0008, "Development, Qualification and Use of Scientific, Engineering, and Safety Analysis Software" [43], provides requirements for development, qualification and use of Scientific, Engineering and Safety Analysis software in design, analysis or support of the continued operation of OPG Nuclear stations. Particular attention is given to software that meets the CSA N286.7, "Quality Assurance of Analytical, Scientific and Design Computer Programs for Nuclear Power Plants" definition of scope as follows:

- Used in design activity associated with a system, structure or component associated with the facility Safety Related Systems list.
- Used in deterministic or probabilistic safety analysis or reliability study for the same systems.
- Used in reactor physics or fuel management calculations.
- Used in data transfer or pre- or post-processing calculations associated with any of the three activities above.

The Quality Assurance (QA) requirements for safety analysis and its associated computer software or data sets are defined in N-MAN-03600-10005, "Nuclear Safety Analysis" [44], which establishes the following:

- Defines general principles used to ensure that the likelihood of errors or omissions in safety analysis is small and continually reduced;
- Identifies quality attributes of the safety analysis and the computer software or data sets used;
- Identifies and reflects those external bases and standards incorporated in the safety analysis QA plan, which contribute to its adequacy and continuing improvement; and
- Stipulates the attitudes, values, perceptions, competencies and behaviours expected of all staff in conducting safety analysis work of high quality.

As a substantial amount of work is directly performed by external suppliers under contract to OPG, N-STD-MP-0014, "Managing Contracted Nuclear Safety Services" [45], defines requirements for procurement of Safety Analysis services, including Safety Analysis QA requirements.

Conclusion:

The conclusion of this Review Task assessment is that adequate guidelines for Deterministic Safety Analysis exist. The intent of Review Task #4 is met and therefore Pickering NGS is compliant.

4.1.5 Review Task #5: Design Extension Conditions

Evaluate the supporting analyses for design extension conditions to confirm that the arrangements aimed at preventing or mitigating severe core damage meet regulatory requirements.

N-STD-MP-0019, "Beyond Design Basis Accident Management" [46], defines the requirements for OPG's BDBA management. BDBA management is focussed primarily on the identification and implementation of operational strategies to:

- Mitigate the consequences of BDBAs and prevent progression to a SA, and thereby preclude or limit fuel and core damage;
- Terminate the progression and mitigate the consequences of SAs, and thereby minimize fuel and core damage;
- Maintain the integrity of the containment envelope;
- Limit both on-site and off-site releases; and
- Achieve a stable plant configuration as soon as possible and implement measures to sustain this state.

Collectively, the written guidance which implements these strategies is referred to as Emergency Mitigating Equipment Guidelines (EMEG)[11] and SAMG. Application of EMEG and SAMG may require temporary changes to permit operation of specific SSCs or EME to implement EMEG/SAMG objectives. For permanent changes to SSCs to facilitate EMEG/SAMG, N-GUID-01130-10000, "Modifications for Beyond Design Basis Accidents" [47], provides guidance related to the design, modification, procurement, operation and testing of SSCs, for managing the progression to BDBAs. Consistent with current regulatory direction (e.g. CNSC REGDOC-2.4.1 [14]), N-GUID-01130-10000 [47] refers to "BDBA" when referring to station conditions that can arise from low frequency events not considered in the plant design. Also, the term Design Extension Condition (DEC) is introduced (as per CNSC REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants" [48]) to describe a sub-set of BDBA conditions for which specific SSCs, referred to as Complementary Design Features, are provided for mitigation.

<u>Supporting Analysis</u>

Following the March 11, 2011 accident at Fukushima Daiichi Nuclear Power Plant, the CNSC requested all Canadian utilities to complete an assessment to review the impact of a similar event (i.e., earthquake and tsunami resulting in a total loss of power, subsequently resulting in a total loss of heat sinks to cool the fuel post-shutdown) at their respective stations. P-REP-03490-10012, "Fukushima Daiichi – Total Loss of Heat Sink Assessment for Pickering A and Pickering B" [49], consolidates the results from various evaluations performed and establishes the time line for progression of a total loss of heat sink event. P-REP-03490-10012 [49] also identifies mitigating provisions which could be put in place to prevent progression to a SA. Provisions to maintain or

[11] The term Emergency Mitigating Equipment (EME) was adopted to categorize the equipment that provides additional lines of defence to maintain critical safety functions.

re-establish the control, cool, contain and monitoring safety functions were examined to determine those that are most practical to implement and also meet specified requirements.

N-BDB-03600-00002, "OPG Emergency Mitigating Equipment for Beyond Design Basis Accidents: Technical Basis Document" [50], summarizes the technical basis for EME including:

- The bounding BDBA event sequence and associated analyses;
- The overall functional requirements for the EME; and
- Other information relevant to EME specification, design and procurement.

NA44-GUID-03600-00001, "Beyond Design Basis Functional Safety Requirements for Pickering 1-4 Nuclear Generating Station" [51] and NK30-GUID-03600-00001, "Beyond Design Basis Functional Safety Requirements for Pickering 5-8 Nuclear Generating Station" [52], identify the BDBA functional safety requirements for SSCs credited to manage and/or mitigate BDBAs, or to prevent progression to SAs. This includes:

- The functional safety requirements for temporary/portable BDBA mitigation equipment (e.g., EME);
- The functional safety requirements for permanent station equipment/connections having a BDBA mitigation function, and
- The incremental functional safety requirements for existing station SSCs that are part of BDBA mitigation strategies.

The BDBA Functional Safety Requirements for both Pickering 1,4 and 5-8 [51], [52], each possess a corresponding compliance table. The compliance tables identify the safety limits, surveillance requirements, preventative maintenance, impairment condition/response and any identified gaps for any of the equipment (e.g., EME pumps, EME generator and EME connection points)

From a Probabilistic Safety Assessment (PSA)[12] perspective, the Pickering NGS 1,4 and 5-8 Level 1[13] At-Power Internal Events Risk Assessment [53], [54] has been updated to take into account the Fukushima-related enhancements (e.g., BDBA procedures/guides and EME)[14]. One of the primary objectives for this update of the Level 1 PSA was to ensure that the PSA is consistent with the current station design and operation, which includes EME implementation. Likewise, the Level 2[15] PSA for Pickering NGS 1,4 and 5-8 has incorporated the risk benefits gained from the Fukushima-related enhancements [55], [56].

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[12] Note, Probabilistic Risk Assessment is now referred to as PSA.
[13] For a Level 1 PSA, the frequency of damage to fuel in the core is estimated (i.e. core damage frequency).
[14] The utilities developed and implemented action plans to enhance the safety and capability of CANDU reactors in response to the Fukushima accident.
[15] A Level 2 PSA extends the analysis of the Level 1 PSA to include containment performance and the frequency of release to the environment.

Conclusion:

The conclusion of this Review Task assessment is that supporting analyses for design extension conditions exist and the arrangements aimed at preventing or mitigating severe core damage meet regulatory requirements. The intent of Review Task #5 is met and therefore Pickering NGS is compliant.

4.1.6 Review Task #6: Equipment Failures and Human Errors

Confirm that the impact of equipment failures and human errors, as well as the adequacy of engineering and administrative measures to prevent and mitigate accidents, have been analyzed and documented.

As outlined in the COG "Principles and Guidelines for Deterministic Safety Analysis" [40], the safety analysis framework consists of:

- the design and beyond design basis deterministic safety analysis documented in the Safety Report, PSA, SAMG and design documentation;
- the analysis acceptance criteria;
- the processes for event identification and classification;
- analysis procedures and quality assurance requirements; and
- the SOE which defines operational limits and conditions based on design basis analysis to ensure operation is in accordance with the safety analysis.

The safety analysis framework provides high confidence in safe operation following any upset/accident starting from any credible configuration and provides confirmation that the impact of equipment failures, as well as the adequacy of engineering and administrative measures to prevent and mitigate accidents, have been analyzed and documented. For example, the accident analysis of the Safety Report assesses equipment failures and demonstrates the adequacy of the engineered mitigating system (e.g., shutdown system effectiveness). Administrative measures such as the SOE, establish the safe operating limits and conditions (SOE limits) of operability. The SOE limits are captured in station operating documentation, which provide operators with the information required to ensure safe operation of the plant in conformance with the requirements of the Safety Analysis.

Events in the current Pickering 1,4 and 5-8 Safety Reports [18], [19] are classified as single or dual failures (based on frequency) and are all considered design basis, based on the requirements of the Siting Guide [39]. Operator error does not normally impact plant response with respect to deterministic safety analysis for DBA events [40]. For example, the development and testing of procedures, intensive operator training and conservative operator action times reduce the probability of operator failure sufficiently to preclude the need for introducing operator error assumptions. However, the PSA includes human interaction events in the fault tree model for significant human interface related events that could lead to an accident. Examples of such human interaction events include:

- Failure to perform a required task;
- Performing an incorrect operation; or
- Failure to detect an alarmed component failure.

Conclusion:

The conclusion of this Review Task assessment is that the impact of equipment failures and human errors, as well as the adequacy of engineering and administrative measures to prevent and mitigate accidents, have been analyzed and documented. The intent of Review Task #6 is met and therefore Pickering NGS is compliant.

4.1.7 Review Task #7: Capabilities of the Plant in its Current State

Confirm that the capabilities of the plant in its current state, and where relevant with account taken of planned safety improvements, have been demonstrated to be within regulatory requirements and expectations for both normal operation and accident conditions.

In addition, confirm that plans are in place to ensure that forecast operational conditions of the plant will meet acceptance criteria for the design basis, including adequacy of safety margins, throughout the period of PSR2.

As outlined in Part 3 of the Pickering NGS Safety Reports [18], [19], the primary objective of the accident analysis is to demonstrate that the radiological consequences of the event under assessment do not exceed the Siting Guide [39] accident-dependent reference dose limits. Siting Guide dose limits are specified for PIEs involving a single process failure and for events involving a single process failure in conjunction with failure of one of the special safety systems. The dose limits are given in the following table:

	Individual Dose Limit		Population Dose Limit	
	Thyroid Dose (mSv)	Whole Body Dose (mSv)	Thyroid Dose (Person mSv)	Whole Body Dose (Person mSv)
Single Failure	30	5	10^5	10^5
Dual Failure	2500	250	10^7	10^7

Derived acceptance criteria are employed for each of the safety systems. These criteria are sufficient to ensure that the applicable dose limits are not exceeded (as demonstrated in the Part 3 Appendices of the Pickering NGS Safety Reports) and are in compliance with regulatory requirements[16]. Section 3 (Part 3) of the Pickering NGS Safety Reports [18], [19], provides an overview of the accident sequences and the consequences of these accidents (which is further supported by the detailed analysis in the Part 3 Appendices). These concise overviews contain:

- A discussion of safety system effectiveness;
- A discussion of the relevant functional response(s) to the event; and
- A discussion of the impacts of impairments, as appropriate, in the conclusions.

During normal operation, the reactors operate within specified operational limits and conditions, including start-up, power operation, shutting down, shut down, maintenance, testing and refuelling. Analysis of normal operation is part of the deterministic safety analysis that is performed during the design phase of the plant and is not repeated unless significant design or operational changes are made that could impact normal operation. Such analysis demonstrates that the process controls and alarms are effective in avoiding the need for safety system action during normal operation, the safety systems initiate only when needed, and normal operation does not escalate to an accident condition. During normal operation, radiological emissions to the environment are required to be maintained below the applicable regulatory emission limits (Derived Release Limits (DRL)). The DRL for a given radionuclide (specified in Section 10.1 of the LCH) is the release rate that would cause an individual of the most highly exposed group to receive and be committed to a dose equal to the regulatory annual dose limit due to a release to air or surface water during normal operation over the period of a calendar year. Safety Factor Report 14, "Radiological Impact on the Environment" provides additional detail on DRLs [57].

As outlined in N-CORR-00531-07409, "OPG Safety Analysis Improvement and REGDOC-2.4.1 Implementation – Action Item 2014OPG-5461" [29], OPG has developed an action plan to comply with CNSC Regulatory Document REGDOC-2.4.1, "Deterministic Safety Analysis" [14] for both Pickering and Darlington NGS. A systematic process was utilized to prioritize Safety Report Appendices which are to undergo Safety Analysis Improvements based on their associated safety significance. For Pickering NGS, the analyses identified for development are the Common Mode Events Appendices.

Design and Operating margins are managed through the implementation of N-STD-MP-0020, "Margin Management" [58]. The Margin Management standard applies to all SSCs with an important role in safe and reliable plant operation and addresses low

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[16] The current operating licence for Pickering NGS requires the implementation and maintenance of a safety analysis program in accordance with REGDOC-2.4.1, "Deterministic Safety Analysis"; and REGDOC-2.4.2, "Probabilistic Safety Analysis for Nuclear Power Plants". The corresponding Compliance Verification Criteria as documented in the Licence Conditions Handbook, identify the dose limits as specified in the Safety Report. Since the acceptance criteria for the accident analysis appendices ensure dose limits are met, the Pickering NGS Safety Reports demonstrate compliance with regulatory requirements.

margin issues arising from equipment degradation, plant configuration and operating procedure changes, engineering modifications and re-analysis. Safety Factor 4 Report, "Aging", provides further details on acceptance criteria and safety margins for safety related SSCs (refer to Review Task 8 in the Safety Factor 4 Report).

Heat Transport System Aging

An OPG Heat Transport System (HTS) Aging Management Program was initiated in 2000 to evaluate the impact of the HTS component aging on safety margins. The objective was to provide an integrated assessment of the collective effects of the identified aging mechanisms, and to develop effective safety margin management strategies based on the results of the assessment (note, Appendix 11 of the Pickering 1,4 and 5-8 Safety Reports [18],[19], provide details of the integration of HTS aging effects in DBA accident analysis). As identified in Section 4.4 of the Application for Renewal of Pickering NGS Power Reactor Operating Licence [30], the most critical accident scenarios from the perspective of HTS aging impacts were determined be:

- Slow Loss of Regulation (LOR) Accident (referred to as Neutron Overpower Protection (NOP))
  An enhanced NOP methodology (E-NOP) incorporating HTS aging effects was developed to perform NOP trip setpoint analysis at projected aged HTS conditions using Extreme Value Statistics (EVS) theory. As per CNSC correspondence [59], CNSC staff agreed that the stations (i.e., Pickering 1,4, Pickering 5-8 and Darlington NGS) are well protected against the Slow Loss of Regulation events by the required NOP trip setpoints calculated using the E-NOP EVS methodology and that there are adequate defence in depth provisions in place. Hence, there is a negligible risk to the station's physical barriers if an NOP event were to occur.

- Small Break Loss of Coolant Accident (SBLOCA) and Loss of Flow (LOF) Analysis
  For SBLOCA and LOF scenarios, it was determined that the major effects of aging are on dryout predictions as a result of Pressure Tube Diametral Creep (affecting Critical Heat Flux leading to earlier onset of dryout), feeder corrosion and roughness (affecting overall system resistance and subsequent decrease in core flow or flow redistribution between channels), and boiler tube fouling (increasing Reactor Inlet Header temperature). As outlined in the Progress Report on OPG HTS Aging Safety Analysis [60], the Pickering 1,4 analysis results for SBLOCA and LOF demonstrate adequate shutdown system trip coverage and ensure continued safe operation till December 2017 (6010 Effective Full Power Days (EFPD)) for the lead unit (Unit 1). Similarly, for Pickering Units 5-8, the SBLOCA and LOF analysis reports demonstrate adequate trip coverage till December 2018 (10300 EFPD) for the lead unit (Unit 6).

As noted above, the current safety analysis for Pickering 1,4 and 5-8, demonstrates adequate shutdown system trip coverage until 2017 and 2018 respectively for the SBLOCA and LOF scenarios. However, the impact of HTS component aging on the SBLOCA, LOF and Slow LOR accident scenarios, will need to be further assessed in order to demonstrate adequate safety margins exist beyond 2020, and therefore a gap exists for Pickering PSR2 (**Pickering PSR2 Gap SF5-1**). It is noted in Reference [60], that work is currently underway to perform Safety Analysis to support the initiative to extend Pickering commercial operation to 2024, accounting for possible mitigation strategies of life-limiting aging mechanisms. Note, a related gap has been captured in the Aging Safety Factor Report (PSR2 Gap SF4-18).

CANDU Safety Issues

As outlined in the 2014 Regulatory Oversight Report for Canadian Nuclear Power Plants [62], in 2007 the CNSC initiated a project to systematically reassess the status of potential design and safety analysis issues for CANDU reactors and to categorize them in order of safety significance (this project complemented the ongoing work at that time on Generic Action Items). These design and safety analysis issues became known as CANDU Safety Issues (CSIs), which were grouped as either Category 1, 2 or 3[17]. Per Reference [63], four CSIs remain in Category 3, of which three are related to LBLOCAs, while the remaining is non-LBLOCA related. For the LBLOCA CSIs, while the development of the industry's proposed Composite Analytical Approach (CAA) is on-going, the licensing basis of existing CANDU reactors for the LBLOCA scenario will continue to be based on conservative safety analysis for which acceptance criteria are established. For the non-LBLOCA CSI, the industry has applied to re-categorize the issue into a lower category based on analytical evidence and actions taken. Since four CSIs applicable to Pickering NGS (3 LBLOCA / 1 non-LBLOCA) are currently in Category 3 and are undergoing further assessment in order to re-classify into a lower category and address operation past 2020, a gap exists for Pickering PSR2 (**Pickering PSR2 Gap SF5-2**). Note, the 3 LBLOCAs CSIs are also captured as a gap in the PSR2 COP Report (Gap COP-20) [64] as they relate to Pickering B IIP Item I09. The 1 non-LBLOCA CSI is also identified as a gap in the Hazards Analysis Safety Factor Report (Gap SF7-1) [65] as it relates to pipe whip.

Conclusion:

The conclusion of this Review Task assessment is that the plant in its current state, and where relevant with account taken of planned safety improvements, has been demonstrated to be within regulatory requirements and expectations for both normal

[17] Category 1 CSI – The issue has been satisfactorily addressed in Canada; Category 2 CSI – The issue is a concern in Canada. However, the licensees have appropriate control measures in place to address the issue and to maintain safety margins; Category 3 CSI – The issue is a concern in Canada. Measures are in place to maintain safety margins, but further experiments and/or analyses are required to improve knowledge and understanding of the issue, and to confirm the adequacy of the measures.

operation and accident conditions. However, the impact of HTS component aging on the SBLOCA, LOF and Slow LOR accident scenarios, will need to be further assessed in order to demonstrate adequate safety margins exist beyond 2020, and therefore a gap exists for Pickering PSR2 (**Pickering PSR2 Gap SF5-1**) (note, a related gap has been captured in the Aging Safety Factor Report (PSR2 Gap SF4-18)). It is noted in Reference [60], that work is currently underway to perform Safety Analysis to support the initiative to extend Pickering commercial operation to 2024, accounting for possible mitigation strategies of life-limiting aging mechanisms. Also, since four CSIs applicable to Pickering NGS (3 LBLOCA / 1 non-LBLOCA) are currently in Category 3 and are undergoing further assessment in order to re-classify into a lower category and address operation past 2020, a gap exists for Pickering PSR2 (**Pickering PSR2 Gap SF5-2**) (Note, the 3 LBLOCAs CSIs are also captured as a gap in the PSR2 COP Report (PSR2 Gap COP-20) [64] as they relate to Pickering B IIP Item I09. The 1 non-LBLOCA CSI is also identified as a gap in the Hazards Analysis Safety Factor Report (PSR2 Gap SF7-1) [65] as it relates to pipe whip).

4.2 L/R/C/S Reviews

As per Section 2.2 of this report, detailed reviews for ten L/R/C/Ss with content applicable to Safety Factor 5 are provided in References [6], [7] and [8]. Associated findings applicable to Safety Factor 5 are summarized in Table 3 below.

Table 3: PSR2 L/R/C/S Review Results for Safety Factor 5

L/C/R/S Reviewed	PSR2 L/R/C/S Review Results for Safety Factor 5
CSA N286-12, "Management Systems Requirements for Nuclear Facilities"	There are no PSR2 gaps for CSA N286-12. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with N286-12.
CSA N290.15-10 (R2015), "Requirements for the Safe Operating Envelope of Nuclear Power Plants"	There are no PSR2 gaps for CSA N290.15-10 (R2015). Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CSA N290.15-10 (R2015).
CSA N286.7-16, "Quality Assurance of Analytical, Scientific and Design Computer Programs"	There are no PSR2 gaps for CSA N286.7-16. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CSA N286.7-16.
CNSC REGDOC-2.4.1 (2014), "Deterministic Safety Analysis"	There are two PSR2 CNSC REGDOC-2.4.1 (2014) gaps which relate to Safety Factor 5: 1. The REGDOC-2.4.1 Implementation Plan and associated gap assessments capture all gaps related to REGDOC-2.4.1 and incorporate a systematic selection of the scope of work to address the most pertinent gaps in accordance with the graded approach to upgrading existing analyses. REGDOC-2.4.1 compliant analysis activities and progress related to REGDOC-2.4.1 implementation in the Pickering Licence Conditions Handbook are tracked according to the CNSC Compliance Verification Criteria. Since the implementation is in progress, this has been identified

L/C/R/S Reviewed	PSR2 L/R/C/S Review Results for Safety Factor 5
	as a PSR2 gap for Pickering NGS REGDOC-2.4.1 compliance (**Pickering PSR2 Gap SF5-3**). 2. As described in the REGDOC-2.4.1 Implementation Plan: "Limited upgrades are proposed in the Pickering A and B Plan, which has been developed with consideration for demonstration of continued safe operation while accounting for the limited remaining operating life of the Pickering Units". The REGDOC-2.4.1 Implementation Plan for Pickering did not consider operation past 2020 and therefore the need for review and update of the Implementation Plan in the context of operation of Pickering NGS beyond 2020 is identified as a PSR2 gap (**Pickering PSR2 Gap SF5-4**). This will be informed by the timeline of the Darlington REGDOC-2.4.1 Implementation Plan, and the limited additional years of Pickering NGS operation.
CNSC REGDOC-2.5.2 (2014), "Design of Reactor Facilities: Nuclear Power Plants"	There is one PSR2 CNSC REGDOC-2.5.2 (2014) gap which relates to Safety Factor 5: 1. Clauses 4.2.1, 6.4 and 7.3 of REGDOC-2.5.2 introduce new requirements and limits for AOOs, DBAs and BDBAs and include specific dose limits for AOOs and DBAs. Current Pickering Safety Report analyses do not identify and classify events into these categories. Dose limits currently used in Pickering are aligned with the single failure / dual failure limits in accordance with the Pickering Licence Conditions Handbook. This issue has therefore been identified as a PSR2 gap (**Pickering PSR2 Gap SF5-5**). It is being addressed as part of REGDOC-2.4.1 implementation.
CNSC REGDOC-2.3.2 (2015), "Accident Management, Version 2"	For Safety Factor 5, there are no PSR2 gaps for CNSC REGDOC-2.3.2 (2015).
CSA N286.7.1-09, "Guideline for the Application of N286.7-99, Quality Assurance of Analytical, Scientific, and Design Computer Programs for Nuclear Power Plants"	The N286.7.1 guide has been amalgamated into the new (-16) edition of the N286.7 Standard. The N286.7 CSA Impact Statement [9] states: "The CSA N286.7.1 guide will no longer be maintained after this new edition of N286.7 is issued. Any relevant guidance has been put into the new edition of N286.7." As a result, only the review of N286.7-16 has been prepared for PSR2.
CNSC G-144 (2006), "Trip Parameter Acceptance Criteria for the Safety Analysis of CANDU Nuclear Power Plants"	There are no PSR2 gaps for CNSC G-144 (2006). Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with G-144 (2006).
CNSC G-149 (2000), "Computer Programs Used in Design and Safety Analyses of Nuclear Power Plants and Research Reactors"	There are no PSR2 gaps for CNSC G-149 (2000). Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with G-149 (2000).

L/C/R/S Reviewed	PSR2 L/R/C/S Review Results for Safety Factor 5
CSA N288.2-14, "Guidelines for Calculating Radiological Consequences to the Public from a Release of Airborne Radioactive Material for Nuclear Reactor Accidents"	There is one PSR2 CSA N288.2-14 gap which relates to Safety Factor 5: 1. Safety Report upgrades currently underway for Pickering as part of REGDOC-2.4.1 implementation for the period of 2017-2021 will utilize methods consistent with N288.2-14. The REGDOC-2.4.1 Implementation Plan update will consider the incremental implications of Pickering operation beyond 2020, including any considerations of N288.2 revisions. This issue has therefore been identified as a PSR2 gap (**Pickering PSR2 Gap SF5-6**). It is being addressed as part of REGDOC-2.4.1 implementation.

4.3 OPG Program Effectiveness Reviews

The OPG Nuclear Program reviewed for Safety Factor 5 is identified in Table 2, and details of the associated effectiveness review for this N-PROG are provided in Appendix B.

4.4 Additional Review Findings

As discussed in Section 3.4, the PSR2 Safety Factor 5 assessment also included a review of commitments previously made to the CNSC, open CNSC action items, and exemptions granted by the CNSC, as identified in the R04 Pickering LCH [4], to determine if there are any impacts associated with operation of the Pickering Units past 2020. The review also included identification and review of previously identified programmatic Darlington PSR1 gaps related to Safety Factor 5 to determine impacts associated with operation of the Pickering Units past 2020.

Review of the Darlington IIP [13] for gaps that may be applicable in the context of Pickering PSR2 for operation past 2020, identified the following:

- **Gap SF5-7** – Darlington Gap IIP-OI 055 was related to use of the best estimate approach for analysis of operational events at Darlington NGS. The action for Darlington (AR 28175247, Target Completion Date Q1 2020) was to revise OPG governing document N-MAN-03600-10005, "Nuclear Safety Analysis", to require the use of the best estimate approach or a similarly conservative approach for analysis of operational events. This action is also applicable for Pickering NGS and is therefore a gap for Pickering PSR2.

Exemptions and concessions listed in the LCH [4] were reviewed to determine applicability to PSR2. There are no CNSC exemptions and concessions from the LCH that are applicable to Safety Factor 5.

Findings from the review of previously identified PSR1 gaps in the Pickering Units 5-8 Continued Operations Plan [11] are provided in a separate PSR2 COP Review Report. Findings from the review of Fukushima Action Items are provided in a separate PSR2

FAI Review Report. Results from the Continued Operations Plan and Fukushima Action Items reviews will be considered in the Global Assessment process.

There were two PSR2 gaps identified in this Safety Factor 5 report that are already discussed in other Safety Factor Reports. PSR2 Gap SF5-1 identifies that the impact of HTS component aging on the SBLOCA, LOF and Slow LOR accident scenarios will need to be further assessed to demonstrate adequate safety margins beyond 2020. This gap is also identified as part of PSR2 Gap SF4-18 in the Aging Safety Factor Report as the work also relates to aging. In addition, there is one non-LBLOCA CSI captured in PSR2 Gap SF5-2 which relates to pipe whip. Since this CSI is related to hazard analysis, it is also identified as PSR2 Gap SF7-1 in the Hazard Analysis Safety Factor Report.

5.0 RESULTS AND CONCLUSIONS

OPG Governance, Programs, Policies, Procedures, Instructions and Guidelines related to Safety Factor 5 were reviewed for the seven PSR2 Review Tasks in Section 4.1 of this report and resulted in Pickering PSR2 Gaps SF5-1 and SF5-2 below. L/R/C/S and OPG Nuclear Program effectiveness reviews for Safety Factor 5 were prepared per Sections 4.2 and 4.3, respectively, and resulted in PSR2 Gaps SF5-3 to SF5-6 below. Per Section 4.4, this report also included identification and review of previously identified programmatic Darlington PSR1 gaps related to Safety Factor 5 (to ascertain the implications of extending Pickering NGS operation beyond 2020), as well as a review of the R04 Pickering LCH [4] for any impacts of Pickering NGS operation beyond 2020 on: a) OPG commitments previously made to the CNSC, b) open CNSC action items, and c) exemptions granted by the CNSC (all related to Safety Factor 5), which resulted in PSR2 Gap SF5-7.

The seven PSR2 gaps that will need to be addressed as part of Pickering PSR2 are:

- **Gap SF5-1**: Per Review Task #7, the current safety analysis for Pickering 1,4 and 5-8, demonstrates adequate shutdown system trip coverage until 2017 and 2018 respectively for the Small Break Loss of Coolant Accident (SBLOCA) and Loss of Flow (LOF) scenarios. However, the impact of Heat Transport System (HTS) component aging on the SBLOCA, LOF and Slow Loss of Regulation (LOR) accident scenarios will need to be further assessed in order to demonstrate adequate safety margins exist beyond 2020, and therefore a gap exists for Pickering PSR2. It is noted in Reference [60] that work is currently underway to perform Safety Analysis to support the initiative to extend Pickering commercial operation to 2024, accounting for possible mitigation strategies of life-limiting aging mechanisms. Note, a related gap has been captured in the Aging Safety Factor Report (PSR2 Gap SF4-18).

- **Gap SF5-2:** Per Review Task #7, for the Large Break Loss of Coolant Accident (LBLOCA) CANDU Safety Issues (CSIs), while the development of the industry's proposed Composite Analytical Approach (CAA) is on-going, the licensing basis of existing CANDU reactors for the LBLOCA scenario will continue to be based on conservative safety analysis for which acceptance criteria are established. For the Category 3 non-LBLOCA CSI, the industry has applied to re-categorize the issue into a lower category based on analytical evidence and actions taken. Since four CSIs applicable to Pickering NGS (3 LBLOCA / 1 non-LBLOCA) are currently in Category 3 and are undergoing further assessment in order to re-classify into a lower category and address operation past 2020, a gap exists for Pickering PSR2. Note, the 3 LBLOCAs CSIs are also captured as a gap in the PSR2 Continued Operations Plan (COP) Report (PSR2 Gap COP-20) as they relate to Pickering B Integrated Implementation Plan (IIP) Item I09. The 1 non-LBLOCA CSI is also identified as a gap in the Hazards Analysis Safety Factor Report (PSR2 Gap SF7-1) as it relates to pipe whip.

- **<u>Gap SF5-3</u>:** The REGDOC-2.4.1 Implementation Plan and associated gap assessments capture all gaps related to REGDOC-2.4.1 and incorporate a systematic selection of the scope of work to address the most pertinent gaps in accordance with the graded approach to upgrading existing analyses. REGDOC-2.4.1 compliant analysis activities and progress related to REGDOC-2.4.1 implementation in the Pickering Licence Conditions Handbook are tracked according to the CNSC Compliance Verification Criteria. Since the implementation is in progress, this has been identified as a PSR2 gap for Pickering NGS REGDOC-2.4.1 compliance.

- **<u>Gap SF5-4</u>:** As described in the REGDOC-2.4.1 Implementation Plan: "Limited upgrades are proposed in the Pickering A and B Plan, which has been developed with consideration for demonstration of continued safe operation while accounting for the limited remaining operating life of the Pickering Units". The REGDOC-2.4.1 Implementation Plan for Pickering did not consider operation past 2020 and therefore the need for review and update of the Implementation Plan in the context of operation of Pickering NGS beyond 2020 is identified as a PSR2 gap. This will be informed by the timeline of the Darlington REGDOC-2.4.1 Implementation Plan, and the limited additional years of Pickering NGS operation.

- **<u>Gap SF5-5</u>:** Clauses 4.2.1, 6.4 and 7.3 of REGDOC-2.5.2 introduce new requirements and limits for Anticipated Operational Occurrences (AOOs), Design Basis Accidents (DBAs) and Beyond Design Basis Accidents (BDBAs) and include specific dose limits for AOOs and DBAs. Current Pickering Safety Report analyses do not identify and classify events into these categories. Dose limits currently used in Pickering are aligned with the single failure / dual failure limits in accordance with the Pickering Licence Conditions Handbook. This issue has therefore been identified as a PSR2 gap. It is being addressed as part of REGDOC-2.4.1 implementation.

- **<u>Gap SF5-6</u>:** Safety Report upgrades currently underway for Pickering as part of REGDOC-2.4.1 implementation for the period of 2017-2021 will utilize methods consistent with N288.2-14. The REGDOC-2.4.1 Implementation Plan update will consider the incremental implications of Pickering operation beyond 2020, including any considerations of N288.2 revisions. This issue has therefore been identified as a PSR2 gap. It is being addressed as part of REGDOC-2.4.1 implementation.

- **<u>Gap SF5-7</u>**: The Darlington Integrated Implementation Plan (IIP) [13] identified a gap (IIP-OI 055) related to use of the best estimate approach for analysis of operational events at Darlington NGS. The action for Darlington (AR 28175247, Target Completion Date Q1 2020) was to revise OPG governing document N-MAN-03600-10005, "Nuclear Safety Analysis", to require the use of the best estimate approach or a similarly conservative approach for analysis of operational events. This action is also applicable for Pickering NGS and is therefore a gap for Pickering PSR2.

The review of Safety Factor 5 has confirmed that the deterministic safety analysis programs and procedures at OPG are comprehensive, resulting in a systematic and disciplined approach to identifying, prioritizing and addressing any safety analysis related issues.

6.0 REFERENCES

[1] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[2] CNSC REGDOC-2.3.3, *Periodic Safety Reviews*, April 2015.

[3] IAEA Safety Guide No. SSG-25, *Periodic Safety Review for Nuclear Power Plants*, 2013.

[4] CNSC Report, LCH-PNGS-R004, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[5] OPG Report P-REP-03680-00003 R000, *Pickering NGS PSR2: Definition of Safety Factor Review Tasks,* May 2016.

[6] OPG Report, P-REP-03680-0586480 R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 9, 11 and 15*, September 2016.

[7] OPG Report, P-REP-03680-00021 R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 8, 10, 12, 13 and 14*, December 2016.

[8] OPG Report, P-REP-03680-00029 R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 1, 5, 6, and 7,* March 2017.

[9] CSA Impact Statement, *Notification of CSA N286.7 on Quality Assurance of Analytical Scientific, and Design Computer Programs; Product: New Edition; Product Designation: CSA N286.7; Previous Edition Published: 1999, Reaffirmed 2007 and 2012*, Date not provided.

[10] OPG Program, N-PROG-MP-0014 R005, *Reactor Safety Program*, September 16, 2015.

[11] OPG Plan, NK30-PLAN-00531-00001 R005, *Pickering 5-8 Continued Operations Plan*, December 2015.

[12] OPG Procedure, N-PROC-RA-0023 R018*, Fleetview Program Health and Performance Reporting*, August 2013.

[13] OPG Report, NK38-REP-03680-10185 R002, *Darlington NGS Integrated Implementation Plan (IIP)*, April 2015.

[14] CNSC REGDOC-2.4.1, *Deterministic Safety Analysis,* May 2014.

[15] OPG Report, NA44-SR-01320-00001 R015, *Pickering A Safety Report – Part I and Part II,* July 24, 2012.

[16] OPG Report, NK30-SR-01320-00001 R004, *Pickering B Safety Report – Part 1,* October 10, 2012.

[17] OPG Report, NK30-SR-01320-00002 R004, *Pickering B Safety Report – Part 2,* October 10, 2012.

[18] OPG Report, NA44-SR-01320-00002 R004, *Pickering Nuclear 1-4 Safety Report: Part 3 – Accident Analysis,* October 31, 2013.

[19] OPG Report, NK30-SR-01320-00003 R004, *Pickering Nuclear 5-8 Safety Report: Part 3 – Accident Analysis,* October 30, 2014.

[20] OPG Procedure, N-PROC-MP-0086 R004, *Safety Analysis Basis and Safety Report Update,* December 12, 2014.

[21] OPG Instruction, N-INS-09000-10004 R001, *Guidelines for the Control of the Analysis of Record,* January 5, 2012.

[22] OPG Report, NA44-REP-00531.7-10001 R032, *Pickering A Analysis of Record,* November 20, 2015.

[23] OPG Report, NK30-REP-00531.7-00001 R032, *Pickering B Analysis of Record,* November 20, 2015.

[24] CNSC REGDOC-3.1.1 version 2, *Reporting Requirements for Nuclear Power Plants,* April 2016.

[25] OPG Procedure, N-PROC-RA-0094 R006, *Discovery Issue Resolution Process,* June 17, 2015.

[26] OPG Instruction, N-INS-09000-10001 R001, *Processing of Safety Report Analysis Issues: Overview,* January 25, 2012.

[27] OPG Instruction, N-INS-09000-10002 R001, *Guidelines for Evaluating and Prioritizing Safety Report Analysis Issues,* May 6, 2012.

[28] OPG Instruction, N-INS-09000-10005 R001, *Safety Report Issue Database Management,* February 9, 2012.

[29] OPG Letter, N-CORR-00531-07409, *OPG Safety Analysis Improvement and REGDOC-2.4.1 Implementation – Action Item 2014OPG-5461*, October 13, 2015.

[30] OPG Letter, P-CORR-00531-03719, *Application for Renewal of Pickering Nuclear Generating Station Power Reactor Operating Licence,* July 4, 2012.

[31] OPG Standard, N-STD-MP-0016 R002, *Safe Operating Envelope,* June 21, 2012.

[32] OPG Standard, N-ST-08131.02-10000 R02, *Preparation of Operational Safety Requirements,* December 8, 2014.

[33] OPG Standard, N-STI-03602-10000 R001, *SOE Instrument Uncertainty and Allowable Value Calculations,* September 27, 2011.

[34] OPG Instruction, N-INS-03602-10001 R001, *Preparation of Safe Operating Envelope Compliance Tables,* February 9, 2015.

[35] OPG Procedure, N-PROC-AS-0028 R017, *Development, Review and Approval of Technical Procedures,* July 20, 2015.

[36] OPG Program, N-PROG-MP-0001 R014, *Engineering Change Control,* January 13, 2015.

[37] OPG Program, N-PROG-RA-0003 R010, *Corrective Action,* January 9, 2015.

[38] OPG Procedure, N-PROC-MP-0045 R008, *Technical Operability Evaluation,* September 14, 2015.

[39] CNSC Information, AECB-1059, *Reactor Licensing and Safety Requirements (Siting Guide), Paper 72-CAN-102 (Presented at the 12th Annual Conference of the Canadian Nuclear Association),* June 11, 1972.

[40] CANDU Owners Group COG-09-9030 R03, *Principles and Guidelines for Deterministic Safety Analysis,* November 2014.

[41] OPG Manual, N-MAN-03600-10002 R004, *Nuclear Safety Analysis Planning,* March 3, 2015.

[42] OPG Manual, N-MAN-03600-10003 R004, *Nuclear Safety Analysis Execution,* March 31, 2015.

[43] OPG Standard, N-STD-MP-0008 R004, *Development, Qualification and Use of Scientific, Engineering, and Safety Analysis Software,* September 30, 2013.

[44] OPG Manual, N-MAN-03600-10005 R005, *Nuclear Safety Analysis,* March 31, 2015.

[45] OPG Standard, N-STD-MP-0014 R004, *Managing Contracted Nuclear Safety Services,* November 05, 2014.

[46] OPG Standard, N-STD-MP-0019 R001, *Beyond Design Basis Accident Management,* September 23, 2014.

[47] OPG Guideline, N-GUID-01130-10000 R01, *Modifications for Beyond Design Basis Accidents,* February 6, 2015.

[48] CNSC REGDOC-2.5.2, *Design of Reactor Facilities: Nuclear Power Plants,* May 2014.

[49] OPG Report, P-REP-03490-10012 R01, *Fukushima Daiichi – Total Loss of Heat Sink Assessment for Pickering A and Pickering B,* June 14, 2013.

[50] OPG Manual, N-BDB-03600-00002 R00, *OPG Emergency Mitigating Equipment for Beyond Design Basis Accidents: Technical Basis Document,* October 13, 2015.

[51] OPG Guideline, NA44-GUID-03600-00001 R00, *Pickering 1-4 Beyond Design Basis Functional Safety Requirements,* October 29, 2014.

[52] OPG Guideline, NK30-GUID-03600-00001 R00, *Pickering 5-8 Beyond Design Basis Functional Safety Requirements,* October 31, 2014.

[53] OPG Report, NA44-REP-03611-00031 R00, *Pickering NGS A Level 1 At-Power Risk Assessment (PARA-L1P) – Fukushima Action Item Update,* February 2014.

[54] OPG Report, NK30-REP-03611-00025 R00, *Pickering NGS B Level 1 At-Power Internal Events Risk Assessment – Fukushima Action Item Update,* February 2014.

[55] OPG Report, NA44-REP-03611-00032 R00, *Pickering NGS A Level 2 At-Power Internal Events Risk Assessment (PARA-L2P) - Fukushima Action Item Update,* February 2014.

[56] OPG Letter, N-CORR-00531-06704, *OPG Progress Report No. 6 on CNSC Action Plan – Fukushima Action Items,* October 31, 2014.

[57] OPG Report, P-REP-03680-00018 R000, *Pickering NGS PSR2 Safety Factor 14 Report: Radiological Impact on the Environment,* December 12, 2016.

[58] OPG Standard, N-STD-MP-0020 R004, *Margin Management,* June 22, 2016.

[59] OPG Letter, N-CORR-00531-17979 R00, *Darlington and Pickering NGS: Enhanced Neutron Overpower Protection (E-NOP) Extreme Value Statistics (EVS) Methodology, Closure of Action Item 2009OPG-06 – New Action Item 2016-OPG-7349,* January 21, 2016.

[60] OPG Letter, N-CORR-00531-16444, *Progress Report on OPG Heat Transport System Aging Safety Analysis,* February 23, 2016.

[61] OPG Letter, N-CORR-00531-18045, *Darlington and Pickering NGS: Derived Acceptance Criteria for Deterministic Safety Analysis of Slow Events and Re-categorization of the CANDU Safety Issue PF18 Fuel Bundle/Element Behaviour under Post Dryout,* April 21, 2016.

[62] CNSC Report, *Regulatory Oversight Report for Canadian Nuclear Power Plants: 2014,* September 2015.

[63] OPG Letter, N-CORR-00531-18052, *Progress Update on Category 3 CANDU Safety Issues – Implementation of Risk Control Measures,* June 15, 2016.

[64] OPG Report, P-REP-03680-00024 R000, *Pickering 5-8 Continued Operations Plan Review in Support of PNGS Periodic Safety Review 2 (PSR2),* January 12, 2017.

[65] OPG Report, P-REP-03680-00011 R000, *Pickering NGS PSR2 Safety Factor 7 Report: Hazard Analysis,* March 2017.

Appendix A: Nomenclature

AOO	Anticipated Operational Occurrences
AoR	Analysis of Record
BDBA	Beyond Design Basis Accident
CAA	Composite Analytical Approach
CANDU	CANada Deuterium Uranium
CNSC	Canadian Nuclear Safety Commission
COG	CANDU Owners Group
COP	Continued Operations Plan
CSI	CANDU Safety Issue
DBA	Design Basis Accident
DCR	Document Change Request
DEC	Design Extension Condition
DIRP	Discovery Issue Resolution Process
DRL	Derived Release Limit
EFPD	Effective Full Power Day
EMEG	Emergency Mitigating Equipment Guidelines
FAI	Fukushima Action Item
HTS	Heat Transport System
IAEA	International Atomic Energy Agency
IIP	Integrated Implementation Plan
ISR	Integrated Safety Review
IUC	Instrument Uncertainty Calculation
LCH	Licence Conditions Handbook
LBLOCA	Large Break Loss of Coolant Accident
LOF	Loss of Flow
LOR	Loss of Regulation
NGS	Nuclear Generating Station
NOP	Neutron Overpower Protection
OPG	Ontario Power Generation
OSR	Operational Safety Requirements
PARTS	Pickering A Return to Service

PIE	Postulated Initiating Event
PROL	Power Reactor Operating Licence
PSA	Probabilistic Safety Assessment
PSR	Periodic Safety Review
PSR1	Periodic Safety Review 1 (earlier OPG PSR work and other associated assessments)
PSR2	Periodic Safety Review 2 (subsequent PSR per CNSC REGDOC-2.3.3)
QA	Quality Assurance
SA	Severe Accident
SAM	Severe Accident Management
SAMG	Severe Accident Management Guidelines
SBLOCA	Small Break Loss of Coolant Accident
SCR	Station Condition Record
SOE	Safe Operating Envelope
SRU	Safety Report Update
SSC	Structures, Systems and Components
TOE	Technical Operability Evaluation

Appendix B: OPG Program Effectiveness Review Results

B.1 N-PROG-MP-0014, "Reactor Safety Program"

The purpose of the Reactor Safety Program is to define organizational responsibilities and key program elements for the management of issues related to Nuclear Safety Analysis and the following major components of safe operation:

- Safety Analysis Basis;
- Safe Operating Envelope (SOE); and
- Severe Accident Management (SAM).

The Safety Analysis Basis includes nuclear safety assessments performed to ensure safe operation, in particular the Design Basis Accident (DBA) analyses contained in the Safety Report. SOE is defined by the Operational Safety Requirements (OSR), Instrument Uncertainty Calculations (IUC) and other safety related limits and system credits that ensure operation within the Safety Analysis Basis. SAM examines Beyond Design Basis Accidents (BDBAs). SAM guidelines (SAMGs) provide a framework for responding to BDBAs, in order to manage residual risk.

In June 2015, Nuclear Oversight conducted a performance based audit at Pickering and Darlington NGS, NO-2015-021 [B.1.1], to determine whether the Reactor Safety Program activities were being performed effectively and in compliance with program requirements for safe and reliable operations. The audit concluded that the managed system controls were effective. There were opportunities for improvement in the areas of nuclear safety assessment quality assurance, nuclear safety services contracting and the safety report update process.

Three SCRs were initiated to address the above findings, which required corrective actions to be implemented. Two SCRs have been completed (N-2015-10848 and N-2015-10867), while SCR N-2015-10785 (AR 28177966) has actions in place and is expected to be completed by Q2 2017.

The Reactor Safety department completed a self-assessment in April 2014, P14-000401-SA [B.1.2], in order to assess the preparation phase of the P1441 Pickering NGS outage against OPG outage management governance. Minor recommendations were generated; however, no findings/SCRs were initiated as a result of this self-assessment.

The Governance and Services Section at OPG completed a self-assessment in November 2015, BAS15-001614-SA [B.1.3], in order to assess the governance structure of N-PROG-MP-0014, "Reactor Safety Program", which is applicable for both Pickering and Darlington NGS. Minor recommendations were generated; however no findings/SCRs were initiated as a result of this self-assessment.

References

[B.1.1] Nuclear Oversight Audit, N-REP-01070-0542955 (NO-2015-021), *Nuclear Oversight Report – Reactor Safety Program*, June 10, 2015.

[B.1.2] Self-Assessment, P14-000401-SA, *Self-Assessment Report – Pre-Outage P1441 from Reactor Safety Perspective,* April 24, 2014.

[B.1.3] Self-Assessment, BAS15-001614-SA, *Self-Assessment Report –Program Assessment, N-PROG-MP-0014,* November 30, 2015.

ONTARIOPOWER GENERATION	
ACCEPTED	✓
ACCEPTED AS NOTED	
REVISE AND RESUBMIT	
Signature	03 MAR 2017 **Date**
Name: Mike Ruffolo, Manager	
Dept: Pickering Engineering - Aging Management & Strategic Initiatives	
OPG Proprietary	
Doc No.: P-REP-03680-00010	**Rev: 000**
This acceptance does not relieve the contractor from responsibility for errors or omissions or from any obligations or liability under this contract.	

Pickering NGS PSR2 Safety Factor 6 Report: Probabilistic Safety Assessment

PS112/RP/012 R01

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Revision Summary for Amec Foster Wheeler Report PS112/RP/012

Rev	Date	Author	Comments
R00	July 7, 2016	Brandon McLean / Ranil Jayasundera	Initial issue for OPG review and comment.
R01	March 2, 2017	Brandon McLean, Ranil Jayasundera, Ben Hryciw	Updated report addressing OPG comments on R00 Report.

EXECUTIVE SUMMARY

Ontario Power Generation (OPG) is performing a Periodic Safety Review (PSR) to support the possibility of continued operation of Pickering Nuclear Generating Station (NGS) beyond 2020. The PSR (referred to as "PSR2") is a subsequent PSR building on the review basis of earlier OPG Integrated Safety Reviews and other associated assessments. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

Part of PSR2 involves the preparation of Safety Factor reports for each of fifteen major topic areas. Safety Factor reports consist of:

- Assessing compliance against "Review Tasks" identified in the PSR2 Basis Document [1]. These Review Tasks are derived from Canadian Nuclear Safety Commission (CNSC) REGDOC-2.3.3, "Periodic Safety Reviews" [2] and International Atomic Energy Agency (IAEA) SSG-25, "Periodic Safety Review for Nuclear Power Plants" [3];

- Documenting assessments against applicable modern Laws, Regulations, Codes and Standards (L/R/C/Ss) as defined in Reference [1]; and

- Reviewing the effectiveness of applicable OPG Nuclear Programs that support the above assessments, through review of audit and self-assessment results.

The PSR2 review of Safety Factor 6, *Probabilistic Safety Assessment (PSA)*, is presented in this report. OPG Governance, Programs, Policies, Procedures, Instructions and Guidelines related to Safety Factor 6 were reviewed for the six PSR2 Review Tasks specified in Section 4.1 of this report. L/R/C/S and OPG Nuclear Program effectiveness reviews for Safety Factor 6 were prepared per Sections 4.2 and 4.3, respectively. Per Section 4.4, the PSR2 assessment includes a review of previously identified PSR1 gaps related to Safety Factor 6 (to ascertain the implications of extending Pickering NGS operation beyond 2020), as well as a review of the R04 Pickering Licence Conditions Handbook [4] for any impacts of Pickering NGS operation beyond 2020 on: a) OPG commitments previously made to the CNSC, b) open CNSC action items, and c) exemptions granted by the CNSC (all related to Safety Factor 6).

The results of the review of Safety Factor 6 are discussed in Section 5.0. The review of Safety Factor 6 has confirmed that the PSA programs and procedures at OPG are comprehensive, resulting in a systematic and disciplined approach to identifying, prioritizing and addressing any PSA related issues. As discussed in Section 5.0, the review identified five gaps that will need to be addressed further as part of the PSR2 Global Assessment process.

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1.0 INTRODUCTION

Ontario Power Generation (OPG) is performing a Periodic Safety Review (PSR) to support the possibility of continued operation of Pickering Nuclear Generating Station (NGS) beyond 2020.[1] A comprehensive Integrated Safety Review (ISR) was completed for Pickering Units 5 through 8 in 2009 in support of refurbishment and continued operation. Pickering Units 1,4 integrated safety assessments were also performed for Pickering A Return to Service (PARTS) in support of approval to restart Units 1 and 4. In addition to these Pickering-specific studies, the 2013 Darlington ISR performed extensive code and standard reviews that were updated in relation to the versions that were assessed in the 2009 Pickering B ISR.[2] These previous ISRs are considered to constitute the first PSR completed for Pickering (referred to as "PSR1"). The current PSR (referred to as "PSR2") is a subsequent PSR building on the basis of earlier OPG integrated safety assessments through review of the various studies, assessments and licence renewals performed since PSR1. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

PSR2 will support and complement the licence renewal application for Pickering NGS going forward. Fifteen Safety Factors will be assessed as part of the PSR. The purpose of Safety Factor reviews is to confirm that the design, condition and operation of Pickering Units 1,4 and 5-8 (as well as common systems) will support continued safe operation for the period of PSR2 by:

- Assessing compliance against "Review Tasks" identified in the PSR2 Basis Document [1], which were derived from Canadian Nuclear Safety Commission (CNSC) REGDOC-2.3.3 [2] and International Atomic Energy Agency (IAEA) SSG-25 [3];
- Documenting assessments against applicable modern Laws, Regulations, Codes and Standards (L/R/C/Ss) (as defined in Reference [1]); and
- Reviewing the effectiveness of applicable OPG Nuclear Programs that support the above assessments, through review of audit and self-assessment results.

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1 Currently, Pickering Units 5-8 are approved to operate to 247,000 Effective Full Power Hours. This operation limit is expected to be reached on some units in 2020. For the purposes of PSR2, OPG assumes operation of Pickering NGS for up to eight additional years, from 2020 until 2028. OPG will make a decision regarding the permanent shut down dates for the six reactors following the performance of a technical evaluation that will include PSR2, and will communicate it to the CNSC as required by the current Power Reactor Operating Licence (PROL).

2 Much of the compliance assessment and evaluation of Safety Factor health for the Darlington ISR is based on programs and practices that apply across OPG's Nuclear operations. As a result, where Pickering is confirmed to follow the same Nuclear programs and practices as were assessed for Darlington, the Darlington ISR programmatic conclusions are applicable to Pickering. As discussed in Section 1.0, an effectiveness review (effectiveness at Pickering NGS) of OPG programs used to demonstrate compliance with the PSR2 Assessment Basis will be conducted using recent audit and self-assessment results.

As outlined in IAEA SSG-25 [3], the objective of the review of Probabilistic Safety Assessment (PSA) Safety Factor 6 is to determine:

- The extent to which the existing PSA study remains valid as a representative model of the plant;
- Whether the results of the PSA show that the risks are sufficiently low and well balanced for all postulated initiating events and operational states;
- Whether the scope (which should include all operational states and identified internal and external hazards), methodologies and extent (i.e., Level 1, 2 or 3) of the PSA are in accordance with current national and international standards and good practices;
- Whether the existing scope and application of PSA are sufficient.

REGDOC-2.3.3 [2] requires that: "The licensee shall conduct a PSR in accordance with this regulatory document for the period until the next PSR or, if applicable, until the end of commercial operation of the plant."

This report documents the results of the review of Safety Factor 6 for Pickering PSR2. The report is based on the OPG Governance, Programs, data, and material available up to January 15, 2016 which is the freeze date for PSR2.

2.0 SCOPE OF REVIEW

2.1 Review Task Assessments

The Pickering PSR2 Safety Factor 6 Review Tasks are defined in Reference [1]. Details of the derivation of these Review Tasks from CNSC REGDOC-2.3.3 [2] and IAEA SSG-25 [3] are shown in Reference [5]. The Safety Factor 6 Review Tasks are:

1) Confirm existence of a PSA and the assumptions used, the fault schedule, the representations of operator actions and common cause events, the modelled plant configuration and consistency with other aspects of the safety case.

2) Confirm existence of processes to assess the impact of changes in plant design, operation, and plant specific failure data and update the PSA to reflect the current plant status as required.

3) Confirm there are guidelines to account for operator actions, common cause events, cross-link effects, redundancy, and diversity.

4) Confirm that the accident management programs for accident conditions (design basis accident conditions and design extension conditions) are consistent with PSA models and results.

5) Confirm that the results of the PSA show that risks are sufficiently low and well balanced for all postulated initiating events and operational states, and meet relevant probabilistic safety criteria.

6) Review the extent to which hazards are represented in the PSA to verify that omissions are based on site specific justifications and that these omissions do not weaken the overall risk assessment for the plant.

The methodology for the reviews is discussed in Section 3.1. Review Task findings are summarized in Section 4.1 of this Report.

2.2 L/R/C/S Reviews

The applicable Laws, Regulations, Codes and Standards relevant to the Probabilistic Safety Assessment Safety Factor are identified in Reference [1] and are listed in Table 1 below. Table 1 also identifies the modern version and date of each L/R/C/S to be considered, the Safety Factor(s) to which each document is applicable, and the type of review that will be completed in PSR2.

All of the Safety Factor 6 L/R/C/S reviews are incremental in nature. The definition of an Incremental Review is as follows:

- <u>Incremental Review</u>: For L/R/C/Ss that have been reviewed in PSR1 but have had revisions since the last review, a topical review will be performed of the changes.

The methodology for the reviews is discussed in Section 3.2. A detailed assessment for each L/R/C/S is provided in References [6], [7] and [8]. Associated findings are summarized in Section 4.2 of this report.

Table 1: L/R/C/Ss Reviewed for the Probabilistic Safety Assessment Safety Factor 6

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
L/R/C/Ss Referenced in Pickering NGS PROL 48.02/2018						
1	CSA N286	Management System Requirements for Nuclear Facilities	N286-12	5, 6, 9, 10, 11	Incremental	N286 addressed as part of Pickering B and Darlington ISRs.
2	CSA N286.7	Quality Assurance Of Analytical, Scientific And Design Computer Programs	N286.7-16	1, 5, 6, 7, 10	Incremental	N286.7 addressed as part of Pickering B and Darlington ISRs.
3	CNSC REGDOC-2.4.2	Probabilistic Safety Assessment (PSA) for Nuclear Power Plants	2014	6, 7	Incremental	S-294 addressed as part of Pickering B and Darlington ISRs. Implementation plan in place.
Additional L/R/C/Ss						
4	CNSC REGDOC-2.5.2	Design of Reactor Facilities: Nuclear Power Plants	2014	1, 5, 6, 7	Incremental	RD-337 and NS-R-1 (precursors to REGDOC-2.5.2) addressed as part of Darlington ISR. NS-R-1 also addressed as part of Pickering B ISR.
5	CNSC REGDOC-2.3.2	Accident Management, Version 2	2015	1, 5, 6, 7, 8, 10	Incremental	REGDOC-2.3.2 addressed as part of Darlington ISR.
6	CSA N286.7.1	Guideline for the Application of N286.7-99, Quality Assurance of Analytical, Scientific, and Design Computer Programs for Nuclear Power Plants	N286.7.1-09	1, 5, 6, 7, 10	N/A [3]	N286.7.1 not addressed as part of Pickering B or Darlington ISRs.

[3] The N286.7.1 guide has been amalgamated into the new (-16) edition of the N286.7 Standard. The N286.7 CSA Impact Statement [9] states: "The CSA N286.7.1 guide will no longer be maintained after this new edition of N286.7 is issued. Any relevant guidance has been put into the new edition of N286.7." As a result, only the review of N286.7-16 has been prepared for PSR2.

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
7	CNSC G-149	Computer Programs Used in Design and Safety Analyses of Nuclear Power Plants and Research Reactors	2000	1, 5, 6, 7	Incremental	G-149 addressed as part of Pickering B and Darlington ISRs.

2.3 OPG Program Effectiveness Reviews

The OPG Nuclear Program (N-PROG) reviewed for Safety Factor 6 is listed in Table 2 below.[4] The methodology for the effectiveness reviews is discussed in Section 3.3. The assessment results for the N-PROG in Table 2 are provided in Appendix B, and findings are summarized in Section 4.3.

Table 2: OPG Program Reviewed for Safety Factor 6

Document Number	Document Title
N-PROG-RA-0016 [10]	Risk and Reliability Program

2.4 Additional Reviews

The PSR2 Safety Factor 6 report includes a review of the R04 Pickering Licence Conditions Handbook (LCH) [4] for any impacts of Pickering NGS operation beyond 2020 on the following (all related to Safety Factor 6):

- OPG commitments previously made to the CNSC;
- Open CNSC action items; and
- Exemptions granted by the CNSC.

The PSR2 assessment includes identification and review of previously identified PSR1 gaps related to Safety Factor 6 to ascertain the implications of extending Pickering NGS operation beyond 2020. The methodology for these reviews is described in Section 3.4. Any PSR2 gaps identified as a result of programmatic Darlington PSR1 gaps related to Safety Factor 6 are discussed in Section 4.4 of this report. The review of Pickering PSR1 gaps previously identified in the Pickering Units 5-8 Continued Operations Plan (COP) [11] is provided in a separate PSR2 COP Review Report.

[4] The list of Nuclear Programs to be assessed for effectiveness for PSR2 was derived from review of current OPG Governance. Although there may be content in Nuclear Programs that is applicable to multiple Safety Factors, N-PROG reviews are only provided in one Safety Factor report and are not duplicated.

In addition, Fukushima Action Items (FAIs) were reviewed to identify implications of extending operation beyond 2020 (if any). This review is presented in a separate PSR2 FAI Review Report.

Any PSR2 gaps identified as a result of the Safety Factor 6 review which are relevant to other Safety Factors are discussed in Section 4.4 of this report.

3.0 METHODOLOGY

The sub-sections below summarize the methodology used to assess Review Tasks, L/R/C/Ss, and Nuclear Program effectiveness for the Probabilistic Safety Assessment Safety Factor.

3.1 Review Tasks

As discussed earlier, the Safety Factor Review Tasks are derived from CNSC REGDOC-2.3.3 [2] and IAEA SSG-25 [3], taking into consideration the Review Tasks used in the Pickering B and Darlington ISRs (as derived in [5]).

For each Safety Factor 6 Review Task identified in Section 2.1, a confirmation of the existence of applicable OPG Governance, Programs, Policies and Procedures (as well as Instructions and Guidelines, as applicable) was performed. Compliance against Review Tasks is also assessed by reference to applicable Condition Assessments, safety analyses and operating experience, as required.

The Review Task assessments identify Compliances and Gaps as defined below:

- Compliance: Compliance indicates that either the safety requirement or the intent of the Review Task is met.
- Gap: A Gap indicates that the intent of the Review Task is not met.

3.2 L/R/C/S Reviews

The process to identify the modern L/R/C/Ss that are applicable to the PSR2 Assessment Basis involved first creating a broad list from multiple sources (potential candidate L/R/C/Ss) and then filtering it to identify those that are most significant and that are applicable to the PSR2 scope. The identification and selection criteria are detailed in Reference [1]. The result of the identification and selection process was a set of modern L/R/C/Ss that became part of the "PSR2 Assessment Basis".

PSR2 is focused on the extension of Pickering NGS operations beyond 2020, and will conduct reviews against a baseline of past PSR1 work. As a subsequent PSR, PSR2 focuses on changes in requirements, plant conditions, operating experience and new information. Since PSR2 is an update of previous ISRs, it incorporates reviews of L/R/C/Ss that have occurred as new versions have been issued. Since this assessment is a subsequent PSR, the focus is on identifying differences between what was previously assessed and what is now different within the current Pickering PSR2 Assessment Basis. In general, these differences relate to:

- More recent (new or revised) L/R/C/S versions than what was previously assessed as part of PSR1; [5]
- Safety significant differences between Pickering and Darlington, if the Darlington ISR is the basis for the earlier assessment;
- Implications of extending Pickering NGS operation beyond 2020; and
- Safety significant differences between Pickering Units 1,4 and Units 5-8.

As described in Reference [1], L/R/C/S review types are clause-by-clause, high level or incremental. Most of the L/R/C/Ss in the PSR2 Assessment Basis receive incremental reviews since PSR2 is an update of previous PSR1 assessments and clause-by-clause or high level reviews for the majority of the L/R/C/Ss in the PSR2 Assessment Basis have already been completed. Implementation plans (including gap analyses or code-over-code reviews) also exist for the latest editions of many L/R/C/Ss. As a result, incremental review is also used in circumstances where a L/R/C/S in the PSR2 Assessment Basis was not assessed in previous PSR1 reviews but an implementation plan currently exists for compliance.

The PSR2 incremental reviews in this report include an assessment of the intent of recent changes to the L/R/C/Ss on a topic or subject-matter basis where there is potential to impact nuclear safety. Incremental reviews provide:

- A summary of the purpose of the L/R/C/S;
- Pertinent background information about the current revision of the L/R/C/S that is being considered;
- Identification of which Safety Factor(s) are applicable to the current revision of the L/R/C/S;
- A description of which version(s) of the L/R/C/S were assessed for PSR1 (i.e., Darlington ISR (for programmatic content), Pickering B ISR and PARTS code reviews);
- Identification of whether the current version of the L/R/C/S is an update of a previous version of the L/R/C/S that was assessed in PSR1 (and if so, a description of the major changes in the latest revision is provided as discussed below);

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[5] "New" refers to a code or standard that was not previously considered in the context of earlier assessments. "Revised" refers to an updated version of a code or standard that was previously considered in the context of earlier assessments. Where a document has a new number/type, but addresses the same topic from the same organization, it is a "revised", not "new", document (e.g., if a REGDOC replaces a CNSC G or RD document).

- An assessment of the applicability of PSR1 assessment findings (gaps and conclusions), including the implications of extending Pickering NGS operation beyond 2020 if any;

- An assessment of the applicability of assessment findings that address more recent (post-PSR1) editions of the L/R/C/S, including any implementation or transition plans that are already committed to by OPG; and

- Where PSR1 and post-PSR1 assessments are not sufficient to address changes in the latest edition of the L/R/C/S, an assessment of the changes from the previously assessed edition of the L/R/C/S (including identification of any safety significant PSR2 gaps which result).

High Level reviews provide the same information as above, where applicable, in a similar format. However, given that High Level L/R/C/Ss generally have not received past assessment during PSR1, the Incremental review content is augmented by a high level, section-by-section assessment of the degree of conformance of Pickering NGS with the L/R/C/S (demonstrating, with supporting evidence, whether the intent of the requirements stipulated in the document are met).

There are currently no L/R/C/S clause-by-clause reviews identified in the PSR2 Assessment Basis.

The Safety Factor 6 L/R/C/S reviews identify Compliances and Gaps as defined below:[6]

- <u>Compliance</u>:

    - Where an Incremental review has been performed, Compliance indicates that the change in the safety requirement, per the topical review, is met.

    - Where a High Level review has been performed, Compliance indicates that the intent of the safety requirement is met. (Note: No High Level reviews were performed as part of Safety Factor 6.)

    - For Clause-by-Clause reviews of modern L/R/C/Ss, Compliance indicates that the safety requirement is met. (Note: No Clause-by-Clause reviews were performed as part of Safety Factor 6.)

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[6] Safety Factor assessments for Review Tasks and L/R/C/S reviews make use of: a) OPG Governance, Programs, Policies and Procedures which support the assessment arguments, b) Commitments previously made to the CNSC, open CNSC action items, and exemptions granted by the CNSC (all related to the Safety Factor under review), as identified in the R04 Pickering LCH [4], c) Identification of previously identified Pickering-specific or programmatic PSR1 gaps related to the Safety Factor under review and the status of OPG's improvement plan(s) or other dispositions to address these, and d) Assessments and reviews performed since the PSR1 documents were completed.

- Gap:
    - Where an Incremental review has been performed, a Gap indicates that the change in the safety requirement, per the topical review, is not met.
    - Where a High Level review has been performed, a Gap indicates that the intent of the safety requirement is not met. (Note: No High Level reviews were performed as part of Safety Factor 6.)
    - For Clause-by-Clause reviews of modern L/R/C/Ss, a Gap indicates that the safety requirement is not met. (Note: No Clause-by-Clause reviews were performed as part of Safety Factor 6.)

The reviews assume that use of the word:

- "Shall" is used in an L/R/C/S to express a requirement, i.e., a provision that the licensee is obliged to satisfy in order to comply with the standard;
- "Should" is used to express a recommendation or that which is advised but not required;
- "May" is used to express an option or that which is permissible within the limits of the standard; and
- "Can" is used to express possibility or capability.

3.3 OPG Program Effectiveness Reviews

As discussed earlier, effectiveness reviews (effectiveness at Pickering NGS) of OPG programs used to demonstrate compliance with the PSR2 Assessment Basis were conducted using recent applicable audit and self-assessment results:

- OPG Nuclear Oversight independent performance-based Program audits (typically performed in 1 to 5 year cycles) and self-assessments. This includes review of associated Station Condition Records and Action Requests to determine the status of any resulting corrective actions; and
- CNSC "Type I" and "Type II" inspections of the effectiveness and performance of OPG programs, where discussed in OPG audits or self-assessments.

There are many audits and self-assessments that are performed to assess the effectiveness of important aspects of each program. A sample of audits and self-assessments has been summarized for each program in order to demonstrate that program effectiveness is being assessed on an ongoing basis. The focus of these reviews was on effectiveness of the programs at Pickering NGS, where specific information is available. Results from these audits and self-assessments will be considered in the Global Assessment process. It is noted that audits and self-assessments are, by their nature, self-critical and are used to drive excellence in

performance. As a result, the broad review scope of program audits focuses on identifying improvement opportunities rather than presenting a balanced picture of program performance.

Program effectiveness is also monitored and addressed through the Fleetview Program Health and Performance Reporting process [12]. This process involves direct oversight by the Chief Nuclear Officer, and includes participation by the Nuclear Executive Committee members. Programs are reviewed, senior oversight is provided, and improvement plans are generated.

The list of Nuclear Programs to be assessed for each Safety Factor was derived from review of current OPG Governance, and has used the most recent version of these documents as of the PSR2 freeze date of January 15, 2016.

3.4 Additional Reviews

A review of the R04 Pickering LCH [4] was performed to determine if there are any impacts associated with operation of the Pickering Units past 2020 on the following (all related to Safety Factor 6):

- Commitments previously made to the CNSC;
- Open CNSC action items; and
- Exemptions granted by the CNSC.

The PSR2 assessment includes identification and review of previously identified Pickering-specific or programmatic PSR1 gaps related to Safety Factor 6 (as identified in the Darlington ISR Integrated Implementation Plan [13] and Pickering Units 5-8 Continued Operations Plan [11]) to ascertain the status of OPG's improvement plan(s) or other dispositions to address these and the implications of extending operation beyond 2020 (if any)[7].

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[7] PSR2 includes consideration and confirmation that the findings of PSR1 remain valid, as applicable, for the operation period. This includes assessment of PSR1 conclusions against implications resulting from extended operation. In particular, Pickering PSR1 results are applicable to PSR2 if there was a PSR1 gap that is still open, or if a closed PSR1 gap could be affected by extended operation. If so these gaps are carried forward into PSR2 for consideration in the Global Assessment. (When references to PSR1 are made, the source document is identified and the relevant text from that source document is summarized in the context of PSR2.) With respect to the Darlington ISR, much of the evaluation of Safety Factor health is based on programs and practices that apply across OPG's nuclear operations. As a result, Darlington ISR programmatic conclusions are applicable to Pickering PSR2 where Pickering is confirmed to follow the same nuclear programs and practices that were assessed for Darlington. Darlington PSR1 results are applicable to Pickering PSR2 if there are Darlington PSR1 gaps that are found to be relevant to Pickering PSR2.

Fukushima Action Items were reviewed to identify implications of extending operation beyond 2020 (if any). The methodology for this review is provided a separate PSR2 FAI Review Report.

Any PSR2 gaps identified as a result of the Safety Factor 6 review which are relevant to other Safety Factors are also discussed.

4.0 REVIEW FINDINGS

4.1 Review Tasks

The sub-sections below provide an assessment of the adequacy of applicable OPG Governance, Programs, Policies and Procedures (as well as Instructions and Guidelines, as applicable) in demonstrating compliance against the Safety Factor 6 Review Tasks.

4.1.1 Review Task #1: Current Probabilistic Safety Assessments and Assumptions

Confirm existence of a PSA and the assumptions used, the fault schedule, the representations of operator actions and common cause events, the modelled plant configuration and consistency with other aspects of the safety case.

OPG Standard N-STD-RA-0034 [14], "Preparation, Maintenance and Application of Probabilistic Risk Assessment", outlines the guidance for the preparation and maintenance of the Pickering Probabilistic Safety Assessment (PSA) program. This PSA standard ensures that the intent of the OPG Nuclear Safety Policy for the appropriate use of PSA in operating the plant, in compliance with CNSC Regulatory Standard S-294 [60] Probabilistic Safety Assessment for Nuclear Power Plants, is met at all OPG Nuclear Stations.

The R04 Pickering Licence Conditions Handbook [4] was revised to reflect "OPG request for PROL amendment [PROL 48.02/2018] to replace S-294 with REGDOC-2.4.2", effective December 18th, 2015. Per Section 5.1 of the R04 LCH:

> *REGDOC-2.4.2 Probabilistic Safety Assessment (PSA) for Nuclear Power Plants: Implementation Strategy: OPG will update the Pickering A PSA and Pickering B PSA using a graded approach as permitted by REGDOC-2.4.2. The PSA elements created as part of S-294 will be updated and the updated requirements of REGDOC-2.4.2, such as Irradiated Fuel Bay (IFB) risk assessment, will be addressed.*
>
> *The next Pickering B PSA update will be completed in 2017, including detailed risk re-quantification, in accordance with S-294. Similarly, the next Pickering A PSA update will be completed in 2018, including detailed risk re-quantification, in accordance with S-294.*
>
> *All the Pickering A PSA and Pickering B PSA updates extended to 2020 will be solely focused on the additional updated requirements of REGDOC-2.4.2 going beyond S-294 requirements, including for example, IFB risk assessment, and which are risk contributors of less significance. The updated requirements of REGDOC-2.4.2 may be dealt with through alternative methods to PSA for which guidance is currently being developed by industry.*

The results of the PSR2 review of REGDOC-2.4.2 are summarized in Section 4.2.

To support implementation of the PSA standard N-STD-RA-0034 [14], PSA Guides, N-GUID-03611-10001 Volumes 1 through 10 ([15] through [23]) provide for the preparation, maintenance and application of OPG PSAs including methodologies and assumptions used. Further station specific guidelines for Pickering Units 1,4 have also been prepared and are available in references [24], [25], [26], [27], [28] and [29]. The PSA for Pickering Units 5-8 did not require station specific guides since the N-GUIDs described above were applicable. Using these guidelines, OPG submitted S-294 compliant Pickering Units 5-8 and Units 1,4 PSAs to the CNSC (in 2012 and 2014 respectively) consisting of the following PSA documentation as outlined in [4]: [8]

- NA44-REP-03611-00011, Hazards Screening Analysis – Pickering A, [30]
- NA44-REP-03611-00012, Pickering NGS A Level 1 At-Power Internal Events Risk Assessment (PARA-L1P), [31]
- NA44-REP-03611-00013, Pickering NGS A Level -2 At-Power Internal Events Risk Assessment (PARA-L2P), [32]
- NA44-REP-03611-00014, Pickering NGS A Level -1 Outage Internal Events Risk Assessment (PARA-L1O), [33]
- NA44-REP-03611-00021, Pickering NGS A Internal Flood Probabilistic Risk Assessment (PARA Flood), [34]
- NA44-REP-03611-00022, Pickering NGS A PRA- Based Seismic Margin Assessment (PARA Seismic), [35]
- NA44-REP-03611-00023, Pickering NGS A Level 1 High Wind Probabilistic Risk Assessment, [36]
- NA44-REP-03611-00038, Pickering NGS A Probabilistic Risk Assessment (PRA) – Internal Fire Report, [37]
- NK30-REP-03611-00008, Hazards Screening Analysis – Pickering B, [38]
- NK30-REP-03611-00006, Pickering NGS B Level 1 At-power Internal Events Risk Assessment, [39]
- NK30-REP-03611-00010, Pickering NGS B At-Power Level 2 Probabilistic Risk Assessment (PRA) for Internal Initiating Events, [40]
- NK30-REP-03611-00009, Pickering NGS B Level 1 Outage Internal Events Risk Assessment, [41]
- NK30-REP-03611-00011, Probabilistic Risk Assessment Level 2 Outage Report – Pickering B, [42]
- NK30-REP-03611-00012, Pickering NGS B Probabilistic Risk Assessment - Internal Fire Report, [43]
- NK30-REP-03611-00013, Pickering NGS B (PNGS-B) PRA Based Seismic Margin Assessment (SMA), [44]
- NK30-REP-03611-00014, Pickering NGS B Internal Flood Probabilistic Risk Assessment (PBRA Flood), [45]
- NK30-REP-03611-00020, Pickering NGS B High Wind Probabilistic Risk Assessment, [46]

[8] Note that Probabilistic Risk Assessment (PRA) is now referred to as Probabilistic Safety Assessment (PSA).

Supplementary PSA analyses were also completed, as warranted, to address the Pickering Licence Hold Point, which among other things credit the use of Emergency Mitigating Equipment (EME) that was installed as part of the lessons learned from the Fukushima event. These supplementary analyses will be incorporated into the regulatory compliant PSAs as part of the regular PSA review and update cycles for each station, and have been listed here for completeness.

- NA44-REP-03611-00031 R000, Pickering NGS A Level-1 At-power Risk Assessment (PARA-L1P) - FAI Update, [47]
- NA44-REP-03611-00032 R000, Pickering NGS A Level-2 At-power Internal Events Risk Assessment (PARA-L2P) - Fukushima Action Item (FAI) Update, [48]
- NA44-REP-03611-00033 R000, Pickering NGS A High Wind Probabilistic Risk Assessment - FAI Update, [49]
- NA44-REP-03611-00034 R000, Pickering NGS A PRA-based Seismic Margin Assessment (para-seismic) - FAI Update, [50]
- NA44-REP-03611-00035 R000, Pickering NGS A Internal Flood Probabilistic Risk Assessment (PARA-flood) - FAI Update, [51]
- NK30-REP-03611-00025 R000, Pickering NGS B Level 1 At-power Internal Events Risk Assessment Fukushima Action Item Update, [52]
- NK30-REP-03611-00022 R000, Pickering NGS B Level 2 At-power Internal Events Risk Assessment Fukushima Action Item Update, [53]
- NK30-REP-03611-00027 R000, Pickering NGS 'B' Level-1 High Wind Probabilistic Risk Assessment - FAI Update, [54]
- NK30-REP-03611-00028 R000, Pickering NGS B Probabilistic Risk Assessment For Internal Fires - Fukushima Action Item Update, [55]

<u>The Fault Schedule</u>

The list of initiating events (i.e., the fault schedule) is developed and documented for each OPG station as per the direction provided in Section 2.0 of [15] and [24]. The most current listings for Pickering are provided in:

- NK30-REP-03611-00021 R00, Pickering B Risk Assessment Summary Report [57]
- NA44-REP-03611-00036 R00, Pickering A Risk Assessment Summary Report [58]

<u>Representation of Operator Actions</u>

Section 2.6 of N-GUID-03611-10001 Volume 1 [15], "OPG Probabilistic Risk Assessment (PRA) Guide - Level 1 (At-Power)", and similarly Section 2.6 of [24], provides the requirement to account for operator actions during preparation of Level 1 PSA studies. Additional guidance relevant to specific PSA elements is provided in the PSA Guide volumes for those elements (e.g., [17] and [26] for outage PSA).[9]

[9] The OPG Level 1 At-Power PSA Guides [15] [24] provide general methodologies that are generic to the PSA models for all hazards and operating states. Other volumes of the PSA Guide provide specific additional guidance relevant to the analysis for that specific hazard or operating state, but refer back to the Level 1 At-Power Guides for the generic requirements. This Safety Factor 6 report therefore generally refers to the generic requirements described in the Level 1 At-Power PSA Guides.

Specifically Section 2.6.8.1 of [15] [24], "Methodology for Human Interaction Re-Quantification", outlines a detailed process used to re-quantify any human operator interactions in the station PSA that are identified as being potentially significant contributors to risk. The process includes a formal review of significant human interactions with station operations staff and/or simulator training staff to confirm that the modelled operator tasks have been properly understood. Furthermore the PSA Guides for Level 1 and 2 PSA preparation [15] [16] [24] [25] also mandate that current operator actions/responses be captured via an in depth review of current station documentation and supporting references.

Representation of Common Cause Events

External and internal common cause hazards are assessed according to the guidelines specified in [21] (external), and [22] (internal). These guidelines outline a series of generic hazards comprised of 66 external natural hazards, 8 external human induced hazards and 16 internal hazard categories, including such examples as electromagnetic interference, volcanic activity and static electricity. They outline a precise methodology for screening each event for its inclusion in the PSA for any given station and include general principles and approaches for conducting both an initial qualitative and subsequent quantitative screening. The qualitative screening is based on the impact of the hazard on the plant's safe operation and the associated consequences while the quantitative screening is based on the probability of occurrence (frequency) of the hazards.

A list of screened internal and external common cause events compliant with these guidelines was developed as part of the Hazard Screening Analysis completed in 2012 for both Pickering Units 1,4 and Units 5-8 [30], [38]. Certain events such as fire, flood, high wind and seismic are, however, not contained in these guides as they are already included in the PSA. Fire, flood, high wind and seismic PSAs have each been prepared in accordance with their respective PSA guidelines as documented above.

For the hazards that have been included in the PSA models, the PSA Guides ([15] to [37]) provide guidance on the methodologies to be used in modelling common cause events. Specifically, Section 2.5 "Dependency Failure Analysis" of the Level 1 At-Power PSA Guide [15] [24] provides guidance on modelling of common cause events, which is applicable in the PSAs for all hazards and operating states. This includes guidance for addressing functional dependencies (e.g., failures involving shared sub-systems or support systems), physical interaction dependencies (e.g., involving hostile conditions in a shared environment), human interaction dependencies, and implicit residual common cause (e.g., for redundant equipment with common design features, common manufacturer, etc.).

Modelled Plant Configuration and Consistency with Safety Case

OPG Nuclear Program N-PROG-RA-0016 [10] states that "*risk information used in safety decision-making should be based to the extent practical on data and models that reflect the characteristics of the facility concerned*". Similarly, Section 1.3.6 of Nuclear Standard N-STD-RA-0034 [14] mandates that the response of the system to each initiating event of concern be modelled in a fault tree analysis, with the response as specified in Design Manuals, the Safety Report and the Abnormal Incidents Manuals (AIM).

The specific instructions for the preparation and maintenance of both Level 1 and Level 2 PSAs, contained in Sections 1.9 and 1.8 of N-GUID-03611-10001 Volume 1 [15] and Volume 2 [16] respectively (and similarly in [24] and [25]), dictate that the development of the PSAs rely on supporting references including but not limited to the stations' Safety Report, existing safety analysis, Operational Safety Requirements (OSRs), AIMs and System Operating Manuals. Furthermore these sections outline precisely how station documentation for each credited mitigating system and support system will be used to support the OPG PSAs, and that *"operational data including system alignment, safety related system testing, maintenance outage for plant systems and processes will be referenced to support as-built and as-operated system states for "realistic" input to the OPG PRAs."*, hence ensuring consistency with each station's safety case and current design.

A PSA based process of modelling, assessing and managing nuclear safety risk that results from maintenance, during either planned unit outages (outage risk) or at-power reactor operation (on-line risk), is documented in N-STD-RA-0030 "Risk Management for Outage Planning and On-Line Maintenance" [56]. This standard outlines requisite reactor safety assessments of work schedules, risk-informed advice to work planners and the quantification of the risk of individual maintenance configurations using an appropriate risk monitoring software platform. It therefore ensures that temporary maintenance configurations are adequately modelled and conform to PSA policies and principles.

Conclusion:

The conclusion of this Review Task assessment is that the existence of a PSA and the assumptions used, the fault schedule, the representations of operator actions and common cause events, the modelled plant configuration and consistency with other aspects of the safety case is confirmed. The intent of Review Task #1 is met and therefore Pickering NGS is compliant.

4.1.2 Review Task #2: Impact of Changes in Plant Design, Operation and Failure Data

Confirm existence of processes to assess the impact of changes in plant design, operation, and plant specific failure data and update the PSA to reflect the current plant status as required.

OPG Nuclear Program N-PROG-RA-0016 [10], "Risk and Reliability Program", states that "*risk information used in safety decision-making should be based to the extent practical on data and models that reflect the characteristics of the facility concerned*". The program specifies that facility-specific PSA updates are completed during regular review cycles or when warranted by design change [10], hence, ensuring the current condition of each station is reflected in all PSA related analyses.

Specifically, N-STD-RA-0034 [14], "Preparation, Maintenance and Application of Probabilistic Risk Assessment", provides requirements for the preparation, maintenance and application of PSA at OPG Nuclear facilities including the required standard elements which have been developed to meet the intent of CNSC Regulatory Standard S-294[10], Probabilistic Safety Assessment for Nuclear Power Plants. N-STD-RA-0034 further clarifies that the "*PRA shall reflect the current configuration of the facility*" and that "*PRA maintenance shall ensure that changes to the design, operation and maintenance of the facility are reflected in the PRA*".

Plant Design and As-Built Configuration

Design changes at both Pickering Units 1,4 and Units 5-8 are managed through the Engineering Change Control program, N-PROG-MP-0001, [59]. It requires that the changes be controlled such that plant configuration is maintained in conformance with the Safe Operating Envelope.

OPG Nuclear Program N-PROG-RA-0016 [10] states that any proposed changes to plant operation, configuration or procedures that may either significantly increase or decrease risks be reviewed and their impact on risk be quantified. It further mandates that the preparation, revision and maintenance of PSAs be conducted in accordance with OPG Nuclear Standard N-STD-RA-0034 [14] to reflect current design, operation basis and reliability data.

Clause 5.0 (3) of S-294[10] [60] requires that the PSA models reflect the plant's as-built and as-operated condition, as closely as reasonably achievable within the limitations of PSA technology and consistent with risk impact. The overall program direction contained in OPG Nuclear Standard N-STD-RA-0034 [14] and the specific instructions for the preparation and application of PSA at both Pickering Units 1,4 and Units 5-8 contained in [15] through [29] are compliant with the requirements of Clause 5.0 (3).

Operations

N-PROG-RA-0016 [9], "Risk and Reliability Program", states that any proposed changes to plant operation, configuration or procedures that may either significantly increase or decrease risks be reviewed and their impact on risk be quantified. It further mandates that the preparation, revision and maintenance of PSAs be conducted in accordance with OPG Nuclear Standard N-STD-RA-0034 [14] to reflect current design, operation basis and reliability data.

More specifically Section 2.6 of N-GUID-03611-10001 Volume 1 [15], "OPG Probabilistic Risk Assessment (PRA) Guide - Level 1 (At-Power)", and similarly Section 2.6 of [24], provides a requirement to account for, quantify and re-quantify applicable operator actions during preparation of the Level 1 PSA. These generic requirements similarly apply to the PSAs for other hazards and operating states. The preliminary

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10 Note OPG request for PROL amendment [PROL 48.02/2018] to replace S-294 with REGDOC-2.4.2, effective December 18th, 2015 per the discussion included in Section 4.1.1 of this document. The results of the PSR2 review of REGDOC-2.4.2 are summarized in Section 4.2.

quantification is performed as part of the development of the fault tree while potentially significant human interactions are subsequently re-quantified using more detailed methods. Section 2.6.8.1 Methodology for Human Interaction Re-Quantification, outlines the detailed process used to re-quantify any human operator interactions in the station PSA identified as being potentially significant contributors to risk. This section includes a formal review of significant human interactions with station operations staff and/or simulator training staff to confirm that operator tasks modelled by the human interactions have been properly understood. The Level 1 and Level 2 PSAs also consider station operation impacts via an in depth review of current station documentation and supporting references. Per N-GUID-03611-10001 Volume 2 [16] (and similarly in [25]), "*Station documentation for each credited mitigating system and support system will be used to support the PSA. Specifically, operational data including system alignment, safety related system testing, maintenance outage for plant systems, operator actions/responses and processes will be referenced to support as-built and as-operated system states…*"

<u>Plant Specific Reliability Data</u>

N-PROG-RA-0016 [9] states that "component reliability data be compiled, analyzed, and applied to maintain risk and unavailability models". Consistent with the overall industry approach that the model reflect the plant as closely as reasonably achievable, the OPG Nuclear Standard N-STD-RA-0034 [14] similarly requires that the component reliability data and initiating event frequency data used in the PSA be updated on a regular basis and be representative of expected future performance at the facility (internal OPEX). It further requires that this data be quantified using information from the facility being studied to the extent possible; however, it states that generic data from other sources (external OPEX) may be used where facility-specific data is not available. It identifies sources for the reliability data including but not limited to, best available facility specific data, e.g., from the Annual Risk and Reliability Reports. The updated plant-specific reliability data is incorporated into the PSAs as part of the 5-year update cycle, using the detailed methodologies described in the OPG PSA Guides ([15] to [29]).

The requirements for reliability monitoring and reporting of the Systems Important to Safety are provided in OPG Nuclear Standard N-STD-RA-0033, Reliability Monitoring and Reporting of Systems Important to Safety [61].

Conclusion:

The conclusion of this Review Task assessment is that there are currently processes to assess the impact of changes in plant design, operation, and plant specific failure data and update the PSA to reflect the current plant status. The intent of Review Task #2 is met and therefore Pickering NGS is compliant.

4.1.3 Review Task #3: Guidelines for Probabilistic Safety Assessment

Confirm there are guidelines to account for operator actions, common cause events, cross-link effects, redundancy, and diversity.

OPG Nuclear Program N-PROG-RA-0016 [9], "Risk and Reliability Program", establishes the framework for the development and use of PSAs as a means to manage radiological risks and contribute to safe operation of OPG's nuclear generating stations.

N-STD-RA-0034 [14], "Preparation, Maintenance and Application of Probabilistic Risk Assessment", ensures that in conducting PSAs, OPG performs human reliability analysis with a structured approach, identifying potential human failure events and systematically estimating the probability of those events using data, models, or expert judgment.

Section 2 of N-GUID-03611-10001 Volume 1 [15], "OPG Probabilistic Risk Assessment (PRA) Guide - Level 1 (At-Power)", and similarly Section 2 of [24], provides the requirement to account for common cause events, cross-link effects, redundancies, non-diverse equipment (i.e., equipment susceptible to common cause failures) and operator actions (human interactions). These general requirements apply to the PSA models for all hazards and operating states. More specifically common cause events, cross-link effects, redundancy, and diversity considerations are analyzed and modelled in line with the guidelines discussed in Section 2.5 of N-GUID-03611-10001 Volume 1 [15] regarding dependent or common-cause failures[11] while human interactions are primarily discussed as part of Section 2.6 on Human Reliability Analysis.

Human interactions in particular are also included in the Outage PSA Guide N-GUID-03611-10001 Volume 4 [17], and similarly in [26]. Identification and quantification of human interaction initiators are unique in Outage PSA as during an outage, operations and maintenance activities are performed in more dynamic situations with greater requirements for manual control actions. The opportunity for human error to result in an initiating event is higher than in the At-Power PSA. Given the potential for human-based initiators, the Outage PSA explicitly identifies them. This differs from the accepted approach for At-Power PSA, which implicitly captures the pre-initiating human interaction events in the reliability data collected at the component, system and plant level (as opposed to quantifying each human-based initiator independently).

Conclusion:

The conclusion of this Review Task assessment is that there are guidelines in place to account for operator actions, common cause events, cross-link effects, redundancy, and diversity. The intent of Review Task #3 is met and therefore Pickering NGS is compliant.

[11] i.e., failures which can result in simultaneous impairment of multiple systems or reduce the designed redundancy within a system.

4.1.4 Review Task #4: Consistency of PSA with Accident Management Programs

> *Confirm that the accident management programs for accident conditions (design basis accident conditions and design extension conditions) are consistent with PSA models and results.*

The accident management program for OPG's nuclear power plants is defined in the Consolidated Nuclear Emergency Plan (CNEP) [62], its supporting suite of governing documents, and the implementing operational procedures. The planning basis for the CNEP is derived from sources such as the station's deterministic safety analysis (as documented in the Safety Report) and PSA. In particular, PSA studies are the primary tool for assessing the potential consequences of accidents arising from external events (e.g., seismic, high wind, flooding).

For Design Basis Accidents (DBAs), Pickering has a mature accident management program for all internal and external accidents. These events are fully assessed in the Safety Report, PSA and supplementary assessments. PSA consistency with the underlying Safety Analysis (including the Safety Report and by extension the OSRs and AIMs) is assured via the implementation of N-STD-RA-0034 [14], "Preparation, Maintenance and Application of Probabilistic Risk Assessment" which mandates that the PSA be consistent with operator actions, modelled plant configuration and the existing safety case. While procedures for the mitigation of DBAs are derived most directly from the deterministic safety analysis for the station, operator responses as specified in these procedures are modelled in detail in the PSA, as explained under Review Task 1, Section 4.1.1 of this Safety Factor report. This ensures consistency between the PSA and accident management for DBAs. PSA results are in turn used to confirm that credited accident management actions are appropriate, or to identify areas for improvement to reduce the risk contribution from post-accident operator actions.

References [63] and [64], applicable to Pickering Units 1,4 and Units 5-8 respectively, identify the Beyond Design Basis Accident (BDBA) functional safety requirements for Structures, Systems and Components (SSCs) credited to prevent progression to severe accidents. Accident management for BDBAs relies on plant SSCs and on the use of Emergency Mitigating Equipment (EME) that is specifically intended to respond to a scenario involving an extended loss of all AC power, with the consequential total loss of all heat sinks. This scenario was selected, in part, because of its contribution to severe core damage frequency and large release frequency based on Level 2 PSA studies. The EME technical basis [65] specifically uses Pickering PSA results to establish approximate timing(s) for accident progression (assuming no actions are taken) in order to demonstrate an achievable success path for EME to prevent BDBA progression to a severe accident.

The use of EME for beyond design basis accidents (in particular design extension conditions) at Pickering Units 5-8 has been modelled in supplementary PSA analyses ([52] to [55]) performed as part of the removal of the Pickering Licence Hold Point, subsequent to the completion of the 2012 Pickering B PSA. Among other things this analysis credits the use of EME that was installed as part of the lessons learned from the Fukushima event to provide emergency make-up water (e.g., to the steam generators and moderator), the installation of Passive Auto-Catalytic Recombiners (PARs) to reduce containment hydrogen levels, and other accident mitigating features. Similar analyses for Pickering Units 1,4 ([47] to [51]) were also completed alongside the Pickering A PSA in 2014, again as part of resolution of the Pickering Licence Hold Point [66]. PSA results reflect the modelling of certain Fukushima enhancements and are included in Reference [67]. Hence, current PSAs for Pickering are consistent with the accident management provisions that have been implemented to address design extension conditions for BDBAs.

PSA models and results for Pickering have also played a significant role in deriving the Severe Accident Management Guidelines (SAMG) that are in place to manage accident response should a BDBA progress to a severe accident. SAMG documentation has been approved for use under the NA44-SAM-09013-10000 document series for Pickering 1,4 and the NK30-SAM-09013-10000 document series for Pickering 5-8. The physical processes that govern severe accident phenomena are complex and, consequently, SAMG cannot be made highly dependent on detailed analyses because of uncertainties associated with severe accident causes and progression. However, reasonable strategies for coping with severe accident progression can be identified and one of the primary sources of insight used to develop SAMG for OPG stations is the PSA. For example, PSA studies have been used to inform SAMG development by characterizing the potential timing and magnitude of hazards during a severe accident, most notably elevated source terms for fission products and hydrogen generation, and peak pressures inside containment. This information has been used to create computational aids that are used to estimate in-plant conditions as part of SAMG response. In some cases, PSA results have been used to identify additional clarifications, instructions and cautions with respect to the use of certain mitigating strategies that have been incorporated into updates of the stations' SAMG documentation. In addition, Level 2 PSA analyses have been conducted to demonstrate the extent to which specific mitigation measures (e.g., PARs, venting) are expected to be effective to protect containment during severe accident progression. The results of PSA studies have also been used to demonstrate plant habitability and accessibility to perform SAMG actions, and to demonstrate the survivability of equipment and instrumentation used in SAMG. These studies provide confidence that SAMG strategies can be implemented and lead to a successful outcome, and illustrate how accident management for severe accidents relies upon PSA models and results.

Another important aspect of accident management is the need to demonstrate preparedness by periodically performing drills and exercises (in accordance with [68]) that provide a means to test the effectiveness of emergency response procedures, facilities, equipment and training, as well as to provide training for emergency responders and identify opportunities for continuous improvement. To meet these objectives, plausible scenarios with meaningful simulated plant data are required. These are typically derived from Level 2 PSA analyses.

Conclusion:

The conclusion of this Review Task assessment is that the accident management programs for accident conditions (design basis accident conditions and design extension conditions) are consistent with PSA models and results. The intent of Review Task #4 is met and therefore Pickering NGS is compliant.

4.1.5 Review Task #5: Compliance with Safety Criteria

Confirm that the results of the PSA show that risks are sufficiently low and well balanced for all postulated initiating events and operational states, and meet relevant probabilistic safety criteria.

N-PROG-RA-0016 [9], "Risk and Reliability Program" outlines the Ontario Power Generation Risk-Based Safety Goals applicable to both Pickering Units 1,4 and Units 5-8. The safety goals specified in the document are numerical PSA based criteria used to ensure the radiological risks arising from nuclear accidents associated with operation of nuclear reactors are low in comparison to risks to which the public is normally exposed.

N-PROG-RA-0016 [9], provides average risk-based safety goals of 10^{-4} per year for severe core damage, and 10^{-5} per year for a large off-site release, per each operating unit and applicable to each hazard. These safety goals are comparable to industry best practice and represent the accepted risk exposure above which corrective action shall be taken to reduce risk. If the risk cannot be returned to an acceptable level, N-PROG-RA-0016 states that the Chief Nuclear Engineer and the Director, Operations and Maintenance may direct the immediate and orderly shutdown of the affected units or stations. Furthermore the program also specifies administrative safety goal values that are 10 times smaller than their safety goals and represent the desired objective towards which the facility should strive.

PSA Guides [15] through [29] reference the safety goals from N-PROG-RA-0016 [9], as required, and hence ensure that the results of the PSAs remain within the specified safety goals for average risk.

Time-average PSA results are provided in [67] and are presented in Table 3 and Table 4 below for reference. These tables show both the S-294 compliant baseline PSA results submitted to the CNSC in 2012 and 2014, for Pickering Units 5-8 and Units 1,4 respectively, as well as the updated PSA results that include certain Fukushima enhancements as of Spring 2014, completed as part of addressing the Pickering Licence Hold Point.

When each hazard is considered individually by reactor, the time-average Severe Core Damage Frequency PSA results for certain S-294 PSA elements, in Table 3, are above the OPG specified Administrative Safety Goal (10^{-5}/r-yr) but within the OPG Safety Goal (10^{-4}/r-yr). After incorporating the PSA updates that include certain Fukushima enhancements, only the Pickering Units 1,4 at-power fire risk remains above the OPG specified Administrative Safety Goal for Severe Core Damage Frequency. This is therefore identified as a gap for Pickering PSR2 **(Pickering PSR2 Gap SF6-1)**.

When each hazard is considered individually by reactor, the time-average Large Release Frequency PSA results for certain S-294 PSA elements, in Table 4, are above the OPG specified Administrative Safety Goal (10^{-6}/r-yr) but within the OPG Safety Goal (10^{-5}/r-yr). After incorporating the PSA updates that include certain Fukushima enhancements, only the Pickering Units 1,4 at-power internal events and at-power fire risks remain above the OPG specified Administrative Safety Goal for Large Release Frequency. This is therefore identified as a gap for Pickering PSR2 **(Pickering PSR2 Gap SF6-2)**.

OPG has been actively working on risk reduction plans [69] to better address certain higher risk items, such as internal fires at Pickering Units 1,4, with the objective of reducing Pickering Units 1,4 risk for both Severe Core Damage Frequency and Large Release Frequency. These risk reduction plans include further refinements to the PSA analyses to address potential over-conservatisms in the existing results. Work on the risk reduction plans was initiated several years ago, as described in Section 5 of Reference [67], and this work continues at OPG. The PSA results presented in this report do not incorporate these enhancements and therefore are not reflective of the risk reduction activities currently underway as part of the development of a REGDOC-2.4.2 compliant PSA (See Section 4.2). Additionally, work is in progress relating to the treatment of aggregate risk as part of an industry initiative discussed in Section 3.3.2.3 of Reference [70] and in Section 2 of Reference [67].

To maintain safety during temporary outage or maintenance alignments, the impact that these alignments have on the PSA is evaluated in accordance with N-STD-RA-0030 [56]. To ensure that reasonable bounds are placed on these short-term risks (i.e., risks present for significantly less time than the one year time period assumed by the safety goals for average risk), N-PROG-RA-0016 [9], also defines instantaneous safety goals of $3x10^{-4}$ per year for severe core damage, and $3x10^{-5}$ per year for a large off-site release, per operating unit and applicable to each hazard. Operation in a higher risk state, for short periods above these values, requires approval by both the Chief Nuclear Engineer and the Director of Operations and Maintenance. These instantaneous safety goals hence provide a level of assurance that risks are well balanced over the one year time period assumed by the average risk. Although the instantaneous risk safety goals in N-PROG-RA-0016 [9] apply to Level 1 and Level 2 PSA for all hazards, the limitations of some PSA model elements lead to the current practice at Pickering NGS (and other Canadian utilities) where instantaneous risk is only evaluated using the Level 1 at-power internal events and Level 1 outage internal events PSA models. This is therefore identified as a gap for Pickering PSR2 **(Pickering PSR2 Gap SF6-3)**.

Although instantaneous risk is only currently evaluated by all Canadian utilities using the Level 1 internal events PSA models, there are other deterministic rules (e.g., shutdown requirements for impairments of special safety systems and safety related systems) that provide assurance of defense in depth during temporary outage or maintenance alignments. These deterministic considerations apply regardless of the potential hazard, and may often result in greater restrictions than would be imposed by the risk-based safety goals alone. Given the differences in PSA models and tools for some hazards, work is also underway to develop methodologies for application of the PSAs for other hazards (e.g., assessment of instantaneous risk) as part of an industry initiative via the Candu Owner's Group.

Table 3: Severe Core Damage Frequency (SCDF) PSA Results

PSA Element	**Severe Core Damage Frequency**			
	S-294 Baseline PSA		**Fukushima Enhancement Adjusted PSA**	
	P1,4 Per Reactor (x 10^{-5} per r-yr)	**P5-8 Per Reactor** (x 10^{-5} per r-yr)	**P1,4 Per Reactor** (x 10^{-5} per r-yr)	**P5-8 Per Reactor** (x 10^{-5} per r-yr)
At-Power Internal Events	1.63	0.42	0.83	0.08
At-Power Fire	4.73 Note 2	0.38	4.73	0.06
At-Power Flood	1.02	0.07	0.56	0.07 Note 5
At-Power Seismic	0.26	0.10	0.18	0.10 Note 5
At-Power High Wind	2.69 Note 2	0.80	0.30	0.03
Outage Internal Events	0.66	0.10	0.66 Note 3	0.10 Note 5
Outage Fire	Note 1	Note 1	Note 1	Note 1
Outage Flood	Note 1	Note 1	0.15 Note 4	Note 1
Outage Seismic	Note 1	Note 1	0.05 Note 4	Note 1
Outage High Wind	Note 1	Note 1	0.08 Note 4	Note 1

Notes:

1. Outage risk is bounded by all units at-power. SCDF for outage external events was not estimated in detail.
2. This S-294 PSA includes credits for Phase 1 of the Emergency Mitigating Equipment.
3. Bounded by S-294 Level 1 outage PSA for internal events.
4. Estimate of outage SCDF based on Fukushima enhancement results for at-power.
5. Fukushima enhancements SCDF judged to be lower than S-294 SCDF, therefore not calculated in detail.

Table 4: Large Release Frequency (LRF) PSA Results

PSA Element	Large Release Frequency			
	S-294 Baseline PSA		Fukushima Enhancement Adjusted PSA	
	P1,4 Per Reactor (x 10^{-5} per r-yr)	**P5-8 Per Reactor** (x 10^{-5} per r-yr)	**P1,4 Per Reactor** (x 10^{-5} per r-yr)	**P5-8 Per Reactor** (x 10^{-5} per r-yr)
At-Power Internal Events	0.47	0.39	0.17	0.03
At-Power Fire	0.84 Note 2	0.34 Note 6	0.66	0.04
At-Power Flood	0.20	< 0.07	0.09	< 0.07 Note 5
At-Power Seismic	0.26	0.10 Note 7	0.04	< 0.10 Note 5
At-Power High Wind	0.80 Note 2	< 0.80	0.07	< 0.03
Outage Internal Events	< 0.1	< 0.1	< 0.1 Note 3	< 0.1 Note 5
Outage Fire	Note 1	Note 1	Note 1	Note 5
Outage Flood	Note 1	Note 1	0.02 Note 4	Note 5
Outage Seismic	Note 1	Note 1	0.01 Note 4	Note 5
Outage High Wind	Note 1	Note 1	0.02 Note 4	Note 5

Notes:

1. Outage risk is bounded by all units at-power. LRF for outage external events was not estimated in detail.
2. This S-294 PSA includes credits for Phase 1 of the Emergency Mitigating Equipment.
3. Bounded by S-294 Level 2 outage PSA for internal events.
4. Estimate of outage LRF based on Fukushima enhancement results for at-power.
5. Fukushima enhancements LRF judged to be lower than S-294 LRF, therefore not calculated in detail.
6. LRF for PB At-Power Fire per [57].
7. LRF for PB At-Power Seismic Events per [57].

Conclusion:

The results of the latest PSA completed as part of the response to the Pickering Licence Hold Point show that time-average risks are sufficiently low for all postulated initiating events, with the exception of Pickering Units 1,4 at-power internal events (above the Administrative Safety Goal but below the Safety Goal for large release frequency) and Pickering Units 1,4 internal fire events (above the Administrative Safety Goals but below the Safety Goals for both severe core damage and large release). This Safety Factor report also recognized that instantaneous risk Safety Goals are provided for Severe Core Damage Frequency and Large Release Frequency, but that instantaneous risk is only evaluated for severe core damage from internal events and not for large release or for other hazards such as internal fires or seismic events. The PSA results described in this PSR2 report do not incorporate certain proposed analysis enhancements, and do not necessarily reflect the latest risk reduction activities currently underway at OPG. Three gaps, listed above, are therefore identified for Pickering PSR2.

4.1.6 Review Task #6: Impact of Omissions in Probabilistic Safety Assessment

Review the extent to which hazards are represented in the PSA to verify that omissions are based on site specific justifications and that these omissions do not weaken the overall risk assessment for the plant.

The requisite screening conditions which must be satisfied to omit specific hazards from the PSA are documented in the internal and external hazard screening PSA guides, references [22] and [21] respectively. These documents outline qualitative and quantitative screening criteria for each hazard type as well as for external hazard combinations, and take into account both Canadian and international regulations and standards as well as information on credible hazards at the OPG sites.

Site specific screened lists of internal and external events for both Pickering Units 1,4 and Units 5-8 were completed using these OPG Guides as part of the Hazard Screening Analysis in 2012 [30], [38] and are compliant with CNSC Regulatory Standard S-294.

Certain hazards such as high wind were not able to be screened out and were subsequently included in standalone station PSAs while others such as internal fire and seismic events were already provided in standalone assessments. Other hazards, such as the effects of high or low temperatures, have effects which have been built into either the Level 1 or the Level 2 PSAs. Further, more detailed information regarding hazard assessments can be found in the Hazard Analysis PSR2 Safety Factor Report [71].

Conclusion:

Based on the assessment above, it is confirmed that hazards are represented in the PSA and omissions are based on site specific justifications which do not weaken the overall risk assessment for the plant. The intent of Review Task #6 is met and therefore Pickering NGS is compliant.

4.2 L/R/C/S Reviews

As per Section 2.2 of this report, detailed reviews for seven L/R/C/Ss with content applicable to Safety Factor 6 are provided in References [6], [7] and [8]. Associated findings applicable to Safety Factor 6 are summarized in Table 5 below.

Table 5: PSR2 L/R/C/S Review Results for Safety Factor 6

L/C/R/S Reviewed	PSR2 L/R/C/S Review Results for Safety Factor 6
CSA N286-12, "Management System Requirements for Nuclear Facilities"	There are no PSR2 gaps for CSA N286-12. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with N286-12.
CSA N286.7-16, "Quality Assurance Of Analytical, Scientific And Design Computer Programs"	There are no PSR2 gaps for CSA N286.7-16. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CSA N286.7-16.
CNSC REGDOC-2.4.2 (2014), "Probabilistic Safety Assessment (PSA) for Nuclear Power Plants"	There is one PSR2 CNSC REGDOC-2.4.2 (2014) gap which relates to Safety Factor 6: 1. The REGDOC-2.4.2 Pickering Implementation Plan agreed to with the CNSC did not consider operation beyond 2020 and therefore, the review and update of the Implementation Plan in the context of operation of Pickering NGS beyond 2020 is required. Therefore, this has been identified as a PSR2 gap (**Pickering PSR2 Gap SF6-4**).
CNSC REGDOC-2.5.2 (2014), "Design of Reactor Facilities: Nuclear Power Plants"	There is one PSR2 CNSC REGDOC-2.5.2 (2014) gap which relates to Safety Factor 6: 1. Clause 4.2.2 of REGDOC-2.5.2 introduces new requirements and limits for probabilistic analysis risk limits, such as a core damage frequency limit of $<10^{-5}$ yrs/yr. It has not been demonstrated that these requirements can be achieved. Therefore, this has been identified as a PSR2 gap (**Pickering PSR2 Gap SF6-5**).
CNSC REGDOC-2.3.2 (2015), "Accident Management, Version 2"	For Safety Factor 6, there are no PSR2 gaps for CNSC REGDOC-2.3.2 (2015).
CSA N286.7.1-09, "Guideline for the Application of N286.7-99, Quality Assurance of Analytical, Scientific, and Design Computer Programs for Nuclear Power Plants"	The N286.7.1 guide has been amalgamated into the new (-16) edition of the N286.7 Standard. The N286.7 CSA Impact Statement [9] states: "The CSA N286.7.1 guide will no longer be maintained after this new edition of N286.7 is issued. Any relevant guidance has been put into the new edition of N286.7." As a result, only the review of N286.7-16 has been prepared for PSR2.
CNSC G-149 (2000), "Computer Programs Used in Design and Safety Analyses of Nuclear Power Plants and Research Reactors"	There are no PSR2 gaps for CNSC G-149 (2000). Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with G-149 (2000).

4.3 OPG Program Effectiveness Reviews

The OPG Nuclear Program reviewed for Safety Factor 6 is identified in Table 2 and details of the associated effectiveness review for this N-PROG are provided in Appendix B.

4.4 Additional Review Findings

As discussed in Section 3.4, the PSR2 Safety Factor 6 assessment also included a review of commitments previously made to the CNSC, open CNSC action items, and exemptions granted by the CNSC, as identified in the R04 Pickering LCH [4], to determine if there are any impacts associated with operation of the Pickering Units past 2020. The review also included identification and review of previously identified programmatic Darlington PSR1 gaps related to Safety Factor 6 to determine impacts associated with operation of the Pickering Units past 2020. This assessment did not identify any gaps for Safety Factor 6.

Findings from the review of previously identified PSR1 gaps in the Pickering Units 5-8 Continued Operations Plan [11] are provided in a separate PSR2 COP Review Report. Findings from the review of Fukushima Action Items are provided in a separate PSR2 FAI Review Report. Results from the Continued Operations Plan and Fukushima Action Items reviews will be considered in the Global Assessment process.

There were no PSR2 gaps identified in this Safety Factor 6 Report that are relevant to other Safety Factors.

5.0 RESULTS AND CONCLUSIONS

OPG Governance, Programs, Policies, Procedures, Instructions and Guidelines related to Safety Factor 6 were reviewed for the six PSR2 Review Tasks in Section 4.1 of this report and resulted in Pickering PSR2 Gaps SF6-1, SF6-2, and SF6-3 below. L/R/C/S and OPG Nuclear Program effectiveness reviews for Safety Factor 6 were prepared per Sections 4.2 and 4.3, respectively, and resulted in PSR2 Gaps SF6-4 and SF6-5 below. Per Section 4.4, this report also included identification and review of previously identified programmatic Darlington PSR1 gaps related to Safety Factor 6 (to ascertain the implications of extending Pickering NGS operation beyond 2020), as well as a review of the R04 Pickering LCH [4] for any impacts of Pickering NGS operation beyond 2020 on: a) OPG commitments previously made to the CNSC, b) open CNSC action items, and c) exemptions granted by the CNSC (all related to Safety Factor 6), which resulted in no additional PSR2 gaps.

The five PSR2 gaps that will need to be addressed as part of Pickering PSR2 are:

- **Gap SF6-1:** When each hazard is considered individually by reactor, the time-average Severe Core Damage Frequency Probabilistic Safety Assessment (PSA) results for certain S-294 PSA elements, in Table 3, are above the OPG specified Administrative Safety Goal (10^{-5}/r-yr) but within the OPG Safety Goal (10^{-4}/r-yr). After incorporating the PSA updates that include certain Fukushima enhancements, only the Pickering Units 1,4 at-power fire risk remains above the OPG specified Administrative Safety Goal for Severe Core Damage Frequency. The PSA results described in this PSR2 report, however, do not incorporate certain proposed analysis enhancements and do not necessarily reflect the latest risk reduction activities currently underway at OPG.

- **Gap SF6-2:** When each hazard is considered individually by reactor, the time-average Large Release Frequency Probabilistic Safety Assessment (PSA) results for certain S-294 PSA elements, in Table 4, are above the OPG specified Administrative Safety Goal (10^{-6}/r-yr) but within the OPG Safety Goal (10^{-5}/r-yr). After incorporating the PSA updates that include certain Fukushima enhancements, only the Pickering Units 1,4 at-power internal events and at-power fire risks remain above the OPG specified Administrative Safety Goal for Large Release Frequency. The PSA results described in this PSR2 report, however, do not incorporate certain proposed analysis enhancements and do not necessarily reflect the latest risk reduction activities currently underway at OPG.

- **Gap SF6-3:** Although the OPG instantaneous risk Safety Goals apply to Level 1 and Level 2 Probabilistic Safety Assessment (PSA) for all hazards, current practice at Pickering NGS is that instantaneous risk is only evaluated using the Level 1 at-power internal events and Level 1 outage internal events PSA models. This is similar to current practices at other Canadian utilities. There are other deterministic rules at Pickering NGS that provide assurance of defense in depth during temporary outage or maintenance alignments, and these deterministic considerations apply regardless of the potential hazard. Work is underway, via an industry initiative, to develop methodologies for assessment of instantaneous risk from other hazards included in the PSA.

- **Gap SF6-4:** The REGDOC-2.4.2 Pickering Implementation Plan agreed to with the CNSC did not consider operation beyond 2020 and therefore, the review and update of the Implementation Plan in the context of operation of Pickering NGS beyond 2020 is required. Therefore, this has been identified as a PSR2 gap.

- **Gap SF6-5:** Clause 4.2.2 of REGDOC-2.5.2 introduces new requirements and limits for probabilistic analysis risk limits, such as a core damage frequency limit of $<10^{-5}$ yrs/yr. It has not been demonstrated that these requirements can be achieved. Therefore, this has been identified as a PSR2 gap.

The review of Safety Factor 6 has confirmed that the PSA programs and procedures at OPG are comprehensive, resulting in a systematic and disciplined approach to identifying, prioritizing and addressing any PSA related issues.

6.0 REFERENCES

[1] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[2] CNSC REGDOC-2.3.3, *Periodic Safety Reviews*, April 2015.

[3] IAEA Safety Guide No. SSG-25, *Periodic Safety Review for Nuclear Power Plants*, 2013.

[4] CNSC Report, LCH-PNGS-R004, *Pickering NGS: Licence Conditions Handbook*, December 23, 2015.

[5] OPG Report, P-REP-03680-00003 R000, *Pickering NGS PSR2: Definition of Safety Factor Review Tasks*, May 30, 2016.

[6] OPG Report, P-REP-03680-0586480 R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 9, 11, and 15*, September 2016.

[7] OPG Report, P-REP-03680-00021 R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 8, 10, 12, 13, and 14,* December 2016.

[8] OPG Report, P-REP-03680-00029 R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 1, 5, 6, and 7,* March 2017.

[9] CSA Impact Statement, *Notification of CSA N286.7 on Quality Assurance of Analytical Scientific, and Design Computer Programs; Product: New Edition; Product Designation: CSA N286.7; Previous Edition Published: 1999, Reaffirmed 2007 and 2012*, Date not provided.

[10] OPG Nuclear Program, N-PROG-RA-0016 R009, *Risk and Reliability Program*, June 24, 2016.

[11] OPG Plan, NK30-PLAN-00531-00001 R005, *Pickering 5-8 Continued Operations Plan*, December 2015.

[12] OPG Procedure, N-PROC-RA-0023 R018*, Fleetview Program Health and Performance Reporting*, August 2013.

[13] OPG Report, NK38-REP-03680-10185 R002, *Darlington NGS Integrated Implementation Plan (IIP)*, April 30, 2015.

[14] OPG Nuclear Standard, N-STD-RA-0034 R003, *Preparation, Maintenance and Application of Probabilistic Risk Assessment*, October 1, 2013.

[15] OPG Report, N-GUID-03611-10001, Volume 1 R004, *OPG Probabilistic Risk Assessment (PRA) Guide - Level 1 (At-Power)*, October 24, 2014.

[16] OPG Report, N-GUID-03611-10001, Volume 2 R003, *OPG Probabilistic Safety Assessment (PSA) Guide Volume 2 - Level 2 (At-Power)*, May 2015.

[17] OPG Report, N-GUID-03611-10001, Volume 4 R001, *OPG Outage Probabilistic Risk Assessment (PRA) Guide - Level 1*, July 9, 2010.

[18] OPG Report, N-GUID-03611-10001, Volume 5 R001, *OPG Probabilistic Risk Assessment (PRA) Guide – Fire*, June 6, 2011.

[19] OPG Report, N-GUID-03611-10001, Volume 6 R001, *OPG Probabilistic Risk Assessment (PRA) Guide - Internal Flood*, June 3, 2011.

[20] OPG Report, N-GUID-03611-10001, Volume 7 R001, *OPG Probabilistic Risk Assessment (PRA) Guide – Seismic*, March 15, 2011.

[21] OPG Report, N-GUID-03611-10001, Volume 8 R003, *OPG Probabilistic Safety Assessment (PSA) Guide - External Hazards Screening*, December 16, 2014.

[22] OPG Report, N-GUID-03611-10001, Volume 9 R001, *OPG Probabilistic Safety Assessment (PSA) Guide - Internal Hazards Screening*, December 15, 2014.

[23] OPG Report, N-GUID-03611-10001, Volume 10 R000, *OPG Probabilistic Risk Assessment Guide - High Wind Hazard*, August 20, 2012.

[24] OPG Guideline, NA44-GUID-03611-00010 R000, *Pickering NGS 'A' Probabilistic Risk Assessment Guide – Level 1 (At-Power)*, October 2, 2012.

[25] OPG Guideline, NA44-GUID-03611-00011 R000, *Pickering A Probabilistic Risk Assessment (PRA) Guide – Level 2 (At-Power)*, August 27, 2012.

[26] OPG Guideline, NA44-GUID-03611-00012 R000, *Pickering 014 Probabilistic Risk Assessment Guide – Level 1 Outage for Internal Events*, October 30, 2013.

[27] OPG Guideline, NA44-GUID-03611-00013 R000, *Pickering NGS A Probabilistic Risk Assessment Guide – Fire*, November 1, 2012.

[28] OPG Guideline, NA44-GUID-03611-00014 R000, *Pickering NGS A Probabilistic Risk Assessment (PRA) Guide – Internal Flood*, July 24, 2012

[29] OPG Guideline, NA44-GUID-03611-00015 R001, *Pickering NGS A Probabilistic Risk Assessment (PRA) Guide – Seismic*, November 22, 2013.

[30] OPG Report, NA44-REP-03611-00011 R000, *Hazards Screening Analysis – Pickering A*, January 2012

[31] OPG Report, NA44-REP-03611-00012 R000, *Pickering NGS A Level 1 At-Power Internal Events Risk Assessment (PARA-L1P)*, September 2013.

[32] OPG Report, NA44-REP-03611-00013 R000, *Pickering NGS A Level -2 At-Power Internal Events Risk Assessment (PARA-L2P)*, December 2013.

[33] OPG Report, NA44-REP-03611-00014 R000, *Pickering NGS A Level -1 Outage Internal Events Risk Assessment (PARA-L1O)*, February 2014.

[34] OPG Report, NA44-REP-03611-00021 R000, *Pickering NGS A Internal Flood Probabilistic Risk Assessment (PARA Flood)*, December 2013.

[35] OPG Report, NA44-REP-03611-00022 R000, *Pickering NGS A PRA- Based Seismic Margin Assessment (PARA Seismic)*, December 2013.

[36] OPG Report, NA44-REP-03611-00023 R000, *Pickering NGS A Level 1 High Wind Probabilistic Risk Assessment*, December 2013.

[37] OPG Report, NA44-REP-03611-00038 R000, *Pickering NGS A Probabilistic Risk Assessment (PRA) – Internal Fire Report*, April 2014.

[38] OPG Report, NK30-REP-03611-00008 R000, *Hazards Screening Analysis – Pickering B*, January 2012.

[39] OPG Report, NK30-REP-03611-00006 R001, *Pickering NGS B Level 1 At-power Internal Events Risk Assessment*, November 2012.

[40] OPG Report, NK30-REP-03611-00010 R000, *Pickering NGS B At-Power Level 2 Probabilistic Risk Assessment (PRA) for Internal Initiating Events*, December 2012.

[41] OPG Report, NK30-REP-03611-00009 R000, *Pickering NGS B Level 1 Outage Internal Events Risk Assessment*, November 2012.

[42] OPG Report, NK30-REP-03611-00011 R000, *Probabilistic Risk Assessment Level 2 Outage Report – Pickering B*, October 2012.

[43] OPG Report, NK30-REP-03611-00012 R000, *Pickering NGS B Probabilistic Risk Assessment - Internal Fire Report*, December 2012.

[44] OPG Report, NK30-REP-03611-00013 R001, *Pickering NGS B (PNGS-B) PRA Based Seismic Margin Assessment (SMA)*, April 2015.

[45] OPG Report, NK30-REP-03611-00014 R000, *Pickering NGS B Internal Flood Probabilistic Risk Assessment (PBRA Flood)*, December 2012.

[46] OPG Report, NK30-REP-03611-00020 R001, *Pickering NGS B High Wind Probabilistic Risk Assessment*, February 2014.

[47] NA44-REP-03611-00031 R000, *Pickering NGS A Level-1 At-power Risk Assessment (PARA-L1P) - FAI Update*, February 2014.

[48] NA44-REP-03611-00032 R000, *Pickering NGS A Level-2 At-power Internal Events Risk Assessment (PARA-L2P) - Fukushima Action Item (FAI) Update*, February 2014.

[49] NA44-REP-03611-00033 R000, *Pickering NGS A High Wind Probabilistic Risk Assessment - FAI Update*, February 2014.

[50] NA44-REP-03611-00034 R000, *Pickering NGS A PRA-based Seismic Margin Assessment (PARA-seismic) - FAI Update*, February 2014.

[51] NA44-REP-03611-00035 R000, *Pickering NGS A Internal Flood Probabilistic Risk Assessment (PARA-flood) - FAI Update*, February 2014.

[52] NK30-REP-03611-00025 R000, *Pickering NGS B Level 1 At-power Internal Events Risk Assessment Fukushima Action Item Update*, February 2014.

[53] NK30-REP-03611-00022 R000, *Pickering NGS B Level 2 At-power Internal Events Risk Assessment Fukushima Action Item Update*, February 2014.

[54] NK30-REP-03611-00027 R000, *Pickering NGS 'B' Level-1 High Wind Probabilistic Risk Assessment - FAI Update*, February 2014.

[55] NK30-REP-03611-00028 R000, *Pickering NGS B Probabilistic Risk Assessment For Internal Fires - Fukushima Action Item Update*, February 2014.

[56] OPG Standard, N-STD-RA-0030 R003, *Risk Management for Outage Planning and On-Line Maintenance*, November 2013

[57] OPG Report, NK30-REP-03611-00021 R000, *Pickering B Risk Assessment Summary Report*, February 2013.

[58] OPG Report, NA44-REP-03611-00036 R000, *Pickering A Risk Assessment Summary Report*, April 2014.

[59] OPG Nuclear Program, N-PROG-MP-0001 R014, *Engineering Change Control*, January 7, 2015.

[60] Regulatory Standard S-294, *Probabilistic Safety Assessment (PSA) for Nuclear Power Plants*, Canadian Nuclear Safety Commission.

[61] OPG Nuclear Standard, N-STD-RA-0033 R002, *Reliability Monitoring and Reporting of Systems Important to Safety*, October 24, 2013.

[62] OPG Nuclear Program, N-PROG-RA-0001 R014, *Consolidated Nuclear Emergency Plan*, May 2015.

[63] OPG Guideline, NA44-GUID-03600-00001 R000, *Pickering 1-4 Beyond Design Basis Functional Safety Requirements*, October 29, 2014.

[64] OPG Guideline, NK30-GUID-03600-00001 R000, *Pickering 5-8 Beyond Design Basis Functional Safety Requirements*, October 31, 2014.

[65] OPG Manual, N-BDB-03600-00002 R000, *OPG Emergency Mitigating Equipment for Beyond Design Basis Accidents: Technical Basis Document,* October 2015.

[66] OPG Report, N-REP-03600-10003 R007, *Fukushima Action Item Status Report*, November 27, 2015.

[67] OPG Letter, P-CORR-00531-04092, B. Phillips (OPG) to M.A. Leblanc (CNSC), *OPG Material and List of Presenters for Pickering Hold Point Removal Public Hearing – May 7, 2014*, April 30, 2014.

[68] OPG Procedure, N-PROC-RA-0045 R010, *Emergency Preparedness Drills and Exercises*, December 2015.

[69] OPG Letter, P-CORR-00531-04672, B. McGee (OPG) to M. Santini (CNSC), *Pickering NGS: Risk Improvement Plan Update*, February 26, 2016.

[70] CNSC Report, *Regulatory Oversight Report for Canadian Nuclear Power Plants: 2015*, October 2016.

[71] OPG Report, P-REP-03680-00011 R000, *Pickering NGS PSR2 Safety Factor 7 Report: Hazard Analysis*, March 2017.

Appendix A: Nomenclature

AIM	Abnormal Incidents Manuals
BDBA	Beyond Design Basis Accident
CNSC	Canadian Nuclear Safety Commission
COP	Continued Operations Plan
EME	Emergency Mitigating Equipment
FAI	Fukushima Action Item
IAEA	International Atomic Energy Agency
IFB	Irradiated Fuel Bay
ISR	Integrated Safety Review
LCH	Licence Conditions Handbook
L/R/C/Ss	Laws, Regulations, Codes and Standards
LRF	Large Release Frequency
NGS	Nuclear Generating Station
OPEX	Operating Experience
OPG	Ontario Power Generation
OSR	Operational Safety Requirements
PARs	Passive Autocatalytic Recombiners
PARTS	Pickering A Return to Service
PNGS	Pickering Nuclear Generating Station
PRA	Probabilistic Risk Assessment
PROL	Power Reactor Operating Licence
PSA	Probabilistic Safety Assessment
PSR	Periodic Safety Review
PSR1	Periodic Safety Review 1 (earlier OPG PSR work and other associated assessments)
PSR2	Periodic Safety Review 2 (subsequent PSR per CNSC REGDOC-2.3.3)
SCDF	Severe Core Damage Frequency
SSC	Structures, Systems and Components

Appendix B: OPG Program Effectiveness Review Results

B.1 N-PROG-RA-0016, "Risk and Reliability"

The purpose of the Risk and Reliability program is to establish a framework for the development and use of Probabilistic Safety Assessment (PSA) at OPG Nuclear as a means to manage radiological risks from nuclear accidents and to contribute to safe operation of nuclear reactors. Program elements have been developed to meet the intent of the OPG Nuclear Safety Policy and the applicable Canadian Nuclear Safety Commission (CNSC) regulatory requirements.

Nuclear Oversight conducted a performance based audit, NO-2016-013 [B.1.1], in August 2016 in order to determine compliance with the requirements of the Risk and Reliability program for Pickering and Darlington NGS. The audit identified performance improvement opportunities applicable to Pickering NGS in the areas of training and qualification as well as implementation of some program elements (e.g., the Probabilistic Risk Assessment Issues Database, mission time testing, and documentation updates.)

Two SCRs were initiated to address the above findings (P-2016-20258 and N-2016-20262). Both are now complete.

The Nuclear Safety and Technology Department completed a self-assessment, NO14-000591-SA [B.1.2], in December 2014 in order to determine compliance with the requirements of the Risk and Reliability program for Pickering and Darlington NGS. The self-assessment concluded that Pickering NGS has good alignment with current governance and best industry practices; recent procedural changes are reflected in the work program; and the Pickering Risk and Reliability Program satisfies the intent of the governance. A performance improvement opportunity applicable to Pickering NGS was identified in the area of Reactor Safety training.

AR 28173197 was initiated to track the associated assignments for this corrective action. The final remaining action is expected to be completed by Q2 2017.

References

[B.1.1] Nuclear Oversight Report, N-REP-01070-0606747 T06 (NO-2016-013), *Risk and Reliability Program*, August 12, 2016.

[B.1.2] Self-Assessment Report, NO14-000591-SA R000, *OPG Risk and Reliability Program - Station and Procedural Alignment*, December 15, 2014.

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ACCEPTED AS NOTED	
REVISE AND RESUBMIT	
Signature	Date: 03 MAR 2017
Name: Mike Ruffolo, Manager	
Dept: Pickering Engineering - Aging Management & Strategic Initiatives	
OPG Proprietary	
Doc No.: P-REP-03680-00011	Rev: 000
This acceptance does not relieve the contractor from responsibility for errors or omissions or from any obligations or liability under this contract.	

EM

Pickering NGS PSR2 Safety Factor 7 Report: Hazard Analysis

PS112/RP/008 R01

March 3, 2017

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Revision Summary for Amec Foster Wheeler Report PS112/RP/008

Rev	Date	Author	Comments
R00	July 4, 2016	Brandon McLean / Ranil Jayasundera	Initial issue for OPG review and comment.
R01	March 3, 2017	Brandon McLean / Ranil Jayasundera / Janice Cheng	Updated to address OPG comments on R00 Report.

EXECUTIVE SUMMARY

Ontario Power Generation (OPG) is performing a Periodic Safety Review (PSR) to support the possibility of continued operation of Pickering Nuclear Generating Station (NGS) beyond 2020. The PSR (referred to as "PSR2") is a subsequent PSR building on the review basis of earlier OPG Integrated Safety Reviews and other associated assessments. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

Part of PSR2 involves the preparation of Safety Factor reports for each of fifteen major topic areas. Safety Factor reports consist of:

- Assessing compliance against "Review Tasks", identified in the PSR2 Basis Document [1]. These Review Tasks are derived from Canadian Nuclear Safety Commission (CNSC) REGDOC-2.3.3, "Periodic Safety Reviews" [2] and International Atomic Energy Agency (IAEA) SSG-25, "Periodic Safety Review for Nuclear Power Plants" [3];

- Documenting assessments against applicable modern Laws, Regulations, Codes and Standards (L/R/C/Ss) as defined in Reference [1]; and

- Reviewing the effectiveness of applicable OPG Nuclear Programs that support the above assessments, through review of audit and self-assessment results.

The PSR2 review of Safety Factor 7, *Hazard Analysis*, is presented in this report. OPG Governance, Programs, Policies, Procedures, Instructions and Guidelines related to Safety Factor 7 were reviewed for the three PSR2 Review Tasks specified in Section 4.1 of this report. L/R/C/S and OPG Nuclear Program effectiveness reviews for Safety Factor 7 were prepared per Sections 4.2 and 4.3, respectively. Per Section 4.4, the PSR2 assessment includes a review of previously identified PSR1 gaps related to Safety Factor 7 (to ascertain the implications of extending Pickering NGS operation beyond 2020), as well as a review of the R04 Pickering Licence Conditions Handbook [4] for any impacts of Pickering NGS operation beyond 2020 on: a) OPG commitments previously made to the CNSC, b) open CNSC action items, and c) exemptions granted by the CNSC (all related to Safety Factor 7).

The results of the review of Safety Factor 7 are discussed in Section 5.0. The review of Safety Factor 7 has confirmed the adequacy of protection of Pickering NGS against internal and external hazards, with account taken of plant design (including confirmation that analyses/methods address the condition of Structures, Systems and Components (SSCs) important to safety), site characteristics, and current analytical methods, safety standards and knowledge. As discussed in Section 5.0, the review identified one gap that will need to be addressed further as part of the PSR2 Global Assessment process.

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1.0 INTRODUCTION

Ontario Power Generation (OPG) is performing a Periodic Safety Review (PSR) to support the possibility of continued operation of Pickering Nuclear Generating Station (NGS) beyond 2020.[1] A comprehensive Integrated Safety Review (ISR) was completed for Pickering Units 5 through 8 in 2009 in support of refurbishment and continued operation. Pickering Units 1,4 integrated safety assessments were also performed for Pickering A Return to Service (PARTS) in support of approval to restart Units 1 and 4. In addition to these Pickering-specific studies, the 2013 Darlington ISR performed extensive code and standard reviews that were updated in relation to the versions that were assessed in the 2009 Pickering B ISR.[2] These previous ISRs are considered to constitute the first PSR completed for Pickering (referred to as "PSR1"). The current PSR (referred to as "PSR2") is a subsequent PSR building on the basis of earlier OPG integrated safety assessments through review of the various studies, assessments and licence renewals performed since PSR1. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

PSR2 will support and complement the licence renewal application for Pickering NGS going forward. Fifteen Safety Factors will be assessed as part of the PSR. The purpose of Safety Factor reviews is to confirm that the design, condition and operation of Pickering Units 1,4 and 5-8 (as well as common systems) will support continued safe operation for the period of PSR2 by:

- Assessing compliance against "Review Tasks" identified in the PSR2 Basis Document [1], which were derived from Canadian Nuclear Safety Commission (CNSC) REGDOC-2.3.3 [2] and International Atomic Energy Agency (IAEA) SSG-25 [3];

- Documenting assessments against applicable modern Laws, Regulations, Codes and Standards (L/R/C/Ss) (as defined in Reference [1]); and

- Reviewing the effectiveness of applicable OPG Nuclear Programs that support the above assessments, through review of audit and self-assessment results.

---

1 Currently, Pickering Units 5-8 are approved to operate to 247,000 Effective Full Power Hours. This operation limit is expected to be reached on some units in 2020. For the purposes of PSR2, OPG assumes operation of Pickering NGS for up to eight additional years, from 2020 until 2028. OPG will make a decision regarding the permanent shut down dates for the six reactors following the performance of a technical evaluation that will include PSR2, and will communicate it to the CNSC as required by the current Power Reactor Operating Licence (PROL).

2 Much of the compliance assessment and evaluation of Safety Factor health for the Darlington ISR is based on programs and practices that apply across OPG's Nuclear operations. As a result, where Pickering is confirmed to follow the same Nuclear programs and practices as were assessed for Darlington, the Darlington ISR programmatic conclusions are applicable to Pickering. As discussed in Section 1.0, an effectiveness review (effectiveness at Pickering NGS) of OPG programs used to demonstrate compliance with the PSR2 Assessment Basis will be conducted using recent audit and self-assessment results.

As outlined in IAEA SSG-25 [3], the objective of the review of Hazard Analysis Safety Factor 7 is to: "determine the adequacy of protection of the nuclear power plant against internal and external hazards, with account taken of the plant design, site characteristics, the actual condition of the Structures, Systems and Components (SSCs) important to safety and their predicted state at the end of the period covered by PSR2, and current analytical methods, safety standards and knowledge". REGDOC-2.3.3 [2] requires that: "The licensee shall conduct a PSR in accordance with this regulatory document for the period until the next PSR or, if applicable, until the end of commercial operation of the plant."

This report documents the results of the review of Safety Factor 7 for Pickering PSR2. The report is based on the OPG Governance, Programs, data, and material available up to January 15, 2016 which is the freeze date for PSR2.

2.0 SCOPE OF REVIEW

2.1 Review Task Assessments

The Pickering PSR2 Safety Factor 7 Review Tasks are defined in Reference [1]. Details of the derivation of these Review Tasks from CNSC REGDOC-2.3.3 [2] and IAEA SSG-25 [3] are shown in Reference [5]. The Safety Factor 7 Review Tasks are:

1) Perform an assessment of the existing Deterministic and Probabilistic analyses to confirm existence of hazard analyses for hazards listed below. The following hazards are to be included in the assessment:

   (i) Internal Hazards:
   - Fire, Pipe whip, Steam release, Toxic gas, Flooding, Missiles, Spray, Explosion.

   (ii) External Hazards:
   - Changes in site characteristics, High winds (Tornado), Seismic, Toxic gas, Flooding, Extreme temperatures, Aircraft crash, Explosions.

2) Confirm that the analyses and/or methods take into account the plant design and the condition of SSCs important to safety (both at present and predicted for the end of the period covered by PSR2).

3) For each relevant hazard, verify, by means of current analytical techniques and data, that the frequency of occurrence and/or the consequences of the hazard are sufficiently low so that either no specific protective measures are necessary, or the preventive and mitigatory measures in place are adequate.

The methodology for the reviews is discussed in Section 3.1. Review Task findings are summarized in Section 4.1 of this report.

2.2 L/R/C/S Reviews

The applicable Laws, Regulations, Codes and Standards relevant to the Hazard Analysis Safety Factor are identified in Reference [1] and are listed in Table 1 below. Table 1 also identifies the modern version and date of each L/R/C/S to be considered, the Safety Factor(s) to which each document is applicable, and the type of review that will be completed in PSR2.

All of the Safety Factor 7 L/R/C/S reviews are incremental in nature. The definition of an Incremental Review is as follows:

- <u>Incremental Review</u>: For L/R/C/Ss that have been reviewed in PSR1 but have had revisions since the last review, a topical review will be performed of the changes.

The methodology for the reviews is discussed in Section 3.2. A detailed assessment for each L/R/C/S is provided in References [6] and [7]. Associated findings are summarized in Section 4.2 of this report.

Table 1: L/R/C/Ss Reviewed for the Hazard Analysis Safety Factor 7

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
L/R/C/Ss Referenced in Pickering NGS PROL 48.02/2018						
1	CSA N286.7	Quality Assurance Of Analytical, Scientific And Design Computer Programs	N286.7-16	1, 5, 6, 7, 10	Incremental	N286.7 addressed as part of Pickering B and Darlington ISRs.
2	CSA N293	Fire Protection for Nuclear Power Plants	N293-12	1, 7, 13	Incremental	N293 addressed as part of Pickering B and Darlington ISRs and PARTS.
3	CNSC REGDOC-2.4.1	Deterministic Safety Analysis	2014	5, 7	Incremental	C-6 addressed as part of the Pickering B ISR and PARTS. S-310 and RD-310 addressed as part of the Pickering B and Darlington ISRs, respectively. Implementation plan in place and gap assessment between REGDOC-2.4.1 and OPG Safety Analysis Program already performed.
4	CNSC REGDOC-2.4.2	Probabilistic Safety Assessment (PSA) for Nuclear Power Plants	2014	6, 7	Incremental	S-294 addressed as part of Pickering B and Darlington ISRs. Implementation plan in place.
Additional L/R/C/Ss						
5	CNSC REGDOC-2.5.2	Design of Reactor Facilities: Nuclear Power Plants	2014	1, 5, 6, 7	Incremental	RD-337 and NS-R-1 (precursors to REGDOC-2.5.2) addressed as part of Darlington ISR. NS-R-1 also addressed as part of Pickering B ISR.
6	CNSC REGDOC-2.3.2	Accident Management, Version 2	2015	1, 5, 6, 7, 8, 10	Incremental	REGDOC-2.3.2 addressed as part of Darlington ISR.

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
7	CSA N286.7.1	Guideline for the Application of N286.7-99, Quality Assurance of Analytical, Scientific, and Design Computer Programs for Nuclear Power Plants	N286.7.1-09	1, 5, 6, 7, 10	N/A [3]	N286.7.1 not addressed as part of Pickering B or Darlington ISRs.
8	CNSC G-149	Computer Programs Used in Design and Safety Analyses of Nuclear Power Plants and Research Reactors	2000	1, 5, 6, 7	Incremental	G-149 addressed as part of Pickering B and Darlington ISRs.

2.3 OPG Program Effectiveness Reviews

The OPG Nuclear Program (N-PROG) reviewed for Safety Factor 7 is listed in Table 2 below.[4] The methodology for the effectiveness reviews is discussed in Section 3.3. The assessment results of the N-PROG in Table 2 are provided in Appendix B, and findings are summarized in Section 4.3.

Table 2: OPG Program Reviewed for Safety Factor 7

Document Number	Document Title
N-PROG-RA-0012 [9]	Fire Protection

2.4 Additional Reviews

The PSR2 Safety Factor 7 Report includes a review of the R04 Pickering Licence Conditions Handbook (LCH) [4] for any impacts of Pickering NGS operation beyond 2020 on the following (all related to Safety Factor 7):

- OPG commitments previously made to the CNSC;
- Open CNSC action items; and
- Exemptions granted by the CNSC.

The PSR2 assessment includes identification and review of previously identified PSR1 gaps related to Safety Factor 7 to ascertain the implications of extending Pickering

[3] The N286.7.1 guide has been amalgamated into the new (-16) edition of the N286.7 Standard. The N286.7 CSA Impact Statement states [8]: "The CSA N286.7.1 guide will no longer be maintained after this new edition of N286.7 is issued. Any relevant guidance has been put into the new edition of N286.7." As a result, only the review of N286.7-16 has been prepared for PSR2.

[4] The list of Nuclear Programs to be assessed for effectiveness for PSR2 was derived from review of current OPG Governance. Although there may be content in Nuclear Programs that is applicable to multiple Safety Factors, N-PROG reviews are only provided in one Safety Factor report and are not duplicated.

NGS operation beyond 2020. The methodology for these reviews is described in Section 3.4. Any PSR2 gaps identified as a result of programmatic Darlington PSR1 gaps related to Safety Factor 7 are discussed in Section 4.4 of this report. The review of Pickering PSR1 gaps previously identified in the Pickering Units 5-8 Continued Operations Plan (COP) [10] is provided in a separate PSR2 COP Review Report.

In addition, Fukushima Action Items (FAIs) were reviewed to identify implications of extending operation beyond 2020 (if any). This review is presented in a separate PSR2 FAI Review Report.

Any PSR2 gaps identified as a result of the Safety Factor 7 review which are relevant to other Safety Factors are discussed in Section 4.4 of this report.

3.0 METHODOLOGY

The sub-sections below summarize the methodology used to assess Review Tasks, L/R/C/Ss, and Nuclear Program effectiveness for the Hazard Analysis Safety Factor.

3.1 Review Tasks

As discussed earlier, the Safety Factor Review Tasks are derived from CNSC REGDOC-2.3.3 [2] and IAEA SSG-25 [3], taking into consideration the Review Tasks used in the Pickering B and Darlington ISRs (as derived in [5]).

For each Safety Factor 7 Review Task identified in Section 2.1, a confirmation of the existence of applicable OPG Governance, Programs, Policies and Procedures (as well as Instructions and Guidelines, as applicable) was performed. Compliance against Review Tasks is also assessed by reference to applicable Condition Assessments, safety analyses and operating experience, as required.

The Review Task assessments identify Compliances and Gaps as defined below:

- Compliance: Compliance indicates that either the safety requirement or the intent of the Review Task is met.

- Gap: A Gap indicates that the intent of the Review Task is not met.

3.2 L/R/C/S Reviews

The process to identify the modern L/R/C/Ss that are applicable to the PSR2 Assessment Basis involved first creating a broad list from multiple sources (potential candidate L/R/C/Ss) and then filtering it to identify those that are most significant and that are applicable to the PSR2 scope. The identification and selection criteria are detailed in Reference [1]. The result of the identification and selection process was a set of modern L/R/C/Ss that became part of the "PSR2 Assessment Basis".

PSR2 is focused on the extension of Pickering NGS operations beyond 2020, and will conduct reviews against a baseline of past PSR1 work. As a subsequent PSR, PSR2 focuses on changes in requirements, plant conditions, operating experience and new information. Since PSR2 is an update of previous ISRs, it incorporates reviews of L/R/C/Ss that have occurred as new versions have been issued. Since this assessment is a subsequent PSR, the focus is on identifying differences between what was previously assessed and what is now different within the current Pickering PSR2 Assessment Basis. In general, these differences relate to:

- More recent (new or revised) L/R/C/S versions than what was previously assessed as part of PSR1;[5]

- Safety significant differences between Pickering and Darlington, if the Darlington ISR is the basis for the earlier assessment;

- Implications of extending Pickering NGS operation beyond 2020; and

- Safety significant differences between Pickering Units 1,4 and Units 5-8.

As described in Reference [1], L/R/C/S review types are clause-by-clause, high level or incremental. Most of the L/R/C/Ss in the PSR2 Assessment Basis receive incremental reviews since PSR2 is an update of previous PSR1 assessments and clause-by-clause or high level reviews for the majority of the L/R/C/Ss in the PSR2 Assessment Basis have already been completed. Implementation plans (including gap analyses or code-over-code reviews) also exist for the latest editions of many L/R/C/Ss. As a result, incremental review is also used in circumstances where a L/R/C/S in the PSR2 Assessment Basis was not assessed in previous PSR1 reviews but an implementation plan currently exists for compliance.

The PSR2 incremental reviews in this Report include an assessment of the intent of recent changes to the L/R/C/Ss on a topic or subject-matter basis where there is potential to impact nuclear safety. Incremental reviews provide:

- A summary of the purpose of the L/R/C/S;

- Pertinent background information about the current revision of the L/R/C/S that is being considered;

- Identification of which Safety Factor(s) are applicable to the current revision of the L/R/C/S;

- A description of which version(s) of the L/R/C/S were assessed for PSR1 (i.e., Darlington ISR (for programmatic content), Pickering B ISR and PARTS code reviews);

- Identification of whether the current version of the L/R/C/S is an update of a previous version of the L/R/C/S that was assessed in PSR1 (and if so, a description of the major changes in the latest revision is provided as discussed below);

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[5] "New" refers to a code or standard that was not previously considered in the context of earlier assessments. "Revised" refers to an updated version of a code or standard that was previously considered in the context of earlier assessments. Where a document has a new number/type, but addresses the same topic from the same organization, it is a "revised", not "new", document (e.g., if a REGDOC replaces a CNSC G or RD document).

- An assessment of the applicability of PSR1 assessment findings (gaps and conclusions), including the implications of extending Pickering NGS operation beyond 2020 if any;

- An assessment of the applicability of assessment findings that address more recent (post-PSR1) editions of the L/R/C/S, including any implementation or transition plans that are already committed to by OPG; and

- Where PSR1 and post-PSR1 assessments are not sufficient to address changes in the latest edition of the L/R/C/S, an assessment of the changes from the previously assessed edition of the L/R/C/S (including identification of any safety significant PSR2 gaps which result).

High Level reviews provide the same information as above, where applicable, in a similar format. However, given that High Level L/R/C/Ss generally have not received past assessment during PSR1, the Incremental review content is augmented by a high level, section-by-section assessment of the degree of conformance of Pickering NGS with the L/R/C/S (demonstrating, with supporting evidence, whether the intent of the requirements stipulated in the document are met).

There are currently no L/R/C/S clause-by-clause reviews identified in the PSR2 Assessment Basis.

The Safety Factor 7 L/R/C/S reviews identify Compliances and Gaps as defined below[6]:

- <u>Compliance</u>:

    - Where an Incremental review has been performed, Compliance indicates that the change in the safety requirement, per the topical review, is met.

    - Where a High Level review has been performed, Compliance indicates that the intent of the safety requirement is met. (Note: No High Level reviews were performed as part of Safety Factor 7.)

    - For Clause-by-Clause reviews of modern L/R/C/Ss, Compliance indicates that the safety requirement is met. (Note: No Clause-by-Clause reviews were performed as part of Safety Factor 7.)

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6 Safety Factor assessments for Review Tasks and L/R/C/S reviews make use of: a) OPG Governance, Programs, Policies and Procedures which support the assessment arguments, b) Commitments previously made to the CNSC, open CNSC action items, and exemptions granted by the CNSC (all related to the Safety Factor under review), as identified in the R04 Pickering LCH [4], c) Identification of previously identified Pickering-specific or programmatic PSR1 gaps related to the Safety Factor under review and the status of OPG's improvement plan(s) or other dispositions to address these, and d) Assessments and reviews performed since the PSR1 documents were completed.

- Gap:
    - Where an Incremental review has been performed, a Gap indicates that the change in the safety requirement, per the topical review, is not met.
    - Where a High Level review has been performed, a Gap indicates that the intent of the safety requirement is not met. (Note: No High Level reviews were performed as part of Safety Factor 7.)
    - For Clause-by-Clause reviews of modern L/R/C/Ss, a Gap indicates that the safety requirement is not met. (Note: No Clause-by-Clause reviews were performed as part of Safety Factor 7.)

The reviews assume that use of the word:

- "Shall" is used in an L/R/C/S to express a requirement, i.e., a provision that the licensee is obliged to satisfy in order to comply with the standard;
- "Should" is used to express a recommendation or that which is advised but not required;
- "May" is used to express an option or that which is permissible within the limits of the standard; and
- "Can" is used to express possibility or capability.

3.3 OPG Program Effectiveness Reviews

As discussed earlier, effectiveness reviews (effectiveness at Pickering NGS) of OPG programs used to demonstrate compliance with the PSR2 Assessment Basis were conducted, using recent applicable audit and self-assessment results:

- OPG Nuclear Oversight independent performance-based Program audits (typically performed in 1 to 5 year cycles) and self-assessments. This includes review of associated Station Condition Records (SCRs) and Action Requests (ARs) to determine the status of any resulting corrective actions; and
- CNSC "Type I" and "Type II" inspections of the effectiveness and performance of OPG programs, where discussed in OPG audits or self-assessments.

There are many audits and self-assessments that are performed to assess the effectiveness of important aspects of each program. A sample of audits and self-assessments has been summarized for each program in order to demonstrate that program effectiveness is being assessed on an ongoing basis. The focus of these reviews was on effectiveness of the programs at Pickering NGS, where specific information is available. Results from these audits and self-assessments will be considered in the Global Assessment process. It is noted that audits and self-assessments are, by their nature, self-critical and are used to drive excellence in

performance. As a result, the broad review scope of program audits focuses on identifying improvement opportunities rather than presenting a balanced picture of program performance.

Program effectiveness is also monitored and addressed through the Fleetview Program Health and Performance Reporting process [11]. This process involves direct oversight by the Chief Nuclear Officer, and includes participation by the Nuclear Executive Committee members. Programs are reviewed, senior oversight is provided, and improvement plans are generated.

The list of Nuclear Programs to be assessed for each Safety Factor was derived from review of current OPG Governance, and has used the most recent version of these documents as of the PSR2 freeze date of January 15, 2016.

3.4 Additional Reviews

A review of the R04 Pickering LCH [4] was performed to determine if there are any impacts associated with operation of the Pickering Units past 2020 on the following (all related to Safety Factor 7):

- Commitments previously made to the CNSC;
- Open CNSC action items; and
- Exemptions granted by the CNSC.

The PSR2 assessment includes identification and review of previously identified Pickering-specific or programmatic PSR1 gaps related to Safety Factor 7 (as identified in the Darlington ISR Integrated Implementation Plan [12] and Pickering Units 5-8 Continued Operations Plan [10]) to ascertain the status of OPG's improvement plan(s) or other dispositions to address these and the implications of extending operation beyond 2020 (if any).[7]

[7] PSR2 includes consideration and confirmation that the findings of PSR1 remain valid, as applicable, for the operation period. This includes assessment of PSR1 conclusions against implications resulting from extended operation. In particular, Pickering PSR1 results are applicable to PSR2 if there was a PSR1 gap that is still open, or if a closed PSR1 gap could be affected by extended operation. If so these gaps are carried forward into PSR2 for consideration in the Global Assessment. (When references to PSR1 are made, the source document is identified and the relevant text from that source document is summarized in the context of PSR2.) With respect to the Darlington ISR, much of the evaluation of Safety Factor health is based on programs and practices that apply across OPG's Nuclear operations. As a result, Darlington ISR programmatic conclusions are applicable to Pickering PSR2 where Pickering is confirmed to follow the same Nuclear programs and practices that were assessed for Darlington. Darlington PSR1 results are applicable to Pickering PSR2 if there are Darlington PSR1 gaps that are found to be relevant to Pickering PSR2.

Fukushima Action Items were reviewed to identify implications of extending operation beyond 2020 (if any). The methodology for this review is provided in a separate PSR2 FAI Review Report.

Any PSR2 gaps identified as a result of the Safety Factor 7 review which are relevant to other Safety Factors are also discussed.

4.0 REVIEW FINDINGS

4.1 Review Tasks

The sub-sections below provide an assessment of the adequacy of applicable OPG Governance, Programs, Policies and Procedures (as well as Instructions and Guidelines, as applicable) in demonstrating compliance against the Safety Factor 7 Review Tasks.

4.1.1 Review Task #1: Internal and External Hazards in Deterministic and Probabilistic Analyses

Perform an assessment of the existing Deterministic and Probabilistic analyses to confirm existence of hazard analyses for hazards listed below. The following hazards are to be included in the assessment:

(i) Internal Hazards:
- Fire, Pipe whip, Steam release, Toxic gas, Flooding, Missiles, Spray, Explosion.

(ii) External Hazards:
- Changes in site characteristics, High winds (Tornado), Seismic, Toxic gas, Flooding, Extreme temperatures, Aircraft crash, Explosions.

The scope of initiating events addressed by safety analysis was originally defined deterministically by Atomic Energy Control Board (AECB) Consultative Document C-006-R001 [13]. Deterministic reviews of each of the stations' respective safety analyses against the scope of CNSC Consultative Document C-006-R001 [13] were completed as part of Reference [14], in 2001 for Pickering Units 1,4, and as part of the Pickering B ISR [15] during 2007 for Pickering Units 5-8. These reviews covered over 110 initiating events for each station including fire, earthquakes, explosions, toxic gases, internal and external flooding, aircraft crash and extreme weather including high wind and temperature extremes, and found no significant safety implications for any of these hazards. They further determined all C-006-R001 [13] events to be adequately addressed as being bounded by events explicitly analysed in supplementary reviews, including the Pickering B risk assessment, or the existing Safety Reports [16], [17].

Subsequently, a list of Internal and External Events, for inclusion in the Level 1/Level 2 Probabilistic Safety Assessments (PSAs) for Pickering Units 1,4 and Pickering Units 5-8 was prepared as part of the Hazard Screening Analyses completed in 2012 [18], [19]. This list is compliant with both CNSC Regulatory Standard S-294 [20] and international practice (such as IAEA 50-P-7 [21] and US Nuclear Regulatory Commission NUREG/CR-2300 [22]) and includes discussion of pipe whip, missiles, toxic gas (external), flooding (external), extreme temperatures, aircraft impacts, explosions and various hazard combinations including these, and other hazards, occurring concurrently.

The External Hazards Screenings contained in these reports rely on the concept of Review Level Conditions (RLCs) for external hazards against which the station's current design is reviewed. Originally defined and analysed for various external hazards in the Hazard Screening Analyses [18], [19], the RLCs were further analysed and refined for tornado, seismic, and external flooding hazards in [39] and further applied to modifications in place for Beyond Design Basis accident considerations, as per [40].

When a probabilistic analysis of hazards is required it is conducted in accordance with standard N-STD-RA-0034 [41], which identifies the scope of the PSAs with respect to internal and external hazards as follows:

- Risks associated with internal initiating events shall be addressed using a PSA with a unit operating at high power and while shutdown, with the exception of risks associated with internal fires and internal floods while a unit is shutdown, which may be addressed, with agreement of persons authorized by the Commission, using alternate analysis methods.

- Risks associated with fuelling machine events and Irradiated Fuel Bay (IFB) events may be addressed using a PSA or, with agreement of persons authorized by the Commission, by using alternate analysis methods.

- Risks associated with external events (internal hazards and external hazards) may be addressed using a PSA or, with agreement of persons authorized by the commission, by using alternate analysis methods.

Consistent with the guidance above, the hazards from internal flooding and fire, along with seismic and high wind, have been assessed as part of standalone PSAs for each station. Other hazards, such as extreme temperature (also assessed by the Hazard Screening Analyses [18], [19]) are modelled in the PSAs of both stations [23]. All OPG PSA documents have been completed in accordance with the applicable generic OPG PSA guidelines specified in References [23] through [32], and the applicable Pickering Units 1,4 specific guides [33] through [38].

Further events analysed under this Review Task such pipe whip, steam release and changing site conditions are deterministically considered by inclusion in the most current revisions of both the Pickering Units 1,4 and Pickering Units 5-8 Safety Reports [16], [17], while others including certain internal floods and sprays have been factored into elements of the current EQ program N-PROG-RA-0006 [42]. The Safety Reports for each station are currently being updated to incorporate a common mode event appendix similar to analyses already contained in [18] and [19].

Each of the internal and external hazards are addressed separately below.

INTERNAL HAZARDS

Each internal hazard is identified at a high level in Table 3: Internal Hazards Summary Table, shown on the next page, which summarizes the approach used to address the hazard. A hazard specific discussion follows the summary table. The most recent Internal Hazard Screening Analyses are [18] and [19], for Pickering 1,4 and Pickering 5-8 respectively.

The PSA Internal Hazards Screening Guide [31] (applicable to both Pickering 1,4 and Pickering 5-8) provides guidance on the screening of internal hazards at OPG facilities for inclusion in or exclusion from more detailed PSA analysis. Consequential effects and adverse conditions (e.g., pressure, temperature, humidity, radiation, steam/water spray, vibration, etc.) are identified, when relevant, in the analyses and assessed against relevant mitigating provisions (e.g., environmental qualification/protection and operating capabilities).

Many of the internal hazards have been assessed as part of the Environmental Qualification (EQ) Program, N-PROG-RA-0006 "Environmental Qualification" [42], which provides the EQ governing policy and the overall EQ methodology for the qualification of environmentally qualified equipment/components. It also defines the qualification program for systems and components that must be qualified or protected from the adverse environmental effects of design basis events. Additional details on documents that define the complete framework of the EQ Program are provided in Safety Factor 3 Report: Equipment Qualification (Seismic and Environmental) [43].

Certain hazards also have been assessed in standalone deterministic analyses. One such example includes internal fires, which were analysed in Pickering Fire Safety Assessments [44], [45].

Table 3: Internal Hazards Summary Table

Hazard	Assessment Approach				
	Deterministic		Probabilistic		Notes
	Pickering Units 1,4	Pickering Units 5-8	Pickering Units 1,4	Pickering Units 5-8	
Internal					
Fire	Yes	Yes	Yes	Yes	• Compliant with "fire protection" program N-PROG-RA-0012 [9]. • Part of PSA updates as per [36], [27]. • Pickering Fire Safety Assessments [44], [45]. • Included in existing PSA [46], [47].
Pipe Whip	Yes	Yes	N/R See Note 8	N/R See Note 8	• Discussed in Section 6.5 "release of stored energy" [18], [19]. • Safety Reports [48], [49]. • Included as CNSC CANDU Safety Issue CSI-IH 6 [50].
Steam Release	Yes	Yes	Yes	Yes	• Safety Reports [48], [49] and EQ program [42]. • Steam release events are included in the scope of the Level 1 PSAs [23].
Toxic Gas	See Note 9	See Note 9	N/R See Note 8	N/R See Note 8	• Discussed in various sections of [18], [19].
Flooding	Yes	Yes	Yes	Yes	• Safety Reports [16], [51]. • EQ program [42], [52], [53]. • Part of PSA updates as per [37], [28]. • Included in existing PSA [54], [55].
Missiles	See Note 9	See Note 9	N/R See Note 8	N/R See Note 8	• Mechanical Missile Impacts – Section 6.2 of [18], [19].
Spray	Yes	Yes	Yes	Yes	• Addressed in design as part of the EQ program [42]. • Spray events are captured as part of the PSA flood consequence analysis as per [28].
Explosion	See Note 9	See Note 9	N/R See Note 8	N/R See Note 8	• Explosions within the Generating Station – Section 6.3 of [18], [19].

N/R = Not Required

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[8] The potential for these hazards was assessed in the Hazard Screening Analyses [18], [19] and screened out of the PSA.

[9] The Hazard Screening Analyses are done as part of the PSA program to determine which events need to be subjected to further PSA assessment. However, the hazards are evaluated based on a deterministic evaluation of the consequences and the frequency of occurrence and therefore, the screening assessments also demonstrate that the hazards have been identified and deterministically evaluated.

Fire

An overall program, N-PROG-RA-0012 "Fire Protection" [9], has been established to manage Fire Protection at OPG nuclear plants. This program identifies the processes, overall requirements, and staff accountabilities to ensure that an effective Fire Protection Program is established and maintained, thereby minimizing both the risks and consequences of fire. This document also describes the nuclear governing documents required to achieve the fire protection goals of Canadian Standards Association (CSA) N293 "Fire Protection for Nuclear Power Plants".

Regular fire inspections of accessible areas to ensure conformance with fire prevention regulations are conducted in accordance with N-PROC-RA-0054, Control of Transient Materials and Space Allocation Within the Site (Reference [56]), and N-PROC-RA-0057, Control of Ignition Sources And Hot Work Activities (Reference [57]), along with N-PROG-MA-0004, Conduct of Maintenance (Reference [58]), and N-STD-RA-0026, Fire Protection Surveillance Availability and Compensatory Measures (Reference [59]).

Deterministic fire hazard analyses were completed in the Fire Hazard Assessment (FHA) and the Fire Safe Shutdown Analysis (FSSA), which were both conducted to criteria based on N293-95. The FHA is an area by area assessment of plant fire risks and consequences while the FSSA is an assessment of the impact of fire on nuclear safety. These two reviews were combined into a Fire Safety Assessment and submitted to the CNSC on September 29, 2000 [44] for Pickering Units 1,4 and on December 9, 2011 [45] for Pickering Units 5-8. The initial FHA evaluations reviewed the plant on a fire area / zone basis for in situ fire hazards while the FSSAs evaluated the capability to achieve and maintain the shutdown state with postulated fire damage.

Fire related hazards are also in the scope of both the Pickering Units 1,4 and Pickering Units 5-8 PSAs as documented in general guidance for the preparation of internal Fire PSAs [36], [27]. The most current fire PSA analyses for both stations are documented in [46] and [47].

Pipe Whip

Pipe ruptures in Pickering NGS have been assessed for their potential to cause consequential damage to other components, which could then jeopardize the safe shutdown and continued cooling of the reactor. Pipe-whip effects are assessed as consequential failures in Section 6.5 "Release of Stored Energy" of the Hazard Screening Analyses [18], [19] for Pickering Units 1,4 and Pickering Units 5-8 respectively. The reports conclude that critical components or equipment, particularly for the Primary Heat Transport (PHT), Shutdown System (SDS) and Emergency Coolant Injection (ECI) systems have been analysed for pipe whip potential at varying temperatures as per [60], [61], [62], [63] and [64].

Pipe whip effects are also discussed in the Internal Flooding Probabilistic Risk Assessment (PRA)[10] guide [28]. The Internal Flood PRAs, [54], [55], consider Pipe Whip; however, they defer to deterministic rules for screening distances and deterministic analysis for assessment of consequences. Pipe whip has been deterministically assessed in the analyses discussed below.

Part 3 of the Pickering Units 1,4 and Pickering Units 5-8 Safety Reports [48], [49] address consequential damage resulting from pipe-whip jet forces. The assessments presented similarly confirm that these hazards do not result in unacceptable consequences despite potential damage to in-core structures, damage due to deflection of a postulated failed fuel channel or damage to neighboring fuel channels and impairment of the insertion of the shutoff rods (Section 4.5.4 of the Pickering Units 1,4 and Pickering Units 5-8 Safety Reports [48], [49]).

A CNSC Category 3 CANDU Safety Issue (CSI), CSI-IH 6 regarding Pipe Whip, was determined to require further assessment as per [50]. A methodology for the assessment of pipe-whip and jet-impingement of high-energy piping inside the reactor buildings of Pickering Units 1,4 and Pickering Units 5-8 was subsequently developed [65]. An assessment for Pickering Units 5-8 based on this methodology, Reference [67], was completed in 2016 and confirmed that the layout of the high-energy piping and safety related systems inside of the reactor buildings of Pickering Units 5-8 are in compliance with the practices, expectations, and guidelines in modern standards such as IAEA NS-G-1.11 [68]. To address CNSC CANDU Safety Issue CSI-IH 6, the potential for, and possible impacts of, high-energy piping failures in Pickering Units 1,4 must also be assessed. This assessment is currently underway as per Reference [65] and has a target completion date in mid-2017 [66]. This item is therefore identified as a gap for Pickering PSR2 **(Pickering PSR2 Gap SF7-1)**.

Steam Release

Steam line failures are presented in Appendix 7 of the Pickering Units 1,4 and Pickering Units 5-8 Safety Reports, References [48] and [49] respectively. Various failures are presented that are associated with the steam systems as the result of control system failures, process upsets or, in the extreme, postulated catastrophic piping ruptures. The report presents the results of this analysis of the steam supply system failures in terms of safety system initiation, thermal hydraulic response, powerhouse environment assessment and radiological consequences.

The Environmental Qualification Program N-PROG-RA-0006 "Environmental Qualification" [42] provides assurance that equipment and components will perform their safety-related functions when exposed to harsh environmental conditions resulting from a Design Basis Accident (DBA).

As per the OPG PSA guidelines specified in [23], several pipe breaks, such as LOCAs and steam line breaks, are modelled as dependent initiating events (i.e. events which not only initiate a transient but may also contribute to failure of a mitigating system),

[10] Probabilistic Risk Assessment (PRA) is now referred to as Probabilistic Safety Assessment (PSA).

which impact a wide range of equipment due to the physical interactions with the associated hostile steam environment. The environmental impacts associated with these events are included as part of the station's Level 1 PSAs [69], [70].

Toxic Gas

Internal toxic gas hazards due to on site transportation accidents have been screened out based on low frequency as per Section 6.7.3 of the Hazard Screening Analyses for both stations [18], [19].

Similarly, the release of toxic gases from on-site storage has been assessed in Appendix A Table A1 of Hazard Screening Analyses [18], [19]. Of all the chemicals stored on site, only the five listed below have the potential to result in toxic effects and are stored in sufficiently large quantities such that further analysis was required. As detailed in the Hazard Screening Analyses [18], [19], each substance was then analysed further and screened out independently as follows:

- Sodium Hypochlorite: Screened out as a result of its propensity to react to form a neutral solution (stored in the Chlorine Building, south of the Powerhouse and the Reactor Auxiliary Bay).
- Hydrogen Peroxide: Screened out because only 35% concentrations are used on site while concentrations below 50% are only considered an irritant.
- Morpholine: Screened out based on maximum postulated concentrations in the control room being below the Immediately Dangerous to Life and Health (IDLH) concentrations[11].
- Hydrazine: Screened out based on the fact that advance warning would provide adequate time for individuals in the main control room to evacuate and/or don respiratory protective equipment before the IDLH is exceeded.
- Sodium Metabisulphite: Stored outside in temporary storage tanks (not in the Powerhouse or the Reactor Auxiliary Bay) and is therefore bounded by on site transportation accidents, which are screened out by frequency as per above.

The storage and movement of toxic materials on site are further controlled in accordance with OPG's hazardous materials management procedure [71] and Space Allocation for Transient Material (SATM) requirements [72]. These procedures outline requirements for both the storage and transit of toxic materials on site to minimize the likelihood and consequence of spills and/or dangerous reactions.

[11] As per the Hazard Screening Analyses [18], [19] the IDLH concentrations represent the values at which, in the event of respiratory failure, a worker could evacuate within 30 minutes without experiencing any escape-impairing or irreversible health effects.

Flooding

DBA conditions addressed by the EQ Program are a subset of service conditions that are addressed as part of equipment qualification. As stated in Section B.5.0 of N-PROG-RA-0006 [42], a flooding hazard inside containment for DBAs is considered as part of the equipment qualification program. Post-accident flooding conditions are included in the Room Conditions Manuals (RCMs) of both Pickering Units 1,4 and Pickering Units 5-8, References [52], [53] respectively, as mandated by EQ RCM Nuclear Instruction N-INS-03651-10003 [73].

Per Part 2, Section 1.3.3 of both the Pickering Units 1,4 and Pickering Units 5-8 Safety Reports [16] and [51] respectively, sufficient mitigating systems are designed to operate effectively even if consequential effects of the initiating event pose a threat to their operability. Such effects include harsh environments due to steam and water discharge (causing flooding), thrust forces on piping, and jet impingement, following a pipe break.

Flood related hazards are addressed in the Pickering Units 1,4 and Pickering Units 5-8 PSAs, and have been prepared based on the guidance for the preparation of internal flood PSAs [37], [28] for Pickering Units 1,4 and Pickering Units 5-8 respectively. The most current flood PSAs for both stations reflect flooding due to various sources including high energy secondary side pipe failures and service water pipe failures at different break locations, and are documented in [54] and [55].

External flooding hazards are assessed in the external hazards section below.

Missiles

The consequences of internally generated missiles are not currently discussed in the Pickering Safety Reports but have been assessed in Section 6.2 of the Hazard Screening Analyses [18], [19] for Pickering Units 1,4 and Pickering Units 5-8 respectively. Possible missiles originating from turbine disintegration, the failure of various valves and pumps, and the explosion of acetylene bottles used for cutting and welding were assessed. The failure of turbines, pumps, and valves were screened out due to low impact after considering the separation of independent systems and the inclusion of protective barriers in the Pickering design, while missiles due to acetylene bottle explosions were screened out due to low frequency.

Spray

The spray hazard inside containment following Design Basis Accidents is addressed as part of the equipment qualification program, N-PROG-RA-0006 [42]. Additionally as per Part 2, Section 1.3.3 of both the Pickering Units 1,4 and Pickering Units 5-8 Safety Reports [16] and [51], respectively, sufficient mitigating systems are designed to operate effectively even if consequential effects of the initiating event pose a threat to their operability. Such effects include harsh environments due to steam and water discharge, thrust forces on piping, and jet impingement, following a pipe break and its subsequent spray.

Certain spray events are also captured as part of the flood consequence analysis included in the internal flooding PSA. The internal flooding PRA guide [28] requires that the impact on equipment be evaluated for failures by submergence, jet impingement, spray, humidity, condensation and temperature. It provides further general guidelines to assume that all unprotected electrical equipment will fail within a minimum horizontal distance of 3 m for liquid flood sources (or 6 m for high energy flood sources). The flooding PSAs for both Pickering Units 1,4 and Pickering Units 5-8 conservatively assume a variety of spray induced failures.

Explosion

The consequences of internally generated explosions have been assessed in Section 6.3 of the Hazard Screening Analyses [18], [19] for Pickering Units 1,4 and Pickering Units 5-8 respectively. Possible hydrogen explosions in battery cells and in the turbine generator were analysed, as well as acetylene bottle explosions. Battery cell and acetylene bottle explosions were both determined to be improbable events and screened out by frequency while a hydrogen explosion in the turbine generator was deemed to be inconsequential due to adequate dispersion of hydrogen in the turbine hall and sufficient ventilation.

EXTERNAL HAZARDS

Each external hazard is identified at a high level in the Table 4: External Hazards Summary Table, shown on the next page, which summarizes the approach used to address the hazard, and is then individually addressed in a separate discussion. The most recent External Hazard Screening Analyses are shown in [18] and [19] for Pickering Units 1,4 and Pickering Units 5-8 respectively.

The screening of external hazards relies on the concept of Review Level Conditions (RLCs) against which the station's current design is reviewed. Originally defined and analysed for various external events in the Hazard Screening Analyses [18], [19], the RLCs were further analysed and refined for tornado, seismic, and external flooding hazards in [39].

The PSA External Hazards Screening Guide [30] (applicable to both Pickering Units 1,4 and Pickering Units 5-8) provides guidance on the screening of external hazards at OPG facilities for inclusion in or exclusion from more detailed PSA analysis.

Table 4: External Hazards Summary Table

Hazard	Assessment Approach				
	Deterministic		Probabilistic		Notes
	Pickering Units 1,4	Pickering Units 5-8	Pickering Units 1,4	Pickering Units 5-8	
External					
Changes to Site Characteristics	See Note 12	See Note 12	N/R See Note 13	N/R See Note 13	• Impacts captured in development of models and methodology for hazard analyses. • [18], [19] consider the proximity of industries, shipping lanes, aircraft, etc.
High Winds (Tornado)	N/R See Note 14	N/R See Note 14	Yes	Yes	• Not screened out at either Pickering Units 1,4 or Pickering Units 5-8 [18], [19]. • Part of PSA updates as per [32]. • Included in existing PSA [80], [81].
Seismic	Yes	Yes	Yes	Yes	• Seismic Margin Assessments [77], [78], [82], [83]. • Part of PSA updates as per [29], [38]. • Included in existing PSA [82], [83].
Toxic Gas	See Note 12	See Note 12	N/R See Note 13	N/R See Note 13	• Addressed in various sections of [18], [19].
Flooding	Yes	Yes	N/R See Note 13	N/R See Note 13	• Safety reports [16], [51]. • Assessed in [74] and [75]. • Discussed in Section 4.4 "Flooding" of [18], [19].
Extreme Temperatures	See Note 12	See Note 12	Yes	Yes	• Discussed in Section 4.5 "Meteorological – Extremes" [18], [19]. • Currently modeled into the PSA of both stations.

[12] The Hazard Screening Analyses [18], [19] are done as part of the PSA program to determine which events need to be subjected to further PSA assessment. However, the hazards are evaluated based on a deterministic evaluation of the consequences and the frequency of occurrence and therefore, the screening assessments also demonstrate that the hazards have been identified and deterministically evaluated.

[13] The potential for these hazards was assessed in the Hazard Screening Analyses [18], [19] and screened out of the PSA.

[14] Wind loading was addressed as part of the building code requirements used during the construction of each station but not analysed as a Design Basis Event. High wind events were however, already included in each stations' PSA prior to the creation of the Hazard Screening Analyses [18], [19] and hence not screened out in these reports.

Hazard	Assessment Approach				
	Deterministic		Probabilistic		Notes
	Pickering Units 1,4	Pickering Units 5-8	Pickering Units 1,4	Pickering Units 5-8	
External					
Aircraft Crash	See Note 12	See Note 12	N/R See Note 13	N/R See Note 13	• Discussed in Section 3.1 "Aircraft Impact" [18], [19].
Explosions	See Note 12	See Note 12	N/R See Note 13	N/R See Note 13	• Addressed in various sections of [18], [19].

N/R = Not Required

Change in Site Characteristics

Land use surrounding the Pickering site is discussed in the Safety Reports [16], [17]. Part 1, Section 2 of the Pickering Units 1,4 and Pickering Units 5-8 Safety Report discusses the site location and access, surrounding populations, land use, local agriculture, industries, fishing, recreation and transportation. Any changes in site characteristics with an impact on nuclear safety would be captured during the Safety Report updates. The impacts of changes in the site characteristics would then be captured in the development of models and methodology for external hazard analyses.

The Hazard Screening Analyses [18], [19], also consider the proximity of certain industries to the site and the effects they could have on both the probability of and severity of various hazards. These include:

- the location and use of airports to screen out certain aircraft impact hazards (discussed as a separate hazard below);
- the proximity of wharfs and commercial shipping activities to rule out possible shipping accidents; and
- the proximity of highways, railways or fixed sources of compressed gas and how they influence explosion and toxic gas hazards.

The effects of weather on site characteristics, e.g., soil failures (slope instability, subsidence, soil frost), the path of flood runoff or the build-up of snowpacks, are also considered and screened out within these reports.

High Winds (Tornado)

A tornado was not considered a Design Basis Event during the design of either Pickering Units 1,4 or Pickering Units 5-8, but high winds were addressed as part of the building codes used during the construction of each station.

Tornado and hurricane winds were assessed in the Hazard Screening Analyses [18], [19], for Pickering Units 1,4 and Pickering Units 5-8 respectively. High wind hazards (tornados and straight line winds) are in the scope of both the Pickering Units 1,4 and

Pickering Units 5-8 PSAs as documented in the general guidance for the preparation of high wind hazard PSAs [32], applicable to both Pickering Units 1,4 Pickering Units 5-8. The most current high wind PSA analyses for both stations are documented in [80] and [81].

Seismic

Seismic assessment of Pickering Units 1,4 was performed following the design and construction stages. Although the Pickering Units 1,4 systems were not originally required to be seismically qualified, the equipment was evaluated and a Safe Shutdown Equipment List was defined by the 1998 Seismic Margin Assessment [77], [78], which identified the SSCs required for the seismic success path. A PSA-based seismic margin assessment was subsequently completed for Pickering Units 1,4 in [82].

Seismic qualification of Pickering Units 5-8 was established during the design and construction stages through the implementation of the overall safety design requirements documented in Engineering Design Guide DG-30-68000-2 [79]. A PSA-based seismic margin assessment was subsequently completed for Pickering Units 5-8 in [83].

There are a number of governing documents used to establish and maintain the Seismic Qualification of safety related SSCs. These are:

- N-PROG-MP-0009, "Design Management" [85];
- N-PROG-MP-0001, "Engineering Change Control" [86];
- N-PROG-MA-0004, "Conduct of Maintenance" [58];
- N-PROG-MP-0008, "Integrated Aging Management" [87];
- N-PROC-MA-0024, "System Performance Monitoring" [88]; and
- N-STD-MP-0025, "General Requirements for Seismic Qualification of OPG Nuclear Facilities" [89].

The specific applications for Seismic Qualification as captured in these programs are described in the Safety Factor 3 Report: Equipment Qualification (Seismic and Environmental) [43].

Toxic Gas

Toxic gas releases due to rail transportation, road transportation, fixed sources and onsite vehicle movements have all been assessed in the Hazard Screening Analyses [18] and [19] for Pickering Units 1,4 and Pickering Units 5-8 respectively. Specifically, potential toxic chemical releases originating from the following sources were examined:

- One of the two rail lines that run “east-west” directly north of the Pickering NGS site: The CN Rail mainline runs north of the Pickering nuclear site, approximately 2.5 km north of the two powerhouses and is considered in the hazard screening assessment; the CP Rail mainline is located approximately 6 km north of the site, beyond the screening distance value for rail hazards and is not considered further in [18] and [19].

- Highway 401, which is no closer than 2.5 km away from the Pickering site.

- All possible external fixed sources of hazardous gases identified within a radius of 5 km of the Pickering site.

- Toxic Chemicals regularly transported to the site and stored in bulk on site.

The Hazard Screening Analyses [18], [19] present a conservative assessment of the consequences of possible accidents originating from each of these sources and conclude that any accidents of consequence to either of the Pickering stations can be screened out based on extremely low frequencies.

Flooding

External flooding was considered in the designs of both Pickering 1,4 and Pickering 5-8 per Part 2 Section 2.2 of their respective Safety Reports [16], [51]. The overall flood protection system for the Pickering site consists of the shoreline breakwater works, catch basins, and storm sewers. The ground surface elevation of the Pickering site is above the 1 in 200 year lake water level. In the event that overtopping of flood protection occurs due to a high magnitude storm coupled with high Lake Ontario level, lake water will be collected in the catch basins while the storm sewers will discharge this water either directly back to the lake, or to the intake channel.

Flood assessments in support of Pickering A Return to Service completed in 2002 and 2003 (References [74] and [75] respectively) assessed the potential for flooding of the Pickering site after a 1 in 100 year rainfall and a storm similar to the Hurricane Hazel storm in 1954. These assessments determined that there was potential for some flooding at Pickering NGS under certain extreme weather conditions, such as 21 cm of ponding for about 40 minutes as a result of a 100 year rainfall event coupled with wave overtopping. However, the assessments concluded this flooding was deemed to not represent a hazard to the station given that “all critical components and systems required for safe operation of the plant are either too far away or high above the critical catchment areas to be susceptible to any water damage”.

The consequences of external flooding events were further assessed in Section 4.4 of the Hazard Screening Analyses for both Pickering Units 1,4 and Pickering Units 5-8 [18], [19], which assessed the more severe Probable Maximum Precipitation (PMP) as defined in [93]. The PMP is significantly more severe than those events assessed in the previous studies and is discussed further below.

The hazard screening reports assessed a number of flooding sources and combination of sources for inclusion into the station PSA, including the following:

- Flooding due to runoff;
- Flooding due to river;
- Flooding due to waves;
- Flooding due to seiche (a large standing wave);
- Flooding due to tsunami;
- Flooding due to releases of water from natural or artificial storage; and
- Flooding due to ice-jamming.

The majority of these events were screened out for both Pickering Units 1,4 and Pickering Units 5-8 as a result of these assessments. As a result of the Fukushima event and follow-up, a flood hazard impact assessment for the PMP event was conducted for the Pickering site as described in the Hazard Screening Analyses [18], [19]. Based on this impact assessment, flooding due to runoff could not be screened out for Pickering Units 1,4. This is described further below.

Additional assessments, including site walkdowns, were completed as a follow-up to the Fukushima event in 2011, as discussed in Reference [94]. These assessments identified that the installation of new fences on site had the potential to negatively impact potential flood levels. Based on this, OPG proactively designed and installed flood protection barriers to prevent water ingress to the Standby Generator Fuel Forwarding Pump House, should potential flood conditions materialize. The as needed use of these barriers is now documented in NA44-OP-54600-0016 [91] as called for by N-PROC-RA-0095 [92], which documents severe weather emergency preparedness.

Given the findings of the qualitative impact assessment of the PMP event described above, a quantitative assessment of the PMP event was performed for the area that includes the Pickering Units 1,4 Standby Generator Fuel Forwarding Pump House and Standby Generators. The results demonstrated that flood threshold elevations for the key equipment in the Pickering Units 1,4 SG area will not be exceeded provided the overland flow paths and Intake Forebay outlet are not blocked following the PMP event [94].

The Fukushima Action Item Status Report [95] summarizes the work performed to evaluate external events at Pickering NGS as part of S-294 compliance activities. As part of this work, OPG also provided the rationale for screening out external flooding for Pickering Units 1,4, which was accepted by the CNSC in Reference [96].

In 2015, the industry, in coordination with the CANDU Owners Group (COG) Technical Committee, initiated a work package to revisit the treatment of external flooding analysis methodologies and potentially redefine the PMP event. This work is currently being conducted under COG work package 23207, detailed in Reference [76]. Once complete, the results will be evaluated with respect to impact on Pickering and addressed as required.

Extreme Temperatures

Both high and low temperature extremes have been assessed in Section 4.5 of the Hazard Screening Analyses [18], [19] for Pickering Units 1,4 and Pickering Units 5-8 respectively.

For low temperatures these reports state that per the original versions of the Pickering 1,4 and Pickering 5-8 Safety Reports, the stations were built to withstand -20°C and -22°C respectively, both of which are greater than the RLC temperature of -40.0°C, which defines the minimum potentially credible temperature the stations could experience as per the Hazard Screening Analyses [18], [19]. Nevertheless the assessments in the Hazard Screening Analyses conclude that the effects on plant safety are negligible because the Class III power gas turbine generators are capable of a black-start during the RLC. The station itself would be heated through extraction steam and, in the event sufficient process steam is unavailable, through a fossil fuel based Auxiliary Boiler System powered by off-site Class IV power. Further, if there is a concurrent loss of grid event, the Auxiliary Power System (APS) will be able to supply Class IV power for auxiliary boiler powerhouse heating, thus mitigating the effects from low temperatures [18], [19].

Similarly both Pickering Units 1,4 and Pickering Units 5-8 were built to a maximum temperature of 30°C, equivalent to the National Building Code of Canada requirement, which is below the RLC of 46.6°C, which defines the maximum credible temperature the stations could experience as per the Hazard Screening Analyses [18], [19]. Nevertheless, at this maximum temperature it is expected that the Heating, Ventilation and Air Conditioning (HVAC) systems would provide sufficient cooling to maintain equipment rooms within design limits in both stations. Should a shutdown be required, it is improbable that power available through Class IV, APS, Standby Generators (SGs) and Emergency Power Generators (EPGs) (in the case of Pickering Units 5-8 only) would be insufficient for a safe shutdown and removal of decay heat [18], [19]. The EQ program [42] looks at conditions that are much more severe than the RLC and ensures the availability of safety significant equipment to function.

As per the Hazard Screening Analyses [18], [19], the effects of both high and low temperatures are built into the Level 1 PRA as per N-GUID-03611-10001 Volume 1 [23], Section 2.4.8.3, which provides specific failure probabilities for equipment susceptible to temperature extremes over different temperature ranges.

Aircraft Crash

The major hazard from air transport is the possibility of an accidental impact of a large aircraft on the station. As indicated in the Hazard Screening Analyses [18], [19], a study of this possibility at Pickering NGS has shown that the probability of this event is acceptably small given the current level of air traffic in the area.

Explosions

The consequences of external explosions were analysed in Sections of 3.2 and 3.3 in both Hazard Screening Analyses [18], [19] for Pickering Units 1,4 and Pickering Units 5-8 respectively. Specifically, explosions originating from the following sources were examined:

- One of the two rail lines that run "east-west" directly north of the Pickering NGS sites: the CN Rail mainline runs north of the Pickering nuclear site, approximately 2.5 km north of the two powerhouses and is considered in the hazard screening assessment; the CP Rail mainline is located approximately 6 km north of the site, beyond the screening distance value for rail hazards and so is not considered further in [18] and [19].

- Highway 401, which is no closer than 2.5 km away from the Pickering site.

All events originated in areas outside of their respective Screening Distance Value provided in [30] and were therefore screened out based on distance.

Conclusion:

Based on the assessment above, it is confirmed that hazard analyses exist for the internal and external hazards listed in the Review Task with the following exception:

- To address CNSC Category 3 CANDU Safety Issue CSI-IH 6, the potential for, and possible impacts of, high-energy piping failures in Pickering Units 1,4 must be assessed. This assessment is currently underway [65], [66]. This is therefore identified as a gap for Pickering PSR2 (**Pickering PSR2 Gap SF7-1**).

4.1.2 Review Task #2: Consideration of Plant Design and Condition of SSCs

Confirm that the analyses and/or methods take into account the plant design and the condition of SSCs important to safety (both at present and predicted for the end of the period covered by PSR2).

N-PROC-MP-0086 [90], "Safety Analysis Basis and Safety Report Updates", defines the Safety Analysis Basis and describes the established practices to ensure that the Safety Analysis Basis is maintained. This includes a description of the process for potential changes to the safety analysis portion of the Safety Report, and provides the necessary instructions for raising and initial processing of issues, processing of inputs, prioritizing and performing analysis, physically updating the Safety Reports on a periodic basis ([97] to [100]), closing the issues and reporting to the CNSC. In accordance with the operating license, it is a requirement per CNSC Standard REGDOC-3.1.1 , to update each of the stations' Safety Reports at least once in every five year period [16], [17], hence ensuring the most current analyses and present plant design/condition is reflected.

The current plant design and the condition of SSCs important to safety used in deterministic safety analysis is determined by the implementation of N-PROG-MP-0014, "Reactor Safety Program" [101], and its implementing and interfacing documents. These documents govern management of issues related to nuclear safety analysis and their impact on safe operation. Interfacing documents include, but are not limited to, OPG's Environmental Qualification [42], Integrated Aging Management [87], Engineering Change Control [86], Equipment Reliability [102] and Component and Equipment Surveillance [103] programs, which assure that both the present and predicted condition of SSCs important to safety are maintained as follows:

- N-PROG-MP-0008, "Integrated Aging Management"

  Ensures the condition of critical Nuclear Power Plant equipment is understood and that required activities are in place to ensure the health of these components and systems while the plant ages. The program also requires preparation of life cycle management plans and condition assessments for critical plant equipment.

- N-PROG-RA-0006, "Environmental Qualification"

  Provides auditable assurance that equipment, to which Safe Operating Envelope (SOE) requirements apply, will perform its intended function when exposed to harsh environmental conditions resulting from a DBA, and that this capability is preserved over the life of the station.

- N-PROG-MP-0001, "Engineering Change Control"

  Specifies the requirement for risk evaluation for engineering changes based on reactor safety criteria. Preliminary and detailed safety analysis is required to be conducted under the reactor safety program if the proposed engineering changes would impact the Safety Analysis Basis.

- N-PROG-MA-0026, "Equipment Reliability"

  Establishes requirements for system monitoring and reporting to ensure that equipment, to which SOE requirements apply, will function as intended following DBAs.

- N-PROG-MA-0017, "Component and Equipment Surveillance"

  Establishes an integrated and comprehensive surveillance monitoring process to ensure components and equipment continue to meet the SOE requirements.

Facility-specific PSA updates are completed during regular review cycles, or when warranted such as by a major design change [104], hence ensuring the current condition of each station is reflected in all PSA related analyses. Specifically, N-STD-RA-0034 [41], "Preparation, Maintenance and Application of Probabilistic Risk Assessment", provides requirements for the preparation, maintenance and application of PSA at OPG Nuclear facilities including the required standard elements that have been developed to meet the intent of CNSC Regulatory Standard S-294, Probabilistic Safety Assessment for Nuclear Power Plants [20]. N-STD-RA-0034 further clarifies that "*PRA shall reflect the current configuration of the facility*" and that "*PRA maintenance shall ensure that changes to the design, operation and maintenance of the facility are reflected in the PRA*".

Conclusion:

The conclusion of this Review Task assessment is that the analyses and/or methods take into account the plant design and the condition of SSCs important to safety (both at present and predicted for the end of the period covered by PSR2). The intent of Review Task #2 is met and therefore Pickering NGS is compliant.

4.1.3 Review Task #3: Hazard Frequencies and Consequences

For each relevant hazard, verify, by means of current analytical techniques and data, that the frequency of occurrence and/or the consequences of the hazard are sufficiently low so that either no specific protective measures are necessary, or the preventive and mitigatory measures in place are adequate.

Both internal and external hazards are screened based on probability and/or impact as per OPG PSA guidelines contained in [30], [31] and [35]. As per the detailed discussions included in Section 4.1.1, the most recent Hazard Screening Analyses completed in 2012 for Pickering Units 1,4 and Pickering Units 5-8, [18] and [19] respectively, screened out internal hazards related to toxic gas, missiles and explosions as well as external hazards related to toxic gas, flooding (with the exception of flooding due to runoff at Pickering Units 1,4), aircraft crashes and explosions. Internal fire, internal flooding, seismic, high wind and ambient temperature extremes however are all explicitly addressed in the most current station PSA in accordance with the requirements outlined in PRA standard N-STD-RA-0034 [41], and its subsidiary guidelines [23] through [38]. The current PSA analyses, including [46], [47], [54], [55], [80], [81], [82] and [83], confirm that the current predicted severe core damage frequencies for each hazard are below the OPG safety goals.

The survivability of equipment exposed to steam releases, internal flooding and sprays are factored into elements of the current EQ program N-PROG-RA-0006 [42] while the effects of pipe whip, steam release and changing site conditions are all within each station's safety analysis basis by way of their consideration in the most current revision of the Safety Report for each station.

Furthermore, the Fukushima Action Items and Emergency Mitigating Equipment (EME) provisions documented in [84], will provide additional defence in depth measures (e.g., monitoring, power supplies and fuel cooling) that will further reduce the consequences of many of the hazard analyses discussed above.

Conclusion:

The conclusion of this Review Task assessment is that the frequency of occurrence and/or the consequences of each relevant hazard (with the exception of one PSR2 gap identified in Review Task #1) are sufficiently low such that either no specific protective measures are necessary, or the preventive and mitigatory measures in place are adequate. The intent of Review Task #3 is met and therefore Pickering NGS is compliant.

4.2 L/R/C/S Reviews

As per Section 2.2 of this report, detailed reviews for eight L/R/C/Ss with content applicable to Safety Factor 7 are provided in References [6] and [7]. Associated findings applicable to Safety Factor 7 are summarized in Table 5 below.

Table 5: PSR2 L/R/C/S Review Results for Safety Factor 7

L/C/R/S Reviewed	**PSR2 L/R/C/S Review Results for Safety Factor 7**
CSA N286.7-16, "Quality Assurance Of Analytical, Scientific And Design Computer Programs"	There are no PSR2 gaps for CSA N286.7-16. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CSA N286.7-16.
CSA N293-12, "Fire Protection for Nuclear Power Plants"	For Safety Factor 7, there are no PSR2 gaps for CSA N293-12.
CNSC REGDOC-2.4.1 (2014), "Deterministic Safety Analysis"	For Safety Factor 7, there are no PSR2 gaps for CNSC REGDOC-2.4.1 (2014).
CNSC REGDOC-2.4.2 (2014), "Probabilistic Safety Assessment (PSA) for Nuclear Power Plants"	For Safety Factor 7, there are no PSR2 gaps for CNSC REGDOC-2.4.2 (2014).
CNSC REGDOC-2.5.2 (2014), "Design of Reactor Facilities: Nuclear Power Plants"	For Safety Factor 7, there are no PSR2 gaps for CNSC REGDOC-2.5.2 (2014).
CNSC REGDOC-2.3.2 (2015), "Accident Management, Version 2"	For Safety Factor 7, there are no PSR2 gaps for CNSC REGDOC-2.3.2 (2015).
CSA N286.7.1-09, "Guideline for the Application of N286.7-99, Quality Assurance of Analytical, Scientific, and Design Computer Programs for Nuclear Power Plants"	The N286.7.1 guide has been amalgamated into the new (-16) edition of the N286.7 Standard. The N286.7 CSA Impact Statement states [8]: "The CSA N286.7.1 guide will no longer be maintained after this new edition of N286.7 is issued. Any relevant guidance has been put into the new edition of N286.7." As a result, only the review of N286.7-16 has been prepared for PSR2.
CNSC G-149 (2000), "Computer Programs Used in Design and Safety Analyses of Nuclear Power Plants and Research Reactors"	There are no PSR2 gaps for CNSC G-149 (2000). Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with G-149 (2000).

4.3 OPG Program Effectiveness Reviews

The OPG Nuclear Program reviewed for Safety Factor 7 is identified in Table 2 and details of the associated effectiveness review for this N-PROG are provided in Appendix B.

4.4 Additional Review Findings

As discussed in Section 3.4, the PSR2 Safety Factor 7 assessment also included a review of commitments previously made to the CNSC, open CNSC action items, and exemptions granted by the CNSC, as identified in the R04 Pickering LCH [4], to determine if there are any impacts associated with operation of the Pickering Units past 2020. The review also included identification and review of previously identified programmatic Darlington PSR1 gaps related to Safety Factor 7 to determine impacts associated with operation of the Pickering Units past 2020. This assessment did not find any PSR2 gaps for Safety Factor 7.

Findings from the review of previously identified PSR1 gaps in the Pickering Units 5-8 Continued Operations Plan [10] are provided in a separate PSR2 COP Review Report. Findings from the review of Fukushima Action Items are provided in a separate PSR2 FAI Review Report. Results from the Continued Operations Plan and Fukushima Action Items reviews will be considered in the Global Assessment process.

There is one PSR2 gap identified in this Safety Factor 7 Report that is also discussed in the Safety Factor 5 Report (Deterministic Safety Analysis) [105]. Gap SF7-1 identifies the on-going assessment to address the Category 3 CSI IH-6 relating to pipe whip. This work is also captured in Gap SF5-2, which relates to all Category 3 CSIs applicable to Pickering NGS.

5.0 RESULTS AND CONCLUSIONS

OPG Governance, Programs, Policies, Procedures, Instructions and Guidelines related to Safety Factor 7 were reviewed for the three PSR2 Review Tasks in Section 4.1 of this report and resulted in PSR2 Gap SF7-1 below. L/R/C/S and OPG Nuclear Program effectiveness reviews for Safety Factor 7 were prepared per Sections 4.2 and 4.3, respectively, and resulted in no PSR2 gaps. Per Section 4.4, this report also included identification and review of previously identified programmatic Darlington PSR1 gaps related to Safety Factor 7 (to ascertain the implications of extending Pickering NGS operation beyond 2020), as well as a review of the R04 Pickering LCH [4] for any impacts of Pickering NGS operation beyond 2020 on a) OPG commitments previously made to the CNSC, b) open CNSC action items, and c) exemptions granted by the CNSC (all related to Safety Factor 7), which resulted in no additional PSR2 gaps.

The one PSR2 gap that will need to be addressed as part of Pickering PSR2 is:

- **Gap SF7-1:** To address CNSC CANDU Safety Issue CSI-IH 6, the potential for, and possible impacts of, high-energy piping failures in Pickering Units 1,4 must be assessed. This assessment is currently underway [65], [66]. Therefore, this has been identified as a PSR2 gap.

The review of Safety Factor 7 has confirmed the adequacy of protection of Pickering NGS against internal and external hazards, with account taken of plant design (including confirmation that analyses/methods address the condition of SSCs important to safety), site characteristics, and current analytical methods, safety standards and knowledge.

6.0 REFERENCES

[1] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[2] CNSC, REGDOC-2.3.3, *Periodic Safety Reviews*, April 2015.

[3] IAEA Safety Guide, SSG-25, *Periodic Safety Review for Nuclear Power Plants*, 2013.

[4] CNSC Report, LCH-PNGS-R004, *Pickering NGS: Licence Conditions Handbook*, December 22, 2015.

[5] OPG Report, P-REP-03680-00003 R000, *Pickering NGS Periodic Safety Review 2 (PSR2): Definition of Safety Factor Review Tasks*, May 2016.

[6] OPG Report, P-REP-03680-00029 R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 1, 5, 6, and 7*, March 2017.

[7] OPG Report, P-REP-03680-00021 R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 8, 10, 12, 13, and 14*, December 2016.

[8] CSA Impact Statement, *Notification of CSA N286.7 on Quality Assurance of Analytical, Scientific, and Design Computer Programs; Product: New Edition; Product Designation: CSA N286.7; Previous Edition Published: 1999; Reaffirmed 2007 and 2012*, date not provided.

[9] OPG Nuclear Program, N-PROG-RA-0012 R011, *Fire Protection*, July 2015.

[10] OPG Plan, NK30-PLAN-00531-00001 R005, *Pickering 5-8 Continued Operations Plan*, December 2015.

[11] OPG Procedure, N-PROC-RA-0023 R018*, Fleetview Program Health and Performance Reporting*, August 2013.

[12] OPG Report, NK38-REP-03680-10185 R002, *Darlington NGS Integrated Implementation Plan (IIP)*, April 2015.

[13] AECB Consultative Document, C-006-R001, *Requirements for the Safety Analysis of CANDU Nuclear Power Plants*, June 1980.

[14] OPG Report, NA44-REP-03500-00001 R000, *Pickering NGS A - Review of Safety Analysis Against CNSC Consultative Document C6 (Revision 1), Safety Analysis of CANDU Nuclear Power Plants*, May 2001.

[15] OPG Report, NK30-REP-03680-00005 R000, *Pickering NGS B – Integrated Safety Review – Safety Analysis Review*, June 2007.

[16] OPG Report, NA44-SR-01320-00001 R015, *Pickering A Safety Report, Part 1 and 2*, July 2012.

[17] OPG Report, NK30-SR-01320-00001 R004, *Pickering B Safety Report - Part 1*, October 2012.

[18] OPG Report, NA44-REP-03611-00011 R00, *Hazard Screening Analysis – Pickering A*, January 2012.

[19] OPG Report, NK30-REP-03611-00008 R00, *Hazard Screening Analysis – Pickering B*, January 2012.

[20] CNSC Regulatory Standard, S-294, *Probabilistic Safety Assessment (PSA) for Nuclear Power Plants*, April 2005.

[21] IAEA, 50-P-7, *Treatment of External Hazards in Probabilistic Safety Assessment for Nuclear Power Plants*, 1995.

[22] U.S. NRC, NUREG/CR-2300, *PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants*, January 1983.

[23] OPG Report, N-GUID-03611-10001, Volume 1 R004, *OPG Probabilistic Risk Assessment (PRA) Guide - Level 1 (At-Power)*, October 2014.

[24] OPG Report, N-GUID-03611-10001, Volume 2 R003, *OPG Probabilistic Risk Assessment (PSA) Guide Volume 2 - Level 2 (At-Power)*, May 2015.

[25] OPG Report, N-GUID-03611-10001, Volume 3 R000, *OPG Probabilistic Risk Assessment (PRA) Guide Volume 3 — Level 3*, June 2011.

[26] OPG Report, N-GUID-03611-10001, Volume 4 R001, *OPG Outage Probabilistic Risk Assessment (PRA) Guide - Level 1*, July 2010.

[27] OPG Report, N-GUID-03611-10001, Volume 5 R001, *OPG Probabilistic Risk Assessment (PRA) Guide – Fire*, June 2011.

[28] OPG Report, N-GUID-03611-10001, Volume 6 R001, *OPG Probabilistic Risk Assessment (PRA) Guide - Internal Flood*, June 2011.

[29] OPG Report, N-GUID-03611-10001, Volume 7 R001, *OPG Probabilistic Risk Assessment (PRA) Guide – Seismic*, March 2011.

[30] OPG Report, N-GUID-03611-10001, Volume 8 R003, *OPG Probabilistic Safety Assessment (PSA) Guide - External Hazards Screening*, December 2014.

[31] OPG Report, N-GUID-03611-10001, Volume 9 R001, *OPG Probabilistic Safety Assessment (PSA) Guide - Internal Hazards Screening*, December 2014.

[32] OPG Report, N-GUID-03611-10001, Volume 10 R000, *OPG Probabilistic Risk Assessment Guide - High Wind Hazard*, August 2012.

[33] OPG Guideline, NA44-GUID-03611-00010 R000, *Pickering NGS 'A' Probabilistic Risk Assessment Guide – Level 1 (At-Power)*, October 2012.

[34] OPG Guideline, NA44-GUID-03611-00011 R000, *Pickering A Probabilistic Risk Assessment (PRA) Guide — Level 2 (At-Power)*, August 27, 2012.

[35] OPG Guideline, NA44-GUID-03611-00012 R000, *Pickering 014 Probabilistic Risk Assessment Guide – Level 1 Outage for Internal Events,* October 2013.

[36] OPG Guideline, NA44-GUID-03611-00013 R000, *Pickering NGS A Probabilistic Risk Assessment Guide – Fire*, November 2012.

[37] OPG Guideline, NA44-GUID-03611-00014 R000, *Pickering NGS A Probabilistic Risk Assessment (PRA) Guide – Internal Flood*, July 2012

[38] OPG Guideline, NA44-GUID-03611-00015 R001, *Pickering NGS A Probabilistic Risk Assessment Guide – Seismic*, November 2013.

[39] OPG Report, N-REP-03500-0401509 R000, "*Implications of the Fukushima Daiichi Event on OPG Nuclear Power Plants: A Summary Report*", July 2012.

[40] OPG Report, N-GUID-01130-10000 R001, *Modification for Beyond Design Basis Accidents*, February 2015.

[41] OPG Nuclear Standard, N-STD-RA-0034 R003, *Preparation, Maintenance and Application of Probabilistic Risk Assessment*, October 2013.

[42] OPG Nuclear Program, N-PROG-RA-0006 R008, *Environmental Qualification*, May 2015.

[43] OPG Report, P-REP-03680-00006 R000, *Pickering NGS PSR2 Safety Factor 3 Report: Equipment Qualification (Seismic and Environmental)*, July 2016.

[44] OPG Letter, R.J. Strickert to C.B. Parsons, NA44-CORR-00531-00210, *Pickering A Fire Assessment*, September 2000.

[45] OPG Letter, G. Jager to M. Santini, NK30-CORR-00531-05774, *Pickering NGS 'B' - Request for CNSC Acceptance of the CCR, the Third Party Review of the Inspection, Testing and Maintenance Report for Fixed Fire Protection Systems, the FSSA and the FHA*, December 2011.

[46] OPG Report, NA44-REP-03611-00038 R000, *Pickering NGS A Probabilistic Risk Assessment (PRA) - Internal Fire Report*, April 2014.

[47] OPG Report, NK30-REP-03611-00012 R000, *Pickering NGS B Probabilistic Risk Assessment - Internal Fire Final Report*, December 2012.

[48] OPG Report, NA44-SR-01320-00002 R004, *Pickering Nuclear 1-4 Safety Report: Part 3 – Accident Analysis*, October 2013.

[49] OPG Report, NK30-SR-01320-00003 R004, *Pickering Nuclear 5-8 Safety Report: Part 3 – Accident Analysis*, October 2014.

[50] CNSC Report, *Regulatory Oversight Report for Canadian Nuclear Power Plants: 2014*, September 2015.

[51] OPG Report, NK30-SR-01320-00002 R004, *Pickering B Safety Report - Part 2*, October 2012.

[52] OPG Manual, NA44-MAN-03651-10001 R002, *Environmental Qualification Room Conditions – Pickering A*, October 2014.

[53] OPG Manual, NK30-MAN-03651-10001 R002, *Environmental Qualification Room Conditions – Pickering B*, November 2014.

[54] OPG Report, NA44-REP-03611-00021 R000, *Pickering NGS A Internal Flood Probabilistic Risk Assessment (para Flood)*, December 2013.

[55] OPG Report, NK30-REP-03611-00014 R000, *Pickering NGS B Internal Flood Probabilistic Risk Assessment (PBRA Flood)*, December 2012.

[56] OPG Nuclear Procedure, N-PROC-RA-0054 R015, *Control of Space Allocation for Transient Materials and Extended Storage of Material within the Site*, April 2015.

[57] OPG Nuclear Procedure, N-PROC-RA-0057 R008, *Control of Ignition Sources and Hot Work Activities*, April 2015.

[58] OPG Nuclear Program, N-PROG-MA-0004 R011, *Conduct of Maintenance*, April 2015.

[59] OPG Nuclear Standard, N-STD-RA-0026 R006, *Fire Protection Surveillance, Availability, and Compensatory Measures*, May 2012.

[60] AECL Report, 44RS-21000-ASD-002 Rev 0, *Pickering "A" Return to Service - Review of PHT Pipe whip Effects on RMD Area (C1-007-03-05-002)*, March 2001.

[61] OPG Correspondence, N-CORR-N0257054, *Pickering NGS A – Pipewhip Restraints*, December 1975.

[62] AECL Report, 44RS-03650-IG-004 Rev 2, *Pickering 'A' Return to Service - Separation Implementation Guide*, May 2002.

[63] OPG Correspondence, NK30-CORR-N0441026, *Pickering NGS "B" Analysis of the Dynamic Effects Associated with the Postulated Pipe Ruptures*, December 1981.

[64] OPG Correspondence, NA44-CORR-33350-84554, *Pickering NGS-A Assessment of ECI Jet Impingement*, October 1993.

[65] OPG Report, P-REP-04960-00001 R002, *Methodology of High-Energy Line Break Assessment for Piping Inside the Pickering Reactor Buildings*, June 2016.

[66] OPG Correspondence, N-CORR-00531-18288 R000, *Re-Categorization Request for CANDU Safety Issue IH6 for Pickering NGS 5-8 and Status for Pickering NGS 1-4*, December 2016.

[67] OPG Report, P-REP-04960-00014 R001, *Pipe-Whip and Jet-Impingement Assessment of Piping inside Reactor Building – Units 5-8*, May 2016.

[68] IAEA Safety Standard, NS-G-1.11, *Protecting against Internal Hazards other than Fires and Explosions in the Design of Nuclear Power Plants*, 2004.

[69] OPG Report, NA44-REP-03611-00012 R000, *Pickering NGS A Level 1 At-Power Internal Events Risk Assessment (PARA-L1P)*, September 2013.

[70] OPG Report, NA44-REP-03611-00014 R000, *Pickering NGS A Level-1 Outage Internal Events Risk Assessment (PARA-L1O)*, February 2014.

[71] OPG Procedure, OPG-PROC-0126 R000, *Hazardous Materials Management*, December 2014.

[72] OPG Nuclear Procedure, N-PROC-RA-0054 R015, *Control of Space Allocation for Transient Material and Extended Storage of Material Within the Site*, April 2015.

[73] OPG Nuclear Instruction, N-INS-03651-10003 R007, *Preparation of the Environmental Qualification Room Conditions Manual*, October 2014.

[74] OPG Letter, G. Preston to T.E. Schaubel, P-CORR-00531-01215, *Potential Flooding of the Site at Pickering Nuclear, Action Item No. 2000-04-07*, September 2002.

[75] OPG Letter, W.R. Robinson to T.E. Schaubel, NA44-CORR-00531-01949, *Potential Flooding of the Site at Pickering Nuclear, Action Item 2000407*, June 2003.

[76] OPG Correspondence, N-CORR-00531-06905 R000, *REGDOC 3.1.1 Research and Development Annual Reporting*, June 2015.

[77] OPG Report, NA44-REP-02004-0073 Volume 2, *Seismic Assessment of Pickering A NGS Summary Report*, February 1998.

[78] OPG Report, NA44-REP-N0622010, Pickering Nuclear Safety Department Report Number 95004, *Pickering NGS-A Seismic Assessment Success Path Description - interim Report*, February 1996.

[79] OPG Design Guide, DG-30-68000-2 R002, *Seismic Qualification of Safety Related Systems*, December 1979.

[80] OPG Report, NA44-REP-03611-00023 R000, *Pickering NGS A Level 1 High Wind Probabilistic Risk Assessment*, December 2013.

[81] OPG Report, NK30-REP-03611-00020 R001, *Pickering NGS B High Wind Probabilistic Risk Assessment*, February 2014.

[82] OPG Report, NA44-REP-03611-00022 R000, *Pickering NGS A PRA Based Seismic Margin Assessment (PARA Seismic)*, December 2013.

[83] OPG Report, NK30-REP-03611-00013 R001, *Pickering NGS B (PNGS-B) PRA Based Seismic Margin Assessment (SMA)*, March 2014.

[84] OPG Manual, N-BDB-03600-00002 R000, *OPG Emergency Mitigating Equipment For Beyond Design Basis Accidents: Technical Basis Document*, October 2015.

[85] OPG Nuclear Program, N-PROG-MP-0009 R011, *Design Management*, December 19, 2014.

[86] OPG Nuclear Program, N-PROG-MP-0001 R014, *Engineering Change Control*, January 2015.

[87] OPG Nuclear Program, N-PROG-MP-0008 R006A, *Integrated Aging Management*, October 2015.

[88] OPG Nuclear Procedure, N-PROC-MA-0024 R015, *System Performance Monitoring*, October 2013.

[89] OPG Nuclear Standard, N-STD-MP-0025 R001, *General Requirements for Seismic Qualification of OPG Nuclear Facilities*, October 2015.

[90] OPG Procedure, N-PROC-MP-0086 R004, *Safety Analysis Basis and Safety Report Update*, December 2014.

[91] OPG Pickering Nuclear Operating Procedure, NA44-OP-54600-0016 R000, *Installation and Removal of Fuel Forwarding Pumphouse Flood Barrier*, April 2013.

[92] OPG Nuclear Procedure, N-PROC-RA-0095 R010, Severe Weather Emergency Preparedness, October 2015.

[93] Ministry of Natural Resources, *Lakes & Rivers Improvement Act Technical Guidelines - Criteria and Standards for Approval*, June 2004.

[94] OPG Report, NA44-REP-10200-00002 R000, *Area 3 Flood Assessment*, May 2013.

[95] OPG Report, N-REP-03600-10003 R007, *Fukushima Action Item Status Report*, November 2015.

[96] OPG Letter, P-CORR-00531-04045 R000, *Pickering NGS Units 1 to 8: Probabilistic Safety Assessment (PSA) – Screening Analysis (Unscreened Hazards)*, January 2014.

[97] OPG Nuclear Instruction, N-INS-09000-10001 R001, *Processing of Safety Report Analysis Issues: Overview*, 2011.

[98] OPG Nuclear Instruction, N-INS-09000-10002 R001, *Guidelines for Evaluating and Prioritizing Safety Report Analysis Issues*, 2012.

[99] OPG Nuclear Instruction, N-INS-09000-10004 R001, *Guidelines for the Control of the Analysis of Record*, 2012.

[100] OPG Nuclear Instruction, N-INS-09000-10005 R001, *Safety Report Issue Database Management*, 2012.

[101] OPG Nuclear Program, N-PROG-MP-0014 R005, *Reactor Safety Program*, September 2015.

[102] OPG Nuclear Program, N-PROG-MA-0026 R002, *Equipment Reliability*, May 2015.

[103] OPG Nuclear Program, N-PROG-MA-0017 R008, *Component and Equipment Surveillance*, June 2015.

[104] OPG Nuclear Program, N-PROG-RA-0016 R009, *Risk and Reliability Program*, June 2016.

[105] OPG Report, P-REP-03680-00009 R000, *Pickering NGS PSR2 Safety Factor 5 Report: Deterministic Safety Analysis*, March 2017.

Appendix A: Nomenclature

AECB	Atomic Energy Control Board
APS	Auxiliary Power System
CANDU	CANada Deuterium Uranium
CNSC	Canadian Nuclear Safety Commission
COG	CANDU Owners Group
COP	Continued Operations Plan
CSA	Canadian Standards Association
CSI	CANDU Safety Issue
DBA	Design Basis Accident
ECI	Emergency Coolant Injection
EME	Emergency Mitigating Equipment
EPG	Emergency Power Generators
EQ	Environmental Qualification
FAI	Fukushima Action Item
FHA	Fire Hazard Assessment
FSSA	Fire Safe Shutdown Analysis
HVAC	Heating, Ventilation and Air Conditioning
IAEA	International Atomic Energy Agency
IDLH	Immediately Dangerous to Life and Health
IFB	Irradiated Fuel Bay
ISR	Integrated Safety Review
LCH	Licence Condition Handbook
L/R/C/S	Laws, Regulations, Codes and Standards
N-PROG	OPG Nuclear Programs
NGS	Nuclear Generating Station
OPG	Ontario Power Generation
PARTS	Pickering A Return to Service
PHT	Primary Heat Transport
PMP	Probable Maximum Precipitation
PRA	Probabilistic Risk Assessment
PROL	Power Reactor Operating Licence

PSA	Probabilistic Safety Assessment
PSR	Periodic Safety Review
PSR1	Periodic Safety Review 1 (earlier OPG PSR work and other associated assessments)
PSR2	Periodic Safety Review 2 (subsequent PSR per CNSC REGDOC-2.3.3)
RCM	Room Conditions Manual
RLC	Review Level Conditions
SATM	Space Allocation for Transient Material
SCR	Station Condition Record
SDS	Shutdown System
SG	Standby Generators
SOE	Safe Operating Envelope
SSC	Structures, Systems and Components

Appendix B: OPG Program Effectiveness Review Results

B.1 N-PROG-RA-0012, "Fire Protection"

The objective of the Fire Protection Program is to minimize the risk of radiological releases to the public due to fire; protect the plant and facility occupants from risks due to fire; and minimize economic loss resulting from fire damage to structures, equipment, and inventories. The specific elements of the Fire Protection Program are based on the requirements of CSA N293, "Fire Protection for CANDU Nuclear Power Plants", as it applies to the Pickering Nuclear Generating Station (NGS). This includes the codes and standards that are referenced within, such as the National Fire Code of Canada, the fire protection requirements of the National Building Code of Canada and applicable National Fire Protection Association standards.

Nuclear Programs and Training completed a self-assessment, NO12-000316-SA [B.1.1], in May 2012 in order to assess the health of N-PROG-RA-0012, "Fire Protection", governance framework, which is applicable to both Darlington and Pickering NGS. This involved a review of SCRs, the governance framework and ownership, revision records and previous program assessments. The self-assessment concluded that the Fire Protection program is in full compliance with the applicable requirements.

The Emergency Management and Fire Protection department completed a self-assessment in June 2016, NO15-001497-SA [B.1.2], in order to determine the causal and human performance factors related to Space Allocation for Transient Material (SATM) issues applicable to both Pickering and Darlington NGS. A performance improvement opportunity applicable to Pickering NGS was identified in the area of SATM training.

Nuclear Oversight completed a performance based audit, NO-2015-016 [B.1.3], in December 2015, in order to assess corrective action effectiveness from previous Nuclear Oversight findings, peer reviews, program changes and initiatives since 2012 applicable to both Pickering and Darlington NGS. The audit identified performance improvement opportunities applicable to Pickering NGS in the areas of SATM requirements and testing for Carbon Dioxide extinguishers. Two SCRs were initiated to address the above findings and both have been completed.

References

[B.1.1] Self-Assessment Report, NO12-000316-SA, *Program Management Assessment – N-PROG-RA-0012*, May 2012.

[B.1.2] Self-Assessment Report, NO15-001497-SA, *Fire Protection*, June 2016.

[B.1.3] Nuclear Oversight Report, N-REP-01070-0572403 T06, *NO-2015-016 Fire Protection*, December 2015.

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ACCEPTED	✓
ACCEPTED AS NOTED	
REVISE AND RESUBMIT	
Signature	**Date** 16 DEC 2016
Name: Mike Ruffolo, Manager	
Dept: Pickering Engineering - Aging Management & Strategic Initiatives	
OPG Proprietary	
Doc No.: P-REP-03680-00012	**Rev: 000**
This acceptance does not relieve the contractor from responsibility for errors or omissions or from any obligations or liability under this contract.	

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Pickering NGS PSR2 Safety Factor 8 Report: Safety Performance

PS112/RP/014 R01

December 16, 2016

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Revision Summary – For Amec Foster Wheeler Report PS112/RP/014

Rev	Date	Author	Comments
R00	July 15, 2016	M. Hoang, R. Jayasundera	Initial issue for OPG review and comment.
R01	December 16, 2016	M. Hoang, R. Jayasundera, J. Cheng	Updated report addressing OPG comments on R00 Report.

EXECUTIVE SUMMARY

Ontario Power Generation (OPG) is performing a Periodic Safety Review (PSR) to support the possibility of continued operation of Pickering Nuclear Generating Station (NGS) beyond 2020. The PSR (referred to as "PSR2") is a subsequent PSR building on the review basis of earlier OPG Integrated Safety Reviews and other associated assessments. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

Part of PSR2 involves the preparation of Safety Factor reports for each of fifteen major topic areas. Safety Factor reports consist of:

- Assessing compliance against "Review Tasks" identified in the PSR2 Basis Document [1]. These Review Tasks are derived from Canadian Nuclear Safety Commission (CNSC) REGDOC-2.3.3, "Periodic Safety Reviews" [2] and International Atomic Energy Agency (IAEA) SSG-25, "Periodic Safety Review for Nuclear Power Plants" [3];

- Documenting assessments against applicable modern Laws, Regulations, Codes and Standards (L/R/C/Ss) as defined in Reference [1]; and

- Reviewing the effectiveness of applicable OPG Nuclear Programs that support the above assessments, through review of audit and self-assessment results.

The PSR2 review of Safety Factor 8, *Safety Performance*, is presented in this report. OPG Governance, Programs, Policies, Procedures, Instructions and Guidelines related to Safety Factor 8 were reviewed for the eight PSR2 Review Tasks specified in Section 4.1 of this report. L/R/C/S and OPG Nuclear Program effectiveness reviews for Safety Factor 8 were prepared per Sections 4.2 and 4.3, respectively. Per Section 4.4, the PSR2 assessment includes a review of previously identified PSR1 gaps related to Safety Factor 8 (to ascertain the implications of extending Pickering NGS operation beyond 2020), as well as a review of the R04 Pickering Licence Conditions Handbook [4] for any impacts of Pickering NGS operation beyond 2020 on: a) OPG commitments previously made to the CNSC, b) open CNSC action items, and c) exemptions granted by the CNSC (all related to Safety Factor 8).

The results of the review of Safety Factor 8 are discussed in Section 5.0 of this report. The review has confirmed that the safety performance indicators and records of operating experience, including the evaluation of root causes of plant events, exist and are utilized to ensure the safe operation of Pickering NGS. As discussed in Section 5.0, the review identified no Pickering PSR2 gaps.

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LIST OF TABLES AND FIGURES

1.0 INTRODUCTION

Ontario Power Generation (OPG) is performing a Periodic Safety Review (PSR) to support the possibility of continued operation of Pickering Nuclear Generating Station (NGS) beyond 2020.[1] A comprehensive Integrated Safety Review (ISR) was completed for Pickering Units 5 through 8 in 2009 in support of refurbishment and continued operation. Pickering Units 1,4 integrated safety assessments were also performed for Pickering A Return to Service (PARTS) in support of approval to restart Units 1 and 4. In addition to these Pickering-specific studies, the 2013 Darlington ISR performed extensive code and standard reviews that were updated in relation to the versions that were assessed in the 2009 Pickering B ISR.[2] These previous ISRs are considered to constitute the first PSR completed for Pickering (referred to as "PSR1"). The current PSR (referred to as "PSR2") is a subsequent PSR building on the basis of earlier OPG integrated safety assessments through review of the various studies, assessments and licence renewals performed since PSR1. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

PSR2 will support and complement the licence renewal application for Pickering NGS going forward. Fifteen Safety Factors will be assessed as part of the PSR. The purpose of Safety Factor reviews is to confirm that the design, condition and operation of Pickering Units 1,4 and 5-8 (as well as common systems) will support continued safe operation for the period of PSR2 by:

- Assessing compliance against "Review Tasks" identified in the PSR2 Basis Document [1], which were derived from Canadian Nuclear Safety Commission (CNSC) REGDOC-2.3.3 [2] and International Atomic Energy Agency (IAEA) SSG-25 [3];

- Documenting assessments against applicable modern Laws, Regulations, Codes and Standards (L/R/C/Ss) (as defined in Reference [1]); and

- Reviewing the effectiveness of applicable OPG Nuclear Programs that support the above assessments, through review of audit and self-assessment results.

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1 Currently, Pickering Units 5-8 are approved to operate to 247,000 Effective Full Power Hours. This operation limit is expected to be reached on some units in 2020. For the purposes of PSR2, OPG assumes operation of Pickering NGS for up to eight additional years, from 2020 until 2028. OPG will make a decision regarding the permanent shut down dates for the six reactors following the performance of a technical evaluation that will include PSR2, and will communicate it to the CNSC as required by the current Power Reactor Operating Licence (PROL).

2 Much of the compliance assessment and evaluation of Safety Factor health for the Darlington ISR is based on programs and practices that apply across OPG's nuclear operations. As a result, where Pickering is confirmed to follow the same nuclear programs and practices as were assessed for Darlington, the Darlington ISR programmatic conclusions are applicable to Pickering. As discussed in Section 1.0, an effectiveness review (effectiveness at Pickering NGS) of OPG programs used to demonstrate compliance with the PSR2 Assessment Basis will be conducted using recent audit and self-assessment results.

As outlined in IAEA SSG-25 [3], the objective of the review of Safety Performance Safety Factor 8 is to: "determine whether the plant's safety performance indicators and records of operating experience, including the evaluation of root causes of plant events, indicate any need for safety improvements". REGDOC-2.3.3 [2] requires that: "The licensee shall conduct a PSR in accordance with this regulatory document for the period until the next PSR or, if applicable, until the end of commercial operation of the plant." For this Safety Factor 8 Report, the objective is to confirm that the safety performance indicators and records of operating experience, including the evaluation of root causes of plant events, exist and are utilized to ensure the safe operation of Pickering NGS. Per the Pickering PSR2 Basis Document [1], analysis of gaps and potential safety enhancements for Pickering NGS (including identification of improvements that are reasonable and practicable to implement) is addressed as part of the Global Assessment process. Preparation of a plan for the implementation of safety enhancements is addressed by the PSR2 Integrated Implementation Plan.

This report documents the results of the review of Safety Factor 8 for Pickering PSR2. The report is based on the OPG Governance, Programs, data, and material available up to January 15, 2016 which is the freeze date for PSR2.

2.0 SCOPE OF REVIEW

2.1 Review Task Assessments

The Pickering PSR2 Safety Factor 8 Review Tasks are defined in Reference [1]. Details of the derivation of these Review Tasks from CNSC REGDOC-2.3.3 [2] and IAEA SSG-25 [3] are shown in Reference [5]. The Safety Factor 8 Review Tasks are:

1) Confirm existence of a system for identifying, classifying and recording safety related incidents and operating experience including:

    - Safety related incidents, low level events and near misses;
    - Safety related operational data;
    - Maintenance, inspection and testing;
    - Replacements of Structures, Systems, Components (SSCs) important to safety owing to failure or obsolescence;
    - Modifications, either temporary or permanent, to SSCs important to safety;
    - Unavailability of safety systems;
    - Radiation doses (to workers, including contractors);
    - Off-site contamination and radiation levels;
    - Discharges of radioactive effluents; and
    - Generation of radioactive waste.

2) Confirm that safety related incidents are investigated using root cause analysis and that lessons learned from investigation of these incidents are fed back into the conduct of Operations and Maintenance.

3) Confirm that the results of the root cause analysis are used to minimize the chances of the same incident reoccurring.

4) Confirm that information from trend analysis of safety related incidents is fed back into the conduct of Operations and/or Maintenance.

5) Confirm there is an adequate set of performance indicators that provides a systematic and comprehensive method to record, trend and analyze safety related data including the major system parameters, and maintenance and inspection records. Performance indicators may include:

    - Frequency of unplanned trips while the reactor is critical;
    - Satisfactory performance of safety system tests within required limits;
    - Special Safety System unavailability;
    - Reliability of Systems Important to Safety;
    - Collective annual radiation dose of plant staff;
    - Amount of gaseous and liquid radioactive release relative to permitted limits;
    - Heavy water escape and loss rates;

- Fuel reliability;
- Chemistry index;
- Volume of Low Level radioactive waste;
- Change control index;
- Maintenance backlog;
- Training;
- Environment Index;
- Non-radioactive effluents, including hazardous substances;
- Non-radioactive wastes; and
- Spills.

6) Confirm that for cases where performance indicators show an unsatisfactory trend, corrective action is taken.

7) Review the adequacy of:

    - Records of the integrity of physical barriers for the containment of radioactive material;
    - Records of radiation doses to persons on the site;
    - Records of data from off-site radiation monitoring and records of the quantities of radioactive effluents;
    - Records of non-radioactive effluents, including hazardous substances;
    - Records of radioactive and non-radioactive waste;
    - Records of spills; and
    - Records of other environmental impacts.

8) Consider the effects of any changes in operation at the plant on safety performance. In particular, confirm that current indicators and other safety performance methods continue to be relevant in the context of current and future operations, and confirm that only relevant data and records are used.

The methodology for the reviews is discussed in Section 3.1. Review Task findings are summarized in Section 4.1 of this report.

2.2 L/R/C/S Reviews

The applicable Laws, Regulations, Codes and Standards relevant to the Safety Performance Safety Factor are identified in Reference [1] and are listed in Table 1 below. Table 1 also identifies the modern version and date of each L/R/C/S to be considered, the Safety Factor(s) to which each document is applicable, and the type of review that will be completed in PSR2.

All of the Safety Factor 8 L/R/C/S reviews are high level or incremental in nature. The definitions of a High Level Review and Incremental Review are as follows:

- High Level: New L/R/C/Ss not referenced in Pickering PROL 48.02/2018 but which are in the PSR2 Assessment Basis will be subject to a high level review. In a high level review, the degree of conformance with clauses or groups of clauses in the L/R/C/S is demonstrated by supporting evidence stating whether the intent of the requirements stipulated in the requirement document is met; and,

- Incremental Review: For L/R/C/Ss that have been reviewed in PSR1 but have had revisions since the last review, a topical review will be performed of the changes.

The methodology for the reviews is discussed in Section 3.2. A detailed assessment for each L/R/C/S is provided in References [6] and [7]. Associated findings are summarized in Section 4.2 of this report.

Table 1: L/R/C/Ss Reviewed for Safety Performance Safety Factor 8

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
L/R/C/Ss Referenced in Pickering NGS PROL 48.02/2018						
1	CSA N290.15	Requirements for the Safe Operating Envelope of Nuclear Power Plants	N290.15-10	8	Incremental	N290.15 not addressed as part of Pickering B or Darlington ISRs, but gap analysis has been performed against OPG Governance and N290.15.
2	CSA N288.1	Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities	N288.1-14	8, 14	Incremental	N288.1 addressed as part of Pickering B and Darlington ISRs.
3	CSA N288.4	Environmental Monitoring Program at Class I Nuclear Facilities and Uranium Mines and Mills	N288.4-10	8, 14	Incremental	N288.4 addressed as part of Pickering B and Darlington ISRs.
4	CNSC REGDOC-2.9.1*	Environmental Protection Policies, Programs and Procedures	2013	8, 14	Incremental	REGDOC-2.9.1 addressed as part of Darlington ISR. S-296 also addressed as part of Pickering B and Darlington ISRs.
Additional L/R/C/Ss						
5	CNSC G-129	Keeping Radiation Exposures and Doses "As Low As Reasonably Achievable (ALARA)"	2004	8, 15	Incremental	G-129 addressed as part of Pickering B and Darlington ISRs.

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
6	CNSC G-228	Developing and Using Action Levels	2001	8, 14, 15	Incremental	G-228 addressed as part of Pickering B and Darlington ISRs.
7	SOR/2000-203	The Radiation Protection Regulations	Amended in June 2015	8, 15	Incremental	SOR/2000-203 addressed as part of Pickering B and Darlington ISRs.
8	CSA N288.6	Environment Risk Assessments at Class I Nuclear Facilities and Uranium Mines and Mills	N288.6-12	8, 14	Incremental [3]	N288.6 not addressed as part of Pickering B or Darlington ISRs. Implementation Plan and clause-by-clause review have been prepared for Pickering Environmental Monitoring Program compliance with N288.6.
9	CSA N288.5	Effluent Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills	N288.5-11	8, 14	Incremental [3]	N288.5 not addressed as part of Pickering B or Darlington ISRs. OPG has performed a gap analysis and completed all actions in the implementation plan to satisfy mandatory requirements of N288.5.
10	CNSC REGDOC-2.3.2	Accident Management, Version 2	2015	1, 5, 6, 7, 8, 10	Incremental	REGDOC-2.3.2 addressed as part of Darlington ISR.
11	CNSC REGDOC-2.3.3	Periodic Safety Reviews	2015	8	High Level	REGDOC-2.3.3 not addressed as part of Pickering B or Darlington ISRs. New PSR methodology.

[3] Per Section 3.2.2 of the R02 PSR2 Basis Document [1]: "Table D1 identifies the review type to be applied to each of the Laws, Regulations, Codes and Standards in the PSR2 Assessment Basis. Following further assessment of past work, the review type of a listed modern Law, Regulation, Code or Standard may be changed from Clause-by-Clause or High Level to Incremental." Past assessments of CSA N288.3.4, N288.5 and N288.6 were reviewed and implementation plans with gap assessments were identified. As a result, the Review Type for these three L/R/C/Ss was changed from High Level to Incremental since "… implementation plans exist for many of the codes and standards not addressed in PSR1 and therefore an incremental review will be applied to these documents" [1].

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
12	CSA N288.3.4	Performance Testing of Nuclear Air-Cleaning Systems at Nuclear Facilities	N288.3.4-13	8, 14	Incremental [3]	N288.3.4 addressed as part of Darlington ISR, but not addressed as part of Pickering B ISR. OPG has completed a gap analysis and is developing an implementation plan to satisfy mandatory requirements of N288.3.4.

* Superseding documents to those currently in Pickering NGS PROL 48.02/2018.

2.3 OPG Program Effectiveness Reviews

The OPG Programs reviewed for Safety Factor 8 are listed in Table 2 below.[4] The methodology for the effectiveness reviews is discussed in Section 3.3. The assessment results of each of the N-PROGs and OPG-PROGs in Table 2 are provided in Appendix B, and findings are summarized in Section 4.3.

Table 2: OPG Programs Reviewed for Safety Factor 8

Document Number	Document Title
N-PROG-RA-0002 [8]	Conduct of Regulatory Affairs
N-PROG-RA-0003 [9]	Corrective Action
OPG-PROG-0010 [10]	Health & Safety Management System Program

2.4 Additional Reviews

The PSR2 Safety Factor 8 report includes a review of the R04 Pickering Licence Conditions Handbook (LCH) [4] for any impacts of Pickering NGS operation beyond 2020 on the following (all related to Safety Factor 8):

- OPG commitments previously made to the CNSC;
- Open CNSC action items; and
- Exemptions granted by the CNSC.

[4] The list of Nuclear Programs to be assessed for effectiveness for PSR2 was derived from review of current OPG Governance. Although there may be content in Nuclear Programs that is applicable to multiple Safety Factors, N-PROG reviews are only provided in one Safety Factor report and are not duplicated.

The PSR2 assessment includes identification and review of previously identified PSR1 gaps related to Safety Factor 8 to ascertain the implications of extending Pickering NGS operation beyond 2020. The methodology for these reviews is described in Section 3.4. Any PSR2 gaps identified as a result of programmatic Darlington PSR1 gaps related to Safety Factor 8 are discussed in Section 4.4 of this report. The review of Pickering PSR1 gaps previously identified in the Pickering Units 5-8 Continued Operations Plan (COP) [11] is provided in a separate PSR2 COP Review Report.

In addition, Fukushima Action Items (FAIs) were reviewed to identify implications of extending operation beyond 2020 (if any). This review is presented in a separate PSR2 FAI Review Report.

Any PSR2 gaps identified as a result of the Safety Factor 8 review which need to be addressed in other Safety Factor reports are discussed in Section 4.4 of this report.

3.0 METHODOLOGY

The sub-sections below summarize the methodology used to assess Review Tasks, L/R/C/Ss, and Nuclear Program effectiveness for the Safety Performance Safety Factor.

3.1 Review Tasks

As discussed earlier, the Safety Factor Review Tasks are derived from CNSC REGDOC-2.3.3 [2] and IAEA SSG-25 [3], taking into consideration the Review Tasks used in the Pickering B and Darlington ISRs (as derived in [5]).

For each Safety Factor 8 Review Task identified in Section 2.1, a confirmation of the existence of applicable OPG Governance, Programs, Policies and Procedures (as well as Instructions and Guidelines, as applicable) was performed. Compliance against Review Tasks is also assessed by reference to applicable Condition Assessments, safety analyses and operating experience, as required.

The Review Task assessments identify Compliances and Gaps as defined below:

- Compliance: Compliance indicates that either the safety requirement or the intent of the Review Task is met.

- Gap: A Gap indicates that the intent of the Review Task is not met.

3.2 L/R/C/S Reviews

The process to identify the modern L/R/C/Ss that are applicable to the PSR2 Assessment Basis involved first creating a broad list from multiple sources (potential candidate L/R/C/Ss) and then filtering it to identify those that are most significant and that are applicable to the PSR2 scope. The identification and selection criteria are detailed in Reference [1]. The result of the identification and selection process was a set of modern L/R/C/Ss that became part of the "PSR2 Assessment Basis".

PSR2 is focused on the extension of Pickering NGS operations beyond 2020, and will conduct reviews against a baseline of past PSR1 work. As a subsequent PSR, PSR2 focuses on changes in requirements, plant conditions, operating experience and new information. Since PSR2 is an update of previous ISRs, it incorporates reviews of L/R/C/Ss that have occurred as new versions have been issued. Since this assessment is a subsequent PSR, the focus is on identifying differences between what was previously assessed and what is now different within the current Pickering PSR2 Assessment Basis. In general, these differences relate to:

- More recent (new or revised) L/R/C/S versions than what was previously assessed as part of PSR1;[5]

- Safety significant differences between Pickering and Darlington, if the Darlington ISR is the basis for the earlier assessment;

- Implications of extending Pickering NGS operation beyond 2020; and

- Safety significant differences between Pickering Units 1,4 and Units 5-8.

As described in Reference [1], L/R/C/S review types are clause-by-clause, high level or incremental. Most of the L/R/C/Ss in the PSR2 Assessment Basis receive incremental reviews since PSR2 is an update of previous PSR1 assessments and clause-by-clause or high level reviews for the majority of the L/R/C/Ss in the PSR2 Assessment Basis have already been completed. Implementation plans (including gap analyses or code-over-code reviews) also exist for the latest editions of many L/R/C/Ss. As a result, incremental review is also used in circumstances where a L/R/C/S in the PSR2 Assessment Basis was not assessed in previous PSR1 reviews but an implementation plan currently exists for compliance.

The PSR2 incremental reviews in this report include an assessment of the intent of recent changes to the L/R/C/Ss on a topic or subject-matter basis where there is potential to impact nuclear safety. Incremental reviews provide:

- A summary of the purpose of the L/R/C/S;

- Pertinent background information about the current revision of the L/R/C/S that is being considered;

- Identification of which Safety Factor(s) are applicable to the current revision of the L/R/C/S;

- A description of which version(s) of the L/R/C/S were assessed for PSR1 (i.e., Darlington ISR (for programmatic content), Pickering B ISR and PARTS code reviews);

- Identification of whether the current version of the L/R/C/S is an update of a previous version of the L/R/C/S that was assessed in PSR1 (and if so, a description of the major changes in the latest revision is provided as discussed below);

[5] "New" refers to a code or standard that was not previously considered in the context of earlier assessments. "Revised" refers to an updated version of a code or standard that was previously considered in the context of earlier assessments. Where a document has a new number/type, but addresses the same topic from the same organization, it is a "revised", not "new", document (e.g., if a REGDOC replaces a CNSC G or RD document).

- An assessment of the applicability of PSR1 assessment findings (gaps and conclusions), including the implications of extending Pickering NGS operation beyond 2020 if any;

- An assessment of the applicability of assessment findings that address more recent (post-PSR1) editions of the L/R/C/S, including any implementation or transition plans that are already committed to by OPG; and

- Where PSR1 and post-PSR1 assessments are not sufficient to address changes in the latest edition of the L/R/C/S, an assessment of the changes from the previously assessed edition of the L/R/C/S (including identification of any safety significant PSR2 gaps which result).

High Level reviews provide the same information as above, where applicable, in a similar format. However, given that High Level L/R/C/Ss generally have not received past assessment during PSR1, the Incremental review content is augmented by a high level, section-by-section assessment of the degree of conformance of Pickering NGS with the L/R/C/S (demonstrating, with supporting evidence, whether the intent of the requirements stipulated in the document are met).

There are currently no L/R/C/S clause-by-clause reviews identified in the PSR2 Assessment Basis.

The Safety Factor 8 L/R/C/S reviews identify Compliances and Gaps as defined below:[6]

- Compliance:

    - Where an Incremental review has been performed, Compliance indicates that the change in the safety requirement, per the topical review, is met.

    - Where a High Level review has been performed, Compliance indicates that the intent of the safety requirement is met.

    - For Clause-by-Clause reviews of modern L/R/C/Ss, Compliance indicates that the safety requirement is met. (Note: No Clause-by-Clause reviews were performed as part of Safety Factor 8.)

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[6] Safety Factor assessments for Review Tasks and L/R/C/S reviews make use of: a) OPG Governance, Programs, Policies and Procedures which support the assessment arguments, b) Commitments previously made to the CNSC, open CNSC action items, and exemptions granted by the CNSC (all related to the Safety Factor under review), as identified in the R04 Pickering LCH [4], c) Identification of previously identified Pickering-specific or programmatic PSR1 gaps related to the Safety Factor under review and the status of OPG's improvement plan(s) or other dispositions to address these, and d) Assessments and reviews performed since the PSR1 documents were completed.

- Gap:
    - Where an Incremental review has been performed, a Gap indicates that the change in the safety requirement, per the topical review, is not met.
    - Where a High Level review has been performed, a Gap indicates that the intent of the safety requirement is not met.
    - For Clause-by-Clause reviews of modern L/R/C/Ss, a Gap indicates that the safety requirement is not met. (Note: No Clause-by-Clause reviews were performed as part of Safety Factor 8.)

The reviews assume that use of the word:

- "Shall" is used in an L/R/C/S to express a requirement, i.e., a provision that the licensee is obliged to satisfy in order to comply with the standard;
- "Should" is used to express a recommendation or that which is advised but not required;
- "May" is used to express an option or that which is permissible within the limits of the standard; and
- "Can" is used to express possibility or capability.

3.3 OPG Program Effectiveness Reviews

As discussed earlier, effectiveness reviews (effectiveness at Pickering NGS) of OPG programs used to demonstrate compliance with the PSR2 Assessment Basis were conducted, using recent applicable audit and self-assessment results:

- OPG Nuclear Oversight independent performance-based Program audits (typically performed in 1 to 5 year cycles) and self-assessments. This includes review of associated Station Condition Records and Action Requests to determine the status of any resulting corrective actions; and
- CNSC "Type I" and "Type II" inspections of the effectiveness and performance of OPG programs, where discussed in OPG audits or self-assessments.

There are many audits and self-assessments that are performed to assess the effectiveness of important aspects of each program. A sample of audits and self-assessments has been summarized for each program in order to demonstrate that program effectiveness is being assessed on an ongoing basis. The focus of these reviews was on effectiveness of the programs at Pickering NGS, where specific information is available. Results from these audits and self-assessments will be considered in the Global Assessment process. It is noted that audits and self-assessments are, by their nature, self-critical and are used to drive excellence in performance. As a result, the broad review scope of program audits focuses on

identifying improvement opportunities rather than presenting a balanced picture of program performance.

Program effectiveness is also monitored and addressed through the Fleetview Program Health and Performance Reporting process [12]. This process involves direct oversight by the Chief Nuclear Officer, and includes participation by the Nuclear Executive Committee members. Programs are reviewed, senior oversight is provided, and improvement plans are generated.

The list of Nuclear Programs to be assessed for each Safety Factor was derived from review of current OPG Governance, and has used the most recent version of these documents as of the PSR2 freeze date of January 15, 2016.

3.4 Additional Reviews

A review of the R04 Pickering LCH [4] was performed to determine if there are any impacts associated with operation of the Pickering Units past 2020 on the following (all related to Safety Factor 8):

- Commitments previously made to the CNSC;
- Open CNSC action items; and
- Exemptions granted by the CNSC.

The PSR2 assessment includes identification and review of previously identified Pickering-specific or programmatic PSR1 gaps related to Safety Factor 8 (as identified in the Darlington ISR Integrated Implementation Plan [13] and Pickering Units 5-8 Continued Operations Plan [11]) to ascertain the status of OPG's improvement plan(s) or other dispositions to address these and the implications of extending operation beyond 2020 (if any).[7]

Fukushima Action Items were reviewed to identify implications of extending operation beyond 2020 (if any). The methodology for this review is provided in a separate PSR2 FAI Review Report.

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[7] PSR2 includes consideration and confirmation that the findings of PSR1 remain valid, as applicable, for the operation period. This includes assessment of PSR1 conclusions against implications resulting from extended operation. In particular, Pickering PSR1 results are applicable to PSR2 if there was a PSR1 gap that is still open, or if a closed PSR1 gap could be affected by extended operation. If so these gaps are carried forward into PSR2 for consideration in the Global Assessment. When references to PSR1 are made, the source document is identified and the relevant text from that source document is summarized in the context of PSR2. With respect to the Darlington ISR, much of the evaluation of Safety Factor health is based on programs and practices that apply across OPG's nuclear operations. As a result, Darlington ISR programmatic conclusions are applicable to Pickering PSR2 where Pickering is confirmed to follow the same nuclear programs and practices that were assessed for Darlington. Darlington PSR1 results are applicable to Pickering PSR2 if there are Darlington PSR1 gaps that are found to be relevant to Pickering PSR2.

Any PSR2 gaps identified as a result of the Safety Factor 8 review which need to be addressed in other Safety Factor reports are also discussed.

4.0 REVIEW FINDINGS

4.1 Review Tasks

The sub-sections below provide an assessment of the adequacy of applicable OPG Governance, Programs, Policies and Procedures (as well as Instructions and Guidelines, as applicable) in demonstrating compliance against the Safety Factor 8 Review Tasks.

4.1.1 Review Task #1: System for Identifying, Classifying, and Recording Safety Related Incidents

Confirm existence of a system for identifying, classifying and recording safety related incidents and operating experience including:

- *Safety related incidents, low level events and near misses;*
- *Safety related operational data;*
- *Maintenance, inspection and testing;*
- *Replacements of SSCs important to safety owing to failure or obsolescence;*
- *Modifications, either temporary or permanent, to SSCs important to safety;*
- *Unavailability of safety systems;*
- *Radiation doses (to workers, including contractors);*
- *Off-site contamination and radiation levels;*
- *Discharges of radioactive effluents;*
- *Generation of radioactive waste.*

The corrective action program, N-PROG-RA-0003, "Corrective Action" [9] establishes the processes to ensure deficiencies, non-conformances, weaknesses with a process, document, or service, or conditions that adversely impact, or may adversely impact plant operations, personnel, nuclear safety, the environment or equipment and component reliability, are promptly identified and corrected or dispositioned. Additionally, the corrective action program provides the processes for ensuring internal and external Operating Experience (OPEX) are evaluated, distributed to appropriate personnel, and applied to implement actions that improve plant safety and reliability. As shown in Figure 1 of N-PROG-RA-0003 [9], there are a number of implementing procedures and interfacing programs. The Corrective Action Program is described in Appendix B.2.

<u>Identifying Incidents</u>

Section 1.1.1 of N-PROG-RA-0003 [9] states that N-PROC-RA-0022, "Processing Station Condition Records" [14] ensures that designs, documents, tools, materials, parts, processes, services, and/or practices that do not meet requirements are promptly identified, documented, reported, analyzed, and corrected.

N-PROC-RA-0022 defines adverse conditions as undesired conditions that are to be screened relative to risk, significance, and consequence. A condition could be adverse if it detracts from the intended function or operation of structures, systems, components, procedures, or practices, and may be caused by inappropriate behaviours, inadequate compliance, acts of nature, or other factors that could impact safety. This definition encompasses adverse conditions related to:

- Safety related incidents, low level events and near misses;
- Safety related operational data;
- Maintenance, inspection and testing;
- Replacements of SSCs important to safety owing to failure or obsolescence;
- Modifications, either temporary or permanent, to SSCs important to safety;
- Unavailability of safety systems;
- Radiation doses (to workers, including contractors);
- Off-site contamination and radiation levels;
- Discharges of radioactive effluents; and
- Generation of radioactive waste.

Adverse conditions identified through interfacing programs (such as N-PROG-MA-0019, "Production Work Management" [15], N-PROG-MP-0004, "Pressure Boundary" [16], N-PROG-MP-0001, "Engineering Change Control" [17], and N-PROG-MA-0026, "Equipment Reliability" [18]), are documented, reviewed, evaluated, corrected and verified in accordance with N-PROC-RA-0022 [14]. As mentioned in P-REP-03680-00017, "Pickering NGS PSR2 Safety Factor 13 Report: Emergency Planning" [19], significant deficiencies identified during Emergency Preparedness drills and exercises also qualify as adverse conditions and are addressed in the Station Condition Record (SCR) process outlined below.

When an adverse condition is discovered, an SCR is initiated and submitted to the initiator's First Line Manager (FLM). The FLM then reviews the SCR to ensure it is in alignment with the SCR initiation process outlined in Section 1.2, "Initiating an SCR", of N-PROC-RA-0022 [14]. The initiator ensures that all required fields have been completed, that personnel names are not included, etc. The initiator then progresses the SCR to "FLM status", and by the end of the shift or within 24 hours, the FLM progresses the entry to "Management Review Meeting" (MRM) status.

Internal OPEX events are identified by the Senior Officer, Performance Improvement – OPEX or Nuclear Support OPEX Coordinator and Evaluating Organization (EO) Manager (or delegate) in accordance with N-GUID-04947.02-10001, "Guidelines for Classifying Internal Events" [20]. The external OPEX process entails identifying CANDU-related OPEX events after the CANDU Owners Group (COG) reviews recent OPEX events from a variety of sources, including World Association of Nuclear Operators (WANO), Institute of Nuclear Power Operators (INPO), and COG. N-GUID-04947.02-10000, "External Events Screening Guide" [21], provides screening guidelines for the review of external OPEX events.

Classifying Incidents

The process for identifying and evaluating degraded station conditions when the ability of SSCs to carry out their defined safety-related functions is questioned is described in the Technical Operability Evaluation (TOE) process in N-PROC-MP-0045, "Technical Operability Evaluation" [22]. The TOE process provides a substantiated engineering verification that a SSC is capable of fulfilling its minimum credited safety function(s). It is required when uncertainty arises with respect to operability of equipment to meet the functional requirements of the defined Safe Operating Envelope (SOE), and is initiated concurrently with an SCR.

The Discovery Issue Resolution Process described in N-PROC-RA-0094, "Discovery Issue Resolution Process" [23] is used when operation of a nuclear facility conforms with its defined SOE, but a problem or a potential problem with the station safety analysis upon which the SOE is based is identified, or when there is a change in analytical basis (such as a gap discovered in the definition of the SOE). This process is initiated to confirm that regulatory limits are met and risk is maintained at an acceptable level. It can also be instigated to put in place mitigating provisions. The Discovery Issue Resolution Process is applied after initiation of an SCR and concurrence is obtained from the appropriate manager, Reactor Safety Engineering Department of the applicable station.

SCR coordinators review all SCRs progressed to MRM status as per Section 1.5, "SCR Coordinator Dispositioning", of N-PROC-RA-0022 [14]. The SCR coordinator determines the Significance Level (S/L) and Resolution Category in accordance with Appendix B, "Station Condition Record Resolution Category Requirements", of N-PROC-RA-0022 [14] and N-LIST-01966-10000, "Station Condition Record Significance Level Criteria" [24].

Table 3, as outlined in N-LIST-01966-10000 [24], shows the various S/Ls for SCRs. S/L is a measure of the impact on Nuclear business deliverables. A detailed list of criteria is available in N-LIST-01966-10000, "Station Condition Record Significance Level Criteria" [24].

Table 3: Definition of Significance Levels 1 to 4

Significance	Description
Level 1	**A highly significant event** or adverse condition or programmatic implementation deficiency that causes a major reduction in the margin of safety to the public or to station personnel and/or which has a major impact on the environment or on production or on other business deliverables.
Level 2	**A significant event** or adverse condition or programmatic implementation deficiency that causes some reduction in the margin of safety to the public or to station personnel and/or which has some impact on the environment or production or on other business deliverables.
Level 3	**An event** or adverse condition or programmatic implementation deficiency, which is not significant by itself but which has the potential to be more significant or which may be the precursor to a more significant event.
Level 4	**A minor condition** adverse to quality, which should help to identify, by means of a trend analysis, those areas that need more attention.

The resolution category applies to the level of effort and investigative rigour that is required to identify and correct the adverse condition. The resolution categories are described in Appendix B of N-PROC-RA-0022 [14] and summarized in Table 4 below.

Table 4: Definition of Resolution Categories A to D

Resolution Category	Description
Category A	An investigation involving multidisciplinary Subject Matter Experts, including an assigned Evaluating Organization Manager accountable for oversight and direction of the team, and a qualified Root Cause evaluator, who will use multiple techniques to determine root cause(s). The causes shall have related Corrective Action(s) that will prevent recurrence.
Category B	An investigation involving a qualified Root Cause Evaluator, with Managers and Subject Matter Experts as required, that will determine root cause(s). The cause(s) will have related Corrective Action(s) that will prevent recurrence.

Resolution Category	Description
Category C	An evaluation of known facts by a person knowledgeable of the issue area that determines the causes and related Corrective Action(s) that will reduce the frequency of recurrence of adverse conditions.
Category D	An isolated incident of a failed barrier, procedural non-compliance or minor condition which can either be fixed or trended to identify areas needing more attention, or the impact associated is low such that no additional corrective action is warranted at this time.

For classifying, reporting, and recording worker and contractor safety incidents and regulatory events within OPG, a procedure is outlined in OPG-PROC-0120, "Safety Incident and Regulatory Event Response" [25]. Per OPG-PROC-0120 [25], line management is responsible for ensuring that all safety incidents are recorded as an SCR according to N-PROC-RA-0022 [14] or an OPG Corporate Report of Incident/Injury. The safety incident is subjected to the safety incident Maximum Reasonable Potential for Harm rating process, in which each incident is reviewed on a case by case basis to determine its potential severity. Upon agreement on the Maximum Reasonable Potential for Harm rating, line management initiates an appropriate investigation and corrective action according to "Safety Incident Investigation and Corrective Actions", OPG-PROC-0121 [26].

<u>Recording Incidents</u>

An SCR contains information fields such as: description of the incident or condition, apparent cause of the condition, recommended resolution, reportability, S/L, trending organization, causal factors (which are used for trending purposes), adverse trend contributing SCRs, and assignments which are part of the Corrective Action Plan (CAP). SCRs are documented through the OPG Station Condition Record system (the Pickering SCR Facility is used for the Pickering station). [8]

Corrective Actions (CAs) from the CAPs are then progressed as per N-PROC-RA-0022 [14], and the status of the SCR is set to IN PROG. Action Tracking Assignments are documented in the OPG document management database Asset Suite. Once all corrective assignments have been completed, the SCR is progressed to COMPLETE.

Procedures are in place for the recording of radiation doses, radioactive effluents, offsite radiation, and radioactive waste. Documentation includes quarterly reports on Safety Performance Indicators. These records are described in detail in Review Task #7.

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[8] Wording has been revised from "SCR database" to "SCR Facility" to reflect the migration of SCR databases into a single web-based platform. "Facility" here does not refer to the physical plant.

The Radiation Protection program, as outlined in N-PROG-RA-0013, "Radiation Protection" [27], implements a series of standards and procedures for the conduct of activities within nuclear sites and with radioactive materials. Events or conditions that indicate real or potential deficiencies identified through the Radiation Protection program are filed as SCRs. The SCRs are subsequently reviewed to determine if reporting to the CNSC in accordance with the applicable CNSC license is necessary.

Further discussion of the use of OPEX is found in P-REP-03680-00013, "Pickering NGS PSR2 Safety Factor 9 Report: Use of Experience from Other Plants and Research Findings" [28].

Conclusion:

The assessment of this Review Task confirms the existence of a system for identifying, classifying, and recording safety related incidents and operating experience. The intent of Review Task #1 is met and therefore Pickering NGS is compliant.

4.1.2 Review Task #2: Safety Related Incidents Investigation – Feedback and Lessons Learned

Confirm that safety related incidents are investigated using root cause analysis and that lessons learned from investigation of these incidents are fed back into the conduct of Operations and Maintenance.

As discussed in Review Task #1, the Corrective Action program, N-PROG-RA-0003 [9], provides the system for identifying, classifying and recording safety related incidents. As part of the Corrective Action program, Root Cause Analyses (RCAs) and Apparent Cause Evaluations (ACEs) are initiated via the SCR process to ensure effective corrective actions are developed and implemented.

Appendix B, "Station Condition Record Resolution Category Requirements", of N-PROC-RA-0022 [14] outlines events that require a RCA or ACE based on an adverse condition's resolution category (A to C). See Review Task #1 for a description of S/L and resolution category. RCAs are required for adverse conditions having resolution category A and B, which are assigned to safety significant events. According to N-STD-RA-0008, "Incident Investigation" [29], the investigation process involves the following key activities:

- Meeting with the EO Manager and establishing terms of reference.
- Assigning team members, developing an investigation plan, meeting with an investigation team and assigning actions.
- Gathering and analyzing information.
- Determining extent of condition, causes, and extent of causes.

- Using operating experience, developing a CAP, and preparing an Incident Investigation Report.

EO Managers, according to N-PROC-RA-0022 [14], are involved in the resolution of all category A, B, and C conditions, and consequently in all RCAs and ACEs. EO Managers are responsible for ensuring:

- The root cause identified in the SCR is addressed with the CAs in accordance to N-PROC-RA-0022 [14], including contributing causes if appropriate.
- CAs from current evaluations address adverse conditions identified in closed out SCRs, and the causes encompass all SCRs closed out to the current evaluation.
- CAPs are developed to balance response with level of significance (risk) of the event or adverse trend.
- Concurrence is obtained from all Responsible Managers assigned responsibility for CA assignments before SCR progresses to "Approve" status.
- An EO Effectiveness Review action is generated for
    - S/L 1 and 2 SCRs.
    - "Corrective Action Review Board" (CARB) - flagged S/L 3 SCRs.
    - OPEX SCRs raised for WANO Significant Operating Experience Report /Significant Event Report and Level 1/2 INPO Event Reports.
- Actions are developed according to "Specific, Measurable, Achievable, Reasonable and Timely" criteria.
- Completion and success criteria are specified for each recurrence control (RC) action assignment.

The Corrective Actions listed in the CAP, in accordance with N-PROG-RA-0003 [9], address the fundamental causes of problems, any generic implications, and the actions to prevent reoccurrence. Further, CAs are documented, communicated to appropriate levels of management, and tracked to completion.

Feedback and Lessons Learned

According to N-STD-RA-0008 [29], RCAs and ACEs are conducted to reduce the probability of similar incidents reoccurring in the future. As outlined in N-PROC-RA-0022 [14], SCR related lessons are transferred and related information is communicated through OPEX mechanisms in accordance with N-PROC-RA-0035, "Operating Experience Process" [30]. This procedure ensures a consistent process is used to evaluate and integrate OPEX information at Ontario Power Generation Nuclear (OPGN). The Internal Information Process (Internal OPEX Process) detailed in N-

PROC-RA-0035 [30], communicates internal events and lessons learned to COG, WANO, and non-incident OPGN sites. Relevant OPEX lessons are also integrated into Training and Nuclear Refurbishment activities in accordance with N-PROC-RA-0035 [30]. In addition, as outlined in N-PROC-OP-0005, "Pre-Job Briefing and Post-Job Debriefing" [31], discussion of relevant OPEX is required during all Pre-job Briefs (PJBs) if the consequence of error is high (Level 3 or 4 PJBs). Internal and external OPEX is also identified in all Level 1 detailed work packages and relevant OPEX is included in all other work, in accordance with N-PROC-MA-0002, "Work Planning" [32]. The level of OPEX discussed during the PJB is determined by the risk of the job.

As stated in N-STD-RA-0008 [29], the status of incident evaluations and the assigned resolution category are presented at a MRM. MRMs, according to Section 1.7, "Management Review Meeting", of N-PROC-RA-0022 [14], ensure the appropriate disposition of SCRs. MRMs ensure sufficient details regarding the adverse condition are provided and ensure the disposition of SCRs is appropriate as outlined in N-PROC-RA-0022 [14]. Additionally, MRMs monitor SCRs to ensure adverse conditions are resolved in a timely manner. For Nuclear Support SCRs, MRMs confirm that any station impacts have been addressed as outlined in N-PROC-RA-0022 [14]. MRMs also determine Site and Department Event Free Day Resets, CAPs to be designated for "CARB Review", and OPEX SCRs requiring CARB Review. MRMs consist of representatives from several organizations, including Operations and Maintenance. In addition, the Chairperson of the Pickering MRMs is the Director of Operations and Maintenance, or their appointed delegate.

Further discussion regarding the OPEX process is provided in P-REP-03680-00013, "Pickering NGS Periodic Safety Review Safety Factor Report: Use of Experience from Other Nuclear Power Plants and Research Findings" [28].

Conclusion:

The assessment of this Review Task confirms that safety related incidents are investigated using root cause analysis and that lessons learned from investigation of these incidents are fed back into the conduct of operations and maintenance. The intent of Review Task #2 is met and therefore Pickering NGS is compliant.

4.1.3 Review Task #3: Minimizing Incident Reoccurrence Using Root Cause Analysis

Confirm that the results of the root cause analysis are used to minimize the chances of the same incident reoccurring.

Solutions to causes identified by Root Cause investigations are designed to prevent reoccurrence, as outlined in Section 1.5.12 of N-STD-RA-0008 [29]. Review Tasks #1 and #2 discuss conditions requiring an RCA, and how lessons are learned and fed back into the conduct of Operations and Maintenance. Reoccurrence of undesirable conditions is prevented by correcting the root cause. CAs are developed to address each identified root cause and, as appropriate, contributing causes in accordance with N-PROC-RA-0022 [14].

Incident investigations are initiated through the SCR process in accordance with N-PROC-RA-0022 [14] or OPG-PROC-0121 [26] for safety events. Appendix B of N-PROC-RA-0022 [14] outlines when to perform an RCA (typically resolution category A and B) or an ACE (typically resolution category C). The SCR process outlined in N-PROC-RA-0022 [14], entails developing a CAP with specific CAs that address each identified root cause. In order to prevent reoccurrence, the SCR process involves identifying adverse conditions, performing RCAs, implementing CAs, and the oversight of an EO Manager and the CARB. For RC, the EO Manager ensures evaluations have at least one RC action to prevent recurrence (for resolution category A and B evaluations) or reduce risk of recurrence (for resolution category C evaluations). See Review Task #1 for a description of S/L and resolution category. SCRs initiated due to safety significant conditions (S/L 1 or 2), as well as CARB-flagged S/L 3 SCRs, are all subject to an Evaluating Organization Effectiveness Review (EOER) as described in N-PROC-RA-0022 [14]. The EOER involves an EO Manager evaluating the effectiveness of S/L 1 and 2 and selected S/L 3 CAPs in accordance with N-GUID-01966-10001, "Evaluating Organization Effectiveness Reviews" [33]. As outlined in Section 1.15, "Evaluation Organization Effectiveness Review", of N-PROC-RA-0022 [14], the EOER should be completed within six months of completion of the last RC action. Further, as stated in Section 1.12 of N-PROC-RA-0022 [14], the Corrective Action Review Board is responsible for assuring the health of the Corrective Action Program. The CARB consists of Directors and senior management, and provides management level oversight of the Corrective Action Program to assure the overall health of the program and verify that significant adverse conditions are being identified and effectively addressed.

As an additional metric that provides fundamental feedback on program effectiveness, potential repeat events are confirmed through MRMs and the Performance Improvement department according to N-PROC-RA-0022 [14]. Repeat events are events previously evaluated via root cause analysis that have any one causal factor group the same and the same terminal condition as previously identified in the last three years.

Conclusion:

The assessment of this Review Task confirms that the results of RCAs are used to minimize the chances of the same incident reoccurring. The intent of Review Task #3 is met and therefore Pickering NGS is compliant.

4.1.4 Review Task #4: Feedback of Trend Analysis into Conduct of Operations and/or Maintenance

Confirm that information from trend analysis of safety related incidents is fed back into the conduct of Operations and/or Maintenance.

Trend Analysis of Safety Related Incidents

Safety related incidents are, as described in Review Task #1, identified, classified and recorded through the SCR process in accordance with N-PROG-RA-0003 [9]. N-PROC-RA-0022 [14] describes the reporting and evaluation process for identified adverse conditions to ensure that:

- Management reviews, screens, and validates identified circumstances of events or adverse conditions;
- Reoccurring or serious nonconforming conditions, potential trends, and significant in-house events are analyzed to determine their causes and generic implications, and corrected to prevent reoccurrence;
- CAs address the fundamental causes of problems, the generic implications, and the actions to prevent reoccurrence of serious nonconforming conditions and potential trends;
- CAs are tracked to completion;
- Overdue CAs are reported to applicable managers on a periodic basis;
- SCR related lessons are shared and communicated through OPEX mechanisms in accordance with N-PROC-RA-0035 [30];
- Trend codes are assigned for applicable trend categories;
- Early identification of potential adverse trends is documented and evaluated; and
- SCR Trend Reports are generated periodically and distributed to management.

When an adverse condition is identified and a subsequent SCR is initiated, the SCR Coordinator applies the appropriate SCR codes as required to perform trend analysis, in accordance with Section 1.5, "SCR Coordinator Dispositioning", of N-PROC-RA-0022 [14]. Trending Organizations are also added or corrected as necessary to identify the most appropriate group to perform trending of the SCR. Up to three Trending Organizations may be added to an SCR if the content of the SCR indicates multiple or cross-functional issues that may be trended and analyzed by additional groups, in accordance with Section 1.2, "Application of SCR Trend Codes", of N-INS-01966.1-10000 [34].

External independent assessments provided by a Nuclear Safety Review Board and the Nuclear Oversight Committee of the Board of Directors, as described in Section 1.1.7, "Independent Assessment", of N-CHAR-AS-0002, "Nuclear Management System" [35], provide a review of the significance of occurrences and trends that may affect nuclear safety and environmental matters. Site cross functional trending and analysis is performed by the department with the program area responsibility or the Performance Improvement Departments for areas not addressed by the program areas. Each quarter, the following station departments perform SCR trend analysis and prepare a Quarterly Performance Improvement Report in adherence to N-INS-01966.1-10000, "Trending and Analysis Instruction and Performance Improvement Reporting" [34]:

- Chemistry;
- Corrective Action;
- Engineering;
- Fuel Handling;
- Human Performance;
- Maintenance;
- Operations;
- Radiation Protection;
- Safety; and
- Work Management.

Other departments identified by an alert group which are assigned as the trending organization for more than 100 SCRs per quarter should perform a quarterly SCR trend analysis and prepare a Quarterly Performance Improvement Report. Line organizations maintain and revise Line Defined Codes as new issues arise that are required to be trended in more detail.

Section 1.5.4 of N-INS-01966.1-10000 [34] outlines the data analysis content included in trend reports at OPG. The reports include:

- SCR Generation Rate;
- Event Based Code Analysis;
- S/L 1 and 2 Analysis;
- Resolution Category Trends;
- Causal Factor Code Analysis;

- Human Performance Code/Event Analysis;
- S-99 Reportable Events; [9]
- Line Defined Code Analysis;
- Observation and Coaching (O&Cs);
- Employee Surveys; and
- Management Focus Areas (Nuclear Safety Culture).

Identification and Characterization of Adverse Trends

N-INS-01966.1-10000 [34] defines an *adverse trend* as follows:

> **Adverse Trend** is a change in performance data that is statistically valid, and that knowledge, experience, and judgement indicate that the performance change is unacceptable.

To facilitate the identification of adverse trends and common issue areas through trending and analysis, SCRs are coded with a minimum of one trend code. N-PROC-RA-0022 [14] states that once an SCR reaches the "MRM" status, the SCR Coordinator assigns any additional trend codes based on an EO Manager's evaluation. The trend codes are selected from the five code sections listed in N-LIST-01966-10001, "Trend Codes Applied to Station Condition Records" [37]. These sections are: Human Performance Trend Codes, Causal Factor Codes, Event Based Codes, Management Focus Area Codes, and Line Defined Codes. A minimum of one Event Based Code is assigned to all SCRs except for non-event SCRs. Non-event SCRs are SCRs that are entered in error, do not represent an adverse condition, are a duplicate of another SCR, or constitute an adverse condition for another facility rather than the originating facility.

For human performance related SCRs, a department Event Free Day Reset code may occur if the requirements described in N-INS-09030-10002, "Site and Department Level Event Free Day Resets" [38], are met. Human performance trend codes listed in Table 1 of N-LIST-01966-10001 [37] describe adverse job site conditions.

As outlined in N-LIST-01966-10001 [37], for all resolution category A, B, and C adverse conditions, at least one Causal Factor Code is required. Causal Factors of adverse conditions at the programmatic level related to Organizational Process and Values (Latent Organizational Weaknesses) are listed in Table 2 of N-LIST-01966-10001 [37].

[9] Per the Pickering License Conditions Handbook [4], the most recent Pickering PROL (48.02/2018) [59] has an amendment to replace S-99 [39] with REGDOC-3.1.1 [36].

For each SCR, at least one Event Based Code is required. Event Based Codes are applied to an SCR to characterize the undesirable incident (event) or observed condition (Job Site Conditions) described in the SCR and are listed in Table 3 of N-LIST-01966-10001 [37].

Reporting of maintenance performance at Pickering NGS is performed quarterly through the Pickering Maintenance Department Performance Improvement Report in accordance with N-INS-01966.1-10000 [34]. SCRs are analyzed to identify new trends, continuing trends, as well as closing trends. For example, in Q4 of 2015, Event Based Codes were applied to SCRs as outlined in P-REP-01966-0575908, "Pickering Maintenance Department Performance Improvement Report, Q4 2015" [40]. This report describes a potential trend identified by Performance Improvement with an Event Based Code. Adverse trend SCR P-2015-29318 was filed to document the potential adverse trend for missing/unavailable tooling and equipment. In response, Crew Event Free Day Resets were applied to SCRs related to missing tools. This allowed tracking on crew level report cards and added focus via review at Crew Management Review Board meetings.

Resolution of Adverse Trends

If analysis identifies an adverse trend, an SCR or Self Assessment is initiated to resolve the issue. An adverse trend SCR is at least resolution category C, as stated in Section 1.6, "Trend Analysis Techniques", of N-INS-01966.1-10000 [34]. As described in N-INS-01966.1-10000 [34], following identification of an adverse trend, the trend performance is tracked until performance is within the acceptable limits for two consecutive quarters. If the implemented actions taken by the EO Manager to reverse a trend are not effective in improving performance over two consecutive quarters, another adverse trend SCR is initiated.

Feedback of Trend Analysis into Conduct of Operations and/or Maintenance

Information from trend analysis is fed back into the conduct of Operations and/or Maintenance through several processes.

The Operations and/or Maintenance line organizations may be selected as one of the Trending Organizations for an SCR. Through the trending process, the Operations and/or Maintenance line organizations inherently share information within the organization.

MRMs involve reviewing all trend SCRs. See Review Task #2 for a description of MRMs. MRMs consist of representatives from several organizations, including Operations and Maintenance. In addition, the Chairperson of the Pickering NGS MRMs is the Director of Operations and Maintenance, or their appointed delegate.

All Performance Improvement Reports are completed and approved by the Department Manager within six weeks following the end of the calendar quarter in accordance with Section 1.3.3 of N-INS-01966.1-10000 [34]. In addition, the CARB reviews Performance Improvement reports to ensure adverse trends are identified and appropriate corrective actions have been identified as outlined in N-PROC-RA-0022 [14].

Conclusion:

The assessment of this Review Task confirms that the information from trend analysis of safety related incidents is fed back into the conduct of operation and/or maintenance. The intent of Review Task #4 is met and therefore Pickering NGS is compliant.

4.1.5 Review Task #5: Adequacy of Performance Indicators

Confirm there is an adequate set of performance indicators that provides a systematic and comprehensive method to record, trend and analyze safety related data including the major system parameters, and maintenance and inspection records.

Performance indicators may include:

- *Frequency of unplanned trips while the reactor is critical*
- *Satisfactory performance of safety system tests within required limits*
- *Special Safety System unavailability*
- *Reliability of Systems Important to Safety*
- *Collective annual radiation dose of plant staff*
- *Amount of gaseous and liquid radioactive release relative to permitted limits*
- *Heavy water escape and loss rates*
- *Fuel reliability*
- *Chemistry index*
- *Volume of Low Level radioactive waste*
- *Change control index*
- *Maintenance backlog*
- *Training*
- *Environment Index*
- *Non-radioactive effluents, including hazardous substances*
- *Non-radioactive wastes*
- *Spills.*

Performance Indicators provide a method of monitoring and measuring the station performance against safety goals and limits. There are two significant sources for measuring performance indicators.

a) In accordance with PROL 48.02/2018 [59], REGDOC-3.1.1 [36] identifies specific performance indicators that must be reported to the CNSC on a quarterly basis. OPG records and reports these in the Quarterly Report on Safety Performance Indicators.

b) OPG monitors performance indicators using the Electronic Performance Reporting (EPR) system to measure the performance of the stations against the business planning goals and targets.

OPGN has implemented a top-down/bottom-up approach to business planning where its leaders establish clear performance targets. The leaders then, in subsequent business plans, identify actions and accountabilities required to achieve these targets over a specified period of time.

The Nuclear business planning framework is a component of the management cycle. It consists of five components: Benchmarking, Setting Strategic Direction, Gap Closure Planning, Developing Detailed Business Plans, and Performance Reporting.

Part of the setting strategic direction phase is to establish the level of performance that the nuclear organization is expected to achieve within the business planning horizon through setting specific targets. Target setting at OPGN is detailed in Section 1.3.4 of N-PROC-AS-0080, "Nuclear Business Planning" [41].

The monitored and measured performance indicators are documented in OPG systems in accordance with N-PROC-AS-0078, "Nuclear Performance Monitoring and Reporting" [42]. N-PROC-AS-0078 [42] defines the requirements for the reporting of OPGN and Nuclear Projects business plan performance results and establishes a process to enable monitoring of performance, based on objective measures to promote and sustain improved performance. Within OPG, a tiered approach for performance indicators has been implemented based on industry best practice. Four tiers are used based on the relevance to the overall Nuclear and site strategic goals and objectives.

- **Tier 1** – Nuclear strategic indicators that tie to the business plan and are benchmarked against industry standards. Tier 1 measures are usually included as part of the Stakeholder Return Program. While Tier 1 measures are of importance to all within Nuclear, they are actively reviewed and managed at the Senior Vice President level and above.

- **Tier 2** – Operational level indicators tied to the business plan to meet strategic Tier 1 goals and are usually found on Nuclear and Site Report Cards. Tier 2 measures are also actively reviewed and managed by Senior Nuclear level management.

- **Tier 3** – Supporting indicators for various functional work groups that tie to the functional area business plans and department/site plans to meet Tier 2 goals. The measures are generally managed and used by the Stratum IV and V level. [10]

- **Tier 4** – Department/site specific indicators established, maintained, and monitored by individual departments, used as precursors to Tier 1, 2 or 3 measures. These measures are the most specific and visible to employees. They may reside in the EPR system or are managed locally by a site or business unit and are generally used by the Stratum III level and below.[11]

The reporting tools used for the collection and transmittal of information are the OPG EPR System and the World Association of Nuclear Operators Data Entry System (WANO DES).

Safety Performance Indicators are reported to the CNSC quarterly and are implemented as specified in Appendix B of REGDOC-3.1.1 [36]. These indicators are:

1) Collective Radiation Exposure
2) Personnel Contamination Events
3) Unplanned Dose/Unplanned Exposure
4) Loose Contamination Events
5) Environmental Releases – Radiological
6) Spills
7) Mispositioning Index
8) Number of Unplanned Transients
9) Reactivity Management Index
10) Unit Capability Factor
11) Unplanned Capability Loss Factor
12) Forced Loss Rate
13) Reactor Trip Rate
14) Corrective Maintenance Backlog
15) Deficient Maintenance Backlog

[10] Director and VP levels.
[11] Manager level.

16) Deferral of Preventive Maintenance

17) Safety System Test Performance

18) Preventive Maintenance Completion Ratio

19) Chemistry Index

20) Chemistry Compliance Index (non-Guaranteed Shutdown State and Guaranteed Shutdown State)

21) Conventional Health and Safety

22) Radiological Emergencies Performance Index

23) Emergency Response Organization Drill Participation Index

24) Emergency Response Resources Completion Index

25) Low- and Intermediate-Level Radioactive Solid Waste Generated

The EPR System is used to report Nuclear business performance management measures. Each EPR measure is assigned a Manager, Verifier and Inputter (MVI) contact at the organizational level to report the measure. The MVI has the responsibility of collecting, inputting, explaining, and verifying information; they must also analyze and initiate actions to correct adverse trends.

WANO DES is used to support the sharing of experience by collecting, trending and disseminating nuclear plant performance in key areas of plant performance. The WANO DES is updated by the DES Inputter. It is the Inputter's job to calculate the performance values each quarter according to WANO specifications.

The detailed process of adding, revising and deleting performance measures is found in N-INS-08115-10000, "Addition, Deletion, and Revision of EPR Performance Measures" [43]. The addition of a new performance measure must be evaluated by the Program Owner (or a delegate), concurred by all affected parties, and approved by the Senior Manager (Planning and Reporting) or Director (Controllership).

Appendix A, "Performance Measure Selection Criteria for Tier Assignment", of N-PROC-AS-0078 [42] identifies the performance measures selection criteria according to the Tier (1, 2, 3, or 4).

The assessment of individual performance indicators is provided in Review Task #6.

Conclusion:

The assessment of this Review Task confirms that there is an adequate set of performance indicators that provides a systematic and comprehensive method to record, trend and analyze safety-related data including the major system parameters and maintenance and inspection records. The intent of Review Task #5 is met and therefore Pickering NGS is compliant.

4.1.6 Review Task #6: Confirmation of Corrective Actions for Unsatisfactory Trends

Confirm that for cases where performance indicators show an unsatisfactory trend, corrective action is taken.

N-PROC-AS-0078 [42] defines the requirements for the reporting of OPGN and Nuclear Projects business plan performance results and establishes a process to enable the monitoring of performance, based on objective measures to promote and sustain improved performance. In particular:

- Section 1.2.4, "Performance Analysis", states that results shall be analyzed and interpreted by focusing on a comparison between the results and the established set of business plan targets. Where performance is not meeting targeted results or is trending adversely, analysis of the data is performed to explain variances and mitigating actions.

- Section 1.5.1, "Data Collection", outlines the required information to analyze, discuss and report for each performance measure by the MVI. This includes a summary of the analysis, year-end and monthly year-to-date performance targets and results, year-end projection, drivers/root causes, mitigating actions completed, and planned actions necessary to close any identified performance gaps.

- Section 1.5.2, "Manager, Verifier, and Inputter", discusses the roles for the MVI. Specifically, subsection (c) states that the Manager is responsible for identifying performance issues and adverse trends, initiating appropriate actions when necessary, and ensuring escalation of performance issues to senior management/Program Owner when adverse performance is not resolved and performance gaps are not closed in a timely manner.

According to N-PROC-AS-0078 [42], the reporting tools for the collection and transmittal of information are the EPR system and the WANO DES. The EPR system and WANO DES are discussed in Review Task #5.

The EPR system standardizes data collection, analysis, and the performance indicator reporting process. It tracks Nuclear, station, and business unit performance measure targets and results in a single repository. Monthly results are rated and colour coded based on performance results versus target. The EPR system, in accordance with N-PROC-AS-0078 [42], is designed to:

- Summarize performance;

- Identify adverse trends and the effectiveness of actions to improve those trends;

- Compare performance between multiple stations and business units on a standardized basis;

- Promote root cause analysis and the development of corrective action plans; and

- Promote ownership and accountability of staff in improving performance.

Section 1.17 of N-PROC-RA-0022 [14] identifies inputs included by the Trend Analysts when performing analysis. This includes corrective action trends, management observations, performance indicators, benchmarking, self-assessment results, and structure, system, equipment and/or component condition data. The trend analysis for these items is performed in accordance with N-INS-01966.1-10000 [34].

The process for initiating and progressing corrective actions when unsatisfactory trends have been identified from trend analysis is described further in Review Task #4.

Conclusion:

The assessment of this Review Task confirms that for cases where performance indicators show an unsatisfactory trend, corrective action is taken. The intent of Review Task #6 is met and therefore Pickering NGS is compliant.

4.1.7 Review Task #7: Adequacy of Records

Review the adequacy of: - *Records of the integrity of physical barriers for the containment of radioactive material.* - *Records of radiation doses to persons on the site.* - *Records of data from off-site radiation monitoring and records of the quantities of radioactive effluents.* - *Records of non-radioactive effluents, including hazardous substances.* - *Records of radioactive and non-radioactive waste.* - *Records of spills.* - *Records of other environmental impacts.*

N-PROG-OP-0006, "Environmental Management" [44] is the program that ensures OPG Nuclear activities are conducted such that adverse environmental effects are prevented or mitigated. The program covers the following activities:

- Requirements of the Ontario Power Generation Environmental Management System that provide the framework for environmental protection within Nuclear; and

- Management's approach to ensure compliance with applicable environmental, legal and other requirements, and conformance with the requirements of the International Organization for Standardization 14001 standard.

As part of the procedures and standards that implement the "Environmental Management" program (N-PROG-OP-0006 [44]), N-STD-OP-0031 "Monitoring of Nuclear and Hazardous Substances in Effluents" [45] establishes minimum requirements for the surveillance and monitoring of nuclear and hazardous substances in airborne and waterborne effluents from OPGN facilities under normal and abnormal operating conditions.

All records and relevant documents are managed through adhering to the standards and procedures outlined in OPG-PROG-0001, "Information Management" [46]. This program lays out requirements for managed systems of activities related to information and documents and establishes uniform, efficient processes for management, maintenance, and final disposition of records and documents throughout Nuclear.

In addition, the following are submitted to CNSC per REGDOC-3.1.1 [36] in accordance with Pickering NGS's Power Reactor Operating Licence, PROL 48.02/2018 [59]:

- Pickering quarterly reports on Safety Performance Indicators, (such as P-CORR-00531-04692 [50] and P-CORR-00531-04625 [51]); and
- An annual environmental protection report.

In OPG, records may be stored in a variety of locations, including the EPR System, Asset Suite, and at Nuclear Records. Each type of record identified in the Review Task is discussed below.

(a) Records of integrity of physical barriers for the containment of radioactive material

N-PROC-MA-0064, "Administrative Requirements for the Periodic Inspection of Nuclear Power Plant Containment Components" [47] provides the administrative process for conducting periodic inspections of CANDU Nuclear Power Plant Containment Components in accordance with Canadian Standards Association (CSA) Standard CAN/CSA-N285.5-08 [48]. Periodic inspection program schedules, inspection records, and inspection reports are stored in Asset Suite. All inspection records from the same inspection campaign are stored electronically and assigned one document number in Asset Suite. This document number is referenced in the inspection report. Further, permanent records and inspection records are maintained in accordance with clauses 12.1 and 12.2 of CSA-N285.5-08 [48], respectively.

In addition, inspection reports are submitted to the CNSC for inspections performed during scheduled maintenance outages (within 90 days of outage completion) and for inspections performed outside a maintenance outage covering a maximum one year period (within 90 days of planned inspection campaign completion).

(b) Records of radiation doses to persons on the site

As described in N-MAN-03416-10000, "Radiation Dosimetry Program-General Requirements" [49], all dose records for Nuclear Energy Workers (both employees and visitors) are transmitted to the National Dose Registry at least once quarterly. The Pickering quarterly reports on Safety Performance Indicators, such as P-CORR-00531-04692 [50] and P-CORR-00531-04625 [51], also contain records of worker collective radiation doses as part of the Collective Radiation Exposure Safety Performance Indicator and are stored in Asset Suite.

(c) Records of data from off-site radiation monitoring and records of the quantities of radioactive effluents.

Information related to the Environment Index, including tritium emissions, is stored in the EPR System. The Pickering quarterly reports on Safety Performance Indicators, such as P-CORR-00531-04692 [50] and P-CORR-00531-04625 [51], also contain records of the quantity of routinely-discharged radioactive effluents and hazardous substances as part of the Environmental Releases Safety Performance Indicator and are stored in Asset Suite.

(d) Records of non-radioactive effluents, including hazardous substances.

The amount of hazardous substances (including concentrations, flow rates, and loadings) released to the environment is monitored as part of OPG's effluent/emission monitoring program and stored in Asset Suite. The amount is also measured in the environment as part of the Environmental Monitoring Program. This information is recorded in the annual report on environmental protection, as outlined in REGDOC-3.1.1 [36].

(e) Records of radioactive and non-radioactive waste.

Records of the amount of low- and intermediate-level radioactive solid waste generated are documented in quarterly reports on Safety Performance Indicators and are stored in Asset Suite, as outlined in REGDOC-3.1.1 [36].

Records of hazardous chemical waste shipments are contained in Ontario Ministry of Environment and Climate Change (MOECC) Waste Manifests, as outlined in P-INS-79000-00010, "Completion of Ministry of the Environment and Climate Change Waste Manifests" [52] for Pickering NGS. These records are sent to the MOECC and stored by Pickering NGS for a minimum of 2 years as required by provincial regulation.

In accordance with N-LIST-00500-10000, "Routine Environment Regulatory Reports/Correspondence" [53], changes in polychlorinated biphenyl (PCB) waste inventories are reported to the MOECC in annual Pickering reports such as "PCB

Waste Inventory Change for Pickering Nuclear", P-CORR-00541-00824 [54], as well as in annual online reports to Environment Canada.

(f) Records of spills.

Reportable spills are immediately reported, as described in N-STD-OP-0026 [55], "Spill Management", to the Ministry of the Environment (now known as MOECC) and other stakeholders. In addition, follow-up written reports are submitted to the regulators within the time limits stipulated by the regulators in accordance with N-PROC-RA-0005, "Written Reporting to Regulatory Agencies" [56]. Further, reportable spills are recorded in the EPR database under Categories A, B, and C as appropriate. Exempted and potential spills are recorded in the EPR database as Category D. Spills may be reported in quarterly safety performance indicator reports if conditions of the spill satisfy definitions outlined in Appendix B, Section 6 of REGDOC-3.1.1 [36].

(g) Records of other environmental impacts.

In accordance with REGDOC-3.1.1 [36], as part of the annual report on environmental protection submitted to CNSC, the amount of nuclear substances measured in the environment is recorded as part of the licensee's Environmental Monitoring Program and stored in Asset Suite.

Conclusion:

The assessment of this Review Task confirms the adequacy of:

- records of the integrity of physical barriers for the containment of radioactive material;
- records of radiation doses to persons on the site;
- records of data from off-site radiation monitoring and records of the quantities of radioactive effluents;
- records of non-radioactive effluents;
- records of radioactive and non-radioactive wastes;
- records of spills; and
- records of other environmental impacts.

The intent of Review Task #7 is met and therefore Pickering NGS is compliant.

4.1.8 Review Task #8: Effect of Changes in Plant Operation on Safety Performance

Consider the effects of any changes in operation at the plant on safety performance. In particular, confirm that current indicators and other safety performance methods continue to be relevant in the context of current and future operations, and confirm that only relevant data and records are used.

Adverse conditions, such as those caused by changes (whether intentional or unintentional) in the operating environment at the plant, are undesired, and are detractions from the intended function or operation of structures, systems and components, and the implementation of procedures and practices. The process for identifying, classifying, and recording these incidents with SCRs has been described in Review Task #1. Additionally, all safety incidents affecting OPG workers and contractors as a result of OPG operations are investigated according to "Safety Incident Investigation and Corrective Actions", OPG-PROC-0121 [26], and are subject to the appropriate CAs. Root cause evaluation tools, such as the fault trees outlined in "Analysis Techniques for Apparent and Root Cause Evaluations", N-GUID-01966-10002 [57], guide evaluators in the analysis process to consider operational problems when determining root causes. In these processes, performance indicators provide the quantitative indications of nuclear plant safety and reliability, plant efficiency, and personnel safety. Performance indicators are therefore used during the data analysis and trending of SCRs, and must be relevant in the context of current and future operations. Appropriate performance indicators are established and produced in support of Nuclear plant operations in accordance with two procedures:

- N-PROC-AS-0078, "Nuclear Performance Monitoring and Reporting" [42], which outlines monitoring of performance based on measures to promote and sustain improved performance.
- N-PROC-RA-0023, "Fleetview Program Health and Performance Reporting" [12], which describes the process for performing a program health and performance review to monitor and routinely report on overall program effectiveness.

According to N-PROC-RA-0022 [14], when initiating SCRs, initiators are responsible for describing adverse conditions clearly and with relevant facts and documents. Similarly, as described in N-PROC-AS-0078 [42], data collected by the Inputter for each performance measure in the WANO DES must be consistent with approved definitions and terminology. Performance measures must be used and applied for the appropriate timeline, as well as reported in a timely manner in the approved format. Data collection, analysis, and the performance indicator reporting process are recorded in the EPR system. The Nuclear performance monitoring and reporting process is based on the accurate and timely reporting of performance results and analysis for a set of performance measures identified during the business planning process.

Performance measures continue to be updated as appropriate against targets established during the business planning process in order to improve system performance. Benchmarking and use of external oversight support the effort to ensure performance indicators are current and useful. The performance level that the nuclear organization is expected to achieve is established during target setting meetings, according to N-PROC-AS-0080 [41]. Through this business planning cycle, irrelevant data and records are omitted from the target setting process and, therefore, are not a part of the process for creating performance measures. Specific targets, such as operational targets, continue to be updated and revised in the first two years of the business planning horizon. During Integration Meetings, attendees are responsible for assessing the potential operational and financial impact of new initiatives intended for the achievement of these targets. To further define and categorize the relevance of performance indicators, N-PROC-AS-0078 [42] indicates that relevance to the overall Nuclear and site strategic goals and objectives is used to rank performance measures into four tiers. For instance, tying directly to the Nuclear Business Plan are Tier 2 performance level measures, meaning they are categorized between corporate level measures and functional level measures in the tiered approach ranking. Tier 2 measures are also selected because they include leading measures, which provide insight on changing performance before the changes significantly impact overall plant operation. A more detailed description of the targets setting process and method of monitoring and measuring station performance is provided in Review Task #5.

N-PROC-MA-0024, "System Performance Monitoring" [58], establishes a process for the effective monitoring, maintenance and enhancement of system performance and reliability. All Pickering safety systems (including systems important to safety, safety related systems, and safety support systems) are monitored in accordance with N-PROC-MA-0024. As part of this process, system performance goals and target values are defined and updated based on sources such as station and department goals, OPEX, and design and safety documentation. Direct and indirect performance indicators of system health are specified to support the collection and analysis of relevant data and records to measure success against these targets. Results of system performance monitoring are used to identify proactive actions to address trends in system health. Performance goals, targets and indicators are documented in System Performance Monitoring Plans, which are reviewed every 2 years (at a minimum) to ensure monitoring objectives are being met and to validate adequacy of equipment surveillance. As shown in Figure 1 of N-PROC-MA-0024, the feedback process for system performance monitoring ensures that performance measures are maintained relevant for current and future operations.

Conclusion:

The assessment of this Review Task confirms that current indicators and other safety performance methods continue to be relevant in the context of current and future operations, and confirm that only relevant data and records are used. The intent of Review Task #8 is met and therefore Pickering NGS is compliant.

4.2 L/R/C/S Reviews

As per Section 2.2 of this report, detailed reviews for twelve L/R/C/Ss with content applicable to Safety Factor 8 are provided in References [6] and [7]. Associated findings applicable to Safety Factor 8 are summarized in Table 5 below.

Table 5: PSR2 L/R/C/S Review Results for Safety Factor 8

L/C/R/S Reviewed	PSR2 L/R/C/S Review Results for Safety Factor 8
CSA N290.15-10, "Requirements for the Safe Operating Envelope of Nuclear Power Plants"	There are no PSR2 gaps for CSA N290.15-10 (R2015). Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CSA N290.15-10 (R2015).
CSA N288.1-14, "Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities"	There are no PSR2 gaps for CSA N288.1-14. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CSA N288.1-14.
CSA N288.4-10, "Environmental Monitoring Program at Class I Nuclear Facilities and Uranium Mines and Mills"	There are no PSR2 gaps for CSA N288.4-10. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CSA N288.4-10.
CNSC REGDOC-2.9.1 (2013), "Environmental Protection Policies, Programs and Procedures"	There are no PSR2 gaps for CNSC REGDOC-2.9.1 (2013). Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CNSC REGDOC-2.9.1 (2013).
CNSC G-129 (2004), "Keeping Radiation Exposures and Doses 'As Low As Reasonably Achievable (ALARA)'"	There are no PSR2 gaps for CNSC G-129 Revision 1 (2004). Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CNSC G-129 Revision 1 (2004).
CNSC G-228 (2001), "Developing and Using Action Levels"	There are no PSR2 gaps for CNSC G-228 (2001). Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CNSC G-228 (2001).
SOR/2000-203 (Amended June 2015), "The Radiation Protection Regulations"	There are no PSR2 gaps for the Radiation Protection Regulations (Amended June 2015). Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with the Radiation Protection Regulations (Amended June 2015).
CSA N288.6-12, "Environment Risk Assessments at Class I Nuclear Facilities and Uranium Mines and Mills"	There are no PSR2 gaps for CSA N288.6-12. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CSA N288.6-12.

L/C/R/S Reviewed	PSR2 L/R/C/S Review Results for Safety Factor 8
CSA N288.5-11, "Effluent Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills"	There are no PSR2 gaps for CSA N288.5-11. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CSA N288.5-11.
CNSC REGDOC-2.3.2 (2015), "Accident Management, Version 2"	For Safety Factor 8, there are no PSR2 gaps for CNSC REGDOC-2.3.2 (2015).
CNSC REGDOC-2.3.3 (2015), "Periodic Safety Reviews"	There are no PSR2 gaps for CNSC REGDOC-2.3.3 (2015). Per the definition of Compliance for a High Level review, Pickering has a PSR2 Compliance associated with CNSC REGDOC-2.3.3 (2015).
CSA N288.3.4-13, "Performance Testing of Nuclear Air-Cleaning Systems at Nuclear Facilities"	There are no PSR2 gaps for CSA N288.3.4-13. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CSA N288.3.4-13.

4.3 OPG Program Effectiveness Reviews

The OPG Programs reviewed for Safety Factor 8 are identified in Table 2 and details of the associated effectiveness reviews for each of the N-PROGs are provided in Appendix B.

4.4 Additional Review Findings

As discussed in Section 3.4, the PSR2 Safety Factor 8 assessment also included a review of commitments previously made to the CNSC, open CNSC action items, and exemptions granted by the CNSC, as identified in the R04 Pickering LCH [4], to determine if there are any impacts associated with operation of the Pickering Units past 2020. The review also included identification and review of previously identified programmatic Darlington PSR1 gaps related to Safety Factor 8 to determine impacts associated with operation of the Pickering Units past 2020. This assessment did not find any PSR2 gaps for Safety Factor 8.

Findings from the review of previously identified PSR1 gaps in the Pickering Units 5-8 Continued Operations Plan [11] are provided in a separate PSR2 COP Review Report. Findings from the review of Fukushima Action Items are provided in a separate PSR2 FAI Review Report. Results from the Continued Operations Plan and Fukushima Action Items reviews will be considered in the Global Assessment process.

There were no PSR2 gaps identified in this Safety Factor 8 Report that require discussion in other Safety Factor Reports.

5.0 RESULTS AND CONCLUSIONS

OPG Governance, Programs, Policies, Procedures, Instructions and Guidelines related to Safety Factor 8 were reviewed for the eight PSR2 Review Tasks in Section 4.1 of this report and resulted in no Pickering PSR2 Gaps. L/R/C/S and OPG Nuclear Program effectiveness reviews for Safety Factor 8 were prepared per Sections 4.2 and 4.3, respectively, and resulted in no PSR2 gaps. Per Section 4.4, this report also included identification and review of previously identified programmatic Darlington PSR1 gaps related to Safety Factor 8 (to ascertain the implications of extending Pickering NGS operation beyond 2020), as well as a review of the R04 Pickering LCH [4] for any impacts of Pickering NGS operation beyond 2020 on a) OPG commitments previously made to the CNSC, b) open CNSC action items, and c) exemptions granted by the CNSC (all related to Safety Factor 8), which resulted in no PSR2 gaps.

The review of Safety Factor 8 has confirmed that the safety performance indicators and records of operating experience, including the evaluation of root causes of plant events, exist and are utilized to ensure the safe operation of Pickering NGS.

6.0 REFERENCES

[1] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[2] CNSC REGDOC-2.3.3, *Periodic Safety Reviews*, April 2015.

[3] IAEA Safety Guide No. SSG-25, *Periodic Safety Review for Nuclear Power Plants*, March 2013.

[4] CNSC Report, LCH-PNGS-R004, *Pickering NGS: Licence Conditions Handbook*, December 23, 2015.

[5] OPG Report, P-REP-03680-00003 R000, *Pickering NGS PSR2: Definition of Safety Factor Review Tasks*, May 2016.

[6] OPG Report, P-REP-03680-0586480 R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 9, 11, and 15*, September 2016.

[7] OPG Report, P-REP-03680-00021 R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 8, 10, 12, 13, and 14*, December 2016.

[8] OPG Program, N-PROG-RA-0002 R008, *Conduct of Regulatory Affairs*, February 2015.

[9] OPG Program, N-PROG-RA-0003 R010, *Corrective Action*, January 2015.

[10] OPG Program, OPG-PROG-0010 R003, *Health and Safety Management Program*, January 2015.

[11] OPG Plan, NK30-PLAN-00531-00001 R005, *Pickering 5-8 Continued Operations Plan*, November 2015.

[12] OPG Procedure, N-PROC-RA-0023 R018, *Fleetview Program Health and Performance Reporting*, August 2013.

[13] OPG Report, NK38-REP-03680-10185 R002, *Darlington NGS Integrated Implementation Plan (IIP)*, April 2015.

[14] OPG Procedure, N-PROC-RA-0022 R032, *Processing Station Condition Records*, November 2014.

[15] OPG Program, N-PROG-MA-0019 R009, *Production Work Management*, December 2014.

[16] OPG Program, N-PROG-MP-0004 R016, *Pressure Boundary*, November 2015.

[17] OPG Program, N-PROG-MP-0001 R014, *Engineering Change Control*, January 2015.

[18] OPG Program, N-PROG-MA-0026 R002, *Equipment Reliability*, June 2015.

[19] OPG Report, P-REP-03680-00017 R000, *Pickering NGS PSR2 Safety Factor 13 Report: Emergency Planning*, December 2016.

[20] OPG Guideline, N-GUID-04947.02-10001 R003, *Guidelines for Classifying Internal Events (WANO, COG, & Fleetwide)*, July 2015.

[21] OPG Guideline, N-GUID-04947.02-10000 R001, *External Events Screening Guide,* December 2012.

[22] OPG Procedure, N-PROC-MP-0045 R008, *Technical Operability Evaluation*, September 2015.

[23] OPG Procedure, N-PROC-RA-0094 R006, *Discovery Issue Resolution Process*, June 2015.

[24] OPG List, N-LIST-01966-10000 R008, *Station Condition Record Significance Level Criteria*, June 2014.

[25] OPG Procedure, OPG-PROC-0120 R002, *Safety Incident and Regulatory Event Response*, March 2015.

[26] OPG Procedure, OPG-PROC-0121 R001, *Safety Incident Investigation and Corrective Actions*, October 2014.

[27] OPG Program, N-PROG-RA-0013 R009, *Radiation Protection*, January 2015.

[28] OPG Report, P-REP-03680-00013 R000, *Pickering NGS PSR2 Safety Factor 9 Report: Use of Experience from Other Nuclear Power Plants and Research Findings*, October 2016.

[29] OPG Standard, N-STD-RA-0008 R013, *Incident Investigation*, November 2014.

[30] OPG Procedure, N-PROC-RA-0035 R018, *Operating Experience Process*, October 2014.

[31] OPG Procedure, N-PROC-OP-0005 R012, *Pre-Job Briefing and Post-Job Debriefing*, June 2013.

[32] OPG Procedure, N-PROC-MA-0002 R028, *Work Planning*, April 2014.

[33] OPG Guideline, N-GUID-01966-10001 R003, *Evaluating Organization Effectiveness Reviews*, March 2015.

[34] OPG Instruction, N-INS-01966.1-10000 R006, *Trending and Analysis Instruction and Performance Improvement Reporting*, April 2014.

[35] OPG Charter, N-CHAR-AS-0002 R018, *Nuclear Management System*, March 2015.

[36] CNSC, REGDOC-3.1.1, *Reporting Requirements for Nuclear Power Plants*, May 2014.

[37] OPG List, N-LIST-01966-10001 R010, *Trend Codes Applied to Station Condition Records*, January 2015.

[38] OPG Instruction, N-INS-09030-10002 R008, *Site and Department Level Event Free Day Resets,* June 2015.

[39] CNSC Regulatory Standard S-99, *Reporting Requirements for Operating Nuclear Power Plants*, March 2003.

[40] OPG Report, P-REP-01966-0575908 R00, *Pickering Maintenance Department Performance Improvement Report, Q4 2015*, February 2016.

[41] OPG Procedure, N-PROC-AS-0080 R003, *Nuclear Business Planning*, December 2013.

[42] OPG Procedure, N-PROC-AS-0078 R004, *Nuclear Performance Monitoring and Reporting*, May 2014.

[43] OPG Instruction, N-INS-08115-10000 R002, *Addition, Deletion, and Revision of EPR Performance Measures*, June 2014.

[44] OPG Program, N-PROG-OP-0006 R018, *Environmental Management*, April 2015.

[45] OPG Standard, N-STD-OP-0031 R006, *Monitoring of Nuclear and Hazardous Substances in Effluents*, October 2014.

[46] OPG Program, OPG-PROG-0001 R009, *Information Management*, September 2015.

[47] OPG Procedure, N-PROC-MA-0064 R005, *Administrative Requirements for the Periodic Inspection of Nuclear Power Plant Containment Components*, October 2013.

[48] CSA Standard, CAN/CSA-N285.5-08, Update 1, *Periodic Inspection of CANDU Nuclear Power Plant Containment Components,* January 2011.

[49] OPG Manual, N-MAN-03416-10000 R001, *Radiation Dosimetry Program – General Requirements*, July 22, 2013.

[50] OPG Letter, P-CORR-00531-04692 R000, K. Dehdashtian to M. Santini, *Pickering Quarterly Report on Safety Performance Indicators – Fourth Quarter of 2015*, March 30, 2016.

[51] OPG Letter, P-CORR-00531-04625 R000, K. Dehdashtian to M. Santini, *Pickering Quarterly Report on Safety Performance Indicators – Third Quarter 2015*, December 15, 2015.

[52] Pickering Instruction, P-INS-79000-00010 R003, *Completion of Ministry of the Environment and Climate Change Waste Manifests*, June 2015.

[53] OPG List, N-LIST-00500-10000 R04, *Routine Environment Regulatory Reports/Correspondence*, September 2014.

[54] OPG Letter, P-CORR-00541-00824 R000, D. Ginter to C. Dugas, *PCB Waste Inventory Change for Pickering Nuclear*, January 21, 2016.

[55] OPG Standard, N-STD-OP-0026 R008, *Spill Management*, March 2014.

[56] OPG Procedure, N-PROC-RA-0005 R015, *Written Reporting to Regulatory Agencies*, January 2015.

[57] OPG Guideline, N-GUID-01966-10002 R000, *Analysis Techniques for Apparent and Root Cause Evaluations*, November 2010.

[58] OPG Procedure, N-PROC-MA-0024 R015, *System Performance Monitoring*, October 2013.

[59] CNSC, PROL-48.02/2018, *Nuclear Power Reactor Operating Licence: Pickering Nuclear Generating Station*, December 18, 2015.

Appendix A: Nomenclature

ACE	Apparent Cause Evaluations
CA	Corrective Action
CANDU	CANada Deuterium Uranium
CAP	Corrective Action Plan
CARB	Corrective Action Review Board
CNSC	Canadian Nuclear Safety Commission
COG	CANDU Owners Group
COP	Continued Operations Plan
CSA	Canadian Standards Association
EO	Evaluating Organization
EOER	Evaluating Organization Effectiveness Review
EPR	Electronic Performance Reporting
FAI	Fukushima Action Item
FLM	First Line Manager
IAEA	International Atomic Energy Agency
INPO	Institute of Nuclear Power Operators
ISR	Integrated Safety Review
LCH	Licence Conditions Handbook
MOECC	Ministry of Environment and Climate Change
MRM	Management Review Meeting
MVI	Manager, Verifier and Inputter
NGS	Nuclear Generating Station
OPEX	Operating Experience
OPG	Ontario Power Generation
OPGN	Ontario Power Generation Nuclear
PARTS	Pickering A Return to Service
PCB	Polychlorinated Biphenyl
PJB	Pre-Job Brief
PROL	Power Reactor Operating Licence
PSR	Periodic Safety Review
PSR1	Periodic Safety Review 1 (earlier OPG PSR work and other associated assessments)

PSR2	Periodic Safety Review 2 (subsequent PSR per REGDOC-2.3.3)
RC	Recurrence Control
RCA	Root Cause Analyses
SAP	Stabilization Activity Plan
SCR	Station Condition Record
S/L	Significance Level
SOE	Safe Operating Envelope
SSC	Structures, Systems, and Components
TOE	Technical Operability Evaluation
WANO	World Association of Nuclear Operators
WANO DES	World Association of Nuclear Operators Data Entry System

Appendix B: OPG Program Effectiveness Review Results

B.1 N-PROG-RA-0002, "Conduct Of Regulatory Affairs"

The purpose of the Conduct of Regulatory Affairs program is to ensure Ontario Power Generation (OPG) complies with regulatory requirements in an effective and efficient manner. The program describes the procedures related to licensing, regulatory interpretations, event reporting, regulatory approvals, Canadian Nuclear Safety Commission (CNSC) inspections, issue management, and communications.

The program includes broad guidelines for managing the interface with regulatory agencies, primarily the CNSC, to ensure effective and efficient compliance with regulatory requirements and to ensure open, honest and timely communications. Successful interface with regulatory agencies is critical in meeting OPG Nuclear objectives.

Nuclear Oversight conducted an audit of the of the Regulatory Affairs and Safeguards programs in July 2012, NO-2012-001 [B.1.1], for both Pickering and Darlington NGS. The objective of the Audit was to confirm that the Conduct of Regulatory Affairs program (as well as the Safeguards program) are effectively managed and in compliance with governing documents. This audit found the Regulatory Affairs program to be effective and consistently met OPG Nuclear requirements. An improvement opportunity was identified regarding Regulatory Affairs training deficiencies. SCR N-2012-03670 (AR 28146389) was initiated to address this and has since been closed with corrective actions completed to address the underlying issues.

Nuclear Regulatory Affairs completed a self-assessment in May 2012, NO12-000075-SA [B.1.2], in order to confirm the adequacy of the Nuclear Regulatory Affairs governance and program implementation for Pickering and Darlington NGS. The self-assessment determined that a number of Nuclear Regulatory Affairs governance documents were beyond their review cycle. SCR N-2012-03002 was initiated to address this and has since been closed with corrective actions completed to address the underlying issues.

The Operations and Maintenance Support department completed a self-assessment in April 2013, NO13-000230-SA [B.1.3], in order to assess the health of N-PROG-RA-0002, "Conduct of Regulatory Affairs". This involved a review of related SCRs, governance framework, revision records and previous program assessment reports. No findings/SCRs were initiated as a result of this self-assessment.

References

[B.1.1] OPG Nuclear Oversight Report, N-REP-01070-0414435 T06 (NO-2012-001), *Audit OPGN NO-2012-001, Regulatory Affairs and Safeguards*, July 26, 2012.

[B.1.2] Self-Assessment Report, NO12-000075-SA, *Conduct of Regulatory Affairs Governance Compliance,* May 1, 2012.

[B.1.3] Self-Assessment Report, NO13-000230-SA, *Program Management Assessment-N-PROG-RA-0002,* April 11, 2013.

B.2 N-PROG-RA-0003, "Corrective Action"

The Corrective Action program establishes the processes to ensure deficiencies; non-conformances; weaknesses with a process, document, or service, or conditions that adversely impact, or may potentially adversely impact plant operations; personnel; nuclear safety; the environment or equipment; and component reliability, are promptly identified and corrected or dispositioned. For those deficiencies considered significant or repetitive in nature, these processes ensure appropriate levels of management are notified, causes identified and actions taken to minimize or prevent recurrence and actions taken to address the identified issues are verified to be complete and effective.

Utilizing Operating Experience (OPEX) from within OPG Nuclear and the industry is an integral part of the Corrective Action Program. Hence, this program also provides the processes to ensure internal and external OPEX is evaluated, distributed to appropriate personnel, and applied to implement actions that improve plant safety and reliability.

Nuclear Oversight conducted a performance based audit of the OPG Nuclear Corrective Action Program in December 2013, NO-2013-023 [B.2.1], for both Pickering and Darlington NGS. The purpose of the audit was to determine the effectiveness of the OPG Nuclear Corrective Action program across the fleet. The audit concluded that performance improvement opportunities applicable to Pickering NGS existed associated with Corrective Action program expectations, management oversight for SCR dispositioning, trending and analysis, and monitoring of staff training qualifications.

Four SCRs (N-2014-02323, N-2014-02324, P-2014-02325 and N-2014-02329) were initiated during this audit, which required corrective actions to be implemented. The necessary corrective actions were completed to address the underlying issues and these SCRs have since been closed.

Nuclear Oversight completed a self-assessment in May 2013, NO13-000038-SA [B.2.2], in order to assess the effectiveness of the 2012 and 2013 Corrective Action program Improvement Plans. This self-assessment is applicable to both Pickering and Darlington NGS. It was concluded that substantial improvements were made to the Corrective Action program, however there were opportunities to improve in the areas of self-assessment approval, governance streamlining, and training and qualification updates.

All the necessary corrective actions were completed to address the underlying issues and the associated SCR (N-2013-02155) has since been closed.

References

[B.2.1] Nuclear Oversight Report, N-REP-01070-0435159 T06 (NO-2013-023), *OPGN Corrective Action Program (CAP) Audit*, January 2014.

[B.2.2] Self-Assessment Report, N013-000038-SA, *OPGN Corrective Action Program for NOA-2013-023,* May 2013.

B.3 OPG-PROG-0010, "Health and Safety Management System Program"

The Ontario Power Generation (OPG) Health and Safety Management System program establishes the process requirements that must be implemented and maintained to ensure that health and safety risks to workers are being mitigated. It also outlines the responsibilities of the various levels of the organization to ensure these activities are carried out. The Health and Safety Management System includes:

- Occupational conditions and factors that could affect the health and safety of workers, in all workplaces or from work-related activities under the control of OPG.
- Non-occupational health-related conditions and factors that could affect the health of OPG workers where it impacts achievement of OPG's business objectives.
- Contractor safety.

Nuclear Oversight completed a self-assessment in March of 2013, NO13-000121-SA [B.3.1], in preparation for Nuclear Oversight's Conventional Safety Program[12] audit, which is applicable for both Darlington and Pickering NGS. The scope of the self-assessment included a review of previous Nuclear Oversight audit corrective actions associated with gaps/findings in order to confirm completion status as well as a review of specific conventional safety hazard/risk areas (i.e., Asbestos Management, Electrical Safety and On-Site Driving). All previous Nuclear Oversight audit corrective actions were found to be completed or on-track for completion as committed and the conventional safety hazard/risk areas assessed were found to meet or exceed applicable requirements. No findings/SCRs were initiated as result of this self-assessment.

Nuclear Oversight conducted a performance based audit of the Conventional Safety Program in October 2013, NO-2013-027 [B.3.2], for Pickering and Darlington NGS. The objective of the audit was to ensure that applicable requirements are being met and that risks to workers and members of the public are appropriately mitigated. The audit concluded that performance improvement opportunities applicable to Pickering NGS existed in the areas of Conventional Safety training, use of performance improvement tools by the Conventional Safety group, documentation to support Personal Protective Equipment and scaffold storage racks, and Ontario Health and Safety Act notice boards compliance.

Five SCRs were initiated to address the above findings which required corrective actions to be implemented, of which four have been completed (SCRs N-2013-14928,

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12 Note, this self-assessment was based on the requirements of N-PROG-HR-0004 R003, "Conventional Safety" which has been superseded by OPG-PROG-0010, "Health and Safety Management System Program". Since the fundamental Health and Safety requirements have not changed between the two documents, a discussion on findings related to N-PROG-HR-0004 (as opposed to OPG-PROG-0010) is considered appropriate.

N-2013-14930, N-2013-14931 and N-2013-14934), while the remaining (SCR N-2013-14933) is expected to be completed by Q4 2017.

References

[B.3.1] Self-Assessment Report, NO13-000121-SA, *Industrial Safety*, March 2013.

[B.3.2] OPG Nuclear Oversight Report, N-REP-01070-0435163 T06, *Audit OPGN NO-2013-027, Conventional Safety Program Audit*, October 10, 2013.

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ONTARIO POWER GENERATION

ACCEPTED	✓
ACCEPTED AS NOTED	
REVISE AND RESUBMIT	
Signature	21 Oct 2016 Date
Name: Mike Ruffolo, Manager	
Dept: Pickering Engineering - Aging Management & Strategic Initiatives	
OPG Proprietary	
Doc No.: P-REP-03680-00013	Rev: 000
This acceptance does not relieve the contractor from responsibility for errors or omissions or from any obligations or liability under this contract.	

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Pickering NGS PSR2 Safety Factor 9 Report: Use of Experience from Other Nuclear Power Plants and Research Findings

PS112/RP/002 R03

October 20, 2016

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Revision Summary – For Amec Foster Wheeler Report PS112/RP/002

Rev	Date	Author	Comments
R00	October 28, 2015	S. B. Harvey	Initial issue for OPG review and comment.
R01	July 22, 2016	S. B. Harvey	Addressed OPG comments and aligned with updated PSR2 Basis Document (P-REP-03680-00001 R02)
R02	September 26, 2016	S. B. Harvey	Addressed OPG comments, and incorporated Fukushima Action Item assessment in Appendix D.
R03	October 20, 2016	S. B. Harvey	Addressed OPG comments, and moved Fukushima Action Item assessment to a stand-alone report. Discussion of COP actions was also moved to a stand-alone report.

EXECUTIVE SUMMARY

Ontario Power Generation (OPG) is performing a Periodic Safety Review (PSR) to support the possibility of continued operation of Pickering Nuclear Generating Station (NGS) beyond 2020. The PSR (referred to as "PSR2") is a subsequent PSR building on the review basis of earlier OPG Integrated Safety Reviews and other associated assessments. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

Part of PSR2 involves the preparation of Safety Factor reports for each of fifteen major topic areas. Safety Factor reports consist of:

- Assessing compliance against "Review Tasks" identified in the PSR2 Basis Document [1]. These Review Tasks are derived from Canadian Nuclear Safety Commission (CNSC) REGDOC-2.3.3, "Periodic Safety Reviews" [2] and International Atomic Energy Agency (IAEA) SSG-25, "Periodic Safety Review for Nuclear Power Plants" [3];

- Documenting assessments against applicable modern Laws, Regulations, Codes and Standards (L/R/C/Ss) as defined in Reference [1]; and

- Reviewing the effectiveness of applicable OPG Nuclear Programs that support the above assessments, through review of audit and self-assessment results.

The PSR2 review of Safety Factor 9, *Use of Experience from Other Nuclear Power Plants (NPPs) and Research Findings* is presented in this report. OPG Governance, Programs, Policies, Procedures, Instructions and Guidelines related to Safety Factor 9 were reviewed for the six PSR2 Review Tasks specified in Section 4.1 of this report. L/R/C/S and OPG Nuclear Program audit and self-assessment reviews for Safety Factor 9 were prepared per Sections 4.2 and 4.3, respectively. Per Section 4.4, the PSR2 assessment includes a review of previously identified PSR1 gaps related to Safety Factor 9 (to ascertain the implications of extending Pickering NGS operation beyond 2020), as well as a review of the R04 Pickering Licence Conditions Handbook [4] for any impacts of Pickering NGS operation beyond 2020 on: a) OPG commitments previously made to the CNSC, b) open CNSC action items, and c) exemptions granted by the CNSC (all related to Safety Factor 9).

The results of the review of Safety Factor 9 are discussed in Section 5.0 of this report. The review has confirmed for Pickering NGS that there is adequate feedback of relevant experience from other nuclear power plants and from findings of research, and that this is used to introduce reasonable and practicable safety improvements at the plant or in the operating organization. As discussed in Section 5.0, the review identified no Pickering PSR2 gaps.

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1.0 INTRODUCTION

Ontario Power Generation (OPG) is performing a Periodic Safety Review (PSR) to support the possibility of continued operation of Pickering Nuclear Generating Station (NGS) beyond 2020.[1] A comprehensive Integrated Safety Review (ISR) was completed for Pickering Units 5 through 8 in 2009, in support of refurbishment and continued operation. Pickering Units 1,4 integrated safety assessments were also performed for Pickering A Return to Service (PARTS) in support of approval to restart Units 1 and 4. In addition to these Pickering-specific studies, the 2013 Darlington ISR performed extensive code and standard reviews that were updated in relation to the versions that were assessed in the 2009 Pickering B ISR.[2] These previous ISRs are considered to constitute the first PSR completed for Pickering (referred to as "PSR1"). The current PSR (referred to as "PSR2") is a subsequent PSR building on the basis of earlier OPG integrated safety assessments through review of the various studies, assessments and licence renewals performed since PSR1. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

PSR2 will support and complement the licence renewal application for Pickering NGS going forward. Fifteen Safety Factors will be assessed as part of the PSR. The purpose of Safety Factor reviews is to confirm that the design, condition and operation of Pickering Units 1,4 and 5-8 (as well as common systems) will support continued safe operation for the period of PSR2 by:

- Assessing compliance against "Review Tasks" identified in the PSR2 Basis Document [1], which were derived from Canadian Nuclear Safety Commission (CNSC) REGDOC-2.3.3 [2] and International Atomic Energy Agency (IAEA) SSG-25 [3];

- Documenting assessments against applicable modern Laws, Regulations, Codes and Standards (L/R/C/Ss) (as defined in Reference [1]); and

- Reviewing the effectiveness of applicable OPG Nuclear Programs that support the above assessments, through review of audit and self-assessment results.

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1 Currently, Pickering Units 5-8 are approved to operate to 247,000 Effective Full Power Hours. This operation limit is expected to be reached on some units in 2020. For the purposes of PSR2, OPG assumes operation of Pickering NGS for up to eight additional years, from 2020 until 2028. OPG will make a decision regarding the permanent shut down dates for the six reactors following the performance of a technical evaluation that will include PSR2, and will communicate it to the CNSC as required by the current Power Reactor Operating Licence (PROL).

2 Much of the compliance assessment and evaluation of Safety Factor health for the Darlington ISR is based on programs and practices that apply across OPG's nuclear operations. As a result, where Pickering is confirmed to follow the same nuclear programs and practices as were assessed for Darlington, the Darlington ISR programmatic conclusions are applicable to Pickering. As discussed in Section 1.0, an effectiveness review (effectiveness at Pickering NGS) of OPG programs used to demonstrate compliance with the PSR2 Assessment Basis will be conducted using recent audit and self-assessment results.

As outlined in IAEA SSG-25 [3], the objective of the review of Use of Experience from Other Nuclear Power Plants (NPPs) and Research Findings, Safety Factor 9, is to: "determine whether there is adequate feedback of relevant experience from other nuclear power plants and from the findings of research and whether this is used to introduce reasonable and practicable safety improvements at the plant or in the operating organization." REGDOC-2.3.3 [2] requires that: "The licensee shall conduct a PSR in accordance with this regulatory document for the period until the next PSR or, if applicable, until the end of commercial operation of the plant."

This report documents the results of the review of Safety Factor 9 for Pickering PSR2. The report is based on the OPG Governance, Programs, data, and material available up to January 15, 2016 which is the freeze date for PSR2.

2.0 SCOPE OF REVIEW

2.1 Review Task Assessments

The Pickering PSR2 Safety Factor 9 Review Tasks are defined in Reference [1]. Details of the derivation of these Review Tasks from CNSC REGDOC-2.3.3 [2] and IAEA SSG-25 [3] are shown in Reference [5]. The Safety Factor 9 Review Tasks are:

1) Confirm existence and adequacy of a program for the sending and receiving of experience relevant to safety to and from other nuclear power plants and relevant nonnuclear plants. ("Other nuclear power plants" specifically include the IAEA, OECD/NEA[3], WANO[4], INPO[5] as well as CANDU[6] Owners Group[7] (COG) and experience within OPG at Darlington.)

2) Confirm existence of a program for receiving of information on the findings of relevant research programs.

3) Confirm there is a process for assessing the significance of operating experience from other plants and incorporating the lessons learned into improving safety performance at the station.

4) Confirm that there is a process for assessing the significance of research findings and technology developments and for incorporating relevant improvements into the station's design and operation.

5) Review adequacy and effectiveness of the feedback arrangements and timely implementation of assessment findings. (Assess program audit results).

6) List the major OPEX[8] events and resulting plant changes that have resulted since PSR1 was completed.

The methodology for the reviews is discussed in Section 3.1. Review Task findings are summarized in Section 4.1 of this Report.

2.2 L/R/C/S Reviews

The applicable Laws, Regulations, Codes and Standards relevant to the Use of Experience from Other NPPs and Research Findings Safety Factor are identified in

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3 Organization for Economic Cooperation and Development, Nuclear Energy Agency
4 World Association of Nuclear Operators
5 Institute of Nuclear Power Operations
6 CANada Deuterium Uranium
7 CANDU Owners Group (COG)
8 "Operating Experience" (OPEX) is a term that encompasses Experience from Other NPPs and Research Findings

Reference [1] and are listed in Table 1 below. Table 1 also identifies the modern version and date of each L/R/C/S to be considered, the Safety Factor(s) to which each document is applicable, and the type of review that will be completed in PSR2.

All of the Safety Factor 9 L/R/C/S reviews are incremental in nature. The definition of an Incremental Review is as follows:

- Incremental Review: For L/R/C/Ss that have been reviewed in PSR1 but have had revisions since the last review, a topical review will be performed of the changes.

The methodology for the reviews is discussed in Section 3.2. A detailed compliance assessment for each L/R/C/S is provided in Reference [6]. Associated findings are summarized in Section 4.2 of this report.

Table 1: L/R/C/Ss Reviewed for Use of Experience from Other NPPs and Research Findings Safety Factor 9

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
L/R/C/Ss Referenced in Pickering NGS PROL 48.02/2018						
1	CSA N286	Management System Requirements for Nuclear Facilities	N286-12	5, 6, 9, 10, 11	Incremental	N286 addressed as part of Pickering B and Darlington ISRs.

2.3 Audit and Self-Assessment Reviews of OPG Programs

There are no OPG Nuclear Programs assessed for the Use of Experience from Other NPPs and Research Findings in this report. Audit and self-assessment results for N-PROG-RA-0003 [7], "Corrective Action" are provided in P-REP-03680-00012, "Pickering NGS PSR2 Safety Factor 8 Report: Safety Performance" [8]. The methodology for the audit and self-assessment reviews is discussed in Section 3.3.

2.4 Additional Reviews

The PSR2 Safety Factor 9 report includes a review of the R04 Pickering Licence Conditions Handbook (LCH) [4] for any impacts of Pickering NGS operation beyond 2020 on the following (all related to Safety Factor 9):

- OPG commitments previously made to the CNSC;
- Open CNSC action items; and
- Exemptions granted by the CNSC.

The PSR2 assessment includes identification and review of previously identified PSR1 gaps related to Safety Factor 9 to ascertain the implications of extending Pickering NGS operation beyond 2020. The methodology for these reviews is described in Section 3.4. Any PSR2 gaps identified as a result of programmatic Darlington PSR1 gaps related to Safety Factor 9 are discussed in Section 4.4 of this report. The review of Pickering PSR1 gaps previously identified in the Pickering Units 5-8 Continued Operations Plan [9] is provided in Reference [10].

In addition, Fukushima Action Items were reviewed to identify implications of extending operation beyond 2020 (if any). This review is presented in Reference [11].

Any PSR2 gaps identified as a result of the Safety Factor 9 review which need to be addressed in other Safety Factor Reports are discussed in Section 4.4 of this report.

3.0 METHODOLOGY

The sub-sections below summarize the methodology used to assess Review Task and L/R/C/S compliance, and Nuclear Program effectiveness for Use of Experience from Other NPPs and Research Findings Safety Factor.

3.1 Review Tasks

As discussed earlier, the Safety Factor Review Tasks are derived from CNSC REGDOC-2.3.3 [2] and IAEA SSG-25 [3], taking into consideration the Review Tasks used in the Pickering B and Darlington ISRs (as derived in [5]).

For each Safety Factor 9 Review Task identified in Section 2.1, a confirmation of the existence of applicable OPG Governance, Programs, Policies and Procedures (as well as Instructions and Guidelines, as applicable) was performed. Compliance against Review Tasks is also assessed by reference to applicable Condition Assessments, safety analyses and operating experience, as required.

The Review Task assessments identify Compliances and Gaps as defined below:

- Compliance: Compliance indicates that either the safety requirement or the intent of the Review Task is met.
- Gap: A Gap indicates that the intent of the Review Task is not met.

3.2 L/R/C/S Reviews

The process to identify the modern L/R/C/Ss that are applicable to the PSR2 Assessment Basis involved first creating a broad list from multiple sources (potential candidate L/R/C/Ss) and then filtering it to identify those that are most significant and that are applicable to the PSR2 scope. The identification and selection criteria are detailed in Reference [1]. The result of the identification and selection process was a set of modern L/R/C/Ss that became part of the "PSR2 Assessment Basis".

PSR2 is focused on the extension of Pickering NGS operations beyond 2020, and will conduct reviews against a baseline of past PSR1 work. As a subsequent PSR, PSR2 focuses on changes in requirements, plant conditions, operating experience and new information. Since PSR2 is an update of previous ISRs, it incorporates reviews of L/R/C/Ss that have occurred as new versions have been issued. Since this assessment is a subsequent PSR, the focus is on identifying differences between what was previously assessed and what is now different within the current Pickering PSR2 Assessment Basis. In general, these differences relate to:

- More recent (new or revised) L/R/C/S versions than what was previously assessed as part of PSR1;[9]
- Safety significant differences between Pickering and Darlington, if the Darlington ISR is the basis for the earlier assessment;
- Implications of extending Pickering NGS operation beyond 2020; and
- Safety significant differences between Pickering Units 1,4 and Units 5-8.

As described in Reference [1], L/R/C/S review types are clause-by-clause, high level or incremental. Most of the L/R/C/Ss in the PSR2 Assessment Basis receive incremental reviews since PSR2 is an update of previous PSR1 assessments and clause-by-clause or high level reviews for the majority of the L/R/C/Ss in the PSR2 Assessment Basis have already been completed. Implementation plans (including gap analyses or code-over-code reviews) also exist for the latest editions of many L/R/C/Ss. As a result, incremental review is also used in circumstances where a L/R/C/S in the PSR2 Assessment Basis was not assessed in previous PSR1 reviews but an implementation plan currently exists for compliance.

The PSR2 incremental reviews in this Report include an assessment of the intent of recent changes to the L/R/C/Ss on a topic or subject-matter basis where there is potential to impact nuclear safety. Incremental reviews provide:

- A summary of the purpose of the L/R/C/S;
- Pertinent background information about the current revision of the L/R/C/S that is being considered;
- Identification of which Safety Factor(s) are applicable to the current revision of the L/R/C/S;
- A description of which version(s) of the L/R/C/S were assessed for PSR1 (i.e., Darlington ISR (for programmatic content), Pickering B ISR and PARTS code reviews);
- Identification of whether the current version of the L/R/C/S is an update of a previous version of the L/R/C/S that was assessed in PSR1 (and if so, a description of the major changes in the latest revision is provided as discussed below);

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9 "New" refers to a code or standard that was not previously considered in the context of earlier assessments. "Revised" refers to an updated version of a code or standard that was previously considered in the context of earlier assessments. Where a document has a new number/type, but addresses the same topic from the same organization, it is a "revised", not "new", document (e.g., if a REGDOC replaces a CNSC G or RD document).

- An assessment of the applicability of PSR1 assessment findings (gaps and conclusions), including the implications of extending Pickering NGS operation beyond 2020 if any;

- An assessment of the applicability of assessment findings that address more recent (post-PSR1) editions of the L/R/C/S, including any implementation or transition plans that are already committed to by OPG; and

- Where PSR1 and post-PSR1 assessments are not sufficient to address changes in the latest edition of the L/R/C/S, an assessment of the changes from the previously assessed edition of the L/R/C/S (including identification of any safety significant PSR2 gaps which result).

High Level reviews provide the same information as above, where applicable, in a similar format. However, given that High Level L/R/C/Ss generally have not received past assessment during PSR1, the incremental review content is augmented by a high level, section-by-section assessment of the degree of conformance of Pickering NGS with the L/R/C/S (demonstrating, with supporting evidence, whether the intent of the requirements stipulated in the document are met).

There are currently no L/R/C/S clause-by-clause reviews identified in the PSR2 Assessment Basis.

The Safety Factor 9 L/R/C/S reviews identify Compliances and Gaps as defined below:[10]

- <u>Compliance</u>:

  - Where an Incremental review has been performed, Compliance indicates that the change in the safety requirement, per the topical review, is met.

  - Where a High Level review has been performed, Compliance indicates that the intent of the safety requirement is met. (Note: No High Level reviews were performed as part of Safety Factor 9.)

  - For Clause-by-Clause reviews of modern L/R/C/Ss, Compliance indicates that the safety requirement is met. (Note: No Clause-by-Clause reviews were performed as part of Safety Factor 9.)

[10] Safety Factor compliance assessments for Review Tasks and L/R/C/S reviews make use of: a) OPG Governance, Programs, Policies and Procedures which support the compliance arguments, b) Commitments previously made to the CNSC, open CNSC action items, and exemptions granted by the CNSC (all related to the Safety Factor under review), as identified in the R04 Pickering LCH [4], c) Identification of previously identified Pickering-specific or programmatic PSR1 gaps related to the Safety Factor under review and the status of OPG's improvement plan(s) or other dispositions to address these, and d) Assessments and reviews performed since the PSR1 documents were completed.

- Gap:
    - Where an Incremental review has been performed, a Gap indicates that the change in the safety requirement, per the topical review, is not met.
    - Where a High Level review has been performed, a Gap indicates that the intent of the safety requirement is not met. (Note: No High Level reviews were performed as part of Safety Factor 9.)
    - For Clause-by-Clause reviews of modern L/R/C/Ss, a Gap indicates that the safety requirement is not met. (Note: No Clause-by-Clause reviews were performed as part of Safety Factor 9.)

The reviews assume that use of the word:

- "Shall" is used in an L/R/C/S to express a requirement, i.e., a provision that the licensee is obliged to satisfy in order to comply with the standard;
- "Should" is used to express a recommendation or that which is advised but not required;
- "May" is used to express an option or that which is permissible within the limits of the standard; and
- "Can" is used to express possibility or capability.

3.3 Audit and Self-Assessment Reviews

As discussed earlier, there are no OPG Nuclear Programs assessed for the Use of Experience from Other NPPs and Research Findings in this report. Effectiveness reviews (effectiveness at Pickering NGS) of OPG programs used to demonstrate compliance with the PSR2 Assessment Basis were conducted in other PSR2 Safety Factor reports, using recent applicable audit and self-assessment results:

- OPG Nuclear Oversight independent performance-based Program audits (typically performed in 1 to 5 year cycles) and self-assessments. This includes review of associated Station Condition Records and Action Requests to determine the status of any resulting corrective actions; and
- CNSC "Type I" and "Type II" inspections of the effectiveness and performance of OPG programs, where called-up by OPG audits or self-assessments.

There are many audits and self-assessments that are performed to assess the effectiveness of important aspects of each program. A sample of audits and self-assessments has been summarized for each program in order to demonstrate that program effectiveness is being assessed on an ongoing basis. The focus of these reviews was on effectiveness of the programs at Pickering NGS, where specific information is available. Results from these audits and self-assessments will be considered in the Global Assessment process. It is noted that audits and self-assessments are, by their nature, self-critical and are used to drive excellence in performance.

The list of Nuclear Programs to be assessed for each Safety Factor was derived from review of current OPG Governance, and has used the most recent version of these documents as of the PSR2 freeze date of January 15, 2016.

3.4 Additional Reviews

A review of the R04 Pickering LCH [4] was performed to determine if there are any impacts associated with operation of the Pickering Units past 2020 on the following (all related to Safety Factor 9):

- Commitments previously made to the CNSC;
- Open CNSC action items; and
- Exemptions granted by the CNSC.

The PSR2 assessment includes identification and review of previously identified Pickering-specific or programmatic PSR1 gaps related to Safety Factor 9 (as identified in the Darlington ISR Integrated Implementation Plan [12] and Pickering Units 5-8 Continued Operations Plan [9]) to ascertain the status of OPG's improvement plan(s) or other dispositions to address these and the implications of extending operation beyond 2020 (if any).[11]

Fukushima Action Items were reviewed to identify implications of extending operation beyond 2020 (if any). The methodology for this review is provided in Reference [11].

Any PSR2 gaps identified as a result of the Safety Factor 9 review which need to be addressed in other Safety Factor Reports are also discussed.

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[11] PSR2 includes consideration and confirmation that the findings of PSR1 remain valid, as applicable, for the operation period. This includes assessment of PSR1 conclusions against implications resulting from extended operation. In particular, Pickering PSR1 results are applicable to PSR2 if there was a PSR1 gap that is still open, or if a closed PSR1 gap could be affected by extended operation. If so these gaps are carried forward into PSR2 for consideration in the Global Assessment. (When references to PSR1 are made, the source document is identified and the relevant text from that source document is summarized in the context of PSR2.) With respect to the Darlington ISR, much of the evaluation of Safety Factor health is based on programs and practices that apply across OPG's nuclear operations. As a result, Darlington ISR programmatic conclusions are applicable to Pickering PSR2 where Pickering is confirmed to follow the same nuclear programs and practices that were assessed for Darlington. Darlington PSR1 results are applicable to Pickering PSR2 if there are Darlington PSR1 gaps that are found to be relevant to Pickering PSR2.

4.0 REVIEW FINDINGS

4.1 Review Tasks

The sub-sections below provide an assessment of the adequacy of applicable OPG Governance, Programs, Policies and Procedures (as well as Instructions and Guidelines, as applicable) in demonstrating compliance against the Safety Factor 9 Review Tasks.

4.1.1 Review Task #1: Program for Sending and Receiving OPEX

Confirm existence and adequacy of a program for the sending and receiving of experience relevant to safety to and from other nuclear power plants and relevant nonnuclear plants. ("Other nuclear power plants" specifically include the IAEA, OECD/NEA, WANO, INPO as well as CANDU Owners Group and experience within OPG at Darlington.)

Overview

The OPG process for the Use of Experience from Other NPPs and of Research Findings is briefly described below.

A weekly COG OPEX screening meeting, facilitated and administered by COG, serves as an initial screening forum to review event reports from CANDU stations, nuclear industry and non-nuclear sources for applicability and significance to CANDU units. Committee members include representatives from all CANDU facilities (including OPG Pickering), vendors, research organizations and World Association of Nuclear Operators (WANO).

Prior to the weekly meeting, the Senior Officer, OPEX screens the recent events at their site, and selects those events that they believe may be of relevance to other sites for review at the COG OPEX Weekly Screening Meeting. On behalf of the utilities, COG provides the initial screening of the international nuclear industry reports and relevant non-industry events.

OPEX representatives at Pickering, Darlington, and Nuclear Support each submit relevant OPG significant events to be presented at the COG OPEX Weekly Screening Meeting as per N-PROC-RA-0035, "Operating Experience Process" [13]. When potentially significant, these events have been investigated according to N-STD-RA-0008, "Incident Investigation" [14], entered into the OPG Station Condition Record (SCR) process and evaluated according to N-PROC-RA-0022, "Processing Station Condition Records" [15].

Upon completion of the COG OPEX Weekly Screening Meeting, plant OPEX staff in consultation with line staff and/or appropriate OPEX Single Point of Contacts (SPOCs) performs a further screening of the Weekly Screening Meeting events from other utilities/non utilities. Items believed to be applicable and actionable at the plant are dispositioned in an Action Request (AR) and if a significant gap is identified an SCR is created and the gap is evaluated according to N-PROC-RA-0022 [15]. This process includes not only consideration of potential weaknesses of the plant equipment and operation, but also opportunities for improvement and utilization of research findings. Items for which applicability cannot be determined in a timely manner due to Subject Matter Expert or System Responsible Engineer unavailability, awaiting vendor information, or awaiting testing results, are reviewed via the AR process (tracked by AR mechanism until complete), and an SCR is filed if applicability is confirmed.

When an external operating experience item is entered into the electronic SCR system, it is analyzed by the relevant line department as an OPEX SCR, and processed and stored according to the Corrective Action Program specified in N-PROG-RA-0003 [7].

Any finding related to safety analysis that may affect the Safe Operating Envelope (SOE) goes through the Discovery Issue Resolution Process (DIRP) per N-PROC-RA-0094, "Discovery Issue Resolution Process" [16] to confirm that regulatory limits are met and the risk is maintained at an acceptable level, or to put in place mitigating provisions.

The Research and Development (R&D) at OPG is primarily governed by N-STD-MP-0023, "Technology & Research" [17]. This standard specifies the essential elements that are used when processing R&D issues and research findings derived from collaboration with COG R&D members. An R&D Program Advisory Review Team, comprised of senior managers associated with the Nuclear R&D program, provides leadership, support and oversight for the effective development and implementation of Nuclear R&D programs. As specified in N-PROC-MP-0092, "Technology and Research Program Management" [18], at the identification phase of a potential R&D issue, the use of OPEX from other plants and research findings is evaluated to determine if an R&D issue should be developed. R&D issues that progress beyond the identification phase go through a planning and development phase. Upon receipt of the R&D research deliverables, the R&D issue is progressed to the implementation and closeout phase.

The adequacy of the program for the sending and receiving of experience relevant to safety is monitored and assessed through audits and self-assessment prescribed by N-PROG-RA-0010, "Independent Assessment" [19], N-PROC-RA-0048, "Conducting Performance Based Audits and Assessments" [20] and N-PROC-RA-0097, "Self-Assessment and Benchmarking" [21]. Findings are documented in audit reports and self-assessment reports. Any adverse conditions are documented in SCRs per N-PROC-RA-0022 [15] to be resolved through corrective actions per N-PROG-RA-0003 [7].

The hierarchy of OPG governance is defined in N-CHAR-AS-0002, "Nuclear Management System" [22]. The charter identifies requirements for sustaining and improving station performance that include utilization of internal and industry OPEX to improve human, plant and equipment performance and design, procurement, construction, commissioning, and operating requirements and practices. Specifically with regard to OPEX, the charter gives direction to the Corrective Action Program specified in N-PROG-RA-0003 [7]. This program identifies the processes that ensure in-house and industry OPEX are distributed to appropriate personnel, and are applied to implement actions that improve plant safety and reliability.

The Corrective Action Program is implemented through a number of procedures including N-PROC-RA-0022 [15] and N-PROC-RA-0035 [13].

Lessons learned from significant in-house events and equipment problems are shared with the industry in accordance with N-PROC-RA-0035 [13].

The specific details of the process for sending and receiving of experience relevant to safety are specified in N-PROC-RA-0035 [13]. These are described below.

Receiving OPEX

There is a substantial amount of industry experience (including research findings) generated and made available to OPG from a variety of sources. Some of the information contains lessons that, if applied within OPG, may have a significant impact on reducing repeat events and on improving safety, productivity, and performance.

N-PROC-RA-0035 [13] provides details on how and from where the OPEX information from other NPPs and relevant non-nuclear plants is received by OPG. Sources of OPEX external to OPG include but are not limited to:

- WANO Significant Operating Experience Reports (SOERs);
- WANO Significant Event Reports (SERs);
- Level 1 INPO Event Reports (IERs);
- Level 2 IERs;
- COG OPEX;
- WANO Event Reports; and
- Level 3 and 4 IERs.

Pickering also considers applicability of experience within OPG from other sites (e.g. Darlington). This is addressed in Section 1.3.1 of Reference [13] which describes the process and responsibilities for identification of OPEX events from other OPG sites.

A wide range of sources of external OPEX are considered. While there is no specific reference to IAEA or OECD/NEA as listed in the Review Task, such sources are used since sources in N-PROC-RA-0035 [13] are meant to be illustrative rather than exhaustive. Evidence of use of IAEA and OECD/NEA OPEX was found in the COG screened events database. For example, in March 2015 there were two IAEA reports and two NEA reports assessed.

Sending OPEX

Sharing of OPG experience with the industry is required to ensure industry cooperation and is also a requirement of membership in WANO. Lessons learned from significant in-house events and equipment problems are shared with the industry in accordance with N-PROC-RA-0035 [13]. In order to share this information, COG OPEX Weekly Screening Meetings are held, chaired by COG and attended by representatives of COG members and of WANO. OPG shares selected events with other utilities, vendors and research facilities through these COG Weekly Screening Meetings. This forum provides an additional review to ensure that any events that meet the WANO reporting criteria are shared with WANO.

OPEX representatives at Pickering, Darlington, and Nuclear Support each submit relevant OPG internal events to be presented at the COG OPEX Weekly Screening Meeting.

Conclusion

In conclusion, OPG procedures for sending and receiving experience relevant to safety from other nuclear power plants and relevant non-nuclear plants are in place. The intent of Review Task #1 is met and therefore Pickering NGS is compliant.

4.1.2 Review Task #2: Program for Receiving Research Findings

Confirm existence of a program for receiving of information on the findings of relevant research programs.

As specified in N-PROC-RA-0035 [13], OPEX information that includes research information, is dispensed to relevant line organizations for further screening and/or analysis. This process is described in Section 4.1.1 of this report. Whenever such research information is found that is contradictory or supplemental to the knowledge used in the safety or design analysis, an SCR is issued. Opportunities for improvement are also assessed and evaluated either through the SCR or the AR process.

Any finding related to safety analysis that may affect the SOE goes through the DIRP per N-PROC-RA-0094 [16] to confirm that regulatory limits are met and risk is maintained at an acceptable level, or to put in place mitigating provisions when required.

The external information received through the OPEX program is used to identify the need for specific research for OPG NPPs. The R&D at OPG is primarily governed by N-STD-MP-0023 [17]. As specified in N-PROG-MP-0092 [18], the R&D program incorporates issue identification, R&D program development, R&D program implementation, and application of R&D results.

OPG guide, N-GUID-08800-10000, "Technology & Research Department Management Guideline" [23] outlines requirements and processes for the management of the Technology and Research Program. It describes the process of receiving information on the findings of relevant research programs from R&D suppliers, typically through COG or University Network of Excellence in Nuclear Engineering, managing the development and implementation of the R&D program, monitoring of program effectiveness and the dissemination of relevant technical R&D information and results. OPG is a member of the Electric Power Research Institute which is also a source of useful information.

Conclusion

In conclusion, OPG procedures for receiving information on the findings of relevant research programs and for distribution of internal and external research findings are in place. The intent of Review Task #2 is met and therefore Pickering NGS is compliant.

4.1.3 Review Task #3: Assessing and Incorporating Operating Experience

Confirm there is a process for assessing the significance of operating experience from other plants and incorporating the lessons learned into improving safety performance at the station.

Safety Significance Classification Process

In support of N-PROC-RA-0035 [13], the significance of OPEX events is assessed in accordance with:

- N-GUID-04947.02-10000, "External Events Screening Guide" [24]. WANO Guideline GL 2003-01, "Guidelines for Operating Experience at Nuclear Power Plants" [25] provides factors and criteria to be considered in screening. Items assessed to be applicable and where a vulnerability to the plant is identified, are entered into the plant's SCR process and evaluated within a specified time allowance as per N-PROC-RA-0022 [15] and N-PROG-RA-0003 [7] to assess the significance and remedial actions as required. The determination of SCR safety significance level is performed by a SCR Coordinator or by Management Review Meeting (MRM) in accordance with N-PROC-RA-0022 [15].

- N-GUID-04947.02-10001, "Guideline for Classifying Internal Events (WANO, COG, Fleetwide)" [26] provides guidance to the Senior Officer OPEX, Nuclear Support OPEX Coordinator, and Evaluating Organization (EO) Manager, for selecting SCRs to be shared as OPEX with WANO, COG and within OPG Nuclear (designated 'Fleetwide' sharing).

Incorporating Lessons Learned

Items from the COG OPEX Weekly Screening Meeting that are assessed to be applicable and where a potential vulnerability to the plant is identified, are entered into the plant's SCR process and evaluated according to N-PROC-RA-0022 [15] and N-GUID-04947.02-10000 [24]. In particular, the MRM:

- Reviews SCRs with a focus on safety, reportability, and operability resulting in an impact on the facility or business unit;
- Confirms that any immediate or interim station impacts are documented; and
- Establishes a screening meeting to facilitate efficient review of SCRs prior to MRM review, including the review for potentially significant issues of reportability, operability, radiological or conventional safety requiring immediate attention.

If an SCR identifies an adverse condition in which a design basis requirement is potentially not met, the MRM, a Department Manager or Shift Manager will identify an Operability issue. The Technical Operability Evaluation (TOE) process described in N-PROC-MP-0045, "Technical Operability Evaluation" [27] will be applied. A formal TOE provides a substantiated engineering verification that a Structure, System, or Component is capable of fulfilling its minimum credited safety function(s).

Subsequently, the EO ensures that a corrective action plan is prepared with appropriate corrective actions identified and assigned to Action Managers. Specifically, the EO Manager ensures that all Significance Level 1, 2, and 3 SCRs (i.e. the most significant SCRs) have at least one corrective action to prevent reoccurrence. As required, corrective action plans are presented to the Corrective Action Review Board (CARB) for review. Once the corrective action plan is approved, the line organization tracks the completion of corrective actions through Action Tracking. After assignments in Action Tracking have been completed, the EO Manager reviews the adequacy of the completion notes and closes the SCR.

Review of items for applicability may be deferred due to Subject Matter Expert or System Responsible Engineer unavailability, awaiting vendor information, or awaiting testing results. Such deferred items are reviewed via the AR process. This process also has a specified time allowance to ensure a timely review. If an item does not apply to the plant, the personnel making that determination document their reasons in the AR closure notes. If a potential applicable item is determined to create a vulnerability to the plant, then an OPEX SCR is issued [15]. Events that are categorized as "For Information" may also be reviewed further by OPEX SPOCs and communicated to appropriate line personnel for awareness. They are stored in COG's database for future utilization in pre job briefs, tailboards, or training.

In N-PROC-RA-0035 [13], OPG Nuclear staff is directed to review and use OPEX, as required, for the work activities they are responsible for including:

- Pre-job briefings;
- Engineering changes;
- Preparation and delivery of training;
- Root Cause investigations; and
- Assessing.

OPEX information is available for all employees through internal and external OPEX search tools which are conveniently available from the OPEX website. These include:

- COG database (external OPEX);
- WANO database (external OPEX);
- INPO database (external OPEX);
- PowerSearch for SCRs (internal OPEX);
- Darlington/Pickering SERs which are internal legacy documents that pre-date SCRs that were introduced in May 1998. These are different from external WANO / INPO SERs;
- Just in Time (JIT) Briefings (for pre-job briefings); and
- WANO SOERs and SERs.

OPG incorporates external lessons learned into training, per N-INS-08920-10029, "Incorporating Operating Experience into Training" [28].

Design verification activities, as detailed in N-PROC-MP-0047, "Design Verification" [29], require past experience with operation and maintenance of the system or component being modified (i.e., OPEX) to be considered.

Other than the direct use of OPEX issues described above, there are two other mechanisms of incorporating the OPEX from other plants and lessons learned into improving safety performance at the Pickering station:

- JIT briefings contain lessons learned from a number of similar events associated with a specific evolution or type of activity. They were developed for use in pre-job briefings and other applications to assist station staff in performing error free activities. Each JIT briefing contains a list of questions or reminders to consider when planning the work. Hundreds of JIT briefings have been prepared by staff of CANDU stations, COG and WANO, and are listed by title in alphabetical order, and in groups according to the subject content. They can be easily downloaded from the COG online website. The review of OPEX in the pre-job briefings and lessons learned in a post-job debriefing are input into N-FORM-11056, "Engineering Pre-Job Brief and Post-Job Debriefing Checklist" [30], as specified by N-GUID-01900-10000, "Human Performance Event Free Tools for Knowledge Work" [31].

- The Risk Based Modification Process, governed by N-PROG-MP-0001, "Engineering Change Control" [32] incorporates OPEX and lessons learned.

Conclusion

In conclusion, processes for assessing the significance of OPEX from other plants and incorporating the lessons learned into improving safety performance at the station are in place. The intent of Review Task #3 is met and therefore Pickering NGS is compliant.

4.1.4 Review Task #4: Assessing and Incorporating Research Findings

Confirm that there is a process for assessing the significance of research findings and technology developments and for incorporating relevant improvements into the station's design and operation.

Safety Significance Classification Process

Section 4.1.2 of this report describes the process for receiving internal and external research OPEX information and findings.

When operation of a nuclear facility conforms with its defined SOE (as determined through a TOE), but the Safety Analysis upon which the SOE is based is itself suspect, or when a gap is discovered in the definition of the SOE, then the DIRP [16] is initiated to determine whether the regulatory limits are met and if the risk is at an acceptable level, or to put in place mitigating provisions if not. An SCR is raised to further process issues related to either station operation or the DIRP, including determination of safety significance level in accordance with N-PROC-RA-0022 [15].

For internal research findings, N-STD-MP-0023 [17] provides direction for identification and selection of R&D issues or new technology opportunities.

Incorporating Relevant Improvements into the Station's Design and Operation

Research findings are communicated to the Design and Operation organization through the OPEX process via the Corrective Action Program, DIRP and Technology & Research Program.

The Corrective Action Program specified in N-PROG-RA-0003 [7] provides the processes to ensure in-house and industry OPEX is evaluated, distributed to appropriate personnel, and applied to implement actions that improve plant safety and reliability.

Research findings and technology developments with potential applicability to the plant and communicated through OPEX are formally reviewed by line staff per N-PROC-RA-0035 [13]. If an item is determined to have applicability, and is significant to safety then it is entered into the plant's SCR system. Subsequently, an overall corrective action plan is developed within the prescribed time from the date of the MRM where the item was discussed. The corrective action plan includes assignments for appropriate organizations with due dates for assignments specified in accordance with N-PROC-RA-0022 [15]. Improvements to the station's design and/or operations are performed through the Risk Based Modification Process, governed by N-PROG-MP-0001 [32].

The DIRP [16] specifies that upon discovery of a problem or a potential problem with Safety Analysis, not only an SCR is raised, but the Safety Report Update process is initiated and, as required, Senior Management, CNSC, and industry partners are informed and updated. Subsequently it is assessed during the Engineering Judgment Phase whether the Regulatory Dose Limit or Risk Increase Limit might have been exceeded and confirmed through further assessment. Significant DIRP issues usually affect more than one CANDU station, thus COG becomes involved.

Conclusion

In conclusion, processes are in place for assessing the significance of research findings and technology developments and for incorporating relevant improvements into the station's design and operation. The intent of Review Task #4 is met and therefore Pickering NGS is compliant.

4.1.5 Review Task #5: Adequacy, Effectiveness and Timely Implementation

Review adequacy and effectiveness of the feedback arrangements and timely implementation of assessment findings. (Assess program audit results)

Feedback Arrangements

N-PROG-RA-0003 [7] provides the processes to ensure in-house and industry OPEX is evaluated, distributed to appropriate personnel, and applied to implement actions that improve plant safety and reliability. This program gives direction to both N-PROC-RA-0035 [13] and N-PROC-RA-0022 [15] for managing incorporation of improvements into the station's design and operation.

N-PROG-RA-0003 [7] is an essential element for a Learning Organization as it drives the organization to learn from minor incidents to prevent significant events. This improvement process includes the identification of adverse conditions and practices, the investigation of selected incidents and adverse trends to identify their causes, the development and timely implementation of actions to address these causes, and verification that the actions have been effective. The organization's effectiveness at implementing the Corrective Action Program is monitored by the Corrective Action Program Health Indicators, periodic self-assessments and audits.

The lessons learned from OPEX items are fed back to all parts of the organization, including management, engineering, operations and maintenance staff. For example, methods such as JIT reviews (refer to Section 4.1.3) of applicable OPEX during pre-job and pre-evolution activities are used to remind managers, engineers, supervisors, and other workers of lessons learned and how to apply them to prevent similar problems.

Timely Implementation of Findings

N-PROC-RA-0035 [13] provides guidelines for initiation of an SCR for OPEX findings. For OPEX determined to be very significant to OPG Nuclear, an OPEX SCR is required to be generated within 5 working days. N-PROC-RA-0022 [15] provides guidelines for processing an SCR and implementing corrective actions. This procedure provides detailed timelines for initiating an SCR, for approval by the first line manager, for disposition by the SCR coordinator and the MRM, for preparation of OPEX Significant Operating Experience Reports, for approval of the corrective actions and for completion of corrective actions.

N-PROC-RA-0094 [16] provides guidelines for time allocated for issue classification and for required actions. For example, it has to be determined within 7 days whether the Regulatory Dose Limit or Level 1 Risk Increase Limit associated with safety analysis could be exceeded. Immediate corrective action is initiated to return within the Regulatory Dose Limit or Level 1 Risk Increase Limit, up to and including a safe and timely shutdown.

Actions arising from SOERs are periodically assessed to confirm they remain effective. The key goal of conducting periodic reviews of SOERs (and other significant external OPEX) is to detect the inadvertent weakening over time of barriers put in place following the initial OPEX evaluation [33].

Effectiveness

As described in P-CORR-00531-03719, "Application for Renewal of Pickering Nuclear Generating Station Power Reactor Operating Licence" [34]:

- During two station assessments by external industry peers in 2011, use of OPEX by the line organizations in their daily work activities was observed and received as positive feedback.

- In Q4 2011, an internal Nuclear Oversight surveillance audit was conducted on processing of industry experience, processing of internal experience, sharing of OPEX events with COG and other nuclear industry associations, process for integration of OPEX into training and processing of nuclear industry SOERs. The audit results concluded the managed system controls for the OPEX process are effective.

- In 2010, a training needs analysis was performed for Root Cause and Apparent Cause training for EO managers. The result was delivery of EO Manager gap training for existing managers, and EO Manager initial training for new managers in 2011.

- In 2011, a training needs analysis was performed for Apparent Cause Evaluator training. Initial training content was enhanced to include a practical classroom component. Continuing training was implemented and included a computer-based training module with a proof of practice to validate that the evaluator skill level is being maintained.

- In March 2012, Pickering Nuclear implemented corrective actions to increase the effectiveness of the Corrective Action Program. Significant among the changes implemented were: revised criteria to reduce duplication of SCR reporting, revised evaluation requirements and a new root cause report format to improve report quality. SCR Screening Meeting and MRM frequency was increased to daily meetings in order to improve the timeliness of processing SCRs. An industry benchmarked "repeat event" metric was implemented to measure Corrective Action Program effectiveness (i.e., to avoid repeat events which would indicate the OPEX program had not been effective).

- Changes to the trending and analysis process and the new tools resulted from external benchmarking performed by Pickering in 2009. A new trend report template N-INS-01966.1-10000 was implemented in 2010 at Pickering to focus on various aspects of organizational learning feedback received from SCRs and other sources. The reporting template was adopted by the OPG fleet as a standardized practice in 2011.

- Improvements in Corrective Action Program training were identified including:

    - Apparent cause evaluator training was revised to include the new causal analysis techniques to improve quality of analysis and determination of corrective actions;

    - CARB member training was conducted to review industry guidelines on CARB roles and responsibilities as they relate to Corrective Action Program oversight and effectiveness, and communicate upcoming changes to the Corrective Action Program; and

    - MRM member training was conducted to align the MRM members with CARB expectations, roles and responsibilities and communicate upcoming changes to the Corrective Action Program.

As identified during the Darlington ISR [35], there were corrective actions, audits and self-assessments performed to ensure that the process for feedback arrangements and timely implementation of findings is effective and efficient. All associated corrective actions were completed.

Review Task #6 (Section 4.1.6) provides further evidence of the adequacy and effectiveness of the feedback arrangements and timely implementation of assessment findings.

Conclusion

In conclusion, processes are in place for reviewing the adequacy and effectiveness of the feedback arrangements, and for timely implementation of assessment findings. The intent of Review Task #5 is met and therefore Pickering NGS is compliant.

4.1.6 Review Task #6: Application of OPEX

List the major OPEX events and resulting plant changes that have resulted since PSR1 was completed.

External OPEX

SOERs represent the most significant external OPEX that is reviewed by OPG as described in Section 4.1.1. SOERs are key industry events and lessons learned of continuing importance. An SOER Module of the Observation and Coaching Self Assessment database is used to document SOERs and how they are addressed.

OPG Procedure N-PROC-RA-0035 [13] requires that corrective actions resulting from SOERs or Level 1 IERs be reviewed every 2 years. These biennial effectiveness reviews ensure that all corrective actions implemented in response to the original OPEX event are still adequate, taking into account any changes to processes or design since the corrective actions were initially implemented [37].

NK38-REP-03680-10104, "Darlington Integrated Safety Review – Final ISR Report" [36] contained a review of the corrective actions taken in response to external safety significant OPEX. The review included safety significant operating experience reports from June 1st, 2001 to May 30th, 2011 [37]. OPG subsequently assessed lessons from major external events including SOERs following the completion of the Darlington ISR in NK38-REP-03680-10207 R000, "Darlington NGS Integrated Safety Review Emerging Issues Report" [37]. That assessment addressed major external events up to the end of 2013 and is applicable to Pickering. An illustrative sample of these event reports is summarized in Appendix C, Table C1. The full list is available in Reference [37]. Appendix C summarizes the OPEX source, and how lessons were applied. WANO SOERs after August 1, 2013 are considered in Appendix C, Table C2. As demonstrated in Appendix C, the changes implemented as a result of the OPEX program provide confidence in the effective implementation of the elements of the OPEX program as summarized in Review Tasks 1 through 5.

The most significant of the external events in the review period was the Fukushima Daiichi accident on March 11, 2011. OPG conducted a rigorous review of the preparedness of its stations to deal with Beyond Design Basis Accidents (BDBA) in N-REP-03500-0401509, "Implications of the Fukushima Daiichi Event on OPG Nuclear Power Plants" [38]. For each of OPG's stations (Darlington, Pickering 1,4 and Pickering 5-8), the areas of review included:

- Design Basis Events;
- External Hazards and Events;
- BDBAs;
- Severe Accident Management; and
- Emergency Preparedness and Response.

The following conclusions and changes have been made:

- The current design and operation of OPG stations are robust and existing design provisions are, in general, sufficient to mitigate beyond design basis external hazards.
- Emergency Mitigating Equipment (EME) has been provided for events resulting in total loss of Alternating Current (AC) power. This EME (i.e., portable pumping capability with mobile diesel powered pumps), is on site and available to supply water to critical loads. This equipment can be fuelled from on-site tanker trucks with sufficient capacities to allow for sustained operation of required equipment. EME also includes portable equipment to provide temporary power supplies for alternate control and monitoring of critical equipment to successfully manage a BDBA and preclude fuel and/or core damage.
- At least one connection point has been provided to supply make up water to steam generators, Heat Transport System, Moderator System and Irradiated Fuel Bays (IFB), and a power supply provides continuous monitoring of critical parameters as needed to maintain the heat sinks, as discussed in N-CORR-00531-06822 R000, "Pickering NGS and Darlington NGS – OPG Specific Action Items Related to Closed Fukushima Action Items – Progress Update No. 4" [39].
- Operational guidelines supporting use of the EME equipment have been developed [34]. Validation has been performed of the effectiveness of EME deployment which includes human factors assessments through site drills and exercises [39].
- BDBA considerations (e.g., EME deployment and severe accident management) have been included in the N-PROG-RA-0001, "Consolidated Nuclear Emergency Plan" [39].

- Improvements to Severe Accident Management Guidelines have been made to improve response to BDBAs including Severe Accidents. Severe Accident Management Guidelines, derived from industry standards, have been issued and authorized staff has been trained in their use.

- Emergency drills and exercises have been expanded to include BDBAs.

- Installation of Passive Autocatalytic Recombiners which do not rely on electrical power, addresses mitigation of the potential for hydrogen hazards inside Containment, as discussed in N-CORR-00531-05574 R000, "Ontario Power Generation Comments on CNSC Staff Action Plan on CNSC Fukushima Task Force Recommendations" [40].

- Improved instructions have been provided to the operations crews for loss of cooling or normal water addition capability in the IFBs [40].

- Enhancements to the water makeup/cooling capability for the IFBs have been implemented [34].

- Additional studies, improvements and actions are being tracked to closure [39].

A review of Fukushima Action Items was performed for PSR2 in Reference [11] to identify implications of extending Pickering NGS operation beyond 2020.

Internal OPEX

Internal OPEX is used to improve performance as demonstrated by the following examples:

- A significant success within the Pump Program is seen through the refinement of the Shutdown Cooling (SDC) pump seal maintenance strategy. This strategy includes the incorporation of station OPEX into operating procedures for SDC pump operation and further improving maintenance practices. Through these activities a reduced forced loss rate due to SDC pump seal failures has been achieved [34].

- Improvements in radiation exposure performance were realized through the implementation of increased line accountability for the control of radiation exposure, including Departmental Dose Reduction Plans. Several initiatives developed during the process, including reducing radiation exposure to Operations staff during resin slurries, improvements to management of scaffolding tasks based on industry OPEX, and improvements in the quality of work performed by Fuel Handling, have resulted in performance better than target [34].

- Innovative uses of shielding, including custom designed shielding (e.g., in-boiler head applications and improved reactor face applications) and enhancements to support structures to decrease installation time (e.g., for boiler drains) have been implemented by the Radiation Protection Department, and OPEX/applications developed at other sites [34].

Conclusion

In conclusion, processes are in place for reviewing major OPEX events and for implementing the resulting plant changes. The intent of Review Task #6 is met and therefore Pickering NGS is compliant.

4.2 L/R/C/S Reviews

As per Section 2.2 of this report, a detailed compliance assessment for one L/R/C/S with content applicable to Safety Factor 9 is provided in Reference [6]. Associated findings applicable to Safety Factor 9 are summarized in Table 2 below.

Table 2: PSR2 L/R/C/S Compliance Assessment Results for Safety Factor 9

L/C/R/S Reviewed	PSR2 Compliance Assessment for Safety Factor 9
CSA N286-12, "Management System Requirements for Nuclear Facilities"	There are no PSR2 gaps for CSA N286-12. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with N286-12.

4.3 Audit and Self-Assessment Reviews

There were no OPG Nuclear Programs reviewed for the Use of Experience from Other NPPs and Research Findings in this report. Audit and self-assessment results for N-PROG-RA-0003 [7], "Corrective Action" are provided in "Pickering PSR2 Safety Factor 8 Report: Safety Performance" [8].

4.4 Additional Review Findings

As discussed in Section 3.4, the PSR2 Safety Factor 9 assessment also included a review of commitments previously made to the CNSC, open CNSC action items, and exemptions granted by the CNSC, as identified in the R04 Pickering LCH [4], to determine if there are any impacts associated with operation of the Pickering Units past 2020. The review also included identification and review of previously identified programmatic Darlington PSR1 gaps related to Safety Factor 9 to determine impacts associated with operation of the Pickering Units past 2020. This assessment did not identify any gaps for Safety Factor 9.

Findings from the review of previously identified PSR1 gaps in the Pickering Units 5-8 Continued Operations Plan [9] are provided in Reference [10]. Findings from the review of Fukushima Action Items are provided in Reference [11]. Results from the

Continued Operations Plan and Fukushima Action Items reviews will be considered in the Global Assessment process.

There were no PSR2 gaps identified in this Safety Factor 9 report that require discussion in other Safety Factor reports.

5.0 RESULTS AND CONCLUSIONS

OPG Governance, Programs, Policies, Procedures, Instructions and Guidelines related to Safety Factor 9 were reviewed for the six PSR2 Review Tasks in Section 4.1 of this report and resulted in compliance with all aspects of the Review Tasks. L/R/C/S reviews for Safety Factor 9 were prepared per Section 4.2 and resulted in no PSR2 gaps. There were no OPG Nuclear Program audit and self-assessment reviews prepared for Safety Factor 9 per Section 4.3. Per Section 4.4, this report also included identification and review of previously identified programmatic Darlington PSR1 gaps related to Safety Factor 9 (to ascertain the implications of extending Pickering NGS operation beyond 2020), as well as a review of the R04 Pickering LCH [4] for any impacts of Pickering NGS operation beyond 2020 on: a) OPG commitments previously made to the CNSC, b) open CNSC action items, and c) exemptions granted by the CNSC (all related to Safety Factor 9), which resulted in no PSR2 gaps.

The review of Safety Factor 9 has confirmed for Pickering NGS that there is adequate feedback of relevant experience from other nuclear power plants and from findings of research, and that this is used to introduce reasonable and practicable safety improvements at the plant or in the operating organization.

6.0 REFERENCES

[1] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[2] CNSC REGDOC-2.3.3, *Periodic Safety Reviews*, April 2015.

[3] IAEA Safety Guide No. SSG-25, *Periodic Safety Review for Nuclear Power Plants*, 2013.

[4] CNSC Report, LCH-PNGS-R004, *Pickering NGS: Licence Conditions Handbook*, December 23, 2015.

[5] OPG Report, P-REP-03680-00003 R000, *Pickering NGS PSR2: Definition of Safety Factor Review Tasks*, May 30, 2016.

[6] OPG Report, P-REP-03680-0586480 R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 9, 11, and 15*, to be issued.

[7] OPG Program, N-PROG-RA-0003 R010, *Corrective Action*, January 9, 2015.

[8] OPG Report, P-REP-03680-00012 R000, *Pickering NGS PSR2 Safety Factor 8 Report: Safety Performance*, to be issued.

[9] OPG Plan, NK30-PLAN-00531-00001 R005, *Pickering 5-8 Continued Operations Plan*, December 2015.

[10] OPG Report, P-REP-03680-00024 R000, *Pickering 5-8 Continued Operations Plan Review in Support of PNGS Periodic Safety Review 2 (PSR2),* to be issued.

[11] OPG Report, P-REP-03680-00022 R000, *Fukushima Action Item Review in Support of PNGS Periodic Safety Review 2 (PSR2)*, to be issued.

[12] OPG Report, NK38-REP-03680-10185 R002, *Darlington NGS Integrated Implementation Plan (IIP)*, April 30, 2015.

[13] OPG Procedure, N-PROC-RA-0035 R018, *Operating Experience Process*, October 10, 2014.

[14] OPG Standard, N-STD-RA-0008 R013, *Incident Investigation*, November 28, 2014.

[15] OPG Procedure, N-PROC-RA-0022 R032, *Processing Station Condition Records*, November 13, 2014.

[16] OPG Procedure, N-PROC-RA-0094 R006, *Discovery Issue Resolution Process*, June 17, 2015.

[17] OPG Standard, N-STD-MP-0023 R000, *Technology and Research*, May 29, 2012.

[18] OPG Procedure, N-PROC-MP-0092 R002, *Technology and Research Program Management*, March 11, 2014.

[19] OPG Program, N-PROG-RA-0010 R013, *Independent Assessment*, April 30, 2014.

[20] OPG Procedure, N-PROC-RA-0048 R017A, *Conducting Performance Based Audits and Assessments*, May 28, 2015.

[21] OPG Procedure, N-PROC-RA-0097 R008, *Self-Assessment and Benchmarking*, December 17, 2014.

[22] OPG Charter, N-CHAR-AS-0002 R018, *Nuclear Management System*, March 6, 2015.

[23] OPG Guideline, N-GUID-08800-10000 R001, *Technology & Research Department Management Guideline*, January 12, 2010.

[24] OPG Guideline, N-GUID-04947.02-10000 R001, *External Events Screening Guide*, December 1, 2012.

[25] WANO Guideline, GL 2003-01 Rev 1, *Guidelines for Operating Experience at Nuclear Power Plants*, June 2015.

[26] OPG Guideline, N-GUID-04947.02-10001 R003, *Guidelines for Classifying Internal Events (WANO, COG & Fleetwide)*, July 10, 2015.

[27] OPG Procedure, N-PROC-MP-0045 R008, *Technical Operability Evaluation*, September 14, 2015.

[28] OPG Instruction, N-INS-08920-10029 R002, *Incorporating Operating Experience Into Training*, November 22, 2012.

[29] OPG Procedure, N-PROC-MP-0047 R006, *Design Verification*, April 30, 2015.

[30] OPG Form, N-FORM-11056 R003, *Engineering Pre-Job Brief and Post-Job Debriefing Checklist*, March 30, 2012.

[31] OPG Guideline, N-GUID-01900-10000 R004, *Human Performance Event Free Tools for Knowledge Work*, November 5, 2015.

[32] OPG Program, N-PROG-MP-0001 R014, *Engineering Change Control*, January 7, 2015.

[33] OPG Guideline, N-GUID-01533-10000 R001, *SOER Review And Response Process*, May 29, 2015.

[34] OPG Letter, G. Jager to M.A. Leblanc, P-CORR-00531-03719 R000, *Application for Renewal of Pickering Nuclear Generating Station Power Reactor Operating Licence*, July 4, 2012.

[35] OPG Report, NK38-REP-03680-10075 R001, *Darlington NGS Integrated Safety Review – Use of Experience from Other Plants and Research Findings Safety Factor Report*, May 30, 2011.

[36] OPG Report, NK38-REP-03680-10104-R000, *Darlington NGS Integrated Safety Review (ISR) – Final ISR Report*, October 2011.

[37] OPG Report, NK38-REP-03680-10207 R000, *Darlington NGS Integrated Safety Review Emerging Issues Report*, February 2014.

[38] OPG Report, N-REP-03500-0401509 R000, *Implications of the Fukushima Daiichi Event on OPG Nuclear Power Plants*, July 2012.

[39] OPG Letter, W. S. Woods to M. Santini and F. Rinfret, N-CORR-00531-06822 R000, *Pickering NGS and Darlington NGS – OPG Specific Action Items Related to Closed Fukushima Action Items – Progress Update No. 4*, June 15, 2015.

[40] OPG Letter, W. M. Elliott to R. Jammal, N-CORR-00531-05574 R000, *Ontario Power Generation Comments on CNSC Staff Action Plan on CNSC Fukushima Task Force Recommendations*, February 3, 2012.

[41] OPG Report, P14-000618-SOER, *SOER Cold Body Review Completed (SOER 2013-1 (INPO IER 11-3): Operator Fundamentals Weaknesses)*, December 2014.

[42] OPG Report, P15-001056-SOER, *SOER Cold Body Review Completed: SOER 2015-1 Rev1 – Safety Challenges from Open Phase Events*, July 2015.

Appendix A: Nomenclature

AC	Alternating Current
AR	Action Request
BDBA	Beyond Design Basis Accident
CANDU	CANada Deuterium Uranium
CARB	Corrective Action Review Board
CNSC	Canadian Nuclear Safety Commission
COG	CANDU Owners Group
CSA	Canadian Standards Association
DIRP	Discovery Issue Resolution Process
EME	Emergency Mitigating Equipment
EO	Evaluating Organization
EOER	Evaluating Organization Effectiveness Review
IAEA	International Atomic Energy Agency
IER	INPO Event Report
IFB	Irradiated Fuel Bay
INPO	Institute of Nuclear Power Operations
ISR	Integrated Safety Review
JIT	Just-In-Time (briefing)
LCH	Licence Conditions Handbook
L/R/C/S	Laws, Regulations, Codes and Standards
MRM	Management Review Meeting
NGS	Nuclear Generating Station
NPP	Nuclear Power Plant
OECD/NEA	Organization for Economic Cooperation and Development, Nuclear Energy Agency
OPEX	Operating Experience
OPG	Ontario Power Generation
OPGN	Ontario Power Generation Nuclear
PARTS	Pickering A Return to Service
PROL	Power Reactor Operating Licence
PSR	Periodic Safety Review

PSR1	Periodic Safety Review 1 (earlier OPG PSR work and other associated assessments)
PSR2	Periodic Safety Review 2 (subsequent PSR per CNSC REGDOC-2.3.3)
R&D	Research and Development
SCR	Station Condition Record
SDC	Shutdown Cooling
SER	Significant Event Reports
SFP	Spent Fuel Pool
SOE	Safe Operating Envelope
SOER	Significant Operating Experience Reports
SPOC	Single Point of Contact
TOE	Technical Operability Evaluation
WANO	World Association of Nuclear Operators

Appendix B: Audit and Self-Assessment Results

There were no OPG Nuclear Programs reviewed for the Use of Experience from Other NPPs and Research Findings in this report. Audit and self-assessment results for N-PROG-RA-0003 [7], "Corrective Action" are provided in the "Pickering NGS PSR2 Safety Factor 8 Report: Safety Performance" [8].

Appendix C: OPEX Events

The following table lists a sample of significant OPEX events to demonstrate that plant changes have resulted since the last ISR was completed. The *Darlington NGS Integrated Safety Review Emerging Issues Report* [37] identified 24 significant external events that occurred after the Darlington ISR (May 31, 2011) and up to December 31, 2013. A full listing of outcomes is provided in Appendix A of that report and a sample of that appendix is summarized in the following table. These results are applicable to Pickering as well as to Darlington.

Table C1: Sample of significant external operating experience events from 2011 to 2013 identified in the *Darlington NGS Integrated Safety Review Emerging Issues Report* [37]

Source	Report Title	SCR Title	SCR Status	Outcome *(Summarized from Darlington ISR Emerging Issues Report)* [37]
WANO SOER 2011-3[12]	**Fukushima Daiichi Nuclear Station Spent Fuel Pool/Pond Loss of Cooling and Makeup:** Following a magnitude 9.0 earthquake and subsequent tsunami, on 11 March 2011, power supplies and equipment necessary to provide cooling, monitoring, and makeup to the Spent Fuel Pools (SFPs) became unavailable or inoperable for an extended period. This could have resulted in damage to the fuel stored	OPEX - WANO SOER 2011-3: Fukushima Daiichi Nuclear Station Spent Fuel Pool/Pond Loss of Cooling and Makeup	Closed	*AR 28134711 was created with specific assignments for the Pickering and Darlington stations to address the recommendations in this WANO SOER. A dedicated team has been assembled at OPG to implement actions related to Fukushima events and the EO Manager responsibility for SCR N-2011-04547 was transferred to the Director – Fukushima Project in early November 2011. An Evaluating Organization Effectiveness Review (EOER) is to be completed six months after the completion of the Corrective Action Plan. As per AR# 28134711, a nuclear business level EOER for this SOER has been completed and documented in the Self Assessment database under NO12-000376.* *AR #28145651 has been created to conduct a biennial effectiveness review for this SOER.*

[12] Subsequent to the Darlington ISR Emerging Issues Report, WANO SOER 2011-3 has been superseded by WANO SOER 2011-3 Rev 1 (discussed in Table C2).

Source	Report Title	SCR Title	SCR Status	Outcome *(Summarized from Darlington ISR Emerging Issues Report)* [37]
	in the Fukushima Daiichi nuclear power station SFPs. At the time of the event, Units 1, 2, and 3 were in operation and Units 4, 5, and 6 were shut down for refueling. The Unit 4 SFP contained both spent fuel and the full offloaded core from the most recent operating cycle. This SOER documents immediate actions required as a result of Fukushima Daiichi SFP problems.			*In April 2012, WANO issued additional documents based on the analysis of responses received from the member companies. OPG created SCR N-2012-02658 to assess the impact to OPG and document lessons learned. The industry responses were compared to OPG's response to SOER 2011-3. No gaps were found and no additional actions were required. N-2012-02658 has been completed.* *This WANO SOER was updated with SOER 2011-3 Rev 1 as noted below.*
WANO SOER 2011-4[13]	**Near-Term Actions to Address an Extended Loss of All AC Power:** This WANO SOER is issued as a result of the Fukushima event. Extended loss of all AC power resulted in emergency core cooling systems being unable to prevent fuel damage at	OPEX – WANO SOER 2011-4: Near-Term Actions to Address an Extended Loss of All AC Power-Fukushima Daiichi OPEX – INPO IER[14] L1 11-4:	Closed Closed	*This OPEX contains five recommendations for the member utilities to implement. The recommendations of this SOER call for the development of preplanned contingencies for protection from extended loss of AC power and beyond station blackout events similar to those experienced at Fukushima Daiichi.* *A dedicated team has been assembled at OPG to implement actions related to Fukushima events as communicated to the CNSC via N-CORR-00531-05574, "Ontario Power Generation Comments on CNSC Staff Action Plan on CNSC Fukushima Task*

[13] A review of Fukushima Action Items for applicability to PSR2 is provided in Reference [11]. Subsequent to the Darlington ISR Emerging Issues Report, WANO SOER 2011-4 has been superseded by WANO SOER 2013-2 Rev 1 (discussed in Table C2).

[14] INPO Event Report

Source	Report Title	SCR Title	SCR Status	Outcome *(Summarized from Darlington ISR Emerging Issues Report)* [37]
	three of the six units. This WANO SOER establishes the near term actions to improve margins for beyond design basis loss of AC power events pending longer-term industry actions to address Fukushima Daiichi lessons learned. Recommendations for this SOER assume the reactor is at full power as the initial condition. The recommendations of this SOER call for the development of preplanned contingencies for protection from extended loss of AC power and beyond station blackout events similar to those experienced at Fukushima Daiichi.	Near-Term Actions to Address the Effects of an Extended Loss of All AC Power in Response to the Fukushima Daiichi Event		*Force Recommendations". OPG Report N-REP-03500-0401509, "Implications of the Fukushima Daiichi Event on OPG Stations: A Summary Report" discusses OPG's response and recommendations contained in various Fukushima related WANO SOERs, SERs, and INPO IERs in detail. Section 3.0 of the report deals specifically with the review of the DNGS against Fukushima events and discusses completed or in-progress actions to address recommendations.*[15] *AR 28140560 was raised to track the actions resulting from the recommendations specific to this SOER. All actions including the Final EOER have been completed. The final fleet level assignment under AR# 28140560 provided Enhanced Communications for Emergency Response events.* *This WANO SOER was issued as a result of the INPO Level 1 IER 11-4. Please refer to the IER 11-4 discussion below.*
WANO SOER 2013-1	**Operator Fundamental Weaknesses** Several significant events have occurred that highlight weaknesses in the knowledge, skills,	OPEX – WANO SOER 2013-1: Operator Fundamentals Weaknesses	Closed	*This SOER 13-1 contains recommendations identified in INPO IER 11-3. A line by line comparison of SOER 13-1 and INPO IER 11-3 was completed as per AR and identified some activities that were not addressed in previous self-assessments as well as additional review activities under "Special Considerations for*

[15] The equivalent review of Pickering 1,4 and 5-8 are in Sections 4.0 and 5.0 respectively.

Source	Report Title	SCR Title	SCR Status	Outcome *(Summarized from Darlington ISR Emerging Issues Report)* [37]
	behaviours, and practices essential for operators to operate the plant safely and effectively – operator fundamentals. In some cases, individuals caused events during operations activities. In other instances, individuals did not mitigate the effects of power transients. Events include reactor trip, loss of reactor coolant system inventory, unplanned reactivity additions and damage to plant equipment. This SOER establishes actions to help members to self-assess the effectiveness of operator fundamentals and training programmes at their stations. This SOER also establishes actions to ensure operator fundamentals are well			*Review". Further corrective actions have been initiated as required.* *Self-Assessment, D14-000127*[16] *was performed to provide a snapshot status review of the station initiatives against SOER 2013-1 recommendations, identify gaps and/or weaknesses, and develop action plans to meet the objectives of the SOER recommendations.* *Reviews performed as part of the self-assessment identified that action plans are in place or have been implemented to address the SOER recommendations. Additionally, action plans have been developed to address areas requiring improvement.* *Corrective actions have been raised to ensure the successful implementation of Operator Fundamentals. These actions include the creation of an action plan to improve operator's risk awareness, performing a follow up self-assessment on training effectiveness in addressing operator fundamentals, and conducting an EOER.*

[16] Self-assessment PNGS P14-000618-SOER [41] was performed at Pickering in December 2014. It was a cold body review of SOER 2013-1 (INPO IER 11-3). The assessment is tracked via AR# 28156277-01 for the 2014 Pickering (014 and 058) review of this SOER. Improvement actions were identified. Fleet actions are being tracked via AR# 28172928. Pickering actions are being tracked via AR# 28167726.

Source	Report Title	SCR Title	SCR Status	Outcome *(Summarized from Darlington ISR Emerging Issues Report)* [37]
	ingrained in and rigorously applied by operators.			
WANO SOER 2013-2[17]	**Post-Fukushima Daiichi Nuclear Accident Lessons Learned** This SOER documents actions required as a result of problems experienced at Fukushima Daiichi, following the 11 March 2011 earthquake and tsunami. The purpose of this SOER is to discuss and provide recommendations for significant lessons from the event. These lessons were drawn from a review of the March 2011 Fukushima Daiichi event and a similar event at the Fukushima Daini site. In addition, it provides a central location for all previously issued Fukushima-related SOERs and recommendations.	OPEX – WANO SOER 2013-2: Post-Fukushima Daiichi Nuclear Accident Lessons Learned;	Closed	*This OPEX contained 11 major recommendations, which address the leadership, organisational, cultural, resource and training issues that contributed to the event and that detracted from an effective emergency response.* *These recommendations had been reviewed through separate related Fukushima Daiichi OPEX. OPG performed a review of all 11 major recommendations and where existing measures or planned actions were determined to be insufficient to address specific recommendations, further actions have been identified.* *This WANO SOER was updated with SOER 2013-2 Rev 1 as noted below.*

[17] A review of Fukushima Action Items for applicability to PSR2 is provided in Reference [11]. Subsequent to the Darlington ISR Emerging Issues Report, WANO SOER 2013-2 has been superseded by WANO SOER 2013-2 Rev 1 (discussed in Table C2).

Source	Report Title	SCR Title	SCR Status	Outcome *(Summarized from Darlington ISR Emerging Issues Report)* [37]
	This SOER and the associated recommendations address the leadership, organisational, cultural, resource and training issues that contributed to the event and that detracted from an effective emergency response.			
INPO IER L1-13-10	**Nuclear Accident at the Fukushima Daiichi**	OPEX - INPO IER L1-13-10: Nuclear Accident at the Fukushima Daiichi Nuclear Power Station SCR# N-2013-01618	Closed	*N-2013-01618, issued in response to INPO IER L1-13-10, has been closed to SCR N-2013-01709, which was issued in response to WANO SOER 2013-2. See WANO SOER for discussion of the assessment against this INPO IER.*

Table C2 lists WANO SOERs since August 1, 2013 and the status of actions to implement plant changes based on OPEX lessons.

Table C2: WANO SOER events from August 2013 to January 2016

Source	Report Title/Scope	SCR Title	SCR Status	Outcome
WANO SOER 2011-3 Rev 1	**Spent Fuel Facility Degradation, Loss of Cooling or Makeup**	OPEX - WANO SOER 2011-3 UPDATE (R1): Spent Fuel Facility Degradation, Loss of Cooling or Makeup (Fukushima Daiichi) SCR# N-2013-22082	Closed	Updates were made to two recommendations from the earlier version of the SOER listed in Table C1 and one new recommendation was made. This includes a new requirement for monitoring of irradiated fuel bay level and temperature and radiation levels from an accessible location and requirements for assessment and monitoring of Dry Fuel Storage areas. The Corrective Action Plan was developed and implemented. Pickering has successfully completed a review against SOER 2011-3 Rev 1 during its 2015 WANO Review.
WANO SOER 2013-2 Rev 1	**Post-Fukushima Daiichi Nuclear Accident Lessons Learned** This SOER builds on earlier Fukushima SOERs and experience based on implementation and evaluation of the Fukushima event. The purpose of this SOER is two-fold. First, it discusses and provides recommendations for significant lessons from the	OPEX - WANO SOER 2013-2: Post-Fukushima Daiichi Nuclear Accident Lessons Learned SCR# N-2013-01709	Closed	The Corrective Action Plan developed for this SCR reflects the actions and activities undertaken as part of OPG's response to the Fukushima SOERs and other Fukushima-related OPEX. This includes the following: • WANO SOER 2011-02 (SCR N-2011-01646) • WANO SOER 2011-02 Addendum (SCR N-2012-02403) • WANO SOER 2011-03 (SCR N-2011-04547) • WANO SOER 2011-04 (SCR N-2011-06587) • INPO 11-005 Addendum (SA N012-000545) Because of the large amount of work already undertaken by OPG at its facilities since the

Source	Report Title/Scope	SCR Title	SCR Status	Outcome
	event. These lessons were drawn from a review of the March 2011 Fukushima Daiichi event and a similar event at the Fukushima Daini site. Second, it provides a central location for all previously issued Fukushima-related SOERs and recommendations.			Fukushima event, and in particular in response to the OPEX items above, many of the recommendations were already at least partly satisfied. Each of the 11 recommendations has been reviewed. Where existing measures, or planned actions were determined to be insufficient to address a specific recommendation, further actions were identified and these were the basis of the Corrective Action Plan for this SCR, which has been completed and implemented. Pickering has successfully completed a review against SOER 2013-2 Rev 1 during its 2015 WANO Review.
WANO SOER 2015-1 Rev 1 [42]	Safety Challenges from Open Phase Events[18] Undetected conditions during open phase events have resulted in the inoperability of important components, losses of shutdown or maintenance cooling, and the potential to affect all trains of safety systems from a common mode failure.	WANO SOER 2015-1 Rev 1: Safety Challenges from Open Phase Events SCR# N-2015-03998	Approved	Pickering is now compliant with all of the recommendations in this SOER.

[18] An open phase event is defined where power in one (or possibly two) of the three phases of an offsite power source feed is lost for a long duration.

Source	Report Title/Scope	SCR Title	SCR Status	Outcome
WANO SOER 2015-2	Risk Management Challenges Some events demonstrated that individuals exhibited at-risk behaviours based on overconfidence from past successful performances, perceived time pressures or complacency. Like individuals, the decisions made by committees, forums or groups of supervisors and managers can be adversely influenced by business pressures, where plant and corporate leaders may inadvertently display a non-conservative risk tolerance. Several events occurred as a result of weaknesses in risk management during the project or modification process, especially for large first-of-a-kind or first-in-a-while projects.	OPEX - WANO SOER 2015-2: Risk Management Challenges SCR# N-2015-28710	Approved	A comprehensive evaluation on risk awareness was performed under N-2016-04704 resulting in the following actions: 1. AR# 28188287-01 - Create OPGN Risk Management Governance 2. AR# 28188287-04 - Select Standardized Risk Management Tool for OPGN 3. AR# 28188287-05 - Implement a systematic approach to delivering risk management training to the Nuclear Division 4. AR# 28188287-08 - Risk Mitigation Plan Workshop for OPG Leadership All of these ARs are in progress with due dates up to February 28, 2017.

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ONTARIO POWER GENERATION	
ACCEPTED	✓
ACCEPTED AS NOTED	
REVISE AND RESUBMIT	
Signature: [signature]	Date: 13 Dec 2016
Name: Mike Ruffolo, Manager	
Dept: Pickering Engineering - Aging Management & Strategic Initiatives	
OPG Proprietary	
Doc No.: P-REP-03680-00014	Rev: 000
This acceptance does not relieve the contractor from responsibility for errors or omissions or from any obligations or liability under this contract.	

Pickering NGS PSR2 Safety Factor 10 Report: Organization, Management System, and Safety Culture

PS112/RP/015 R01

December 12, 2016

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Revision Summary – For Amec Foster Wheeler Report PS112/RP/015

Rev	Date	Author	Comments
R00	July 29, 2016	R. Jayasundera, M. Hoang and L. Macdonald	Initial issue for OPG review and comment.
R01	December 12, 2016	R. Jayasundera, M. Hoang, L. Macdonald and J. Cheng	Updated report addressing OPG comments on R00 Report.

EXECUTIVE SUMMARY

Ontario Power Generation (OPG) is performing a Periodic Safety Review (PSR) to support the possibility of continued operation of Pickering Nuclear Generating Station (NGS) beyond 2020. The PSR (referred to as "PSR2") is a subsequent PSR building on the review basis of earlier OPG Integrated Safety Reviews and other associated assessments. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

Part of PSR2 involves the preparation of Safety Factor reports for each of fifteen major topic areas. Safety Factor reports consist of:

- Assessing compliance against "Review Tasks" identified in the PSR2 Basis Document [1]. These Review Tasks are derived from Canadian Nuclear Safety Commission (CNSC) REGDOC-2.3.3, "Periodic Safety Reviews" [2] and International Atomic Energy Agency (IAEA) SSG-25, "Periodic Safety Review for Nuclear Power Plants" [3];

- Documenting assessments against applicable modern Laws, Regulations, Codes and Standards (L/R/C/Ss) as defined in Reference [1]; and

- Reviewing the effectiveness of applicable OPG Nuclear Programs that support the above assessments, through review of audit and self-assessment results.

The PSR2 review of Safety Factor 10, *Organization, Management System, and Safety Culture* is presented in this report. OPG Governance, Programs, Policies, Procedures, Instructions and Guidelines related to Safety Factor 10 were reviewed for the fifteen PSR2 Review Tasks specified in Section 4.1 of this report. L/R/C/S and OPG Nuclear Program effectiveness reviews for Safety Factor 10 were prepared per Sections 4.2 and 4.3, respectively. Per Section 4.4, the PSR2 assessment includes a review of previously identified PSR1 gaps related to Safety Factor 10 (to ascertain the implications of extending Pickering NGS operation beyond 2020), as well as a review of the R04 Pickering Licence Conditions Handbook [4] for any impacts of Pickering NGS operation beyond 2020 on: a) OPG commitments previously made to the CNSC, b) open CNSC action items, and c) exemptions granted by the CNSC (all related to Safety Factor 10).

The results of the review of Safety Factor 10 are discussed in Section 5.0. The review has confirmed that the Pickering NGS organization, management system and safety culture are adequate and effective for ensuring the safe operation of the plant. As discussed in Section 5.0, the review identified no Pickering PSR2 gaps.

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LIST OF TABLES AND FIGURES

1.0 INTRODUCTION

Ontario Power Generation (OPG) is performing a Periodic Safety Review (PSR) to support the possibility of continued operation of Pickering Nuclear Generating Station (NGS) beyond 2020.[1] A comprehensive Integrated Safety Review (ISR) was completed for Pickering Units 5 through 8 in 2009 in support of refurbishment and continued operation. Pickering Units 1,4 integrated safety assessments were also performed for Pickering A Return to Service (PARTS) in support of approval to restart Units 1 and 4. In addition to these Pickering-specific studies, the 2013 Darlington ISR performed extensive code and standard reviews that were updated in relation to the versions that were assessed in the 2009 Pickering B ISR.[2] These previous ISRs are considered to constitute the first PSR completed for Pickering (referred to as "PSR1"). The current PSR (referred to as "PSR2") is a subsequent PSR building on the basis of earlier OPG integrated safety assessments through review of the various studies, assessments and licence renewals performed since PSR1. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

PSR2 will support and complement the licence renewal application for Pickering NGS going forward. Fifteen Safety Factors will be assessed as part of the PSR. The purpose of Safety Factor reviews is to confirm that the design, condition and operation of Pickering Units 1,4 and 5-8 (as well as common systems) will support continued safe operation for the period of PSR2 by:

- Assessing compliance against "Review Tasks" identified in the PSR2 Basis Document [1], which were derived from Canadian Nuclear Safety Commission (CNSC) REGDOC-2.3.3 [2] and International Atomic Energy Agency (IAEA) SSG-25 [3];

- Documenting assessments against applicable modern Laws, Regulations, Codes and Standards (L/R/C/Ss) (as defined in Reference [1]); and

- Reviewing the effectiveness of applicable OPG Nuclear Programs that support the above assessments, through review of audit and self-assessment results.

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[1] Currently, Pickering Units 5-8 are approved to operate to 247,000 Effective Full Power Hours. This operation limit is expected to be reached on some units in 2020. For the purposes of PSR2, OPG assumes operation of Pickering NGS for up to eight additional years, from 2020 until 2028. OPG will make a decision regarding the permanent shut down dates for the six reactors following the performance of a technical evaluation that will include PSR2, and will communicate it to the CNSC as required by the current Power Reactor Operating Licence (PROL).

[2] Much of the compliance assessment and evaluation of Safety Factor health for the Darlington ISR is based on programs and practices that apply across OPG's Nuclear operations. As a result, where Pickering is confirmed to follow the same Nuclear programs and practices as were assessed for Darlington, the Darlington ISR programmatic conclusions are applicable to Pickering. As discussed in Section 1.0, an effectiveness review (effectiveness at Pickering NGS) of OPG programs used to demonstrate compliance with the PSR2 Assessment Basis will be conducted using recent audit and self-assessment results.

As outlined in IAEA SSG-25 [3], the objective of the review of the Organization, Management System and Safety Culture Safety Factor 10 is to: "determine whether the organization, Management System and safety culture are adequate and effective for ensuring the safe operation of the plant." REGDOC-2.3.3 [2] requires that: "The licensee shall conduct a PSR in accordance with this regulatory document for the period until the next PSR or, if applicable, until the end of commercial operation of the plant."

This report documents the results of the review of Safety Factor 10 for Pickering PSR2. The report is based on the OPG Governance, Programs, data, and material available up to January 15, 2016 which is the freeze date for PSR2.

2.0 SCOPE OF REVIEW

2.1 Review Task Assessments

The Pickering PSR2 Safety Factor 10 Review Tasks are defined in Reference [1]. Details of the derivation of these Review Tasks from CNSC REGDOC-2.3.3 [2] and IAEA SSG-25 [3] are shown in Reference [5]. The Safety Factor 10 Review Tasks are:

1) Review organization and administrative procedures to ensure they play a significant role in defining safety culture and evaluate the adequacy of safety culture indicators.

2) Establish existence of a safety policy to ensure that safety takes precedence over production where a conflict between these two requirements exists.

3) Identify the method for setting performance targets and confirm that these targets are regularly and systematically reviewed. Confirm that appropriate actions are initiated if safety targets are not met.

4) Confirm that the published Nuclear organization, including any recent changes made to the organization, clearly defines the roles and responsibilities of all individuals and work groups who are involved in activities that could influence the safe operation of the station. Ensure that this organization is understood and that adequate and effective procedures are in place to ensure the availability of these resources and control changes to this organization.

5) Establish that mechanisms for maintaining configuration control of the plant and its documentation are effective and up-to-date.

6) Confirm that there are formal arrangements for employing external technical, maintenance or other specialist staff, and confirm that the contracting procedures ensure that contract employees are qualified to do the work assigned to them.

7) Confirm that there is an approved Quality Assurance program and that regular Quality Assurance audits are conducted involving both internal and independent assessors.

8) Confirm that a program for self-assessment and continuous improvement has been adequately and effectively implemented including feedback of experience relating to organizational and management failures.

9) Confirm there is a system to ensure that comprehensive, easily retrievable, and auditable records exist of baseline design information, and operational and maintenance history.

10) Confirm there is an effective framework in place to support the management of regulatory affairs.

11) Confirm that the organization and Management System include the processes and supporting information that explain how work is to be specified, prepared, reviewed, performed, recorded, assessed and improved.

12) Confirm there is control of purchasing of equipment and services where this affects plant safety.

13) Confirm there are comprehensive communication policies in place.

14) Confirm that a questioning attitude exists and conservative decision making is undertaken in the organization.

15) Verify that there is a process in place for prioritization of safety issues, with realistic objectives and timescales that ensures that these issues receive proper resources.

The methodology for the reviews is discussed in Section 3.1. Review Task findings are summarized in Section 4.1 of this report.

2.2 L/R/C/S Reviews

The applicable Laws, Regulations, Codes and Standards relevant to the Organization, Management System and Safety Culture Safety Factor are identified in Reference [1] and are listed in Table 1 below. Table 1 also identifies the modern version and date of each L/R/C/S to be considered, the Safety Factor(s) to which each document is applicable, and the type of review that will be completed in PSR2.

All of the Safety Factor 10 L/R/C/S reviews are incremental in nature. The definition of an Incremental Review is as follows:

- Incremental Review: For L/R/C/Ss that have been reviewed in PSR1 but have had revisions since the last review, a topical review will be performed of the changes.

The methodology for the reviews is discussed in Section 3.2. A detailed assessment for each L/R/C/S is provided in References [6] and [7]. Associated findings are summarized in Section 4.2 of this report.

Table 1: L/R/C/Ss Reviewed for Organization, Management System and Safety Culture Safety Factor 10

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
L/R/C/Ss Referenced in Pickering NGS PROL 48.02/2018						
1	CSA N286	Management System Requirements for Nuclear Facilities	N286-12	5, 6, 9, 10, 11	Incremental	N286 addressed as part of Pickering B and Darlington ISRs.
2	CSA N286.7	Quality Assurance Of Analytical, Scientific And Design Computer Programs	N286.7-16	1, 5, 6, 7, 10	Incremental	N286.7 addressed as part of Pickering B and Darlington ISRs.
3	CNSC RD-204	Certification of Persons Working at Nuclear Power Plants	2008	10	Incremental	RD-204 addressed as part of Darlington ISR.
4	CNSC REGDOC-3.1.1	Reporting Requirements for Nuclear Power Plants	2014	10	Incremental	CNSC S-99 (precursor to REGDOC-3.1.1) addressed as part of Pickering B and Darlington ISRs.
Additional L/R/C/Ss						
5	CNSC G-323	Ensuring the Presence of Sufficiently Qualified Staff at Class I Nuclear Facilities-Minimum Shift Complement	2007	10, 12	Incremental	G-323 addressed as part of Darlington ISR.
6	S.C. 1997, C.9	Nuclear Safety and Control Act (NSCA) and its associated Regulations	Amended in February 2015	10	Incremental	S.C. 1997, C.9 addressed as part of Darlington ISR.
7	SOR/2000-202	The General Nuclear Safety and Control Regulations	Amended in June 2015	10, 15	Incremental	SOR/2000-202 addressed as part of Pickering B and Darlington ISRs.
8	CNSC REGDOC-2.2.2	Personnel Training	2014	10	Incremental	REGDOC-2.2.2 not addressed as part of Pickering B or Darlington ISRs. Transition Plan and gap analysis has been prepared for REGDOC-2.2.2.
9	CNSC REGDOC-2.3.2	Accident Management, Version 2	2015	1, 5, 6, 7, 8, 10	Incremental	REGDOC-2.3.2 addressed as part of Darlington ISR.

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
10	CSA N286.7.1	Guideline for the Application of N286.7-99, Quality Assurance of Analytical, Scientific, and Design Computer Programs for Nuclear Power Plants	N286.7.1-09	1, 5, 6, 7, 10	N/A [3]	N286.7.1 not addressed as part of Pickering B or Darlington ISRs.

2.3 OPG Program Effectiveness Reviews

The OPG Nuclear Programs (N-PROGs) reviewed for Safety Factor 10 are listed in Table 2 below.[4] The methodology for the effectiveness reviews is discussed in Section 3.3. The assessment results of each of the N-PROGs in Table 2 are provided in Appendix B, and findings are summarized in Section 4.3.

Table 2: OPG Programs Reviewed for Safety Factor 10

Document Number	Document Title
N-PROG-AS-0001 [9]	Managed Systems
N-PROG-AS-0002 [10]	Human Performance
N-PROG-RA-0010 [11]	Independent Assessment

2.4 Additional Reviews

The PSR2 Safety Factor 10 report includes a review of the R04 Pickering Licence Conditions Handbook (LCH) [4] for any impacts of Pickering NGS operation beyond 2020 on the following (all related to Safety Factor 10):

- OPG commitments previously made to the CNSC;
- Open CNSC action items; and
- Exemptions granted by the CNSC.

The PSR2 assessment includes identification and review of previously identified PSR1 gaps related to Safety Factor 10 to ascertain the implications of extending Pickering

[3] The N286.7.1 guide has been amalgamated into the new (-16) edition of the N286.7 Standard. The N286.7 CSA Impact Statement states [8]: "The CSA N286.7.1 guide will no longer be maintained after this new edition of N286.7 is issued. Any relevant guidance has been put into the new edition of N286.7." As a result, only the review of N286.7-16 has been prepared for PSR2.

[4] The list of Nuclear Programs to be assessed for effectiveness for PSR2 was derived from review of current OPG Governance. Although there may be content in Nuclear Programs that is applicable to multiple Safety Factors, N-PROG reviews are only provided in one Safety Factor report and are not duplicated.

NGS operation beyond 2020. The methodology for these reviews is described in Section 3.4. Any PSR2 gaps identified as a result of programmatic Darlington PSR1 gaps related to Safety Factor 10 are discussed in Section 4.4 of this report. The review of Pickering PSR1 gaps previously identified in the Pickering Units 5-8 Continued Operations Plan (COP) [12] is provided in a separate PSR2 COP Review Report.

In addition, Fukushima Action Items (FAIs) were reviewed to identify implications of extending operation beyond 2020 (if any). This review is presented in a separate PSR2 FAI Review Report.

Any PSR2 gaps identified as a result of the Safety Factor 10 review which need to be addressed in other Safety Factor Reports are discussed in Section 4.4 of this report.

3.0 METHODOLOGY

The sub-sections below summarize the methodology used to assess Review Tasks, L/R/C/Ss, and Nuclear Program effectiveness for the Organization, Management System and Safety Culture Safety Factor.

3.1 Review Tasks

As discussed earlier, the Safety Factor Review Tasks are derived from CNSC REGDOC-2.3.3 [2] and IAEA SSG-25 [3], taking into consideration the Review Tasks used in the Pickering B and Darlington ISRs (as derived in [5]).

For each Safety Factor 10 Review Task identified in Section 2.1, a confirmation of the existence of applicable OPG Governance, Programs, Policies and Procedures (as well as Instructions and Guidelines, as applicable) was performed. Compliance against Review Tasks is also assessed by reference to applicable Condition Assessments, safety analyses and operating experience, as required.

The Review Task assessments identify Compliances and Gaps as defined below:

- Compliance: Compliance indicates that either the safety requirement or the intent of the Review Task is met.
- Gap: A Gap indicates that the intent of the Review Task is not met.

3.2 L/R/C/S Reviews

The process to identify the modern L/R/C/Ss that are applicable to the PSR2 Assessment Basis involved first creating a broad list from multiple sources (potential candidate L/R/C/Ss) and then filtering it to identify those that are most significant and that are applicable to the PSR2 scope. The identification and selection criteria are detailed in Reference [1]. The result of the identification and selection process was a set of modern L/R/C/Ss that became part of the "PSR2 Assessment Basis".

PSR2 is focused on the extension of Pickering NGS operations beyond 2020, and will conduct reviews against a baseline of past PSR1 work. As a subsequent PSR, PSR2 focuses on changes in requirements, plant conditions, operating experience and new information. Since PSR2 is an update of previous ISRs, it incorporates reviews of L/R/C/Ss that have occurred as new versions have been issued. Since this assessment is a subsequent PSR, the focus is on identifying differences between what was previously assessed and what is now different within the current Pickering PSR2 Assessment Basis. In general, these differences relate to:

- More recent (new or revised) L/R/C/S versions than what was previously assessed as part of PSR1;[5]

- Safety significant differences between Pickering and Darlington, if the Darlington ISR is the basis for the earlier assessment;

- Implications of extending Pickering NGS operation beyond 2020; and

- Safety significant differences between Pickering Units 1,4 and Units 5-8.

As described in Reference [1], L/R/C/S review types are clause-by-clause, high level or incremental. Most of the L/R/C/Ss in the PSR2 Assessment Basis receive incremental reviews since PSR2 is an update of previous PSR1 assessments and clause-by-clause or high level reviews for the majority of the L/R/C/Ss in the PSR2 Assessment Basis have already been completed. Implementation plans (including gap analyses or code-over-code reviews) also exist for the latest editions of many L/R/C/Ss. As a result, incremental review is also used in circumstances where a L/R/C/S in the PSR2 Assessment Basis was not assessed in previous PSR1 reviews but an implementation plan currently exists for compliance.

The PSR2 incremental reviews in this report include an assessment of the intent of recent changes to the L/R/C/Ss on a topic or subject-matter basis where there is potential to impact nuclear safety. Incremental reviews provide:

- A summary of the purpose of the L/R/C/S;

- Pertinent background information about the current revision of the L/R/C/S that is being considered;

- Identification of which Safety Factor(s) are applicable to the current revision of the L/R/C/S;

- A description of which version(s) of the L/R/C/S were assessed for PSR1 (i.e., Darlington ISR (for programmatic content), Pickering B ISR and PARTS code reviews);

- Identification of whether the current version of the L/R/C/S is an update of a previous version of the L/R/C/S that was assessed in PSR1 (and if so, a description of the major changes in the latest revision is provided as discussed below);

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5 "New" refers to a code or standard that was not previously considered in the context of earlier assessments. "Revised" refers to an updated version of a code or standard that was previously considered in the context of earlier assessments. Where a document has a new number/type, but addresses the same topic from the same organization, it is a "revised", not "new", document (e.g., if a REGDOC replaces a CNSC G or RD document).

- An assessment of the applicability of PSR1 assessment findings (gaps and conclusions), including the implications of extending Pickering NGS operation beyond 2020 if any;

- An assessment of the applicability of assessment findings that address more recent (post-PSR1) editions of the L/R/C/S, including any implementation or transition plans that are already committed to by OPG; and

- Where PSR1 and post-PSR1 assessments are not sufficient to address changes in the latest edition of the L/R/C/S, an assessment of the changes from the previously assessed edition of the L/R/C/S (including identification of any safety significant PSR2 gaps which result).

High Level reviews provide the same information as above, where applicable, in a similar format. However, given that High Level L/R/C/Ss generally have not received past assessment during PSR1, the Incremental review content is augmented by a high level, section-by-section assessment of the degree of conformance of Pickering NGS with the L/R/C/S (demonstrating, with supporting evidence, whether the intent of the requirements stipulated in the document are met).

There are currently no L/R/C/S clause-by-clause reviews identified in the PSR2 Assessment Basis.

The Safety Factor 10 L/R/C/S reviews identify Compliances and Gaps as defined below:[6]

- <u>Compliance</u>:

    - Where an Incremental review has been performed, Compliance indicates that the change in the safety requirement, per the topical review, is met.

    - Where a High Level review has been performed, Compliance indicates that the intent of the safety requirement is met. (Note: No High Level reviews were performed as part of Safety Factor 10.)

    - For Clause-by-Clause reviews of modern L/R/C/Ss, Compliance indicates that the safety requirement is met. (Note: No Clause-by-Clause reviews were performed as part of Safety Factor 10.)

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[6] Safety Factor assessments for Review Tasks and L/R/C/S reviews make use of: a) OPG Governance, Programs, Policies and Procedures which support the assessment arguments, b) Commitments previously made to the CNSC, open CNSC action items, and exemptions granted by the CNSC (all related to the Safety Factor under review), as identified in the R04 Pickering LCH [4], c) Identification of previously identified Pickering-specific or programmatic PSR1 gaps related to the Safety Factor under review and the status of OPG's improvement plan(s) or other dispositions to address these, and d) Assessments and reviews performed since the PSR1 documents were completed.

- Gap:
  - Where an Incremental review has been performed, a Gap indicates that the change in the safety requirement, per the topical review, is not met.
  - Where a High Level review has been performed, a Gap indicates that the intent of the safety requirement is not met. (Note: No High Level reviews were performed as part of Safety Factor 10.)
  - For Clause-by-Clause reviews of modern L/R/C/Ss, a Gap indicates that the safety requirement is not met. (Note: No Clause-by-Clause reviews were performed as part of Safety Factor 10.)

The reviews assume that use of the word:

- "Shall" is used in an L/R/C/S to express a requirement, i.e., a provision that the licensee is obliged to satisfy in order to comply with the standard;
- "Should" is used to express a recommendation or that which is advised but not required;
- "May" is used to express an option or that which is permissible within the limits of the standard; and
- "Can" is used to express possibility or capability.

3.3 OPG Program Effectiveness Reviews

As discussed earlier, effectiveness reviews (effectiveness at Pickering NGS) of OPG programs used to demonstrate compliance with the PSR2 Assessment Basis were conducted, using recent applicable audit and self-assessment results:

- OPG Nuclear Oversight independent performance-based Program audits (typically performed in 1 to 5 year cycles) and self-assessments. This includes review of associated Station Condition Records and Action Requests to determine the status of any resulting corrective actions; and
- CNSC "Type I" and "Type II" inspections of the effectiveness and performance of OPG programs, where discussed in OPG audits or self-assessments.

There are many audits and self-assessments that are performed to assess the effectiveness of important aspects of each program. A sample of audits and self-assessments has been summarized for each program in order to demonstrate that program effectiveness is being assessed on an ongoing basis. The focus of these reviews was on effectiveness of the programs at Pickering NGS, where specific information is available. Results from these audits and self-assessments will be considered in the Global Assessment process. It is noted that audits and self-assessments are, by their nature, self-critical and are used to drive excellence in

performance. As a result, the broad review scope of program audits focuses on identifying improvement opportunities rather than presenting a balanced picture of program performance.

Program effectiveness is also monitored and addressed through the Fleetview Program Health and Performance Reporting process [13]. This process involves direct oversight by the Chief Nuclear Officer, and includes participation by the Nuclear Executive Committee members. Programs are reviewed, senior oversight is provided, and improvement plans are generated.

The list of Nuclear Programs to be assessed for each Safety Factor was derived from review of current OPG Governance, and has used the most recent version of these documents as of the PSR2 freeze date of January 15, 2016.

3.4 Additional Reviews

A review of the R04 Pickering LCH [4] was also performed to determine if there are any impacts associated with operation of the Pickering Units past 2020 on the following (all related to Safety Factor 10):

- Commitments previously made to the CNSC;
- Open CNSC action items; and
- Exemptions granted by the CNSC.

The PSR2 assessment includes identification and review of previously identified Pickering-specific or programmatic PSR1 gaps related to Safety Factor 10 (as identified in the Darlington ISR Integrated Implementation Plan [14] and Pickering Units 5-8 Continued Operations Plan [12]) to ascertain the status of OPG's improvement plan(s) or other dispositions to address these and the implications of extending operation beyond 2020 (if any).[7]

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[7] PSR2 includes consideration and confirmation that the findings of PSR1 remain valid, as applicable, for the operation period. This includes assessment of PSR1 conclusions against implications resulting from extended operation. In particular, Pickering PSR1 results are applicable to PSR2 if there was a PSR1 gap that is still open, or if a closed PSR1 gap could be affected by extended operation. If so these gaps are carried forward into PSR2 for consideration in the Global Assessment. (When references to PSR1 are made, the source document is identified and the relevant text from that source document is summarized in the context of PSR2.) With respect to the Darlington ISR, much of the evaluation of Safety Factor health is based on programs and practices that apply across OPG's Nuclear operations. As a result, Darlington ISR programmatic conclusions are applicable to Pickering PSR2 where Pickering is confirmed to follow the same Nuclear programs and practices that were assessed for Darlington. Darlington PSR1 results are applicable to Pickering PSR2 if there are Darlington PSR1 gaps that are found to be relevant to Pickering PSR2.

Fukushima Action Items were reviewed to identify implications of extending operation beyond 2020 (if any). The methodology for this review is provided a separate PSR2 FAI Review Report.

Any PSR2 gaps identified as a result of the Safety Factor 10 review which need to be addressed in other Safety Factor Reports are also discussed.

4.0 REVIEW FINDINGS

4.1 Review Tasks

The sub-sections below provide an assessment of the adequacy of applicable OPG Governance, Programs, Policies and Procedures (as well as Instructions and Guidelines, as applicable) in demonstrating compliance against the Safety Factor 10 Review Tasks.

4.1.1 Review Task #1: Role of Procedures in Defining Safety Culture

Review organization and administrative procedures to ensure they play a significant role in defining safety culture and evaluate the adequacy of safety culture indicators.

N-POL-0001, "Nuclear Safety Policy" [15], which takes its authority from OPG-POL-0032, "Safe Operations Policy" [16] and is approved by the OPG Board of Directors, states in part:

> *Nuclear Safety shall be the overriding priority in all activities performed in support of OPG nuclear facilities. Nuclear Safety shall have clear priority over schedule, cost and production.*
>
> *Everyone shall demonstrate respect for nuclear safety by:*
>
> - *Knowing how your work impacts on **C**ontrol the power, **C**ool the fuel, and **C**ontain radioactivity (3C's).*
> - *Applying Event-Free tools and defences to prevent events.*
> - *Reporting adverse conditions so they can be corrected.*
>
> *Everyone shall conduct themselves in a manner consistent with the following Traits for a Healthy Nuclear Safety Culture:*
>
> 1. *Personal Accountability*
> 2. *Questioning Attitude*
> 3. *Effective Safety Communication*
> 4. *Leadership Safety Values and Actions*
> 5. *Decision-Making*
> 6. *Respectful Work Environment*
> 7. *Continuous Learning*

8. *Problem Identification and Resolution*

9. *Environment for Raising Concerns*

10. *Work Processes*

The Chief Nuclear Officer is accountable to the Chief Executive Officer and the Board of Directors to establish a management system that fosters nuclear safety as the overriding priority. This policy shall be reviewed at an appropriate interval.

As described by OPG in the 2013 PROL renewal application, Section 3.1.1, "Relevance and Management", of P-CORR-00531-03860 [17], safety culture at OPG is characterized by the following attributes:

- Safety Demonstrated as a Value - Policies, objectives, and standards are established and communicate the overriding importance of safety;

- Effective Management Systems - Documented processes and procedures establish accountabilities for safety and control work activities;

- Appropriate Behaviours - Clear expectations are established and appropriate tools and methods are provided to ensure that leaders and staff adopt behaviours which foster safety;

- Continuous Learning - Processes are in place for staff observations and concerns, for applying internal and external operating experience, for effective self-assessment and oversight, and for generating and completing appropriate corrective actions; and

- Good Safety Performance - Performance measures and results reflect a healthy regard for safety.

A safety culture program has been integrated in all areas of operation to ensure that individuals at all levels of the organization consider safety as the overriding priority, per N-POL-0001, "Nuclear Safety Policy" [15]. A healthy safety culture underpins high levels of performance in all 14 Safety and Control Areas (SCAs). The Management System SCA and Human Performance SCA are particularly impacted by safety culture. According to P-CORR-00531-03860 [17], in order to monitor and assess the health of safety culture, Pickering NGS was in the process of implementing new management oversight meetings, including Continuous Improvement and the Nuclear Safety Culture Monitoring Panel. Per N-PROC-AS-0083, "Nuclear Safety Culture Monitoring Panels" [18], the Nuclear Safety Culture Monitoring Panel has since been implemented to monitor process inputs that are indicative of the health of the organization's nuclear safety culture in order to identify strengths and potential concerns. As described in P-CORR-08120-0453327, "Pickering Oversight Meeting Structure" [19], continuous improvement is incorporated into "Ready & Reliable – A Strategy for Continuous Improvement" meetings. It is also addressed in Nuclear Safety Oversight Committee reviews of self-assessment and audit results.

A nuclear safety culture assessment is conducted at Pickering NGS every 3 years, in accordance with N-PROC-AS-0077, "Nuclear Safety Culture Assessment" [20].

Corporate Commitment

The Board of Directors takes an active role in communicating the importance of safety. This is demonstrated through N-POL-0001 [15].

N-CHAR-AS-0002, "Nuclear Management System" [21], which takes authority from the Nuclear Safety Policy, N-POL-0001 [15], gives authority to the OPG Nuclear safety processes and defines responsibilities. It specifies that the Chief Nuclear Officer (CNO) is accountable for:

> *"the effectiveness of the overall Nuclear Management System in ensuring our Nuclear facilities are operated and maintained using sound Nuclear safety and defense-in-depth practices to ensure radiological risks to workers, the public, and environment are as low as reasonably achievable, and in keeping with the Nuclear Safety Policy, and the best practices of the international Nuclear community."*

The OPG Nuclear Management System is the framework that establishes the processes and programs required to ensure that OPG and Pickering NGS achieve their safety objectives and continuously monitor their performance against these objectives.

Management Level

OPG has a well-defined organizational structure and strong lines of authority. N-CHAR-AS-0002 [21], describes the Nuclear quality program, while N-STD-AS-0020, "Nuclear Management Systems Organizations" [22] outlines its implementation. N-STD-AS-0020 [22] establishes the lines of authority and definition of duties. It provides a summary of the interfacing organizations that own programs supporting the Nuclear Management System. Operation and maintenance of the station as per regulatory requirements and Nuclear standards for public safety are directed by the Director, Operations and Maintenance, who also coordinates with the centre-led organizations to effectively use resources to achieve performance targets. The quality and quantity of services provided by these centre-led support organizations is monitored by the site Senior Vice President, who holds responsibility for establishing site requirements and priorities, and the Nuclear Executive Committee, which has a key oversight role. Position specific role documents, N-MAN-08131-10000 (numerous sheets) [23], describe the duties, authorities and accountabilities of the positions described in the standard.

Additional guidance is given in N-PROG-OP-0001, "Nuclear Operations" [24], N-PROG-MA-0004, "Conduct of Maintenance" [25], and N-PROG-MP-0007, "Conduct of Engineering" [26]. The training and qualification description document, N-TQD-601-00001, "Leadership and Management Training and Qualification Description" [27] provides qualification and professional development requirements for supervisors and managers which includes Safety Culture for Managers.

Managers ensure that tasks are executed as defined through N-PROG-AS-0002, “Human Performance” [10]. This program is specifically designed to achieve higher levels of nuclear and industrial safety, higher unit reliability, and reduced operating costs through event-free operation. This performance is accomplished through pre-job briefings, post-job debriefings, self-checking programs, communications, self-assessments, and an observation and coaching program.

Qualified Resources

Resources

OPG develops and funds resources through N-PROG-AS-0005, “Business Planning” [28]. The program requires that the business plans consist of generation, costs, and staff plans. Business plans are detailed by each division. Other inputs to this process include relevant work programs for cost categorization and risk management, defined benchmarking with other nuclear utilities, demographic profiles, and minimum shift complement staffing requirements as specified in the plant operating licence and OPG instruction P-INS-09100-00003, “Pickering Minimum Shift Complement” [29]. Strict limits on hours of work are imposed via N-PROC-HR-0002, “Limits of Hours of Work” [30].

OPG also has consultants and contractors available to support the operating organizations.

Qualification and Training

N-PROG-TR-0005, “Training” [31] describes the Training Program for regular staff, contractors, temporary personnel, and other staff assigned work at Nuclear. It includes the structure, processes and tools for defining, developing, implementing, documenting, assessing and improving the training required to ensure Nuclear staff have the appropriate knowledge, skill, and attitudes for safe and efficient plant operation.

As described in Section 1.6.3 of P-CORR-00531-03719, “Application For Renewal Of Pickering Nuclear Generating Station Power Reactor Operating Licence” [32], OPG identifies qualified and competent individuals for key positions. With career development and succession planning being key elements in the management capability strategy, a corporate succession plan ensures that individuals with high leadership potential are identified to help continue excellence in nuclear safety.

Self-regulation

Nuclear safety oversight is established to ensure that the requirements of N-POL-0001, “Nuclear Safety Policy” [15], and N-CHAR-AS-0002, “Nuclear Management System” [21] are implemented throughout Nuclear. The framework, accountabilities for nuclear safety oversight, as well as the external and internal processes used for oversight and assessment of nuclear safety are summarized in N-STD-AS-0023, “Nuclear Safety Oversight” [33].

Nuclear safety oversight is conducted in a manner consistent with the "Traits of a Healthy Nuclear Safety Culture" defined in N-POL-0001 [15]. A variety of internal and external oversight forums and processes are used to review, evaluate and critique the nuclear safety performance of the organization.

Deficiencies, non-conformances, and weaknesses with a process, document, service, or condition are treated as learning experiences and captured in the "Corrective Action Program", N-PROG-RA-0003 [34]. Individuals are encouraged to identify conditions adverse to safety or quality, and notify appropriate levels of management in order to identify causes and minimize or prevent recurrence. This process is well-defined and sufficiently controlled so that concealment of errors is actively discouraged.

<u>External Oversight and Assessments</u>

The Nuclear Safety Review Board (NSRB), composed of senior external nuclear experienced individuals, provides the CNO with an annual independent assessment of OPG Nuclear activities at each station that may impact nuclear safety and performance. The scope and terms of reference for the operation of the NSRB are described in N-STD-RA-0035, "Nuclear Safety Review Board" [35].

OPG also invites the World Association of Nuclear Operators (WANO) to conduct Peer Reviews at each Nuclear site every 2 years. An external team of experts reviews all key functional and organizational areas against the WANO Performance Objectives and Criteria. These assessments identify areas for improvement at the host plant and note strengths that could be useful to share with other plants.

Both the NSRB and WANO reviews provide valuable insight on opportunities for improvement. Utilizing industry peers in these evaluations provides fresh perspectives on plant issues and useful improvement opportunities.

<u>Internal Oversight and Assessment</u>

As described in N-STD-AS-0023, "Nuclear Safety Oversight" [33], critical oversight is conducted internally on all activities affecting nuclear safety in order to identify and report adverse conditions for correction, as well as to strengthen nuclear safety and improve performance.

The Nuclear Oversight Committee (replaced by the Generation Oversight Committee, after the PSR2 documentation freeze date) is a committee of the OPG Board of Directors. The committee is responsible for oversight of safe and efficient operations of the OPG Nuclear business, regulatory compliance of facilities, review of reports from independent oversight of operations, reviews of management and organizational matters, security of OPG's nuclear facilities and substances, and oversight of nuclear waste and decommissioning liabilities and management. The OPG Nuclear Oversight group proactively assesses changes expected in Canadian Standards Association (CSA) N286-12, "Management System Requirements for Nuclear Power Plants" [36] and their impact on the Nuclear Management System.

Per N-STD-AS-0023 [33], the Nuclear Executive Committee (NEC) reviews nuclear safety performance, safety margins, and potential threats to nuclear safety as part of its normal business reviews. This review is fully integrated into the business/operating reports, and takes account of related information such as independent evaluations, internal self-assessments and international operating experience. NEC members participate in the Fleetview Program Health and Performance reporting oversight process per N-PROC-RA-0023 [13]. As outlined in NK38-REF-09701-0561144, "Nuclear Executive Committee Terms of Reference" [37], NEC members include the Chief and Deputy Nuclear Officers, Chief Nuclear Engineer, Chief Supply Officer, Senior Vice Presidents for both Pickering and Darlington stations, as well as a number of Vice Presidents.

N-PROC-RA-0048, "Conducting Performance Based Audits and Assessments" [38] establishes the methodology and requirements for planning, scheduling, staffing, preparing, performing, reporting and follow up of audits and assessments performed by Nuclear Oversight. This procedure takes direction from N-PROG-RA-0010, "Independent Assessment" [11].

N-PROC-RA-0023, "Fleetview Program Health and Performance Reporting" [13], specifies requirements for performing a Program health and performance review and reporting on overall effectiveness of the Management System to support the management assessment of the effectiveness requirements of CSA N286-05 [39] [8]. This procedure takes direction from N-PROG-AS-0001, "Managed Systems" [9].

Nuclear Safety Oversight is established at each of the nuclear operating sites. As per N-STD-AS-0023, "Nuclear Safety Oversight" [33], its purpose is to review station performance related to nuclear safety and initiate appropriate action as required, consistent with plant initiatives and goals.

Reactivity management practices are implemented in accordance with N-STD-OP-0009, "Reactivity Management" [44] and are consistent with good industry practices including performance targets, such that reactivity of the reactor core is always respected and controlled.

N-STD-OP-0036, "Operational Decision Making" [45] provides principles for effective operational decision making and a systematic approach for the application of these principles enabling operational decisions that support safe and reliable plant operation, both in the near and long term. Conservative decision-making is one of the event-free tools used during operational decision-making.

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[8] Pickering NGS is required to comply with CSA N286-05 per the Pickering NGS PROL [40] while Darlington NGS is required to comply with CSA N286-12 per the Darlington NGS PROL [41]. As a result of the Darlington NGS PROL renewal, OPG was compliant with N286-12 by December 31, 2015 [42]. As discussed in NK38-CORR-00531-06780 [43], there are a large number of OPG documents that need to be revised to reflect the numbering change from N286-05 to N286-12. This administrative revision will be completed over the course of a few years and may not be reflected in OPG Governance as of the PSR2 freeze date.

N-INS-09030.2-10000, “Event Free Challenge Process” [46] provides direction for conducting an Event Free Challenge Meeting, which allows a task execution group to perform a final oversight of preparedness prior to the execution of a task. The process is used for situations where the likelihood and consequence of error are high, to put appropriate defenses/barriers in place to ensure safe execution of work.

N-STD-MA-0016, “Reactor Safety Support of Outages” [47] provides requirements for reactor safety support and oversight for nuclear unit outage planning and execution. Shutdown safety is maintained by working in compliance with a well-developed and reviewed outage schedule.

Periodic safety report updates are performed per the CNSC Regulatory document REGDOC-3.1.1, “Reporting Requirements for Nuclear Power Plants” [48] in accordance with the OPG procedure N-PROC-MP-0086, “Safety Analysis Basis and Safety Report Updates” [49]. Probabilistic Safety Assessments are implemented in accordance with N-PROG-RA-0016, “Risk and Reliability Program” [50].

Third party activities, which contribute to the technical basis of plant safety, are formally qualified through N-PROC-MM-0021, “Supply Inspection” [51]. These suppliers are added to the list of approved nuclear suppliers via N-PROC-MM-0010, “Establishing and Maintaining OPG Approved Suppliers List” [52].

Individuals

A questioning attitude is instilled in all staff performing safety related tasks. Active participation, demonstrated through asking questions, is expected during briefings, as stated in N-PROC-OP-0005, “Pre-Job Briefing and Post-Job Debriefing” [53]. For tasks with novel content, additional requirements are specified in N-PROC-OP-0001, “Conduct of Infrequently Performed Tests or Evolutions” [54].

The tools for rigorous and prudent approach are all documented in N-PROG-AS-0002, “Human Performance” [10] and more specifically in N-STD-AS-0002, “Procedure Use and Adherence” [55].

Detailed guidance on the importance of precise communication and tools for implementation are described in N-STD-OP-0002, “Communications” [56].

Nuclear Safety Performance Measures

The Nuclear Safety Policy N-POL-0001 [15] requires that nuclear safety undergoes constant examination through a variety of monitoring techniques to ensure the organization is alert to detect and respond to indicators that signal declining performance.

International Nuclear Safety Advisory Group, Safety Culture, Safety Series No. 75-INSAG-4, IAEA, Vienna (1991) Appendix [57] was used as a guide to ensure that appropriate and sufficient safety culture indicators are in place.

N-PROC-RA-0023, "Fleetview Program Health and Performance Reporting" [13] is a fleet-wide functional review and reporting process for monitoring and reporting on overall program effectiveness.

Corporate Policy Level

As discussed above, a corporate safety policy exists and is reinforced periodically with staff.

Nuclear Executive Level

As per N-INS-08920-10017, "Training Committees" [58], the Nuclear Training Oversight Committee meets on a biannual basis to review performance of the Nuclear Training Program. These meetings may be part of a scheduled NEC meeting. These NEC meetings also normally include the Fleetview Program Health and Performance report review. Quality and completeness of the Fleetview report are reviewed critically by NEC members, and acceptance or rejection of the reports are decided in meetings.

N-STD-AS-0023, "Nuclear Safety Oversight" [33] summarizes the framework, accountabilities for nuclear safety oversight, as well as the external and internal processes used for oversight and assessment of nuclear safety. This standard applies to all aspects of nuclear operations and to all work and other activities undertaken at or in support of the stations.

Power Plant Level

Nuclear safety oversight is established at each of the nuclear operating sites in accordance with N-STD-AS-0023, "Nuclear Safety Oversight" [33]. Nuclear safety performance measures are incorporated with various monitoring techniques to ensure that declining performance is detected and responded to. N-STD-AS-0023 [33] specifically describes the Nuclear Performance Index as a measure to be reviewed. This metric is reported quarterly and is a weighted composite of WANO Performance Indicators related to safety and production performance reliability. This metric is also used to trend the performance and monitor the effectiveness of various improvement programs and allows OPG Nuclear to benchmark against other nuclear plants worldwide.

Other metrics related to the measure of nuclear safety performance include:

- Reactivity Management Index: This index includes all reactivity management events over a 12 month rolling average. Calculating and reporting of this metric are conducted in accordance with criteria in "Reactivity Management", N-STD-OP-0009 [44].

- Site Event Free Day Resets: The Event Free Day Reset program is one of the tools used to identify human performance events. This indicator is reported monthly at each site and reflects the effectiveness of management in reducing errors and improving organizational processes and activities to reduce the significance and frequency of human performance events. Performance improvement is accomplished by identifying human performance events, investigating the causes to determine corrective actions, performing trending and analysis to identify reoccurring and common issue areas, and communicating the results throughout all levels of the organization. The guidance and criteria for application and administration of this metric are identified in N-INS-09030-10002, "Site and Department Level Event Free Day Resets" [59].

- Integrated Risk Management: N-GUID-03611-10005, "Integrated Risk Management Guidelines" [60] establishes the administrative controls, responsibilities, duties for direction, control, conduct, and oversight of risk significant activities (Nuclear Safety, Conventional Safety, Environmental, Generation and Radiological), Infrequently Performed Tests or Evolutions, and Infrequent Maintenance Activities at OPG Nuclear.

N-PROC-AS-0077, "Nuclear Safety Culture Assessment" [20], which derives its authority from N-PROG-AS-0001, "Managed Systems" [9], has the objective to evaluate the strength of an organization's nuclear safety culture against the traits of a healthy nuclear safety culture identified in N-POL-0001 [15]. Nuclear safety culture assessment focuses on Operations, Maintenance and Engineering staff, but also will assess all site staff and is performed every 3 years.

Conclusion:

The conclusion of this Review Task assessment is that OPG organization and administrative procedures are well established and play a significant role in defining safety culture. Safety culture indicators are monitored and adequate. The intent of Review Task #1 is met and therefore Pickering NGS is compliant.

4.1.2 Review Task #2: Safety Takes Precedence Over Production

Establish existence of a safety policy to ensure that safety takes precedence over production where a conflict between these two requirements exists.

OPG has a document hierarchy that addresses the requirements of this Review Task.

1. A statement of corporate safety policy

   N-POL-0001, "Nuclear Safety Policy" [15] approved by the OPG Board of Directors states in part:

   > *Nuclear Safety shall be the overriding priority in all activities performed in support of OPG nuclear facilities. Nuclear Safety shall have clear priority over schedule, cost and production…*

Everyone shall conduct themselves in a manner consistent with the following Traits of a Healthy Nuclear Safety Culture:

1. *Personal Accountability*
2. *Questioning Attitude*
3. *Effective Safety Communication*
4. *Leadership Safety Values and Actions*
5. *Decision-Making*
6. *Respectful Work Environment*
7. *Continuous Learning*
8. *Problem Identification and Resolution*
9. *Environment for Raising Concerns*
10. *Work Processes*

The Chief Nuclear Officer is accountable to the Chief Executive Officer and the Board of Directors to establish a management system that fosters Nuclear safety as the overriding priority. This policy shall be reviewed at an appropriate interval.

2. Processes and responsibilities that exist to implement the policy

N-CHAR-AS-0002, "Nuclear Management System" [21], defines the nuclear safety processes and responsibilities. It also specifies that the CNO is accountable for:

> *"the effectiveness of the overall Nuclear Management System in ensuring our Nuclear facilities are operated and maintained using sound Nuclear safety and defense-in-depth practices to ensure radiological risks to workers, the public, and environment are as low as reasonably achievable, and in keeping with Nuclear Safety Policy, and the best practices of the international Nuclear community."*

N-PROG-AS-0002, "Human Performance" [10] derives its authority from the Charter N-CHAR-AS-0002 [21]. The Human Performance (Hu) program, which describes key accountabilities and core processes associated with Hu management, addresses many elements of the organization and administration portion of Safety Factor 10.

3. Standards that provide implementation tools for management expectations

   N-STD-OP-0012, "Conservative Decision Making" [61], derives its authority from the Human Performance program. It specifically states in Section 1.2.4, "*Safety shall remain the number one priority ahead of production or cost*." N-STD-OP-0012 [61] requires the operators to reduce reactor power, trip the unit, or take necessary actions to place the plant in a safe condition when key plant parameters deviate from expected conditions.

4. Instructions and procedures used in day-to-day operations

   A questioning attitude is instilled in all staff performing safety related tasks. A formal process is defined in N-PROC-OP-0005, "Pre-Job Briefing and Post-Job Debriefing" [53].

   The tools for maintaining a rigorous and prudent approach during daily operations are documented in N-PROG-AS-0002 [10] and more specifically in N-STD-AS-0002, "Procedure Use and Adherence" [55].

Conclusion:

The conclusion of this Review Task assessment is that OPG has a safety policy in place that requires nuclear safety take precedence over production where a conflict between these two requirements exists. The intent of Review Task #2 is met and therefore Pickering NGS is compliant.

4.1.3 Review Task #3: Method for Setting Performance Targets

Identify the method for setting performance targets and confirm that these targets are regularly and systematically reviewed. Confirm that appropriate actions are initiated if safety targets are not met.

The Nuclear Safety Policy N-POL-0001 [15] requires that nuclear safety undergoes constant examination. The Nuclear Management System Charter, N-CHAR-AS-0002 [21], identifies expectations for the organization to develop priorities based on performance indicators and known challenges.

These expectations and targets are achieved through a variety of monitoring techniques to ensure the organization is alert to detect and respond to indicators that signal declining performance. In order to improve performance, a set of key performance indicators is established, measured, and trended. Identified adverse trends are analyzed, forming the basis for development of corrective actions designed to improve plant and equipment performance, operating requirements and practices, and overall human performance. Performance analysis, per N-PROC-AS-0078, "Nuclear Performance Monitoring and Reporting" [62], requires comparison of results and established business plan targets. Where performance is not meeting targeted results or is trending adversely, analysis of the data to explain variances is required and mitigating actions shall be provided.

Nuclear Performance Index

As described in "Nuclear Performance Monitoring and Reporting", N-PROC-AS-0078 [62], this metric is reported quarterly and is a weighted composite of ten WANO Performance Indicators related to safety and production performance reliability. This metric is also used to trend the performance and monitor the effectiveness of various improvement programs and allows OPG Nuclear to benchmark against other nuclear plants worldwide.

Reactivity Management Index

This index includes all reactivity management events over a 12 month rolling average. Calculating and reporting of this metric are conducted in accordance with criteria in "Reactivity Management", N-STD-OP-0009 [44].

Site Event Free Day Resets

This indicator is reported monthly and reflects the effectiveness of management in reducing errors and improving organizational processes and activities to reduce the significance and frequency of human performance events. The criteria for application and administration of this metric are identified in N-INS-09030-10002, "Site and Department Level Event Free Day Resets" [59].

Performance Monitoring and Reporting

OPG monitors performance indicators to measure the performance of the stations against the business planning goals and targets. The monitored and measured indicators are documented in OPG systems in accordance with N-PROC-AS-0078, "Nuclear Performance Monitoring and Reporting" [62]. This document provides the following:

- Requirements and descriptions of performance measures for nuclear and divisional level reporting;

- Specifications for internal and external performance reports; and

- Requirements for performance measure data, OPG Nuclear performance index, and reporting of performance indicators.

Revision of Performance Measures

OPG Instruction N-INS-08115-10000, "Addition, Deletion and Revision of EPR Performance Measures" [63] provides staff with detailed guidelines and instructions for the introduction of new performance measures, and the revision or deletion of existing performance measures in the Electronic Performance Reporting (EPR) system used for Operational Performance Reporting. The document also defines the requirements for setting and revising targets. It states, "Targets are goals set each year during the business planning process in Nuclear to track progress toward achieving Nuclear excellence and becoming a world class organization. Targets are assessed on a monthly basis as they work toward achieving year-end goals."

Performance measures and other metrics are also found outside of the EPR system. While the EPR system is used for reporting Nuclear business performance management measures, the World Association of Nuclear Operators Data Entry System is used for reporting Nuclear Performance Index performance measure results to WANO. WANO collects, checks, and collates data for each operating nuclear plant and reactor unit after each quarter to produce a range of indicators. As described in Section 1.2.3, "Consistency", of N-PROC-AS-0078, "Nuclear Performance Monitoring and Reporting" [62], definitions and methods of calculation and collection for performance measures input to the EPR system and the WANO Data Entry System must be consistent across sites and business units, as well as remain the same year-over-year to act as a basis for comparison and benchmarking. Other measures, such as the Tier 4 measures also defined in N-PROC-AS-0078 [62], are department or site specific. They are established, maintained, and monitored by individual departments, and managed locally by a site or business unit.

A detailed review of OPG processes for assessing performance indicators is presented in the Pickering NGS PSR2 Safety Factor 8 Report (Safety Performance) [64]. The specific task assessed in Section 4.1.5 of the Safety Performance Safety Factor Report [64] that addresses the requirements of this Review Task is:

- Confirm that there is an adequate set of Performance Indicators that provides a systematic and comprehensive method to record, trend and analyze safety-related data including the major system parameters, maintenance and inspection records.

In accordance with the PROL 48.02/2018 [40], OPG is required to comply with the CNSC standard REGDOC-3.1.1 [48] and report to the CNSC on a quarterly basis the specific performance indicators specified in REGDOC-3.1.1 [48].

Corrective Action

OPG Program N-PROG-RA-0003, "Corrective Action" [34] provides requisite measures to ensure non-conformances and abnormal occurrences are promptly identified, documented in sufficient detail, and reported. Non-conformances are evaluated using a graded approach based on significance. When appropriate, the root cause is identified, which forms the basis for development of immediate corrective actions and actions to prevent recurrence. The aspect of using Operating Experience from within OPG Nuclear and the industry is an integral part of this program.

Further details of OPG processes for assessing performance indicators are presented in the Pickering NGS PSR2 Safety Factor 8 Report (Safety Performance) [64].

Conclusion:

The conclusion of this Review Task assessment is that OPG has processes in place for setting, managing, and regularly reviewing performance targets, and for initiating corrective actions if safety targets are not met. The intent of Review Task #3 is met and therefore Pickering NGS is compliant.

4.1.4 Review Task #4: Nuclear Organization

Confirm that the published Nuclear organization, including any recent changes made to the organization, clearly defines the roles and responsibilities of all individuals and work groups who are involved in activities that could influence the safe operation of the station. Ensure that this organization is understood and that adequate and effective procedures are in place to ensure the availability of these resources and control changes to this organization.

Organization

OPG has a well-defined organizational structure and strong lines of authority. OPG Standard N-STD-AS-0020, "Nuclear Management Systems Organizations" [22] outlines the organization and establishes the lines of authority and definition of duties for implementation. Operation and maintenance of the station as per regulatory requirements and Nuclear standards for public safety are directed by the Director, Operations and Maintenance, who also coordinates with the centre-led organizations to effectively use resources to achieve performance targets. The quality and quantity of services provided by these centre-led support organizations is monitored by the site Senior Vice President, who holds responsibility for establishing site requirements and priorities, and the Nuclear Executive Committee, which has a key oversight role. Position specific role documents N-MAN-08131-10000 (numerous sheets) [23], describe the duties, authorities and accountabilities of the positions described in the standard.

Additional guidance is given in N-PROG-OP-0001, "Nuclear Operations" [24], N-PROG-MA-0004, "Conduct of Maintenance" [25], and N-PROG-MP-0007, "Conduct of Engineering" [26]. The training and qualification description document, N-TQD-601-00001, "Leadership and Management Training and Qualification Description" [27] provides qualification and professional development requirements for supervisors and managers. The responsibilities of Duty Managers are documented in N-STD-OP-0008, "Expectations of Duty Managers" [65].

<u>OPG Business Transformation</u>

As described in P-CORR-00531-03719 [32], improvements have been made to the organization structure and accountabilities within OPG Nuclear as part of the OPG initiative on Business Transformation. Changes made in May 2012 in accordance with OPG-PROC-0166, "Organization Design Change" [66], maximize consistent application of OPG programs across the fleet while providing direct support in the plant on an ongoing basis, and improve accountability for organizational outcomes. As outlined in P-CORR-00531-03860 [17], the process included organizational and governance streamlining, as well as shifting from a station reporting structure to a more centre-led structure for certain organizational functions. This centre-led structure describes reporting relationships at the leadership level, and how teams are aligned or grouped together under a function. Although all functions in the organizational structure support operating units, some portions, such as the Commercial Operations and Environment unit, are centre-led under OPG as a whole rather than specifically under Nuclear.

Despite modifications to the reporting line, a safety culture program is integrated in all areas of operation, and individuals at all levels of the organization continue to consider safety as the overriding priority per the Nuclear Safety Policy, N-POL-0001 [15]. The quality and quantity of services provided by these centre-led support organizations is monitored by the site Senior Vice President, who holds responsibility for establishing site requirements and priorities, and the Nuclear Executive Committee, which has a key oversight role.

OPG-POL-0033, "OPG Business Model" [78], defines roles and accountabilities using an organizational structure with centre-led functions supporting operating units. This policy requires that each operating unit and function has a Management System designed to efficiently and effectively achieve the goals and objectives of the operating unit and function. Appendix A of OPG-POL-0033 [78] clearly outlines responsibilities of various business decision areas, including those that affect safe operation of the station such as conventional safety and training. The Decision Rights table is a part of OPG's Integrated Framework, which describes how the company's governance, Management Systems, risk processes, and assurance function interact to guide operation. The Integrated Framework is designed to support the needs of the Board and OPG leadership by providing clear expectations and guidance that ensure the enterprise is meeting policies and requirements while adequately managing risks.

Change Control

Changes in organizational structure are managed and controlled using OPG procedure OPG-PROC-0166, "Organization Design Change" [66]. It sets out a procedure for managing minor and material organizational changes in alignment with Organization Design Principles and Components, and assigns accountabilities and related requirements for preparing, reviewing, approving, and implementing changes to the OPG Nuclear organization structure. This procedure is used to:

- Review impacts to governance (including role documents) and initiate change as required per OPG-PROC-0001, "Process Administrative Governance Documents" [67];
- Create or move an entire organization unit;

- Add new positions to an organization, which increases staff level beyond the approved business plan;
- Create temporary positions that are expected to last longer than 2 years;
- Make material organization changes; and
- Transfer positions which impact Band H/MP6 level and above (i.e., accountabilities and jurisdiction).

For organizations named in N-STD-AS-0020, "Nuclear Management Systems Organizations" [22], OPG-PROC-0166 [66] is also used to:

- Initiate changes that may have an impact on the Nuclear Pressure Boundary quality program; and
- Initiate changes to role documents that require CNSC notification and approval.

The organizational change control procedure identifies requirements for organizational change. In order to ensure organizational changes are controlled, changes must:

- Not affect safe and reliable operation of OPG facilities;
- Give adequate consideration of best practices for organization design and change management;
- Be justified by business objectives;
- Meet license and regulatory requirements;
- Comply with corporate policies and procedures, and collective agreements;
- Be consistent with human resource practices in areas such as labour relations;
- Be consistent with OPG organization design principles; and
- Have approval in accordance with the Organizational Authority Register.

For changes to key management positions in the organization and Operations staff in the Control Room, regulatory approvals are requested before making the changes. Section 2.1 of the Pickering NGS PROL [40] specifies that a Management System is to be implemented and maintained in accordance with CSA N286-05, "Management System Requirements for Nuclear Power Plants" [39].

Adequacy of Resources

OPG develops and directs resources through N-PROG-AS-0005, "Business Planning" [28]. Other inputs to this process include relevant work programs for cost categorization, defined benchmarking with other nuclear utilities, demographic profiles, and minimum staffing levels as specified in the plant operating licence and OPG procedure P-INS-09100-00003, "Pickering Minimum Shift Complement" [29]. Strict limits on hours of work are imposed via N-PROC-HR-0002, "Limits of Hours of Work" [30].

Conclusion:

The conclusion of this Review Task assessment is that the roles and responsibilities of individuals responsible for safe operation are clearly defined and documented in the published Nuclear organization. This organization is understood and effective governance is in place to ensure availability of these resources and control organizational changes. The intent of Review Task #4 is met and therefore Pickering NGS is compliant.

4.1.5 Review Task #5: Configuration Management

Establish that mechanisms for maintaining configuration control of the plant and its documentation are effective and up-to-date.

Configuration Management is the industry accepted method to ensure that the plant documentation is prepared, is consistent with the plant design basis and licensing basis, and matches the physical plant. Pickering NGS implements configuration management through N-PROG-MP-0005, "Configuration Management" [68]. This Nuclear Program provides the overall direction to ensure the plant is operated, maintained, and modified in conformance with the design and licensing basis and to ensure configuration documentation matches the physical plant.

This program and its effectiveness of implementation at Pickering NGS are discussed in detail in Section 4.1.2 and Appendix B.3 to the Pickering NGS PSR2 Plant Design Safety Factor 1 Report. The assessments in this Safety Factor report conclude that Pickering NGS has a configuration management program that conforms to industry best practices.

Conclusion:

The conclusion of this Review Task assessment is that mechanisms for maintaining configuration control of the plant and its documentation exist and are effective and up-to-date. The intent of Review Task #5 is met and therefore Pickering NGS is compliant.

4.1.6 Review Task #6: Arrangements for Employing External Staff

Confirm that there are formal arrangements for employing external technical, maintenance or other specialist staff, and confirm that the contracting procedures ensure that contract employees are qualified to do the work assigned to them.

N-STD-MP-0014, "Managing Contracted Nuclear Safety Services" [69], provides general requirements for procurement, retention, interfacing, monitoring, and control of external suppliers of nuclear safety services for OPG Nuclear. The range of nuclear safety services covered is listed in Appendix A of this standard. The standard also defines the contracting strategy, qualification of the supplier's Quality Assurance (QA) program, and training and qualifications of the contractor's personnel.

OPG-PROC-0160, "Contractor Safety Management" [70] establishes requirements for managing contracted work to ensure the work is performed safely. Guiding principles of this procedure state that OPG shall pre-qualify and select suppliers, general contractors, and specialized contractors based on their ability to manage contracted work to OPG's safety and other applicable requirements. Furthermore, in accordance with OPG-PROC-0160 [70], line management ensures that:

1. Contract workers have the required qualifications and training identified in the contract document before starting work (such as Certificate of Qualification, licenses, and/or external safety training certificates); and

2. External training qualifications are evaluated for equivalency to OPG requirements and/or any applicable regulatory training standards.

The requirements of N-PROG-MP-0001, "Engineering Change Control" [72], ensure that all problems requiring a modification improve or maintain operability, maintainability, radiological and conventional safety, regulatory or license compliance, and production at an acceptable cost. These requirements also apply to contractors and design agencies performing engineering activities on behalf of OPG Nuclear. When participating in the initiation, design, installation, and commissioning of physical changes and controlled document changes associated with Structures, Systems and Components (SSCs), software, and engineering tooling, contractors must comply with N-PROG-MP-0001 [72], unless they act in accordance with a Quality Program approved by OPG Nuclear.

N-PROG-AS-0007, "Project Management" [73], describes the organizational responsibilities, interfaces, and key program elements for managing and executing projects in OPG Nuclear, and includes Project Oversight and Contract Management. N-STD-AS-0030, "Project Oversight Standard" [71], outlines the criteria and behavioural requirements for Project Oversight and the key elements for oversight of projects executed in OPG Nuclear. It is applicable (but not limited to) to procurement, suppliers, and contractors. Key oversight elements are tailored to include modification of the level of oversight to reflect current project performance and changes in risk profile. For instance, levels of oversight increase where suppliers are new or have performed less well than expected on current and previous projects, or during instances of fabrication by sub-contractors. N-STD-AS-0030 [71] requires that oversight is applied in a manner that respects contract terms and conditions. It does not direct the work of suppliers who are performing under their own approved Management System. The oversight plan is reviewed and updated when required to meet the project objectives in alignment with project and supplier performance.

Contract management incorporates oversight of supplier personnel to ensure they meet all safety, quality, cost, schedule, and performance requirements. It enables parties to meet obligations in order to deliver the objectives required from the contract, and involves active monitoring and anticipation of future issues throughout the contract life cycle. N-PROG-AS-0007 [73] also outlines management of supplemental personnel, which includes contractors and vendors who perform work on and off-site. N-STD-AS-0032, "Oversight of Supplemental Personnel" [74], provides the oversight principles and requirements to be applied to work packages initiated and/or executed within OPG by supplemental personnel.

N-PROG-TR-0005, "Training" [31] describes the program for training regular staff, contractors, temporary personnel, and other staff assigned work at OPG Nuclear.

Conclusion:

The conclusion of this Review Task assessment is that OPG governance exists for employing external technical, maintenance, or other specialist staff and ensuring their qualification for the work assigned to them. The intent of Review Task #6 is met and therefore Pickering NGS is compliant.

4.1.7 Review Task #7: Approved Quality Assurance Program

Confirm that there is an approved Quality Assurance program and that regular Quality Assurance audits are conducted involving both internal and independent assessors.

OPG Charter N-CHAR-AS-0002, "Nuclear Management System" [21], and supporting documents referenced in the charter establish the Nuclear Management System for OPG Nuclear, which assures that systems, equipment, and activities are of the required quality throughout the life of the nuclear facilities. Supporting organizations and contractors who do not have a quality program approved by the Nuclear organization follow Nuclear Management System requirements. The following OPG Nuclear programs and procedures are established to ensure that the Nuclear Management System is effectively audited and assessed on a regular basis, by both internal and independent assessors.

Procedure N-PROC-RA-0023, "Fleetview Program Health and Performance Reporting" [13] describes the process for performing a program health and performance review and reporting on overall effectiveness of the Nuclear Management System to support the management assessment of effectiveness requirements of N286-05[8], "Management System for Nuclear Power Plants" [39].

The standard, N-STD-AS-0023, "Nuclear Safety Oversight" [33], describes independent assessment (external and internal) processes used for oversight and assessment of nuclear safety. Internal independent assessments are performed by Nuclear Oversight, in accordance with N-PROC-RA-0048, "Conducting Performance Based Audits and Assessments" [38]. External independent assessments are performed by the NSRB in accordance with N-STD-RA-0035, "Nuclear Safety Review Board" [35].

The implementation of the Nuclear Oversight program is done through OPG procedure, N-PROC-RA-0048, "Conducting Performance Based Audits and Assessments" [38]. This procedure establishes the methodology and requirements for planning, scheduling, staffing, preparing, performing, reporting and follow-up of audits and assessments performed by Nuclear Oversight. The objective of the audit program is to confirm that the overall Nuclear Management System is established and implemented effectively. As per the requirements of N-PROC-RA-0048 [38], audits are carried out at frequencies determined through risk based assessments, with sufficient frequency to confirm conformance with the QA program and related programs, procedures, and instructions. Audit frequency ranges from one to five years, depending on results of the risk based frequency assessment documented in the Nuclear Operations Assurance Map. Objectives of N-PROC-RA-0048 [38] include identification and documentation of conditions adverse to quality in accordance with N-PROG-RA-0003, "Corrective Action" [34], verification of compliance and effectiveness of the Pressure Boundary quality assurance program, as well as verification of compliance and effectiveness of the Pressure Boundary quality assurance manual for testing and repairing relief valves.

Scheduled audits are supplemented by additional independent audits of specific subjects when effectiveness is questioned. The Nuclear Oversight Matrix table found in Appendix A of N-PROC-RA-0048 [38] provides details of the scope and frequency of audits under the Nuclear Oversight Program as defined in this procedure.

Appendix B of this report, in which the results of audits and self-assessments for Nuclear programs (Managed Systems, Human Performance, and Independent Assessment) are discussed, confirms that audits are not only scheduled, but conducted and findings addressed.

Conclusion:

The conclusion of this Review Task assessment is that there is an approved Quality Assurance program and that regular Quality Assurance audits are conducted involving both internal and independent assessors. The intent of Review Task #7 is met and therefore Pickering NGS is compliant.

4.1.8 Review Task #8: Program for Self-Assessment and Continuous Improvement

Confirm that a program for self-assessment and continuous improvement has been adequately and effectively implemented including feedback of experience relating to organizational and management failures.

N-CHAR-AS-0002, "Nuclear Management System" [21] establishes the overall requirements for sustaining and improving station performance. This is accomplished by:

- Establishing and implementing a managed system of governing documents that communicate the essential elements of Nuclear business;
- Reinforcing individual accountability for performance and implementing various self-verification and independent oversight techniques;
- Identifying, documenting, evaluating, and correcting in a timely manner, conditions adverse to quality;
- Utilizing internal and industry Operating Experience to improve human, plant, and equipment performance and design, procurement, construction, commissioning, and operating requirements and practices; and
- Providing information to the people who need it through the managed systems that establish how necessary information is identified, targeted to required users, maintained current, and communicated effectively.

The Charter identifies a requirement to have planned audits and surveillances designed to provide a comprehensive and critical evaluation of activities to meet all regulatory and OPG objectives. To meet this requirement, audits and assessments are carried out at a sufficient frequency by personnel who neither performed nor verified activities being audited. Results of the audit are promptly documented and communicated to affected groups or individuals, and a corrective action plan is developed to ensure issues are resolved.

As outlined in N-PROC-RA-0048, "Conducting Performance Based Audits and Assessments" [38], audits are managed and conducted by the Nuclear Oversight organization. Audits of OPG Nuclear programs are scheduled on a one to five-year cycle with audit frequency determined through risk based assessments. Higher risk-ranked programs are audited more frequently (i.e., annually), with no program being audited less than once every 5 years. Results of the risk based frequency determination are documented in the Nuclear Operations Assurance Map. The results of the audit are presented to program owners and site management for acceptance and corrective actions.

The following OPG Nuclear programs and procedures are established to ensure that OPG Nuclear programs are effectively audited and assessed on a regular basis.

OPG Program N-PROG-RA-0010, "Independent Assessment" [11] provides independent assessment (internal and external) processes and a Management System review process to perform a comprehensive and critical evaluation of all activities affecting OPG Nuclear. Written observations, findings, and recommendations are submitted to the CNO after each assessment and an annual report is presented to the Nuclear Oversight Committee of the Board of Directors (replaced by the Generation Oversight Committee, after the PSR2 documentation freeze date).

Procedure N-PROC-RA-0023, "Fleetview Program Health and Performance Reporting" [13] describes the process for performing a program health and performance review and reporting on overall effectiveness of the Management System to support the management assessment of effectiveness requirements of CSA N286-05[8], "Management System for Nuclear Power Plants" [39].

Above N-PROC-RA-0023 [13] in the governance hierarchy (as outlined on page 3 of N-PROG-RA-0010, "Independent Assessment" [11]) is N-PROG-AS-0001, "Managed Systems" [9]. For all Nuclear Management System programs and supporting activities, N-PROG-AS-0001 [9] establishes a business framework consisting of "Plan", "Do", "Check", and "Act/Adjust" elements. In particular, the "Check" and "Act/Adjust" components involve monitoring and process improvement. The Managed Systems Program ensures that self-assessments and independent assessments are performed to determine the effectiveness of the Nuclear Management System in achieving expected results, including safety objectives. The Managed Systems Program also ensures that non-compliance issues and improvement opportunities are identified and addressed.

N-PROC-RA-0048, "Conducting Performance Based Audits and Assessments" [38] establishes the methodology and requirements for planning, scheduling, staffing, preparing, performing, reporting and follow-up of audits and assessments performed by Nuclear Oversight. The objective of the audit program is to confirm that the overall Nuclear Management System is established and implemented effectively.

N-PROC-RA-0097, "Self-Assessment and Benchmarking" [75] specifies the requirements for self-assessment activities for the functional and line organizations of OPG Nuclear. It defines the elements required to plan, execute, report, and monitor Divisional Level Self-Assessments, Departmental Level Self-Assessments, and Snapshot Self-Assessments. The self-assessments include evaluations of business programs, processes, and performance.

The Program Management assessment of OPG's Independent Assessment program, N-PROG-RA-0010 [11], is performed every three years. Audits and self-assessments of N-PROG-RA-0010 [11] are discussed in Appendix B.3. Results of audits and self-assessments for other programs, including those discussed in Appendix B, help identify opportunities for improvement (which includes those related to organizational and management issues).

Pickering's two-week Nuclear Safety Culture Assessment completed early in 2015, conducted per N-PROC-AS-0077, "Nuclear Safety Culture Assessment" [20], further confirms successful implementation of self-assessment and continuous improvement programs focusing on Nuclear Safety. Although the Nuclear Safety Culture Assessment report is confidential, an email from the Senior Vice President, Pickering Nuclear, to all Pickering staff entitled "Nuclear Safety Culture Assessment – Results" [76] and dated February 25, 2015 is paraphrased below:

> This assessment consisted of a survey, interviews and field observations. The 81-question survey was sent to all Pickering employees and resulted in 1259 survey responses and over 900 comments. In addition, 65 on-site interviews and 18 field observations were completed by a team of 17 individuals composed of both internal and external team members. This yielded an additional 950 comments. In conclusion, the assessment team determined that Pickering has a healthy Nuclear Safety Culture and a respect for nuclear safety not compromised by production priorities.

Conclusion:

The conclusion of this Review Task assessment is that programs for self-assessment and continuous improvement are fully developed and effectively implemented at OPG, including feedback of experience relating to opportunities for improvement (which includes those related to organizational and management issues). The intent of Review Task #8 is met and therefore Pickering NGS is compliant.

4.1.9 Review Task #9: Record Management

Confirm there is a system to ensure that comprehensive, easily retrievable, and auditable records exist of baseline design information, and operational and maintenance history.

OPG Program OPG-PROG-0001, "Information Management" [77] which takes its authority from OPG-POL-0033, "OPG Business Model" [78], establishes a set of standards and procedures for the management of OPG's information throughout its life cycle regardless of media, including electronic systems such as e-mail, SharePoint, and the Intranet to ensure consistent and appropriate use. It defines uniform and efficient processes for management, maintenance, and final disposition of records and documents through OPG. OPG-PROG-0001 [77] also establishes the overall OPG process for governance including electronic filing, approval, distribution, and maintenance of the OPG Governance Framework. Records are retained and stored to ensure essential records, as defined in referenced procedures, and subsequent revisions are legible, retrievable, traceable, routinely inspected, and available to provide documented evidence of implementation of the quality program. OPG-PROC-0001 [67] ensures a current listing of all Nuclear records is identified and maintained, and also provides a uniform retention schedule to ensure that final disposition of records is carried out according to regulatory and business requirements. Revision control and communication is established through an Approved Information Management System (Asset Suite). Asset Suite also acts as an on-line repository for documents, procedures, and maintenance records.

QA Records are defined in OPG-PROC-0179, "Nuclear Quality Assurance Records" [79], as essential records providing evidence of licensing, design, construction, operation, maintenance, testing, and modification of nuclear facilities. Implementing procedure OPG-PROC-0019, "Records and Document Management" [80] provides direction to ensure that records in the custody or control of OPG are consistently managed, protected, and accessible throughout their life cycle. OPG-PROC-0179 [79] ensures that Nuclear QA records and QA Vaults are managed to protect records against damage by fire, flooding, environmental deterioration, theft, and misuse by unauthorized personnel.

OPG-PROC-0179 [79] identifies the requirements for consistent management and quality checks of Nuclear QA records generated or collected by or for OPG Nuclear. The process for managing the life cycle of Controlled Documents (documents with a defined revision process for their entire lifecycle) is outlined in OPG-PROC-0178, "Controlled Document Management" [81]. Quality Checks are performed for the purpose of self-evaluation, and are less formal and narrower in scope than audits. Annual Quality Checks are conducted on a minimum of 5% of document collections where only paper is maintained. Corrective actions are assigned as required based on results of the Quality Check.

OPG-PROC-0179 [79] identifies the requirements for maintaining Life of Facility, permanent, and all Pressure Boundary QA records which are sole-source. This includes maintenance in a temperature and humidity controlled QA vault for protection against unauthorized access, natural disasters, unsafe environmental conditions, and infestation of insects, mould, or rodents.

N-LIST-01300-10000, "Bounded Document Set" [82] defines the configuration documents that should be updated when modifying the physical or station design. Specifically, Appendix A of N-LIST-01300-10000 [82] lists the set of documentation or data that maintains the physical plant in a state consistent with the design requirements.

N-PROG-MP-0001, "Engineering Change Control" [72], ensures that all modifications to OPG Nuclear SSCs are planned, designed, installed, commissioned, placed into service, or removed from service within the Safe Operating Envelope or Safety and Design Envelope, design basis, and licensing conditions. Along with N-PROG-MP-0009, "Design Management" [99], which specifies requirements for documentation identifying and controlling the design basis and outputs, N-PROG-MP-0001 [72] helps describe a system for maintaining baseline design information.

Conclusion:

The conclusion of this Review Task assessment is that there is a system to ensure that comprehensive, easily retrievable, and auditable records exist of baseline design information, and operational and maintenance history. The intent of Review Task #9 is met and therefore Pickering NGS is compliant.

4.1.10 Review Task #10: Management of Regulatory Affairs

Confirm there is an effective framework in place to support the management of regulatory affairs.

N-PROG-RA-0002, "Conduct of Regulatory Affairs" [83] provides the guidelines for managing the interface with regulatory agencies to ensure effective and efficient compliance with regulatory requirements and to ensure open, honest, and timely communications. Successful interface with regulatory agencies is critical in meeting OPG's overall objectives. This program defines a set of processes to ensure these expectations are met. These implementation procedures are described below. The effectiveness of the Regulatory Affairs program is addressed is Appendix B of the Pickering NGS PSR2 Safety Factor 8 Report (Safety Performance) [64].

N-PROC-RA-0005, "Written Reporting to Regulatory Agencies" [84] defines roles, accountabilities, and processes for complying with regulatory requirements for written event reports to regulatory agencies and for scheduled reports to the CNSC. The Regulatory Agencies include the following:

- CNSC;
- Ministry of Environment and Climate Change;
- Environment Canada;
- Ministry of Labour;
- Transport Canada; and
- Electrical Safety Authority.

N-PROC-RA-0006, "Regulatory Action Management" [85] defines the requirements for the identification and tracking of Regulatory Commitments, Regulatory Obligations, and Regulatory Management Actions. Regulatory Commitments and Regulatory Management Actions are initiated by authorized representatives of OPG.

N-PROC-RA-0020, "Preliminary Event Notifications" [86] defines the requirements and processes for verbally notifying facility and offsite organizations, management, and external officials and agencies after a reportable event has occurred.

N-PROC-RA-0028, "Support of CNSC Type I/II Inspections" [87] defines the roles and responsibilities, as well as provides steps and processes, for OPG staff in supporting CNSC Type I and Type II inspections.

OPG Procedure N-PROC-RA-0047, "Communications with the Canadian Nuclear Safety Commission" [88] defines the planning, review, approval, and records required for communications with the CNSC.

N-PROC-RA-0053, "Evaluation of Proposed Changes for Canadian Nuclear Safety Commission Approval, Consent or Notification" [89] defines the steps for evaluating whether CNSC approval or consent of a proposed change must be obtained prior to implementation, or whether notification of the CNSC is required. Also, the Pickering NGS PROL identifies key organizational documents, and requires that any changes to the Nuclear organization be made in accordance with OPG-PROC-0166, "Organization Design Change" [66].

The reporting requirements of REGDOC-3.1.1 [48] required by the PROL are included in OPG governance as per N-PROC-RA-0020 [86]. This procedure identifies the requirements and process for verbally notifying facility and off-site organizations, management, and external officials and agencies, after a reportable event has occurred. Written notifications to the CNSC after an event are directed through N-PROC-RA-0005, "Written Reporting to Regulatory Agencies" [84].

Conclusion:

The conclusion of this Review Task assessment is that there is an effective framework in place to support the management of regulatory affairs. The intent of Review Task #10 is met and therefore Pickering NGS is compliant.

4.1.11 Review Task #11: Processes and Supporting Information Governing Work

Confirm that the organization and management system include the processes and supporting information that explain how work is to be specified, prepared, reviewed, performed, recorded, assessed and improved.

As discussed in Section 4.1.4, OPG has an organization of centre-led functions supporting the operating units. The OPG organization is described in OPG-POL-0033, "OPG Business Model" [78].

Each function and operating unit of the OPG organization has a Management System sufficient to meet its specific accountabilities. Every Management System is based on the principles of Plan - Do - Check - Act and embraces continuous improvement. An Integrated Framework describes how the company's governance, Management Systems, risk processes and assurance function interact to guide the operation of OPG.

The Nuclear Management Systems Organizations are described in N-STD-AS-0020, "Nuclear Management Systems Organizations" [22]. Work performed at Pickering is specified, prepared, reviewed, performed, recorded and improved by the nuclear operating unit's Operations, Maintenance, Engineering, Nuclear Waste Management, Emergency Services, Inspection & Maintenance Services, Work Management and Projects directorates. Work performed at Pickering is retained by a Business and Administrative Services Function unit called Information Management. Work performed at Pickering is self-assessed by the nuclear operating unit's directorates and independently assessed by the assurance directorates Internal Audit and Nuclear Oversight as well as other third party reviews (e.g. WANO, etc.).

The Nuclear Management System work processes are described in N-CHAR-AS-0002 [21], "Nuclear Management System". Sections of governing documents are cross referenced to clauses of CSA N286-05, "Management System Requirements for Nuclear Power Plants", as required by the Pickering NGS PROL [40], in N-LIST-08130-10023, "CSA N286-05 to OPGN Governance Cross Matrix" [90], and CSA N286-12, "Management System Requirements for Nuclear Facilities", as required by the Darlington NGS PROL [41], in N-LIST-08130-10025, "CSA N286-12 to OPGN Governance Cross Matrix" [91]. The sections of the governance such as N-PROG-RA-0003, "Corrective Action" [34], N-PROG-RA-0010, "Independent Assessment" [11], and N-PROG-RA-0013, "Radiation Protection" [92] also contain cross references to the applicable clauses of CSA N286-05 [39] and CSA N286-12 [36]. There are many programs and procedures that include the processes and supporting information that explain how work is to be specified, prepared, reviewed, performed, recorded, assessed and improved.

Examples of Nuclear work specification, preparation, review, performance and record governance from N-LIST-08130-10025 [91] include:

- N-PROG-MA-0004, "Conduct of Maintenance" [25] Sections 1.1 "Work Planning", 1.2 "Work Execution" and 1.3 "Calibration and Tool Control";

- N-PROG-MA-0019, "Production Work Management" [93] Sections 1.4 "Site On-Line Work Management Processes" and 1.5 "Outage Work Management Processes";

- N-PROG-MP-0004, "Pressure Boundary" [94] Sections 1.5.2 "Pressure Boundary Repair and Replacement, N-PROC-MP-0087" and 1.5.3 "Pressure Boundary Modifications, N-PROC-MP-0088"; and

- N-PROG-RA-0013, "Radiation Protection" [92] Section 1.5.5 "Radioactive Work Planning and Execution".

Examples of Nuclear independent assessment governance from N-LIST-08130-10025 [91] include:

- N-CHAR-AS-0002, "Nuclear Management System" [21] Section 1.1.7 "N-PROG-RA-0010, Independent Assessment" [11]; and

- N-PROG-RA-0010, "Independent Assessment" [11] Sections 1.1 "Independent Assessment – Internal" and 1.2 "Independent Assessment - External".

Examples of Nuclear self-assessment and improvement governance including some cross references from N-LIST-08130-10025 [91] include:

- N-CHAR-AS-0002, "Nuclear Management System" [21] Sections 1.1.1 "N-PROG-AS-0001, Managed Systems" and 1.1.6 "N-PROG-RA-0003, Corrective Action";

- N-PROG-AS-0001, "Managed Systems" [9] Sections 1.3 "Monitoring (CHECK)" and 1.4 "Process Improvement (ACT/ADJUST)";
- N-PROG-RA-0003, "Corrective Action" [34] Section 1.1 "Identification and Resolution of Problems", sub-section 1.1.1 "N-PROC-RA-0022, Processing Station Condition Records", 1.2 "N-PROC-RA-0035, Operating Experience Process", 1.3 "N-PROC-AS-0019, Action Item Management", 1.4 "N-PROC-RA-0097, Self-Assessment and Benchmarking" and Section 1.6 "Performance Indicators and Review", which cites N-PROC-RA-0023, "Fleetview Program Health and Performance Reporting" [13]; and
- N-PROG-MA-0004, "Conduct of Maintenance" [25] Section 1.6 "Performance Indicators and Review", which cites N-PROC-RA-0023, "Fleetview Program Health and Performance Reporting" [13] and N-PROC-RA-0097, Self-Assessment and Benchmarking" [75].

Conclusion:

The conclusion of this Review Task assessment is that the organization and Management Systems include the processes and supporting information that explain how work is to be specified, prepared, reviewed, performed, recorded, assessed and improved. The intent of Review Task #11 is met and therefore Pickering NGS is compliant.

4.1.12 Review Task #12: Control of Purchasing of Equipment and Services

Confirm there is control of purchasing of equipment and services where this affects plant safety.

As discussed in Section 4.1.11, sections of governing documents are cross referenced to clauses of CSA N286-05, "Management System Requirements for Nuclear Power Plants" and CSA N286-12, "Management System Requirements for Nuclear Facilities" in N-LIST-08130-10023, "CSA N286-05 to OPGN Governance Cross Matrix" [90] and N-LIST-08130-10025, "CSA N286-12 to OPGN Governance Cross Matrix" [91] respectively. The sections of the governance such as OPG-PROG-0009, "Items and Services Management" [95] also contain cross references to the applicable clauses of CSA N286-05 [39] and CSA N286-12 [36]. There are many programs and procedures that control purchasing of equipment and services where this affects plant safety.

Examples of Nuclear purchasing governance from N-LIST-08130-10025 [91] include:

- N-CHAR-AS-0002, "Nuclear Management System" [21] Section 1.3.3 "N-PROG-MM-0001, Materials Management" [9];

[9] OPG-PROG-0009 [95] revision 2 supersedes N-PROG-MM-0001, "Materials Management".

- OPG-PROG-0009, "Items and Services Management" [95] Sections 1.1 "Process Management", 1.2 "Core Processes", 1.3 "Procurement of Items and Services", 1.4 "Contract Administration", 1.5 "Nuclear Warehousing" and 1.6 "Item Surplus and Disposal";

- N-PROG-MA-0013, "Welding" [96] Section 1.6 "Procurement of Welded Components and Welding Services";

- N-PROG-RA-0012, "Fire Protection" [97] Section 1.12.3 "OPG-PROG-0009, Items and Services Management";

- N-PROG-MP-0006, "Software" [98] Section 1.4(a) "N-PROC-MP-0049, Procurement of Software and Products Containing Software";

- N-PROG-MP-0009, "Design Management" [99] Section 1.2.9 "Procurement Engineering Process"; and

- N-PROG-MA-0026, "Equipment Reliability" [100] Sections 1.1.1(f) "Identifying and predicting aging and obsolescence issues on important components and embedding mitigating strategies and actions into the business plan" and 1.2.10.3 "Critical Spares".

OPG-PROG-0009, "Items and Services Management" [95] Section 2.12 defines the roles and accountabilities of the Manager, Design Engineering to establish requirements for a managed process of creating and maintaining procurement specifications for replacement items, materials, equipment, components, parts, and services. The processes identified in this program ensure that items, services, and nuclear fuel materials and bundles are purchased in accordance with stated requirements and controlled through proper identification, handling, storage, issuance, and shipping to ensure the quality of equipment and components is preserved.

Purchase Orders for safety-related items and services are subject to Purchase Order verification by the Purchasing Agent to ensure the Purchase Order or contract is correct and complete per Section 1.16.1 of OPG-PROC-0058 [101]. Per Section 1.2.2 (d)(1) of OPG-PROG-0009 [95], changes to item description data require evaluation and approval by qualified staff. For items identified as being safety-related, Pressure Boundary or Nuclear Class, such technical changes are performed in accordance with the Procurement Engineering process outlined in N-PROC-MP-0098, "Procurement Engineering Activities" [102], which includes pre-purchase technical review of items and services and support in resolving technical issues related to purchases. Per Section 1.3.3(b)(1) of OPG-PROG-0009 [95], the Purchasing Agent is not permitted to alter the item description data for safety-related items. Per Section 1.3.3(c)(3) of OPG-PROG-0009 [95], the quotation or proposal evaluation performed by the Purchasing Agent confirms that items or services offered meet technical requirements, which may include safety and reliability.

The controls for establishing and maintaining the OPG Approved Supplier List are documented in Section 1.3.2(b) of N-PROC-MM-0010, "Establishing and Maintaining

Ontario Power Generation Approved Suppliers List" [52]. N-PROC-MM-0010 [52] describes the methods used to originate, request, evaluate, qualify, and maintain the qualification of suppliers of items and services requiring Quality Assurance programs or other OPGN defined Quality requirements (e.g., for Nuclear Class items).

OPG-PROG-0009 [95] is not applicable to the handling, storage, or control of nuclear fuel at OPG power facilities, and items or services specifically described in other approved governing programs and supporting documents. The handling, storage and control of those items is in accordance with requirements stated in the applicable interfacing programs shown in Figure 1 of OPG-PROG-0009 [95].

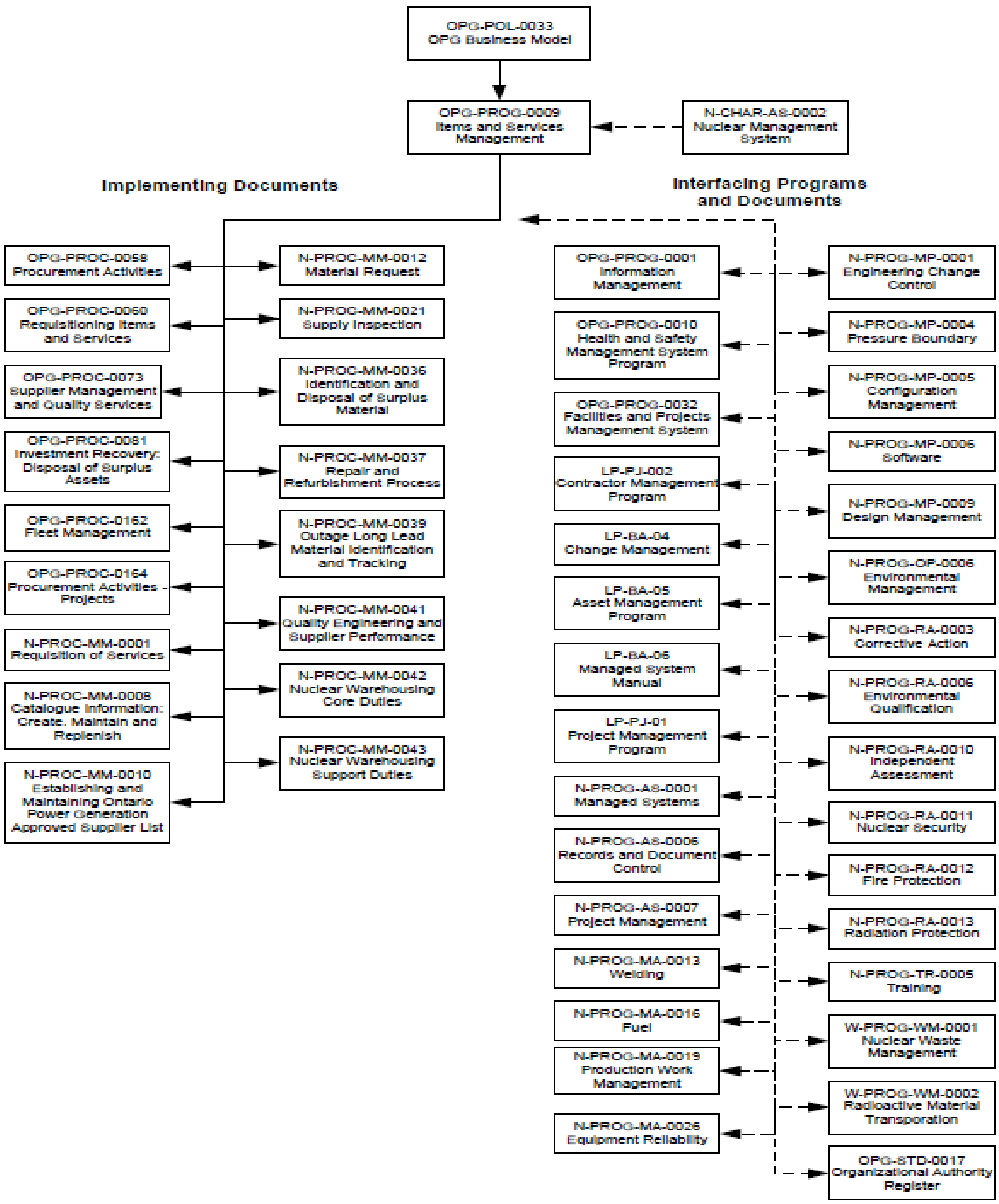

Figure 1: Governing Document Hierarchy for Supply Chain (OPG-PROG-0009 [95])

Conclusion:

The conclusion of this Review Task assessment is that there is control of purchasing of equipment and services where this affects plant safety. The intent of Review Task #12 is met and therefore Pickering NGS is compliant.

4.1.13 Review Task #13: Communication Policies

Confirm there are comprehensive communication policies in place.

N-POL-0001, "Nuclear Safety Policy" [15] states that everyone shall conduct themselves in a manner consistent with the listed Traits for a Healthy Nuclear Safety Culture. One of these traits is effective safety communication.

As discussed in Review Task #11: Processes and Supporting Information Governing Work, sections of governing documents are cross referenced to clauses of CSA N286-05, "Management System Requirements For Nuclear Power Plants" and CSA N286-12, "Management System Requirements For Nuclear Facilities" in N-LIST-08130-10023, "CSA N286-05 to OPGN Governance Cross Matrix" [90] and N-LIST-08130-10025, "CSA N286-12 to OPGN Governance Cross Matrix" [91] respectively. The sections of the governance such as N-PROG-AS-0002, "Human Performance" [10] also contain cross references to the applicable clauses of CSA N286-05 [39] and CSA N286-12 [36]. There are many programs and procedures that implement the communication policies.

Examples of Nuclear governance related to communication from N-LIST-08130-10025 [91] include:

- N-PROG-AS-0002, "Human Performance" [10] Sections 1.2.1 "N-PROC-OP-0005, Pre-Job Briefing and Post-Job Debriefing", 1.2.2 "N-STD-AS-0002, Procedure Use and Adherence", 1.2.3 "N-STD-OP-0002, Communications" and 1.2.10 "N-INS-09030-10004, Observation and Coaching".

N-PROC-AS-0077, "Safety Culture Assessment" [20] Appendix A has survey questions on Safety Communication to assess whether comprehensive communication policies are in place:

(15) Information is freely shared between work groups

(16) There is regular communication about how to work safely

(17) Explanations of operational decisions are promptly communicated

(18) Managers explain the reasoning behind difficult decisions to avoid conflicting messages about safety

(19) There is effective downward flow of information from management

(20) There is effective upward flow of information to management

(21) Management frequently communicates the importance of nuclear safety

(22) Expectations for performing work that can affect safety are communicated to all contractors/vendors

Communication policies applicable to emergencies are described in the Pickering NGS PSR2 Safety Factor 13 Report (Emergency Planning) [120].

In addition, there are OPG corporate level communication policies in place to address nuclear safety, such as communication protocols for nuclear emergencies which include OPG-PROC-0028, "Crisis Management & Communications Centre (CMCC) Procedure" [103] and OPG-PROC-0112, "Corporate Relations and Communications Emergency Preparedness and Response Procedure" [104].

Conclusion:

The conclusion of this Review Task assessment is that there are comprehensive communication policies in place. The intent of Review Task #13 is met and therefore Pickering NGS is compliant.

4.1.14 Review Task #14: Questioning Attitude and Conservative Decision Making

Confirm that a questioning attitude exists and conservative decision making is undertaken in the organization.

N-POL-0001, "Nuclear Safety Policy" [15] states that everyone shall conduct themselves in a manner consistent with the listed Traits for a Healthy Nuclear Safety Culture. Two of these traits are questioning attitude and decision making.

N-PROG-AS-0002, "Human Performance" [10] achieves higher levels of nuclear and industrial safety through event-free operation. It includes Sections 1.2.4 "N-STD-OP-0004, Self-Check" and 1.2.5 "N-STD-OP-0012, Conservative Decision-Making".

N-STD-OP-0004, "Self-Check" [105] describes the features of the Nuclear Self-Check Program:

- Per Section 1.1.2(g), "If uncertain, **stop** and resolve any questions or concerns before proceeding"; and
- Per Appendix A.2(c), "THINK-Understand what should happen when correct action is taken on the correct component. If uncertain, use the questioning-attitude tool".

N-STD-OP-0012, "Conservative Decision-Making" [61] provides management expectations for conservative decision-making in support of a Nuclear Safety culture. Section 4.3 of N-STD-OP-0012 identifies the following performance references:

- N-POL-0001, "Nuclear Safety Policy" [15];
- N-PROG-AS-0002, "Human Performance" [10];
- N-STD-OP-0004, "Self-Check" [105];
- N-STD-OP-0005, "Main Control Room Panel Monitoring, Operation, and Alarm Response" [106]; and
- N-STD-OP-0009, "Reactivity Management" [44].

N-STD-OP-0036, "Operational Decision-Making" [45] provides a systematic approach for the application of principles enabling operational decisions that support safe and reliable plant operation. It states:

- Section 1: Direction
    - *Staff are responsible to ensure individual behaviours and actions are consistent with principles and conservative rules identified in this standard, and applied directly to decision making situations faced on a day-to-day basis.*
    - *Conservative Decision Making is an Event-Free Tool that is used by people when faced with uncertainty or the need to make a decision on the next course of action.*
- Appendix D.1: Operate Conservatively
    - *Conservative operation means you take the 'safe road', i.e., where judgment is required, decisions shall be made which are clearly within established and accepted boundaries of safety.*

N-GUID-01900-10000, "Human Performance Event Free Tools for Knowledge Work" [107] provides guidelines for application of a set of event free tools specifically designed to consider the different work activities in an engineering environment. This guideline outlines when and how engineering staff should apply questioning attitude to help them ensure that planning, judgment and decision making are appropriate for the product in development.

N-PROC-AS-0077, "Safety Culture Assessment" [20] Appendix A has survey questions to assess whether a questioning attitude exists and conservative decision making is undertaken in the organization:

> *(7) Employees understand the unique hazards of nuclear technology.*
>
> *(8) New employees understand working here requires special attention to nuclear safety.*
>
> *(9) Employees stop when they encounter uncertain conditions.*

(10) Employees question unexpected results.

(11) Regardless of position in the organization, people are comfortable questioning each other when they feel something is not correct.

(12) Employees question the assumptions of other employees when the work can affect safety.

(13) Employees maintain a questioning attitude when at work.

(14) Employees remain vigilant for potential problems, even when the plant is operating well.

(39) Employees use a systematic approach when making decisions.

(40) Safety risks are considered in all activities.

(41) Safety is a high priority when decisions are made.

(42) Employees do not justify existing conditions for the sake of completing a task quickly.

(43) When an important nuclear safety decision must be made, it is clear who is responsible to make it.

(44) Important safety decisions are made by the correct person.

As discussed in Section 4.1.8, Pickering completed a Nuclear Safety Culture Assessment in early 2015 in accordance with N-PROC-AS-0077, which concluded that Pickering has a healthy Nuclear Safety Culture.

In addition, the use of questioning attitude and decision making is monitored and reported quarterly in accordance with N-PROC-AS-0083, "Nuclear Safety Culture Monitoring Panels" [18]. For example, in Q1 2015, the application of questioning attitude and decision-making was rated as a strength for several work groups [108].

Conclusion:

The conclusion of this Review Task assessment is that a questioning attitude exists and conservative decision making is undertaken in the organization. The intent of Review Task #14 is met and therefore Pickering NGS is compliant.

4.1.15 Review Task #15: Prioritization of Safety Issues

Verify that there is a process in place for prioritization of safety issues, with realistic objectives and timescales that ensures that these issues receive proper resources.

As discussed in Review Task #11: Processes and Supporting Information Governing Work, sections of governing documents are cross referenced to clauses of CSA N286-05, "Management System Requirements for Nuclear Power Plants" and CSA N286-12, "Management System Requirements for Nuclear Facilities" in N-LIST-08130-10023, "CSA N286-05 to OPGN Governance Cross Matrix" [90] and N-LIST-08130-10025, "CSA N286-12 to OPGN Governance Cross Matrix" [91] respectively. The sections of the governance such as N-PROG-RA-0003, "Corrective Action" [34] also contain cross references to the applicable clauses of CSA N286-05 [39] and CSA N286-12 [36]. There are many programs and procedures that implement safety issue resolution.

Examples of nuclear safety issue prioritization governance including some cross references from N-LIST-08130-10025 [91] include:

- N-PROG-RA-0003, "Corrective Action" [34] Section 1.1 "Identification and Resolution of Problems", sub-sections 1.1.1 "N-PROC-RA-0022, Processing Station Condition Records" and 1.1.2 "N-STD-RA-0008, Incident Investigation", and Section 1.3 "N-PROC-AS-0019, Action Item Management"; and

- N-PROG-MP-0007, "Conduct of Engineering" [26] Sections 1.4.3 "N-STD-MP-0023, Technology and Research" and 1.4.4 "N-PROC-MP-0092, Technology and Research Program Management".

N-PROC-RA-0022 Section 1.5 "SCR Coordinator Dispositioning" [109] has a process for determining the resolution category and significance level for prioritization of safety issues. Sections 1.4 "First Line Manager Review", 1.5.1 "SCR Coordinator Dispositioning", 1.7 "Management Review Meeting", 1.8 "Performance Improvement Department", 1.12 "Corrective Action Review Board" and Appendix A "Corrective Action Process and Status Control Matrix" have timelines for processing Station Condition Records (SCRs), including those related to safety issues.

N-STD-RA-0008, "Incident Investigation" [110], Table 1 has resolution categories for prioritization of safety issues. This standard also addresses timelines and assignment of resources.

N-PROC-MP-0045, "Technical Operability Evaluation" [111] provides a uniform process for identifying and evaluating degraded station conditions when the ability of SSCs to carry out their defined safety-related functions comes into question. It includes a timeline for resolving the issue.

N-PROC-RA-0094, "Discovery Issue Resolution Process" [112] identifies due diligence actions required of staff when issues are identified with the Safety Analysis of an OPG Nuclear station. Discovery issues may result from sources such as new operational information, errors in Safety Analysis development, Research and Development results, and Operating Experience. There are three related governing procedures describing the specific process to be followed when uncertainty arises about safe operation of the plant:

(a) When uncertainty arises with respect to the Deterministic Safety Analysis basis which defines the Safe Operating Envelope, N-PROC-RA-0094 [112] is followed.

(b) When uncertainty arises with respect to operability of equipment to meet the functional requirements of the defined Safe Operating Envelope, N-PROC-MP-0045 [111] is followed.

(c) When a discovery issue challenges the Probabilistic Safety Assessment, N-PROC-RA-0132 "Management of Incremental Risk from Abnormal Plant Configurations" [113] and N-PROC-RA-0131 "Probabilistic Risk Assessment Issues Database Management" [114] apply.

N-PROC-RA-0094 provides a high level guide to key actions required to ensure assessed public risk is maintained within acceptable limits, regulatory limits are respected, and key organizations are involved whenever a Deterministic Safety Analysis issue is discovered. It ensures issues are handled in a managed fashion commensurate with the associated risk.

N-PROC-MA-0008, "Work Initiation, Approval and Prioritization" [115] outlines the processes for identifying, approving, prioritizing, and processing nuclear production work needs, such as work required to address plant equipment deficiencies. Appendix B of N-PROC-MA-0008 provides a Prioritization Matrix that is used to define work priority dependent on the significance of the deficiency and the safety impact. Also in Appendix B, the identified priorities are directly linked to scheduling guidance, i.e., timescale to address the Work Order. Appendix C is used to assign "Importance to Plant" scores to Work Orders, further assisting with prioritization of work in accordance with plant impact, for which safety impact is a large factor. N-PROC-MA-0022, "Integrated On-line Work Schedule" [116] defines the process for scheduling, resourcing, coordinating and tracking on-line work. A high priority scheduling process is provided for urgent work needs with a high impact on the station. N-PROC-MA-0013, "Planned Outage Management" [117] establishes the process for preparation and execution of planned outage work, including schedule development and resource planning. This document confirms that shutdown safety, i.e., ensuring key safety systems and components provide an appropriate margin of plant safety while a unit is shutdown, is the top outage priority.

N-STD-MP-0023, "Technology and Research" [118] establishes the processes for effective management of Research and Development programs for Nuclear in support of safe, reliable and competitive performance of Nuclear Facilities. This standard establishes the scope, requirements and processes that govern consistent inputs, activities and outputs of all stakeholders to form an integrated managed system. Nuclear research and development programs are annually developed and implemented in response to issues that address the plant safety design basis and licensing basis, maintain or improve safe operating margins, minimize radiation and environmental impacts of Nuclear Facilities, and maximize effective utilization of Nuclear assets.

N-PROC-MP-0092, "Technology and Research Program Management" [119] describes the life cycle and associated stakeholder activities for the development and implementation of Research and Development programs for Nuclear, starting from issue identification through program development, execution, application, close-out, and evaluation.

Conclusion:

The conclusion of this Review Task assessment is that there are processes in place for prioritization of safety issues, with realistic objectives and timescales that ensure that these issues receive proper resources. The intent of Review Task #15 is met and therefore Pickering NGS is compliant.

4.2 L/R/C/S Reviews

As per Section 2.2 of this report, detailed reviews for ten L/R/C/Ss with content applicable to Safety Factor 10 are provided in References [6] and [7]. Associated findings applicable to Safety Factor 10 are summarized in Table 3 below.

Table 3: PSR2 L/R/C/S Review Results for Safety Factor 10

L/C/R/S Reviewed	PSR2 L/R/C/S Review Results for Safety Factor 10
CSA N286-12, "Management System Requirements for Nuclear Facilities"	There are no PSR2 gaps for CSA N286-12. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with N286-12.
CSA N286.7-16, "Quality Assurance Of Analytical, Scientific And Design Computer Programs"	There are no PSR2 gaps for CSA N286.7-16. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CSA N286.7-16.
CNSC RD-204 (2008), "Certification of Persons Working at Nuclear Power Plants"	There are no PSR2 gaps for CNSC RD-204 (2008). Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CNSC RD-204 (2008).
CNSC REGDOC-3.1.1 (2014), "Reporting Requirements for Nuclear Power Plants"	There are no PSR2 gaps for CNSC REGDOC-3.1.1 (2014). Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CNSC REGDOC-3.1.1 (2014).

L/C/R/S Reviewed	PSR2 L/R/C/S Review Results for Safety Factor 10
CNSC G-323 (2007), "Ensuring the Presence of Sufficiently Qualified Staff at Class I Nuclear Facilities-Minimum Shift Complement"	There are no PSR2 gaps for CNSC G-323 (2007). Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CNSC G-323 (2007).
S.C. 1997, C.9 (Amended in February 2015), "Nuclear Safety and Control Act (NSCA) and its associated Regulations"	There are no PSR2 gaps for the Nuclear Safety and Control Act (Amended 2015). Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with the NSCA (Amended 2015).
SOR/2000-202 (Amended in June 2015), "The General Nuclear Safety and Control Regulations"	There are no PSR2 gaps for the General Nuclear Safety and Control Regulations (Amended June 2015). Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with the General Nuclear Safety and Control Regulations (Amended June 2015).
CNSC REGDOC-2.2.2 (2014), "Personnel Training"	There are no PSR2 gaps for CNSC REGDOC-2.2.2 (2014). Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CNSC REGDOC-2.2.2 (2014).
CNSC REGDOC-2.3.2 (2015), "Accident Management, Version 2"	For Safety Factor 10, there are no PSR2 gaps for CNSC REGDOC-2.3.2 (2015).
CSA N286.7.1-09, "Guideline for the Application of N286.7-99, Quality Assurance of Analytical, Scientific, and Design Computer Programs for Nuclear Power Plants"	The N286.7.1 guide has been amalgamated into the new (-16) edition of the N286.7 Standard. The N286.7 CSA Impact Statement states [8]: "The CSA N286.7.1 guide will no longer be maintained after this new edition of N286.7 is issued. Any relevant guidance has been put into the new edition of N286.7." As a result, only the review of N286.7-16 has been prepared for PSR2.

4.3 OPG Program Effectiveness Reviews

The OPG Nuclear Programs reviewed for Safety Factor 10 are identified in Table 2, and details of the associated effectiveness reviews for each of the N-PROGs are provided in Appendix B.

4.4 Additional Review Findings

As discussed in Section 3.4, the PSR2 Safety Factor 10 assessment also included a review of commitments previously made to the CNSC, open CNSC action items, and exemptions granted by the CNSC, as identified in the R04 Pickering LCH [4], to determine if there are any impacts associated with operation of the Pickering Units past 2020. The review also included identification and review of previously identified programmatic Darlington PSR1 gaps related to Safety Factor 10 to determine impacts associated with operation of the Pickering Units past 2020. This assessment did not identify any gaps for Safety Factor 10.

Findings from the review of previously identified PSR1 gaps in the Pickering Units 5-8 Continued Operations Plan [12] are provided in a separate PSR2 COP Review Report. Findings from the review of Fukushima Action Items are provided in a separate PSR2 FAI Review Report. Results from the Continued Operations Plan and Fukushima Action Items reviews will be considered in the Global Assessment process.

There were no PSR2 gaps identified in this Safety Factor 10 Report that require discussion in other Safety Factor Reports.

5.0 RESULTS AND CONCLUSIONS

OPG Governance, Programs, Policies, Procedures, Instructions and Guidelines related to Safety Factor 10 were reviewed for the fifteen PSR2 Review Tasks in Section 4.1 of this report and resulted in no PSR2 gaps. L/R/C/S and OPG Nuclear Program effectiveness reviews for Safety Factor 10 were prepared per Sections 4.2 and 4.3, respectively, and resulted in no PSR2 gaps. Per Section 4.4, this report also included identification and review of previously identified programmatic Darlington PSR1 gaps related to Safety Factor 10 (to ascertain the implications of extending Pickering NGS operation beyond 2020), as well as a review of the R04 Pickering LCH [4] for any impacts of Pickering NGS operation beyond 2020 on: a) OPG commitments previously made to the CNSC, b) open CNSC action items, and c) exemptions granted by the CNSC (all related to Safety Factor 10), which resulted in no PSR2 gaps.

The review of Safety Factor 10 has confirmed that the Pickering NGS organization, management system and safety culture are adequate and effective for ensuring the safe operation of the plant.

6.0 REFERENCES

[1] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[2] CNSC REGDOC-2.3.3, *Periodic Safety Reviews*, April 2015.

[3] IAEA Safety Guide No. SSG-25, *Periodic Safety Review for Nuclear Power Plants*, 2013.

[4] CNSC Report, LCH-PNGS-R004, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[5] OPG Report, P-REP-03680-00003 R000, *Pickering NGS PSR2: Definition of Safety Factor Review Tasks*, May 2016.

[6] OPG Report, P-REP-03680-0586480 R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 9, 11, and 15*, September 2016.

[7] OPG Report, P-REP-03680-00021 R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 8, 10, 12, 13, and 14*, December 2016.

[8] CSA Impact Statement, *Notification of CSA N286.7 on Quality Assurance of Analytical Scientific, and Design Computer Programs; Product: New Edition; Product Designation: CSA N286.7; Previous Edition Published: 1999, Reaffirmed 2007 and 2012*, Date not provided.

[9] OPG Program, N-PROG-AS-0001 R017, *Managed Systems*, July 2015.

[10] OPG Program, N-PROG-AS-0002 R015, *Human Performance*, October 2014.

[11] OPG Program, N-PROG-RA-0010 R013, *Independent Assessment,* April 2014.

[12] OPG Plan, NK30-PLAN-00531-00001 R005, *Pickering 5-8 Continued Operations Plan*, December 2015.

[13] OPG Procedure, N-PROC-RA-0023 R018*, Fleetview Program Health and Performance Reporting*, August 2013.

[14] OPG Report, NK38-REP-03680-10185 R002, *Darlington NGS Integrated Implementation Plan (IIP)*, April 2015.

[15] OPG Policy, N-POL-0001 R003, *Nuclear Safety Policy*, March 2014.

[16] OPG Policy, OPG-POL-0032 R002, *Safe Operations Policy*, November 2013.

[17] OPG Letter, P-CORR-00531-03860, *Notice of Participation Pursuant to Rule 18 of CNSC Rules of Procedure – Pickering NGS Licence Renewal Application Hearing – February 20, 2013*, January 21, 2013.

[18] OPG Procedure, N-PROC-AS-0083 R000, *Nuclear Safety Culture Monitoring Panels*, December 2013.

[19] OPG Correspondence, P-CORR-08120-0453327, *Pickering Oversight Meeting Structure*, February 22, 2013.

[20] OPG Procedure, N-PROC-AS-0077 R007, *Nuclear Safety Culture Assessment*, October 2014.

[21] OPG Charter, N-CHAR-AS-0002 R018, *Nuclear Management System*, March 2015.

[22] OPG Standard, N-STD-AS-0020 R013, *Nuclear Management Systems Organizations*, February 2015.

[23] OPG Manual, N-MAN-08131-10000 (numerous sheets), OPG Station Organization Positions Role Documents, *** 137 documents ***

[24] OPG Program, N-PROG-OP-0001 R008, *Nuclear Operations*, November 2015.

[25] OPG Program, N-PROG-MA-0004 R011, *Conduct of Maintenance*, April 2015.

[26] OPG Program, N-PROG-MP-0007 R012, *Conduct of Engineering*, October 2012.

[27] OPG Training and Qualification Description, N-TQD-601-00001 R017, *Leadership and Management Training and Qualification Description*, May 2015.

[28] OPG Program, N-PROG-AS-0005 R005, *Business Planning*, June 2014.

[29] OPG Instruction, P-INS-09100-00003 R009, *Pickering Minimum Shift Complement*, December 2014.

[30] OPG Procedure, N-PROC-HR-0002 R004, *Limit of Hours of Work*, August 2012.

[31] OPG Program, N-PROG-TR-0005 R016, *Training*, January 2016.

[32] OPG Correspondence, P-CORR-00531-03719 R00, *Application For Renewal Of Pickering Nuclear Generating Station Power Reactor Operating Licence*, July 4, 2012.

[33] OPG Standard, N-STD-AS-0023 R008, *Nuclear Safety Oversight*, September 2015.

[34] OPG Program, N-PROG-RA-0003 R010, *Corrective Action*, January 2015.

[35] OPG Standard, N-STD-RA-0035 R004, *Nuclear Safety Review Board*, May 2015.

[36] CSA Standard, N286-12, *Management System Requirements for Nuclear Facilities*, June 2012.

[37] OPG Document, NK38-REF-09701-0561144, *Nuclear Executive Committee Terms of Reference*, October 2013.

[38] OPG Procedure, N-PROC-RA-0048 R017A, *Conducting Performance Based Audits and Assessments*, May 2015.

[39] CSA Standard, N286-05 Update No. 1, *Management System Requirements for Nuclear Power Plants*, November 2007.

[40] CNSC Nuclear Power Reactor Operating Licence, PROL 48.02/2018 (Amendment 2), *Nuclear Power Reactor Operating Licence - Pickering Nuclear Generating Station*, effective date December 18, 2015.

[41] CNSC Nuclear Power Reactor Operating Licence, PROL 13.00/2025, *Darlington Nuclear Generating Station Nuclear Power Reactor Operating Licence*, effective date January 1, 2016.

[42] OPG Letter, NK38-CORR-00531-17206 R000, B. Duncan to M. Leblanc, *Darlington NGS – Additional Information in Support of Application for Renewal of Darlington's Power Reactor Operating Licence (PROL) 13.01/2015*, January 30, 2015.

[43] OPG Letter, NK38-CORR-00531-16780 R000, B. Duncan to F. Rinfret, *Darlington NGS – Updated Application Requirements for Renewal of the Darlington Nuclear Generating Station Power Reactor Operating Licence – Transition Plans for New and Revised Standards and Regulatory Documents*, May 1, 2014.

[44] OPG Standard, N-STD-OP-0009 R009, *Reactivity Management*, April 2015.

[45] OPG Standard, N-STD-OP-0036 R009, *Operational Decision Making*, March 2015.

[46] OPG Instruction, N-INS-09030.2-10000 R001, *Event Free Challenge Process*, January 2014.

[47] OPG Standard, N-STD-MA-0016 R003, *Reactor Safety Support of Outages*, June 2015.

[48] CNSC REGDOC-3.1.1, *Reporting Requirements for Nuclear Power Plants*, May 2014.

[49] OPG Procedure, N-PROC-MP-0086 R004, *Safety Analysis Basis and Safety Report Updates*, December 2014.

[50] OPG Program, N-PROG-RA-0016 R008, *Risk and Reliability Program*, October 2015.

[51] OPG Procedure, N-PROC-MM-0021 R019, *Supply Inspection*, August 2014.

[52] OPG Procedure, N-PROC-MM-0010 R018, *Establishing and Maintaining Ontario Power Generation Approved Suppliers List*, October 2014.

[53] OPG Procedure, N-PROC-OP-0005 R012, *Pre-Job Briefing and Post-Job Debriefing*, June 2013.

[54] OPG Procedure, N-PROC-OP-0001 R007, *Conduct of Infrequently Performed Tests or Evolution*, June 2015.

[55] OPG Standard, N-STD-AS-0002 R015, *Procedure Use and Adherence*, August 2015.

[56] OPG Standard, N-STD-OP-0002 R003, *Communications*, May 2014.

[57] International Nuclear Safety Advisory Group Safety Report, Safety Series No. 75-INSAG-4, *Safety Culture*, Vienna, 1991.

[58] OPG Instruction, N-INS-08920-10017 R005, *Training Committees*, October 2015.

[59] OPG Instruction, N-INS-09030-10002 R008, *Site and Department Level Event Free Day Resets*, June 2015.

[60] OPG Guideline, N-GUID-03611-10005 R004, *Integrated Risk Management Guidelines*, March 2015.

[61] OPG Standard, N-STD-OP-0012 R004, *Conservative Decision-Making*, October 2012.

[62] OPG Procedure, N-PROC-AS-0078 R04, *Nuclear Performance Monitoring and Reporting*, May 2014.

[63] OPG Instruction, N-INS-08115-10000 R002, *Addition, Deletion and Revision of EPR Performance Measures*, June 2014.

[64] OPG Report, P-REP-03680-00012 R00, *Pickering NGS PSR2 Safety Factor 8 Report: Safety Performance*, December 2016.

[65] OPG Standard, N-STD-OP-0008 R007, *Expectations of Duty Managers*, November 2014.

[66] OPG Procedure, OPG-PROC-0166 R002, *Organization Design Change*, June 2015.

[67] OPG Procedure, OPG-PROC-0001 R009, *Process Administrative Governance Documents*, April 2015.

[68] OPG Program, N-PROG-MP-0005 R005, *Configuration Management*, June 2012.

[69] OPG Standard, N-STD-MP-0014 R004, *Managing Contracted Nuclear Safety Services*, November 2014.

[70] OPG Procedure, OPG-PROC-0160 R000, *Contractor Safety Management*, July 2015.

[71] OPG Standard, N-STD-AS-0030 R001, *Project Oversight Standard*, November 2015.

[72] OPG Program, N-PROG-MP-0001 R014, *Engineering Change Control*, January 2015.

[73] OPG Program, N-PROG-AS-0007 R009, *Project Management*, November 2015.

[74] OPG Standard, N-STD-AS-0032 R000, *Oversight of Supplemental Personnel*, August 2015.

[75] OPG Procedure, N-PROC-RA-0097 R008, *Self-Assessment and Benchmarking*, December 2014.

[76] OPG Internal Communication, B. McGee to Staff, *Nuclear Safety Culture Assessment – Results*, February 25, 2015.

[77] OPG Program, OPG-PROG-0001 R009, *Information Management*, September 2015.

[78] OPG Policy, OPG-POL-0033 R004, *OPG Business Model*, July 2015.

[79] OPG Procedure, OPG-PROC-0179 R000, *Nuclear Quality Assurance Records*, September 2015.

[80] OPG Procedure, OPG-PROC-0019 R006, *Records and Document Management*, May 2015.

[81] OPG Procedure, OPG-PROC-0178 R000, *Controlled Document Management*, September 2015.

[82] OPG List, N-LIST-01300-10000 R008, *Bounded Document Set*, November 2014.

[83] OPG Program, N-PROG-RA-0002 R008, *Conduct of Regulatory Affairs*, February 2015.

[84] OPG Procedure, N-PROC-RA-0005 R015, *Written Reporting to Regulatory Agencies*, January 2015.

[85] OPG Procedure, N-PROC-RA-0006 R006, *Regulatory Action Management*, July 2012.

[86] OPG Procedure, N-PROC-RA-0020 R018B, *Preliminary Event Notifications*, January 2015.

[87] OPG Procedure, N-PROC-RA-0028 R005, *Support of Canadian Nuclear Safety Commission Type I/II Inspections*, December 2014.

[88] OPG Procedure, N-PROC-RA-0047 R014*, Communications with the Canadian Nuclear Safety Commission*, April 2015.

[89] OPG Procedure, N-PROC-RA-0053 R006, *Evaluation of Proposed Changes for Canadian Nuclear Safety Commission Approval, Consent or Notification*, December 2015.

[90] OPG List, N-LIST-08130-10023 R003, *CSA N286-05 to OPGN Governance Cross Matrix*, October 2012.

[91] OPG List, N-LIST-08130-10025 R000, *CSA N286-12 to OPGN Governance Cross Matrix*, September 2015.

[92] OPG Program, N-PROG-RA-0013 R009, *Radiation Protection*, January 2015.

[93] OPG Program, N-PROG-MA-0019 R009, *Production Work Management*, December 2014.

[94] OPG Program, N-PROG-MP-0004 R016, *Pressure Boundary*, February 2016.

[95] OPG Program, OPG-PROG-0009 R002, *Items and Services Management*, March 2015.

[96] OPG Program, N-PROG-MA-0013 R009, *Welding*, May 2015.

[97] OPG Program, N-PROG-RA-0012 R011, *Fire Protection*, July 2015.

[98] OPG Program, N-PROG-MP-0006 R009, *Software*, April 2015.

[99] OPG Program, N-PROG-MP-0009 R011, *Design Management*, December 2014.

[100] OPG Program, N-PROG-MA-0026 R002, *Equipment Reliability*, May 2015.

[101] OPG Procedure, OPG-PROC-0058 R009, *Procurement Activities*, May 2015.

[102] OPG Procedure, N-PROC-MP-0098 R008, *Procurement Engineering Activities*, May 2016.

[103] OPG Procedure, OPG-PROC-0028 R005, *Crisis Management & Communications Centre (CMCC) Procedure*, February 2015.

[104] OPG Procedure, OPG-PROC-0112 R001, *Corporate Relations and Communications Emergency Preparedness and Response Procedure*, January 2016.

[105] OPG Standard, N-STD-OP-0004 R004, *Self-Check*, March 2014.

[106] OPG Standard, N-STD-OP-0005 R007, *Main Control Room Panel Monitoring, Operation and Alarm Response*, February 2014.

[107] OPG Guideline, N-GUID-01900-10000 R004, *Human Performance Event Free Tools for Knowledge Work*, November 2015.

[108] OPG Report, P-REP-09030-0540354 R000, *2015 Q1 Pickering Nuclear Safety Culture Monitoring Report*, April 2015.

[109] OPG Procedure, N-PROC-RA-0022 R032, *Processing Station Condition Records*, November 2014.

[110] OPG Standard, N-STD-RA-0008 R013, *Incident Investigation*, November 2014.

[111] OPG Procedure, N-PROC-MP-0045 R008, *Technical Operability Evaluation*, September 2015.

[112] OPG Procedure, N-PROC-RA-0094 R006, *Discovery Issue Resolution Process*, June 2015.

[113] OPG Procedure, N-PROC-RA-0132 R001, *Management of Incremental Risk from Abnormal Plant Configurations*, June 2016.

[114] OPG Procedure, N-PROC-RA-0131 R000, *Probabilistic Risk Assessment Issues Database Management*, August 2014.

[115] OPG Procedure, N-PROC-MA-0008 R021, *Work Initiation, Approval and Prioritization*, December 2015.

[116] OPG Procedure, N-PROC-MA-0022 R022, *Integrated On-Line Work Schedule*, February 2016.

[117] OPG Procedure, N-PROC-MA-0013 R016, *Planned Outage Management*, October 2015.

[118] OPG Standard, N-STD-MP-0023 R000, *Technology and Research*, May 2012.

[119] OPG Procedure, N-PROC-MP-0092 R002, *Technology and Research Program Management*, March 2014.

[120] OPG Report, P-REP-03680-00017 R00, *Pickering NGS PSR2 Safety Factor 13 Report: Emergency Planning*, December 2016.

Appendix A: Nomenclature

CNO	Chief Nuclear Officer
CNSC	Canadian Nuclear Safety Commission
COP	Continued Operations Plan
CSA	Canadian Standards Association
EPR	Electronic Performance Reporting
FAI	Fukushima Action Item
Hu	Human Performance
IAEA	International Atomic Energy Agency
ISR	Integrated Safety Review
LCH	Licence Conditions Handbook
NEC	Nuclear Executive Committee
NGS	Nuclear Generating Station
NIEP	Nuclear Industry Evaluation Program
NSRB	Nuclear Safety Review Board
OPG	Ontario Power Generation
PARTS	Pickering A Return to Service
PROL	Power Reactor Operating Licence
PSR	Periodic Safety Review
PSR1	Periodic Safety Review 1 (earlier OPG PSR work and other associated assessments)
PSR2	Periodic Safety Review 2 (subsequent PSR per CNSC REGDOC-2.3.3)
QA	Quality Assurance
SCA	Safety and Control Area
SCR	Station Condition Record
SSC	Structures, Systems and Components
WANO	World Association of Nuclear Operators

Appendix B: OPG Program Effectiveness Review Results

B.1 N-PROG-AS-0001, "Managed Systems"

The Managed Systems Program establishes a business framework that consists of "Plan", "Do", "Check", "Act/Adjust" elements, that are common to all Nuclear Management System programs and supporting activities. The elements collectively ensure:

- Management system principles of CSA N286, "Management System Requirements for Nuclear Facilities" are consistently and effectively applied to all activities defined in N-CHAR-AS-0002, "Nuclear Management System".
- Nuclear Management System processes and their supporting technologies are standardized to the greatest extent possible.

Nuclear Oversight conducted an audit in May 2012, NO-2012-017 [B.1.1], of the Managed Systems Program, for both Pickering and Darlington NGS. The objective of the audit was to confirm that the Managed System Program was effectively managed and in compliance with regulatory and OPG governance requirements. The audit concluded that the Managed Systems program controls are effective and no findings/SCRs were initiated as a result of the audit.

The Operations and Maintenance Support department completed a self-assessment in February 2014, NO14-000388-SA [B.1.2], in order to assess the health of the Managed Systems governance framework for both Pickering and Darlington NGS. This involved a review of related SCRs, governance framework, Asset Suite and revision records. No findings/SCRs were generated as a result of this self-assessment.

The Engineering Services division completed a self-assessment in October 2014, NO14-000820-SA [B.1.3], in order to confirm that the Pickering NGS Chemistry Laboratory operations comply with the corresponding Quality Management System (N-MAN-01802.1-10000, "Chemistry Laboratory Quality Manual"). Twelve findings were generated (ten of these were against an instrument in the process of being replaced). Corrective actions were captured under AR 28170848, which has since been closed and the necessary corrective actions were completed to address the underlying issues.

References

[B.1.1] Nuclear Oversight Audit, N-REP-01070-0409242 (NO-2012-017), *Audit OPGN NO-2012-017, Managed Systems*, May 28, 2012.

[B.1.2] Self-Assessment, NO14-000388-SA, *Program Assessment – N-PROC-AS-0001, Managed Systems,* February 13, 2014.

[B.1.3] Self-Assessment, NO14-000820-SA, *2014 PND Chemistry Quality Management System (QMS) Self-Assessment,* October 10, 2014.

B.2 N-PROG-AS-0002, “Human Performance”

The objective of the Human Performance (Hu) program is to continually reduce the frequency and severity of events through the systematic reduction of human error and the management of defences in pursuit of zero events of consequence. The Hu program:

- Establishes a systematic framework for Hu management across Ontario Power Generation (OPG) Nuclear;
- Achieves higher levels of nuclear and industrial safety, higher unit reliability and reduced operating costs through event-free operation; and
- Describes key accountabilities, core processes and related activities associated with the conduct of Hu management across all facets of Nuclear.

Nuclear Oversight conducted a performance based audit in December 2015, NO-2015-321 [B.2.1], for both Pickering and Darlington NGS. The objective of the audit was to follow-up the Human Performance Audit NO-2014-012 [B.2.2] and determine if the corrective actions identified at that time had been completed, implemented and sustained. The audit identified additional performance improvement opportunities applicable to Pickering NGS in the areas of Event Free Day Resets, Hu self-assessments and site Hu working and steering committee meeting frequency. SCR N-2015-29665 (AR# 28186262) was initiated to address these performance improvement opportunities with corrective actions. Actions related to the site Hu working and steering committee meeting frequency have been completed, while the remaining actions are targeted to be completed by Q4 2016.

The Governance and Services section completed a self-assessment in December 2015, BAS15-001733-SA [B.2.3], in order to assess compliance with the Human Performance Program, which is applicable to both Pickering and Darlington NGS. No findings/SCRs were initiated as result of this self-assessment.

The Operations and Maintenance Support department completed a self-assessment in December 2015, NO15-000996-SA [B.2.4], in order to confirm worker behaviors are managed through coaching and reinforcement as appropriate for both Pickering and Darlington NGS. The self-assessment concluded that there has been an increase in the supervisor field presence at both sites, however consistency in supervisory oversight and compliance with oversight controls was a finding. SCR P-2015-28988 was initiated to address this finding, which required corrective actions to be implemented. This SCR has since been closed and the necessary corrective actions were completed to address the underlying issues.

References

[B.2.1] Nuclear Oversight Audit, N-REP-01070-0573047 T06 (NO-2015-321), *Follow-up to Human Performance Audit NO-2014-012,* December 18, 2015.

[B.2.2] Nuclear Oversight Audit, N-REP-01070-0409278 T06 (NO-2014-012), *OPGN Human Performance Program Audit,* March 27, 2014.

[B.2.3] Self-Assessment, BAS15-001733-SA, *Program Assessment of N-PROG-AS-0002, Human Performance,* December 2015.

[B.2.4] Self-Assessment, NO15-000996-SA, *Confirm Worker Behaviours are Managed through Coaching and Reinforcement,* December 2015.

B.3 N-PROG-RA-0010, "Independent Assessment"

The Independent Assessment program provides the independent assessment (internal and external) processes to perform a comprehensive and critical evaluation of all activities affecting Ontario Power Generation (OPG) Nuclear. It ensures the Management System under N-CHAR-AS-0002, "Nuclear Management System", is reviewed with sufficient frequency to confirm its continuing effectiveness. The Independent Assessment program is comprised of the following processes:

- Internal independent assessment performed by Nuclear Oversight; and
- External independent assessment performed by the Nuclear Safety Review Board.

Processes conducted under this program are credited in N-STD-AS-0023, "Nuclear Safety Oversight", to assess and report on nuclear safety.

Nuclear Oversight completed a self-assessment in November 2012, NO12-000001-SA [B.3.1], in preparation for a Nuclear Industry Evaluation Program (NIEP) audit of OPG Nuclear's independent oversight functions in 2013. The scope of the self-assessment was to follow-up on findings from the 2010 NIEP Evaluation (NO-2010-069); and to assess if OPG's Nuclear Oversight, Supply Chain Quality Services and Nuclear Safety Review Board organizations are effective in performing independent oversight activities. Performance improvement opportunities were identified in the areas of audit deferrals, audit and assessment scoping, corrective action plan completion, and Nuclear Safety Review board membership.

Four SCRs were initiated to address the above findings (SCRs N-2012-05506, N-2012-05509, N-2012-05510 and N-2012-05513), which required corrective actions to be implemented. These SCRs have since been closed and the necessary corrective actions were completed to address the underlying issues.

The Operations and Maintenance Support department completed a self-assessment in April 2013, NO13-000232-SA [B.3.2], in order to assess the health of the Independent Assessment Program, which is applicable to both Pickering and Darlington NGS. This involved a review of related SCRs, governance framework, revision records and previous program assessment reports. No findings/SCRs were initiated as a result of this self-assessment.

A third party review (as part of the NIEP) of OPG Nuclear' s independent oversight functions for Nuclear Oversight, Supply Chain Quality Services and Off-site Independent Review Process/organizations was conducted in August 2013 [B.3.3]. NIEP reviews are peer evaluations conducted primarily by Nuclear Oversight personnel from other utilities. They are broad comprehensive reviews of the independent oversight function. The audit focused on the performance of Nuclear Oversight functions based on the requirements of CSA N286-05, "Management System Requirements for Nuclear Power Plants", OPG's Nuclear Management System (N-CHAR-AS-0002) as well as industry standards and commitments. The audit concluded that Nuclear Oversight, Supply Chain Quality Services and Off-site Independent Review Process/organizations are effective; however there was an opportunity to improve in the screening of issues identified by external regulatory bodies and the World Association of Nuclear

Operators (WANO), for missed opportunity evaluations as required by Section 10.4 of N-GUID-01070-10001, "Nuclear Oversight Performance Based Audit and Assessment Handbook".

One SCR was initiated to address the above finding (SCR N-2013-13588), which required corrective actions to be implemented. This SCR has since been closed and the necessary corrective actions were completed to address the underlying issues.

References

[B.3.1] Self-Assessment, NO12-000001-SA, *NIEP Evaluation Self-Assessment,* November 2, 2012.

[B.3.2] Self-Assessment, NO13-000232-SA, *Program Management Assessment – N-PROG-RA-0010, Independent Assessment,* April 11, 2013.

[B.3.3] Third Party Review, N-REP-01070-0435208 (NO-2013-029), *NIEP Evaluation – Third Party Review of Independent Oversight Functions,* October 17, 2013.

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ACCEPTED AS NOTED	
REVISE AND RESUBMIT	
Signature	Date 24 Oct-2016
Name: Mike Ruffolo, Manager	
Dept: Pickering Engineering - Aging Management & Strategic Initiatives	
OPG Proprietary	
Doc No.: P-REP-03680-00015	Rev: 000
This acceptance does not relieve the contractor from responsibility for errors or omissions or from any obligations or liability under this contract.	

amec foster wheeler

Pickering NGS PSR2 Safety Factor 11 Report: Procedures

PS112/RP/016 R02

October 20, 2016

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Revision Summary – For Amec Foster Wheeler Report PS112/RP/016

Rev	Date	Author	Comments
R00	July 29, 2016	R. Jayasundera, J. Morris, and J. Cheng	Initial issue for OPG review and comment.
R01	September 23, 2016	R. Jayasundera, J. Morris, and J. Cheng	Updated report addressing OPG comments on R00 report.
R02	October 20, 2016	R. Jayasundera, J. Morris, and J. Cheng	Updated report addressing OPG comments on R01 report.

EXECUTIVE SUMMARY

Ontario Power Generation (OPG) is performing a Periodic Safety Review (PSR) to support the possibility of continued operation of Pickering Nuclear Generating Station (NGS) beyond 2020. The PSR (referred to as "PSR2") is a subsequent PSR building on the review basis of earlier OPG Integrated Safety Reviews and other associated assessments. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

Part of PSR2 involves the preparation of Safety Factor reports for each of fifteen major topic areas. Safety Factor reports consist of:

- Assessing compliance against "Review Tasks" identified in the PSR2 Basis Document [1]. These Review Tasks are derived from Canadian Nuclear Safety Commission (CNSC) REGDOC-2.3.3, "Periodic Safety Reviews" [2] and International Atomic Energy Agency (IAEA) SSG-25, "Periodic Safety Review for Nuclear Power Plants" [3];

- Documenting assessments against applicable modern Laws, Regulations, Codes and Standards (L/R/C/Ss) as defined in Reference [1]; and

- Reviewing the effectiveness of applicable OPG Nuclear Programs that support the above assessments, through review of audit and self-assessment results.

The PSR2 review of Safety Factor 11, *Procedures* is presented in this report. OPG Governance, Programs, Policies, Procedures, Instructions and Guidelines related to Safety Factor 11 were reviewed for the thirteen PSR2 Review Tasks specified in Section 4.1 of this report. L/R/C/S and OPG Nuclear Program audit and self-assessment reviews for Safety Factor 11 were prepared per Sections 4.2 and 4.3, respectively. Per Section 4.4, the PSR2 assessment includes review of previously identified PSR1 gaps related to Safety Factor 11 (to ascertain the implications of extending Pickering NGS operation beyond 2020), as well as a review of the R04 Pickering Licence Conditions Handbook [4] for any impacts of Pickering NGS operation beyond 2020 on: a) OPG commitments previously made to the CNSC, b) open CNSC action items, and c) exemptions granted by the CNSC (all related to Safety Factor 11).

The results of the review of Safety Factor 11 are discussed in Section 5.0 of this report. The review has confirmed that the Pickering NGS processes for managing, implementing and adhering to operating and working procedures and for maintaining compliance with operational limits and conditions and regulatory requirements are adequate and effective and ensure plant safety. As discussed in Section 5.0, the review identified no Pickering PSR2 gaps.

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1.0 INTRODUCTION

Ontario Power Generation (OPG) is performing a Periodic Safety Review (PSR) to support the possibility of continued operation of Pickering Nuclear Generating Station (NGS) beyond 2020.[1] A comprehensive Integrated Safety Review (ISR) was completed for Pickering Units 5 through 8 in 2009 in support of refurbishment and continued operation. Pickering Units 1,4 integrated safety assessments were also performed for Pickering A Return to Service (PARTS) in support of approval to restart Units 1 and 4. In addition to these Pickering-specific studies, the 2013 Darlington ISR performed extensive code and standard reviews that were updated in relation to the versions that were assessed in the 2009 Pickering B ISR.[2] These previous ISRs are considered to constitute the first PSR completed for Pickering (referred to as "PSR1"). The current PSR (referred to as "PSR2") is a subsequent PSR building on the basis of earlier OPG integrated safety assessments through review of the various studies, assessments and licence renewals performed since PSR1. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

PSR2 will support and complement the licence renewal application for Pickering NGS going forward. Fifteen Safety Factors will be assessed as part of the PSR. The purpose of Safety Factor reviews is to confirm that the design, condition and operation of Pickering Units 1,4 and 5-8 (as well as common systems) will support continued safe operation for the period of PSR2 by:

- Assessing compliance against "Review Tasks" identified in the PSR2 Basis Document [1], which were derived from Canadian Nuclear Safety Commission (CNSC) REGDOC-2.3.3 [2] and International Atomic Energy Agency (IAEA) SSG-25 [3];
- Documenting assessments against applicable modern Laws, Regulations, Codes and Standards (L/R/C/Ss) (as defined in Reference [1]); and
- Reviewing the effectiveness of applicable OPG Nuclear Programs that support the above assessments, through review of audit and self-assessment results.

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1 Currently, Pickering Units 5-8 are approved to operate to 247,000 Effective Full Power Hours. This operation limit is expected to be reached on some units in 2020. For the purposes of PSR2, OPG assumes operation of Pickering NGS for up to eight additional years, from 2020 until 2028. OPG will make a decision regarding the permanent shut down dates for the six reactors following the performance of a technical evaluation that will include PSR2, and will communicate it to the CNSC as required by the current Power Reactor Operating Licence (PROL).

2 Much of the compliance assessment and evaluation of Safety Factor health for the Darlington ISR is based on programs and practices that apply across OPG's nuclear operations. As a result, where Pickering is confirmed to follow the same nuclear programs and practices as were assessed for Darlington, the Darlington ISR programmatic conclusions are applicable to Pickering. As discussed in Section 1.0, an effectiveness review (effectiveness at Pickering NGS) of OPG programs used to demonstrate compliance with the PSR2 Assessment Basis will be conducted using recent audit and self-assessment results.

As outlined in IAEA SSG-25 [3], the objective of the review of Procedures Safety Factor 11 is to: "determine whether the operating organization's processes for managing, implementing and adhering to operating and working procedures and for maintaining compliance with operational limits and conditions and regulatory requirements are adequate and effective and ensure plant safety." REGDOC-2.3.3 [2] requires that: "The licensee shall conduct a PSR in accordance with this regulatory document for the period until the next PSR or, if applicable, until the end of commercial operation of the plant."

This report documents the results of the review of Safety Factor 11 for Pickering PSR2. The report is based on the OPG Governance, Programs, data, and material available up to January 15, 2016 which is the freeze date for PSR2.

2.0 SCOPE OF REVIEW

2.1 Review Task Assessments

The Pickering PSR2 Safety Factor 11 Review Tasks are defined in Reference [1]. Details of the derivation of these Review Tasks from CNSC REGDOC-2.3.3 [2] and IAEA SSG-25 [3] are shown in Reference [5]. As noted in the Pickering PSR2 Basis Document [1], the review of this Safety Factor will focus on those procedures that have the highest safety significance. The Safety Factor 11 Review Tasks are:

1) Determine if there is a process for the development, approval, and documenting of all safety related procedures.

2) Confirm there is a formal process for modifying procedures affecting safety, including adequate arrangements for tracking changes.

3) Confirm there is a program for assessing procedures and performance audits to determine if there is regular review and maintenance of these procedures.

4) Confirm that self-assessments are performed to ensure that the procedures are followed.

5) Establish that there is a means for assessing the adequacy of safety related procedures in comparison with industry good practices.

6) Confirm that there are operating procedures that apply comprehensively to normal, abnormal and emergency conditions (including anticipated operational occurrences, design basis accident conditions, post-accident conditions, and design extension conditions).

7) Confirm there is a means for assuring the clarity of procedures taking into account human factors.

8) Evaluate processes to update procedures to allow for changes in the assumptions made and/or the limits and conditions arising from the safety analysis, plant design and operating experience.

9) Verify that the analysis and justification of the accident management procedures are documented.

10) Verify that an appropriate process is in place for the categorization of procedures in accordance with their significance to safety.

11) Examine whether there is adequate involvement in the development of procedures by the staff who will use them.

12) Evaluate the distribution process for the control, copying and removal of obsolete versions of procedures, so that only the last approved edition is used.

13) Evaluate audits, self-assessments, safety performance and events to determine whether there is adequate understanding and acceptance of these procedures by managers and staff.

The methodology for the reviews is discussed in Section 3.1. Review Task findings are summarized in Section 4.1 of this report.

2.2 L/R/C/S Reviews

The applicable Laws, Regulations, Codes and Standards relevant to the Procedures Safety Factor are identified in Reference [1] and are listed in Table 1 below. Table 1 also identifies the modern version and date of each L/R/C/S to be considered, the Safety Factor(s) to which each document is applicable, and the type of review that will be completed in PSR2.

All of the Safety Factor 11 L/R/C/S reviews are incremental in nature. The definition of an Incremental Review is as follows:

- Incremental Review: For L/R/C/Ss that have been reviewed in PSR1 but have had revisions since the last review, a topical review will be performed of the changes.

The methodology for the reviews is discussed in Section 3.2. A detailed compliance assessment for each L/R/C/S is provided in Reference [6]. Associated findings are summarized in Section 4.2 of this report.

Table 1: L/R/C/Ss Reviewed for Procedures Safety Factor 11

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
L/R/C/Ss Referenced in Pickering NGS PROL 48.02/2018						
1	CSA N286	Management System Requirements for Nuclear Facilities	N286-12	5, 6, 9, 10, 11	Incremental	N286 addressed as part of Pickering B and Darlington ISRs.

2.3 Audit and Self-Assessment Reviews of OPG Programs

The OPG Nuclear Program (N-PROG) reviewed for Safety Factor 11 is listed in Table 2 below.[3] The methodology for the audit and self-assessment reviews is discussed in Section 3.3. The assessment results of the N-PROG in Table 2 are provided in Appendix B, and findings are summarized in Section 4.3.

Table 2: OPG Program Reviewed for Safety Factor 11

Document Number	Document Title
N-PROG-OP-0001 [7]	Nuclear Operations

2.4 Additional Reviews

The PSR2 Safety Factor 11 Report includes a review of the R04 Pickering Licence Conditions Handbook (LCH) [4] for any impacts of Pickering NGS operation beyond 2020 on the following (all related to Safety Factor 11):

- OPG commitments previously made to the CNSC;
- Open CNSC action items; and
- Exemptions granted by the CNSC.

The PSR2 assessment includes identification and review of previously identified PSR1 gaps related to Safety Factor 11 to ascertain the implications of extending Pickering NGS operation beyond 2020. The methodology for these reviews is described in Section 3.4. Any PSR2 gaps identified as a result of programmatic Darlington PSR1 gaps related to Safety Factor 11 are discussed in Section 4.4 of this report. The review of Pickering PSR1 gaps previously identified in the Pickering Units 5-8 Continued Operations Plan [8] is provided in Reference [9].

In addition, Fukushima Action Items were reviewed to identify implications of extending operation beyond 2020 (if any). This review is presented in Reference [10].

Any PSR2 gaps identified as a result of the Safety Factor 11 review which need to be addressed in other Safety Factor Reports are discussed in Section 4.4 of this report.

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[3] The list of Nuclear Programs to be assessed for PSR2 through review of audit and self-assessment results was derived from review of current OPG Governance. Although there may be content in Nuclear Programs that is applicable to multiple Safety Factors, N-PROG reviews are only provided in one Safety Factor report and are not duplicated.

3.0 METHODOLOGY

The sub-sections below summarize the methodology used to assess Review Task and L/R/C/S compliance, and Nuclear Program effectiveness for the Procedures Safety Factor.

3.1 Review Tasks

As discussed earlier, the Safety Factor Review Tasks are derived from CNSC REGDOC-2.3.3 [2] and IAEA SSG-25 [3], taking into consideration the Review Tasks used in the Pickering B and Darlington ISRs (as derived in [5]).

For each Safety Factor 11 Review Task identified in Section 2.1, a confirmation of the existence of applicable OPG Governance, Programs, Policies and Procedures (as well as Instructions and Guidelines, as applicable) was performed. Compliance against Review Tasks is also assessed by reference to applicable Condition Assessments, safety analyses and operating experience, as required.

The Review Task assessments identify Compliances and Gaps as defined below:

- Compliance: Compliance indicates that either the safety requirement or the intent of the Review Task is met.

- Gap: A Gap indicates that the intent of the Review Task is not met.

3.2 L/R/C/S Reviews

The process to identify the modern L/R/C/Ss that are applicable to the PSR2 Assessment Basis involved first creating a broad list from multiple sources (potential candidate L/R/C/Ss) and then filtering it to identify those that are most significant and that are applicable to the PSR2 scope. The identification and selection criteria are detailed in Reference [1]. The result of the identification and selection process was a set of modern L/R/C/Ss that became part of the "PSR2 Assessment Basis".

PSR2 is focused on the extension of Pickering NGS operations beyond 2020, and will conduct reviews against a baseline of past PSR1 work. As a subsequent PSR, PSR2 focuses on changes in requirements, plant conditions, operating experience and new information. Since PSR2 is an update of previous ISRs, it incorporates reviews of L/R/C/Ss that have occurred as new versions have been issued. Since this assessment is a subsequent PSR, the focus is on identifying differences between what was previously assessed and what is now different within the current Pickering PSR2 Assessment Basis. In general, these differences relate to:

- More recent (new or revised) L/R/C/S versions than what was previously assessed as part of PSR1;[4]

- Safety significant differences between Pickering and Darlington, if the Darlington ISR is the basis for the earlier assessment;

- Implications of extending Pickering NGS operation beyond 2020; and

- Safety significant differences between Pickering Units 1,4 and Units 5-8.

As described in Reference [1], L/R/C/S review types are clause-by-clause, high level or incremental. Most of the L/R/C/Ss in the PSR2 Assessment Basis receive incremental reviews since PSR2 is an update of previous PSR1 assessments and clause-by-clause or high level reviews for the majority of the L/R/C/Ss in the PSR2 Assessment Basis have already been completed. Implementation plans (including gap analyses or code-over-code reviews) also exist for the latest editions of many L/R/C/Ss. As a result, incremental review is also used in circumstances where a L/R/C/S in the PSR2 Assessment Basis was not assessed in previous PSR1 reviews but an implementation plan currently exists for compliance.

The PSR2 incremental reviews in this report include an assessment of the intent of recent changes to the L/R/C/Ss on a topic or subject-matter basis where there is potential to impact nuclear safety. Incremental reviews provide:

- A summary of the purpose of the L/R/C/S;

- Pertinent background information about the current revision of the L/R/C/S that is being considered;

- Identification of which Safety Factor(s) are applicable to the current revision of the L/R/C/S;

- A description of which version(s) of the L/R/C/S were assessed for PSR1 (i.e., Darlington ISR (for programmatic content), Pickering B ISR and PARTS code reviews);

- Identification of whether the current version of the L/R/C/S is an update of a previous version of the L/R/C/S that was assessed in PSR1 (and if so, a description of the major changes in the latest revision is provided as discussed below);

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[4] "New" refers to a code or standard that was not previously considered in the context of earlier assessments. "Revised" refers to an updated version of a code or standard that was previously considered in the context of earlier assessments. Where a document has a new number/type, but addresses the same topic from the same organization, it is a "revised", not "new", document (e.g., if a REGDOC replaces a CNSC G or RD document).

- An assessment of the applicability of PSR1 assessment findings (gaps and conclusions), including the implications of extending Pickering NGS operation beyond 2020 if any;

- An assessment of the applicability of assessment findings that address more recent (post-PSR1) editions of the L/R/C/S, including any implementation or transition plans that are already committed to by OPG; and

- Where PSR1 and post-PSR1 assessments are not sufficient to address changes in the latest edition of the L/R/C/S, an assessment of the changes from the previously assessed edition of the L/R/C/S (including identification of any safety significant PSR2 gaps which result).

High Level reviews provide the same information as above, where applicable, in a similar format. However, given that High Level L/R/C/Ss generally have not received past assessment during PSR1, the Incremental review content is augmented by a high level, section-by-section assessment of the degree of conformance of Pickering NGS with the L/R/C/S (demonstrating, with supporting evidence, whether the intent of the requirements stipulated in the document are met).

There are currently no L/R/C/S clause-by-clause reviews identified in the PSR2 Assessment Basis.

The Safety Factor 11 L/R/C/S reviews identify Compliances and Gaps as defined below:[5]

- Compliance:
    - Where an Incremental review has been performed, Compliance indicates that the change in the safety requirement, per the topical review, is met.
    - Where a High Level review has been performed, Compliance indicates that the intent of the safety requirement is met. (Note: No High Level reviews were performed as part of Safety Factor 11.)

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[5] Safety Factor compliance assessments for Review Tasks and L/R/C/S reviews make use of: a) OPG Governance, Programs, Policies and Procedures which support the compliance arguments, b) Commitments previously made to the CNSC, open CNSC action items, and exemptions granted by the CNSC (all related to the Safety Factor under review), as identified in the R04 Pickering LCH [4], c) Identification of previously identified Pickering-specific or programmatic PSR1 gaps related to the Safety Factor under review and the status of OPG's improvement plan(s) or other dispositions to address these, and d) Assessments and reviews performed since the PSR1 documents were completed.

- o For Clause-by-Clause reviews of modern L/R/C/Ss, Compliance indicates that the safety requirement is met. (Note: No Clause-by-Clause reviews were performed as part of Safety Factor 11.)

- Gap:

  - o Where an Incremental review has been performed, a Gap indicates that the change in the safety requirement, per the topical review, is not met.

  - o Where a High Level review has been performed, a Gap indicates that the intent of the safety requirement is not met. (Note: No High Level reviews were performed as part of Safety Factor 11.)

  - o For Clause-by-Clause reviews of modern L/R/C/Ss, a Gap indicates that the safety requirement is not met. (Note: No Clause-by-Clause reviews were performed as part of Safety Factor 11.)

The reviews assume that use of the word:

- “Shall" is used in an L/R/C/S to express a requirement, i.e., a provision that the licensee is obliged to satisfy in order to comply with the standard;

- "Should" is used to express a recommendation or that which is advised but not required;

- "May" is used to express an option or that which is permissible within the limits of the standard; and

- "Can" is used to express possibility or capability.

3.3 Audit and Self-Assessment Reviews

As discussed earlier, effectiveness reviews (effectiveness at Pickering NGS) of OPG programs used to demonstrate compliance with the PSR2 Assessment Basis were conducted, using recent applicable audit and self-assessment results:

- OPG Nuclear Oversight independent performance-based Program audits (typically performed in 1 to 5 year cycles) and self-assessments. This includes review of associated Station Condition Records and Action Requests to determine the status of any resulting corrective actions; and

- CNSC “Type I” and “Type II” inspections of the effectiveness and performance of OPG programs, where called-up by OPG audits or self-assessments.

There are many audits and self-assessments that are performed to assess the effectiveness of important aspects of each program. A sample of audits and self-assessments has been summarized for each program in order to demonstrate that program effectiveness is being assessed on an ongoing basis. The focus of these reviews was on effectiveness of the programs at Pickering NGS, where specific information is available. Results from these audits and self-assessments will be considered in the Global Assessment process. It is noted that audits and self-assessments are, by their nature, self-critical and are used to drive excellence in performance.

The list of Nuclear Programs to be assessed for each Safety Factor was derived from review of current OPG Governance, and has used the most recent version of these documents as of the PSR2 freeze date of January 15, 2016.

3.4 Additional Reviews

A review of the R04 Pickering LCH [4] was performed to determine if there are any impacts associated with operation of the Pickering Units past 2020 on the following (all related to Safety Factor 11):

- Commitments previously made to the CNSC;
- Open CNSC action items; and
- Exemptions granted by the CNSC.

The PSR2 assessment includes identification and review of previously identified Pickering-specific or programmatic PSR1 gaps related to Safety Factor 11 (as identified in the Darlington ISR Integrated Implementation Plan [11] and Pickering Units 5-8 Continued Operations Plan [8]) to ascertain the status of OPG's improvement plan(s) or other dispositions to address these and the implications of extending operation beyond 2020 (if any).[6]

Fukushima Action Items were reviewed to identify implications of extending operation beyond 2020 (if any). The methodology for this review is provided in Reference [10].

Any PSR2 gaps identified as a result of the Safety Factor 11 review which need to be addressed in other Safety Factor Reports are also discussed.

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[6] PSR2 includes consideration and confirmation that the findings of PSR1 remain valid, as applicable, for the operation period. This includes assessment of PSR1 conclusions against implications resulting from extended operation. In particular, Pickering PSR1 results are applicable to PSR2 if there was a PSR1 gap that is still open, or if a closed PSR1 gap could be affected by extended operation. If so these gaps are carried forward into PSR2 for consideration in the Global Assessment. (When references to PSR1 are made, the source document is identified and the relevant text from that source document is summarized in the context of PSR2.) With respect to the Darlington ISR, much of the evaluation of Safety Factor health is based on programs and practices that apply across OPG's nuclear operations. As a result, Darlington ISR programmatic conclusions are applicable to Pickering PSR2 where Pickering is confirmed to follow the same nuclear programs and practices that were assessed for Darlington. Darlington PSR1 results are applicable to Pickering PSR2 if there are Darlington PSR1 gaps that are found to be relevant to Pickering PSR2.

4.0 REVIEW FINDINGS

4.1 Review Tasks

The sub-sections below provide an assessment of the adequacy of applicable OPG Governance, Programs, Policies and Procedures (as well as Instructions and Guidelines, as applicable) in demonstrating compliance against the Safety Factor 11 Review Tasks.

4.1.1 Review Task #1: Process for Developing Procedures

Determine if there is a process for the development, approval, and documenting of all safety related procedures.

N-PROG-OP-0001, "Nuclear Operations" [7], establishes safe, uniform, and efficient operating practices and processes within Nuclear facilities that provide nuclear professionals the ability to ensure facilities are operated in such a manner that Reactor Operating License, Operating Policies and Principles, and other applicable regulations and standards are followed. This program applies to all technical procedures which direct the operation, maintenance, or testing of plant structures, systems, or components. This program is applicable to nuclear personnel who are involved in the development, review, validation, verification, approval, implementation, and evaluation of technical procedures.

Technical procedures ensure that the nuclear generating facilities are operated safely, reliably, and in accordance with their respective design and licensing basis. They provide information and instructions for the operation, maintenance and testing of a system, structure, or component. Technical procedures include the following [12]:

- Abnormal Incident Manuals (AIMs);
- Alarm Response Manuals (ARMs);
- Chemistry Control Procedures;
- Chemistry Laboratory Procedures;
- Common Technical Procedures;
- Forms;
- Maintenance Procedures;
- Operating Manuals (OMs);
- Operating Memos (OPMs);
- Operating Procedures;

- Overall Unit Manuals; and
- Safety-Related System Tests (SRSTs).

N-PROC-AS-0028, "Development, Review, and Approval of Technical Procedures" [13], defines the process and establishes requirements for the development, review, validation, verification, and approval of technical procedures.

Procedures are developed in accordance with N-STD-AS-0014, "Requirements for Technical Procedures" [12]. This standard specifies requirements for the structure, minimum content, and format of technical procedures. Compliance with this standard ensures that technical procedures developed and used throughout OPG Nuclear facilities:

- Promote safe and efficient operation;
- Reflect industry best practice; and
- Document compliance with regulatory requirements.

In addition, compliance with N-STD-AS-0014 ensures that procedures are prepared using approved instructions and templates.

Per Section 1.14 of N-PROC-AS-0028, once a procedure has been approved the document owner (or delegate) ensures that the final approved technical procedure is processed through Business Services for issuing, distribution, and retention [13].

Governance documents such as administrative procedures that are related to safety are prepared, reviewed, and approved in accordance with OPG-PROC-0001, "Process Administrative Governance Documents" [14]. These documents generally have a lower safety significance in comparison to technical procedures. Thus, in accordance with Section 2.1, which states that the review of this Safety Factor will focus on those procedures that have the highest safety significance, the supporting discussion in subsequent Review Tasks will be primarily focused on technical procedures.

Conclusion:

The conclusion of this Review Task assessment is that there is governance in place for the development, approval, and documentation of all safety related procedures. The intent of Review Task #1 is met and therefore Pickering NGS is compliant.

4.1.2 Review Task #2: Process for Modifying Procedures

Confirm there is a formal process for modifying procedures affecting safety, including adequate arrangements for tracking changes.

The processes identified in Section 4.1.1, Review Task #1, for the development, approval, and documentation of procedures are also applicable to the revision of existing procedures.

N-PROC-AS-0028 defines the process and establishes requirements for the development, review, validation, verification, and approval of technical procedures [13]. Per Section 1.1 of N-PROC-AS-0028 [13], requests for a revision to an existing technical procedure are initiated through the completion of N-FORM-10014, "Technical Procedure Action Request (TPAR)" [15]. Once the TPAR has been approved, the revision of the procedure, including the necessary verification, validation, review, and approval activities, is completed per N-PROC-AS-0028. All technical procedures are modified in accordance with this procedure, irrespective of their safety significance.

Modifications to procedures are made in accordance with N-STD-AS-0014. Per Section 1.3.3 of N-STD-AS-0014, revisions bars are used to mark new or modified material in a procedure unless the procedure has been modified extensively [12]. Per Section 1.2.10 of N-STD-AS-0014, a summary of changes made or a revision history is included in the revised procedure [12]. If a procedure has been modified extensively, a statement to this effect is included in the summary of changes and revision bars are not used to mark new or modified material.

Per Section 1.14 of N-PROC-AS-0028, once a revised procedure has been approved the document owner (or delegate) ensures that the final approved technical procedure is processed through Business Services for issuing, distribution, and retention [13].

Document Change Requests (DCRs) are used to administer and monitor changes to governance documents that are related to safety. DCRs are processed in accordance with OPG-INS-00700-0001, "Document Change Request Data Administration" [16].

Conclusion:

The conclusion of this Review Task assessment is that there is governance in place for modifying procedures affecting safety, including adequate arrangements for tracking changes. The intent of Review Task #2 is met and therefore Pickering NGS is compliant.

4.1.3 Review Task #3: Regular Review of Procedures

Confirm there is a program for assessing procedures and performance audits to determine if there is regular review and maintenance of these procedures.

Assessing the adequacy of procedures and the initiation of required improvements or corrective actions is part of ongoing efforts to achieve higher levels of nuclear and industrial safety, higher unit reliability, and reduced operating costs through event-free operation. N-CHAR-AS-0002, "Nuclear Management System" [17] establishes the overall requirements for sustaining and improving performance. This is accomplished by the following [17]:

- Establishing and implementing a managed system consisting of governing documents communicating essential elements of Nuclear business;
- Reinforcing individual accountability for performance and implementing various self-verification and independent oversight techniques;
- Identifying, documenting, evaluating, and correcting in a timely manner, conditions adverse to quality;
- Using internal and industry Operating Experience (OPEX) to improve human, plant, and equipment performance and design, procurement, construction, commissioning, and operating requirements and practices; and
- Providing information to the people who need it through the managed systems that establish how necessary information is identified, targeted to required users, maintained current, and communicated effectively.

N-PROG-AS-0002, "Human Performance" [18], establishes a systematic framework for Human Performance management across OPG Nuclear. In accordance with N-PROG-AS-0002, the adequacy of a procedure is assessed each time it is used. The following documents provide further direction on the relevant processes used to evaluate the adequacy of procedures:

- N-PROC-OP-0005, "Pre-Job Briefing and Post-Job Debriefing" [19]; and
- N-STD-AS-0002, "Procedure Use and Adherence" [20].

Prior to starting a task, relevant procedures are identified in accordance with N-STD-AS-0002 [20]. In addition, N-PROC-OP-0005 specifies that a pre-job brief must be delivered prior to performing any task [19]. At a minimum, the pre-job brief will summarize and communicate the task that is to be performed and identify any critical steps associated with that task.

Section 1.4 of N-STD-AS-0002 describes the required actions on how to address challenges encountered regarding procedure adherence [20]. Specifically, if the procedure is unclear, cannot be performed as written, would result in an unsafe condition or result in an unexpected response, or is otherwise inappropriate, the user will stop the activity and contact their supervisor for direction. If the condition that is preventing adherence to the procedure cannot be corrected, then the supervisor will initiate a TPAR in accordance with N-PROC-AS-0028 [13].

Following the completion of the task, a post-job debriefing is held in accordance with N-PROC-OP-0005 [19]. If there are lessons learned or successes need to be recorded, they are documented using the appropriate work process (i.e., a TPAR would be initiated in accordance with N-PROC-AS-0028 to incorporate any procedural improvements/enhancements that were identified during the course of performing the task).

The ongoing assessment of procedures each time they are used is complemented by a range of other processes that ensure there is regular review and maintenance of procedures. These processes are as follows:

- N-PROC-RA-0023, "Fleetwide Program Health and Performance Reporting" [21];
- N-PROC-RA-0048, "Conducting Performance Based Audits and Assessments" [22]; and
- N-PROC-RA-0097, "Self-Assessment and Benchmarking" [23].

N-PROC-RA-0023 [21] describes the process for performing a program health and performance review and for reporting on the overall effectiveness of the Nuclear Management System governance to support the requirements of CSA N286-05, "Management System for Nuclear Power Plants" [24]. Every program implemented either directly or indirectly by N-CHAR-AS-0002 [17] must be reviewed in accordance with N-PROC-RA-0023 [21]. The assessment of the overall health of technical procedures is addressed under the evaluation of N-PROG-OP-0001 [7]. Program health reports for N-PROG-OP-0001 take into consideration findings resulting from audits and self-assessments.

N-PROC-RA-0048 establishes the methodology and requirements for planning, scheduling, staffing, preparing, performing, reporting, and follow-up of audits and assessments performed by OPG Nuclear Oversight [22]. The objective of the performance-based audit and assessment process is to confirm that the overall Nuclear Management System is established and effective. The audit frequency is determined using a risk-based approach, which utilizes industry and OPG Nuclear experience on inherent risk and residual risk, to prioritize programs that constitute the Nuclear Management System. Per Section 1.3.4 of N-PROC-RA-0048 [22], adverse findings identified during audits as requiring attention are documented in a Station Condition Record (SCR), in accordance with N-PROC-RA-0022, "Processing Station Condition Records" [27].

N-PROC-RA-0097 specifies the requirements for self-assessment and benchmarking activities for functional and line organizations of OPG Nuclear [23]. Self-assessment and benchmarking are performance improvement tools and include evaluations of business programs, processes, and performance. They provide a structured method to compare performance with management expectations, industry standards of excellence, and regulatory requirements to identify areas needing improvement. Self-assessments include evaluations of business programs, processes, and performance. Per Section 1.6 of N-PROC-RA-0097, the results of the self-assessment are documented in a Self-Assessment Report, with Action Requests (ARs) generated for improvement opportunities identified and SCRs filed for adverse conditions identified [23].

N-PROG-RA-0010, "Independent Assessment" [25], ensures that audits and self-assessments are carried out at regular intervals as specified by their specific implementation procedures. The purpose of N-PROG-RA-0010 is to ensure that the management system under N-CHAR-AS-0002, including the associated implementation programs that receive their authority from N-CHAR-AS-0002, is reviewed with sufficient frequency to confirm its continuing effectiveness [25]. Per the Pickering PSR2 Basis Document [1], effectiveness reviews (at Pickering NGS) of OPG programs used to demonstrate compliance with the PSR2 Assessment Basis are conducted for PSR2, using recent audit and self-assessment results. Audit and self-assessment results applicable to N-PROG-RA-0010 are summarized in Section 4.3 and Appendix B of P-REP-03680-00014, "Pickering NGS PSR2 Safety Factor 10 Report: Organization, Management System, and Safety Culture" [26].

Conclusion:

The conclusion of this Review Task assessment is that there is governance for assessing procedures and identifying requirements for performance audits to determine if there is regular review and maintenance of these procedures. The intent of Review Task #3 is met and therefore Pickering NGS is compliant.

4.1.4 Review Task #4: Procedural Adherence

Confirm that self-assessments are performed to ensure that the procedures are followed.

The extent to which procedures are to be followed is described in N-STD-AS-0002, which documents the requirements for use of all approved procedures [20]. This standard is applicable to all governance, administrative procedures, and technical procedures.

The level of use for a procedure is determined by considering the risk associated with the task and is specified as Usage Classification that is assigned to the procedure. The three classifications of usage for procedures are continuous, reference, and information [20].

N-STD-AS-0002 documents the minimum requirements for usage of each class of procedures [20]. Per Section 2.0 of N-STD-AS-0002, line supervisors are responsible for applying appropriate oversight of the task to verify procedural adherence, with management having overall responsibility to provide procedures of sufficient quality to accomplish intended tasks and verification of adherence to N-STD-AS-0002 [20].

Compliance with procedural adherence requirements is evaluated in part through the use of self-assessment and benchmarking activities. N-PROC-RA-0097 defines the elements required to plan, execute, report, and monitor self-assessments and benchmarking activities for the functional and line organizations of OPG Nuclear [23]. Self-assessment and benchmarking are performance improvement tools that provide a structured method to compare performance with management expectations, industry standards of excellence, and regulatory requirements to identify areas needing improvement. Self-assessments include evaluations of business programs, processes, and performance. There are three types of self-assessments [23]:

1) Divisional Self-Assessment

    - Requires significant advanced planning, involving a team with membership external to the area being assessed, and typically has a duration of 1-2 weeks.

2) Departmental Self-Assessment

    - Objective and scope is generally within the purview of a department and typically has a duration of 2-7 days.

3) Snapshot Self-Assessment

    - Originates at the department level or lower and is normally performed by one person (i.e., person knowledgeable in the subject area). It has a narrow focused objective and scope and is short in duration.

Per Section 1.4 of N-PROC-RA-0097, Directors and Stratum IV Managers plan and schedule divisional and departmental level self-assessments for the following year by December 15th of the preceding year [23]. Planned self-assessments are reviewed in order to co-ordinate fleetwide self-assessment activities and confirm coordination and scheduling at the site level. Directors and Stratum IV Managers are expected to review and revise the self-assessment schedule periodically due to developing issues or changing business priorities.

N-PROG-RA-0010 ensures that audits and self-assessments are carried out at regular intervals as specified by their specific implementation procedures [25]. Per the Pickering PSR2 Basis Document [1], effectiveness reviews (at Pickering NGS) of OPG programs used to demonstrate compliance with the PSR2 Assessment Basis are conducted for PSR2, using recent audit and self-assessment results. Audit and self-assessment results applicable to N-PROG-RA-0010 are summarized in Section 4.3 and Appendix B of P-REP-03680-00014 [26].

Conclusion:

The conclusion of this Review Task assessment is that there is adequate governance outlining the requirements for performing self-assessments to ensure that procedures are followed. The intent of Review Task #4 is met and therefore Pickering NGS is compliant.

4.1.5 Review Task #5: Procedures Follow Industry Good Practices

Establish that there is a means for assessing the adequacy of safety related procedures in comparison with industry good practices.

As discussed in Review Tasks #1 and #2, Sections 4.1.1 and 4.1.2 respectively, safety-related procedures are prepared and revised following the process documented in N-PROC-AS-0028 [13].

Section 1.4.1 of N-PROC-AS-0028 specifies that for new technical procedures and major revisions to existing technical procedures, the preparation stage of the process must involve a review of OPEX lessons learned, event information, and just-in-time packages to incorporate applicable information [13]. This ensures that industry best practices are incorporated in the procedure development process. In addition, all new procedures and major revisions to existing procedures are validated either in the simulator or in the field prior to use per N-PROC-AS-0028 [13]. The completion of applicable validation activities is consistent with industry best practice.

Per Section 1.4 of N-PROC-AS-0028 [13], technical procedures are prepared or revised by applying the structure and minimum content requirements established in N-STD-AS-0014 [12]. This standard specifies requirements for the structure, minimum content, and format of technical procedures. Compliance with this standard ensures that technical procedures developed and used throughout OPG Nuclear facilities [12]:

- Promote safe and efficient operation;
- Reflect industry best practice; and
- Document compliance with regulatory requirements.

Section 4.3.2 of N-STD-AS-0014 identifies several developmental references that were used to support the development of this standard, including the following documents [12]:

- American Institute of Chemical Engineers: Guidelines for Writing Effective Operating and Maintenance Procedures, New York, Center for Chemical Process Safety, 1996;
- Blake G, Bly RW: The Elements of Technical Writing, New York: Macmillan USA, 1993;

- Campbell JJ, Zimmerman C: Fundamentals of Procedure Writing, Columbia: GP Publishing, Inc., 1988;

- Haramundanis K: The Art of Technical Documentation, Boston: Digital Press, 1998; and

- Wieringa D, Moore C, Barnes V: Procedure Writing, Principles and Practices, Columbus: Battelle Press, 1998.

Collectively, these references demonstrate that industry best practices are applied to the development and documentation of all technical procedures.

As discussed in Review Task #3, Section 4.1.3, regular reviews of approved procedures are used to assess the adequacy of safety-related procedures in comparison with industry best practices. Specifically, pre-job briefings and post-job debriefings are used to assess the adequacy of safety-related procedures. In addition, continuing training on procedures is provided, as appropriate, through classroom, field, and/or simulator training.

Section 1.6 of N-PROC-OP-0005 provides direction on the use of OPEX during pre-job briefings [19]. The review of OPEX prompts staff to consider any industry best practices which may have emerged since the applicable procedure was last revised or used. Following the execution of the procedure, the post-job debriefing process documented in N-PROC-OP-0005 directs staff to document lessons learned or opportunities for improvement using the appropriate work process (i.e., a TPAR would be initiated in accordance with N-PROC-AS-0028 to incorporate any procedural improvements/enhancements that were identified during the course of performing the task) [19].

The adequacy of safety-related procedures is also assessed through the ongoing OPEX process described in N-PROC-RA-0035, "Operating Experience Process", which monitors events around the world to determine if there is any unforeseen event that may have applicability to Pickering NGS [28]. If/when an event is deemed to be applicable, existing procedures are reviewed to identify any vulnerabilities or weaknesses that could result in similar events or problems, with corrective actions initiated as required.

Conclusion:

The conclusion of this Review Task assessment is that there is governance in place that provides a means for assessing the adequacy of safety-related procedures in comparison with industry good practices. The intent of Review Task #5 is met and therefore Pickering NGS is compliant.

4.1.6 Review Task #6: Procedures for Normal, Abnormal and Emergency Conditions

Confirm that there are operating procedures that apply comprehensively to normal, abnormal and emergency conditions (including anticipated operational occurrences, design basis accident conditions, post-accident conditions, and design extension conditions).

N-PROG-OP-0001 implements for OPG Nuclear a series of standards and procedures to ensure the safety of public, environment, plant personnel, and plant equipment. This program establishes safe, uniform, and efficient operating practices and processes within Nuclear facilities that provide nuclear professionals the ability to ensure facilities are operated in such a manner that Reactor Operating License, Operating Policies and Principles, and other applicable regulations and standards are followed [7]. There are a range of operating procedures governed by this program that are used to operate Pickering NGS under normal, abnormal, and emergency operating conditions, as described below.

Normal Conditions

Pickering NGS Operations staff normally operate the plant through the use of approved Operating Manuals (OMs), Operating Procedures (OPs), Alarm Response Manuals (ARMs), and Safety-Related System Tests (SRSTs). OMs are produced for individual plant systems and provide instructions on equipment operation and controls. As specified in N-INS-08120-10000, "Operating Manuals", system OMs follow a standardized format, with each section of the OM representing a stand-alone document that can be revised and issued independently [29].

For normal operating conditions, operation of plant equipment is performed in accordance with the direction provided in Section 4.0, "Standard Operating Conditions" of the applicable system OM. These documents are issued in Asset Suite under the NA44-OM-XX-YYYYY-04.ZZ[7] document series for Pickering Units 1,4. Similarly, OMs for Pickering Units 5-8 are issued in Asset Suite under the NK30-OM-XX-YYYYY-04.ZZ[7] document series.

Abnormal Conditions

Operations staff will initially attempt to respond to an abnormal condition via the system-based OMs or ARMs, as applicable. Specifically, the following sections of OMs are applicable to the initial response to abnormal conditions:

- Section 5.0, "Non-Standard Operating Conditions";
- Section 6.0, "Actions Following Trips and Alarms"; and

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[7] Where XX denotes the applicable unit(s) for the Operating Manual, YYYYY denotes the System Classification Index (SCI), and ZZ denotes the applicable subsection in Section 4.0 of the OM.

- Section 8.0, “Auxiliary Services Failures”.

As discussed above, the system OMs are issued in Asset Suite. Similarly, the ARMs are also issued in Asset Suite.

If it is not possible to address the abnormal condition through the system OMs and/or ARMs, staff will respond per the applicable Abnormal Incident Manual (AIM). AIMs are issued in Asset Suite under the following series of documents:

- NA44-AIM-014-09013-XX[8];
- NK30-AIM-058-09013-XX[8]; and
- P-AIM-018-09013-XX[8].

The AIMs are used to respond to both abnormal and emergency conditions. Of particular relevance to abnormal conditions are Sections 3 and 4 of the AIMs, which address faults (commonly referred to as impairments) that reduce the effectiveness of safety systems or other safety-related systems.

If the abnormal condition is the result of a process upset that results in a reactor trip, sustained reactor setback to less than or equal to 2 %FP, reactor stepback to any power level, or a forced power reduction to less than or equal to 2 %FP, the Critical Safety Parameter (CSP) monitoring and restoration procedures will be entered [31], [32]. These procedures are executed in parallel with the event-based AIMs and are used by operating staff to perform an independent check of key parameters indicative of overall unit health, to determine if any further actions are required.

Collectively, the system OMs and AIMs are used to implement the station response to abnormal conditions, including Anticipated Operational Occurrences (AOOs).

Emergency Conditions

Upon confirmation or diagnosis of emergency conditions, including Design Basis Accidents (DBAs), staff respond per the event-based AIMs and continue to monitor conditions as per the CSP AIM. Continued use of the CSP AIM ensures that operating staff perform independent checks of key parameters indicative of overall unit health, to determine if any further actions are required to complement the event-based AIMs that are in use. The objectives of the event-based AIMs are to:

- Shut down the reactor and maintain it in a shutdown state;
- Maintain adequate heat sinks to cool the fuel and reactor structures; and
- Maintain integrity of the containment system to prevent release of radioactive materials.

8 Where XX denotes the applicable subsection of the AIM.

Once AIM actions to achieve these objectives have been successfully implemented, there is subsequent direction on longer-term actions that need to be completed in order to effectively manage post-accident conditions resulting from the event.

If plant conditions are more severe such that AIM actions, on their own, cannot mitigate the event progression, this implies that the event has progressed to a Beyond Design Basis Accident (BDBA). A BDBA refers to a low frequency event sequence that is not included in the plant design basis (due to the very low probability of occurrence) and is not necessarily bounded by the analyses of the station design basis. If the consequences of such events are significant core degradation, these BDBAs are referred to as Severe Accidents (SAs).

BDBA management strategies are described in N-STD-MP-0019, "Beyond Design Basis Accident Management" and implemented through Emergency Mitigating Equipment Guidelines (EMEGs) and Severe Accident Management Guidance (SAMG) [33]. EMEGs and SAMG are used to respond to all BDBAs, including Design Extension Conditions (DECs) and SAs[9].

EMEGs are entered when entry conditions specified in the event-based AIMs are met. EMEGs have a primary focus on fuel cooling and are used to mitigate accident progression when design basis equipment is unable to provide adequate core cooling. The intent of EMEG use is to prevent a BDBA sequence from progressing to a SA [33].

SAMG is a set of written guidance to implement strategies should a BDBA progress to a SA. SAMG is entered based on entry conditions in the EMEGs, CSPs, or event-based AIMs being met. Entry to SAMG implies that adequate fuel cooling has been lost and severe core damage is either imminent or has occurred. The goals of SAMG are to terminate progression of core damage, if possible, by restoring cooling, and to maintain containment integrity and minimize radioactive releases. The SAMG allows flexibility in application and the SAMG document set is referred to as "guidance" [33]. In addition, the SAMG aids in identifying longer-term actions that will be required to address post-accident conditions once the station has been returned to a controlled, stable state.

Conclusion:

The conclusion of this Review Task assessment is that there are operating procedures that apply comprehensively to normal, abnormal, and emergency conditions (including AOOs, DBAs, post-accident conditions, and DECs). The intent of Review Task #6 is met and therefore Pickering NGS is compliant.

[9] It is recognized that EMEGs and SAMG are guidelines as opposed to procedures. However, within the context of this report the EMEGs and SAMG are treated as procedures as these are the documents that are used to respond to BDBAs.

4.1.7 Review Task #7: Clarity of Procedures

Confirm there is a means for assuring the clarity of procedures taking into account human factors.

As discussed in Review Tasks #1 and #2, Sections 4.1.1 and 4.1.2 respectively, procedures are prepared and revised following the process documented in N-PROC-AS-0028 [13].

Per Section 1.4 of N-PROC-AS-0028 [13], technical procedures are prepared or revised by applying the structure and minimum content requirements established in N-STD-AS-0014 [12]. This standard specifies requirements for the structure, minimum content, and format of technical procedures. In addition, N-STD-AS-0014 ensures that procedures are developed using approved instructions and templates specific to the type of procedure being developed. Compliance with this standard ensures that technical procedures developed and used throughout OPG Nuclear facilities [12]:

- Promote safe and efficient operation;
- Reflect industry best practice; and
- Document compliance with regulatory requirements.

Section 4.3.2 of N-STD-AS-0014 identifies several developmental references that were used to support the development of this standard, which include the following [12]:

- American Institute of Chemical Engineers: Guidelines for Writing Effective Operating and Maintenance Procedures, New York, Center for Chemical Process Safety, 1996;
- Blake G, Bly RW: The Elements of Technical Writing, New York: Macmillan USA, 1993;
- Campbell JJ, Zimmerman C: Fundamentals of Procedure Writing, Columbia: GP Publishing, Inc., 1988;
- Haramundanis K: The Art of Technical Documentation, Boston: Digital Press, 1998; and
- Wieringa D, Moore C, Barnes V: Procedure Writing, Principles and Practices, Columbus: Battelle Press, 1998.

The references cited above relate to best practices for technical writing and procedure writing. Thus, developing procedures in accordance with N-STD-AS-0014 promotes the development of procedures which are sufficiently clear to support safe and efficient operation of the station.

N-PROC-AS-0028 [13] includes direction to ensure that procedures are prepared in accordance with N-STD-AS-0014. Specifically, Step 1.4.1 of N-PROC-AS-0028 prompts the document author to complete N-FORM-10141, "Writer's Guide Review Checklist". The completion of N-FORM-10141 establishes that the procedure has been reviewed for consistent format, presentation, level of detail, attention to detail, and other requirements specified in N-STD-AS-0014 [34]. In addition, N-PROC-AS-0028 directs the document author to determine the required validation, verification, and review activities to support finalizing the procedure [13]. The completion of these activities aids in identifying issues that may impact on the clarity of the procedure from the user's perspective so that such issues can be resolved prior to the procedure being issued for use. The completion of the activities described above ensures that considerations for human factors are taken into account during the procedure development process.

The clarity of existing procedures is also evaluated through regular reviews of procedures, as described in Review Task #3, Section 4.1.3. Per Section 1.4 of N-STD-AS-0002, staff are expected to stop work and contact their supervisor for direction if the procedure they are using is unclear, cannot be performed as written, would result in an unsafe condition or expected response, or is otherwise inappropriate [20]. If issues are identified with regard to the clarity of a procedure, a TPAR is initiated to revise the procedure per the process documented in N-PROC-AS-0028 [13]. Similarly, TPARs may also be issued in response to feedback from continuing training provided to Operations staff where procedures are reviewed for clarity and ease of use.

Adherence to N-PROC-AS-0028 also ensures that TPARs initiated to improve the clarity of a procedure or improve the usability of the procedure from a human factors perspective (i.e., reduce potential for error resulting from use of the procedure) are appropriately prioritized [13]. Per Step 1.3 of N-PROC-AS-0028, TPAR requests are categorized using Appendix C, "TPAR Categorization Criteria" [13]. One of the categories listed in Appendix C of the procedure is "Enhancement", which explicitly acknowledges human factors improvements to procedures that are intended to reduce error potential.

TPARs to improve the clarity of existing procedures can also be initiated from OPEX reviews (per N-PROC-RA-0035 [28]), self-assessments (per N-PROC-RA-0097 [23]) or N-PROG-RA-0003, "Corrective Action" [35].

Conclusion:

The conclusion of this Review Task assessment is that governance exists which provides a means of assuring the clarity of procedures taking into account human factors. The intent of Review Task #7 is met and therefore Pickering NGS is complaint.

4.1.8 Review Task #8: Processes to Update Procedures

> *Evaluate processes to update procedures to allow for changes in the assumptions made and/or the limits and conditions arising from the safety analysis, plant design and operating experience.*

As discussed in previous Review Tasks, the need to update a procedure can be identified through various processes. This Review Task will focus on the following activities which may prompt the initiation of a request to update existing station procedures.

1) Receipt of OPEX that identifies vulnerabilities and/or weaknesses in procedures, which require correction.

2) Modifications to the plant design through the Engineering Change Control (ECC) process.

3) New safety analysis has been performed that introduces new operating limits or revises assumptions associated with previously existing operating limits.

OPEX

Per the OPEX Safety Factor report [36], the CANDU Owners Group (COG) reviews recent OPEX events from a variety of sources (WANO, INPO, COG members, etc.) at the COG OPEX Weekly Screening Meeting.

SCRs are initiated for items from the Weekly Screening Meeting where a potential vulnerability is identified that is assessed to be applicable and actionable for Pickering NGS. SCRs are processed in accordance with N-PROC-RA-0022, "Processing Station Condition Records" [27]. An outcome of the SCR process may include corrective actions to update existing procedures. If required, a TPAR would be initiated to implement the necessary procedural changes in accordance with N-PROC-AS-0028 [13].

Modifications to Plant Design

N-PROG-MP-0001, "Engineering Change Control" ensures all modifications to OPG Nuclear structures, systems, and components (SSCs), including software and engineered tooling, are planned, designed, installed, commissioned, placed into service, or removed from service within the Safe Operating Envelope (SOE) or Safety and Design Envelope (SDE), design basis, and licensing conditions [37].

Per Section 1.9 of N-PROG-MP-0001, SSCs impacting the design basis are modified following a risk-based process outlined in N-PROC-MP-0090, "Modification Process" [38]. Adherence to N-PROC-MP-0090 ensures that plant configuration is controlled and maintained by updating design, operating, maintenance, and training documentation impacted by design changes. Engineering Changes (ECs) include an Affected Documents List (ADL) which identifies engineering, operations, maintenance, and training controlled documents that are affected by the modification and require revision [38]. In accordance with N-PROC-MP-0090 [38], procedures included on the ADL will be revised through the TPAR process documented in N-PROC-AS-0028 [13]. Revised procedures are issued and available for use prior to Operations acceptance of the modification following the completion of installation activities.

Changes to Safety Analysis

N-PROG-MP-0014, "Reactor Safety Program", defines the organizational responsibilities and key program elements for the management of issues related to nuclear safety analysis, including Generic Action Items, and the following major aspects of safe operation that are relevant to the content in existing procedures [39]:

- Safety Analysis Basis (Safety Report and Analysis of Record); and
- Safe Operating Envelope (SOE).

N-PROC-MP-0086, "Safety Analysis Basis and Safety Report Update" defines the Safety Analysis Basis and describes the established practices used to ensure that the Safety Analysis Basis is maintained [40]. It also defines the process and provides the instructions for processing changes to the Safety Analysis part of the Safety Report (i.e., Part 3 of the Safety Report). The resolution of Safety Report issues does not always require changes to station procedures (i.e., resolution of an issue may be limited to revising an analytical assumption that does not change how the plant is operated). However, Step 1.4.3 of N-PROC-MP-0086 [40] specifies that if it becomes apparent in the process of analyzing an issue that the issue is better addressed by making an engineering change to the plant or a change to operating procedures, then work should proceed in accordance with N-PROG-MP-0001 [37]. As described above for the process that is followed to update procedures to reflect design modifications, TPARs would be initiated to revise affected procedures.

In addition to ongoing work to update the Safety Report, discovery issues may be encountered that result in changes to the assumptions, limits, or conditions used in the existing safety analysis. N-PROC-RA-0094, "Discovery Issue Resolution Process" requires that upon discovery of an issue (or potential issue) with deterministic safety analysis, a SCR be raised and the Safety Report Update process be initiated [41]. That is, any changes required to the safety analysis would be processed per N-PROC-MP-0086, as described above.

N-STD-MP-0016, "Safe Operating Envelope", defines the processes, organizational responsibilities, and key program elements to ensure that the SOE is defined and documented in a correct, complete, and consistent manner and reflected as required in the station operating documentation [42]. Figure 1 of N-STD-MP-0016 illustrates the process by which SOE revisions are defined and implemented. This figure illustrates that revisions to the SOE are initiated in accordance with N-PROG-MP-0001 [37], with any gaps identified in station operating documentation addressed through the initiation of TPARs that are processed per N-PROC-AS-0028 [13].

Conclusion:

The conclusion of this Review Task assessment is that there is governance in place to update procedures as required in response to changes in the assumptions made and/or the limits and conditions arising from the safety analysis, plant design and operating experience. The intent of Review Task #8 is met and therefore Pickering NGS is compliant.

4.1.9 Review Task #9: Accident Management Procedures

Verify that the analysis and justification of the accident management procedures are documented.

As discussed in Review Task #6, Section 4.1.6, accident management procedures[10] include the following types of documents:

- OMs/ARMs;
- AIMs/CSPs;
- EMEGs; and
- SAMG.

The appropriateness of accident management procedures is demonstrated and documented through a combination of procedure validation activities, training activities, probabilistic safety analysis, and deterministic safety analysis.

Per Review Task #1, Section 4.1.1, all procedures are developed in accordance with N-PROC-AS-0028 [13]. Adherence to N-PROC-AS-0028 [13] ensures that appropriate validation activities are performed to confirm the usability and technical validity of the procedure, per the validation and review requirements identified from the completion of N-FORM-10212 [43].

Review Task #1 of the Deterministic Safety Analysis Safety Factor report [44] discusses various assumptions that are used in the deterministic safety analysis. One of the generic assumptions discussed in this report is the required operator action credits. More specifically, the report references Tables 1-2 and 1-11 and Tables S.1-1 to S.1-10 in Part 3 of the Pickering 1,4 and 5-8 Safety Reports respectively, as summarizing all required operator action credits. These actions generally correspond to actions that would be taken by operating staff per OMs or AIMs.

[10] As noted in Section 4.1.6, it is recognized that EMEGs and SAMG are guidelines as opposed to procedures. However, within the context of this report the EMEGs and SAMG are treated as procedures as these are the documents that are used to respond to BDBAs.

Similarly, Review Task #2 of the Probabilistic Safety Assessment (PSA) Safety Factor report [45] discusses assumptions and requirements related to the modelling of operator actions in the station PSAs. More specifically, the report references N-GUID-03611-10001 Volume 1, "OPG Probabilistic Risk Assessment (PRA[11]) Guide – Level 1 (At-Power)", which provides a requirement to account for, quantify and re-quantify applicable operator actions during the PSA Level 1 preparation. The Level 1 and Level 2 PSAs also consider station operation impacts via an in-depth review of current station documentation and supporting references, including the EMEGs.

If actions identified in OMs and AIMs are unsuccessful in terminating the accident progression, actions will be taken per the EMEGs to prevent the accident from progressing to a SA. N-BDB-03600-00002, "OPG Emergency Mitigating Equipment for Beyond Design Basis Accidents: Technical Basis Document" [46] summarizes the technical basis for EME including:

- The bounding BDBA event sequence and associated analyses;
- The overall functional requirements for the EME; and
- Other information relevant to EME specification, design, and procurement.

N-BDB-03600-00002 constitutes the analysis basis for the EMEGs, which are used to deploy EME when required as part of the response to an accident at Pickering NGS.

If a BDBA were to progress to a SA, SAMG would be used to respond to the event. SAMG is a set of written guidance that is used to terminate progression of core damage, if possible, by restoring cooling, and to maintain containment integrity and minimize radioactive releases. The physical processes that govern SA phenomena are complex and, consequently, SAMG cannot be made highly dependent on detailed analysis because of limited understanding of certain SA phenomena and other uncertainties associated with SA causes and progression. However, reasonable strategies for coping with SA progression can be identified and developed using "state of the art" reviews, Probabilistic Safety Assessments, and insights on accident behaviours from accident analyses. Each station-specific SAMG document is supported by a station-specific SAMG Background Document that provides the rationale for aspects of the response.

Conclusion:

The conclusion of this Review Task assessment is that supporting documentation exists for the analysis and justification of accident management procedures. The intent of Review Task #9 is met and therefore Pickering NGS is compliant.

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[11] Note that PRA is now referred to as PSA.

4.1.10 Review Task #10: Categorization of Procedures Based on Safety Significance

Verify that an appropriate process is in place for the categorization of procedures in accordance with their significance to safety.

As discussed in Review Task #6, Section 4.1.6, it has been confirmed there are existing procedures that apply comprehensively to normal, abnormal, and emergency conditions (including AOOs, DBAs, post-accident conditions, and DECs). This framework is also applied to categorize new procedures in accordance with their significance to safety.

Procedures that apply to normal and abnormal conditions are typically structured to be used as part of an event-based response. That is, procedures which fall into this category are used to respond to higher frequency, lower consequence events (in comparison to procedures used in certain emergency conditions) where it is practicable to predict the applicable event sequences in advance.

If actions taken through procedures used in response to normal or abnormal conditions are unsuccessful, plant conditions will eventually deteriorate to the point where procedures for emergency conditions are initiated. Procedures which fall into this category are typically structured to be used as part of symptom-based response. That is, critical parameters for monitoring the health of a unit are identified, with mitigating actions taken to return key plant parameters to a desired value.

If a new technical procedure is identified as being required, a TPAR is submitted per N-PROC-AS-0028. Step 1.3.1(d) of N-PROC-AS-0028 provides the following direction on how to process TPARs [13]:

If request is for a new technical procedure, perform the following:

1) *Ensure request does not meet criteria for a Work Plan.*

2) *Determine procedure category.*

3) *Identify appropriate system to be used in procedure number, if applicable (sequence number to be determined by Controlled Document Records personnel).*

Thus, adherence to N-PROC-AS-0028 ensures that as part of approving the TPAR new procedures are categorized based on their significance to safety (i.e., through the determination of the appropriate procedure category).

Following the approval of the TPAR, the document author prepares a draft procedure. Per Step 1.4.1(c) of N-PROC-AS-0028 [13], the draft technical procedure is prepared by applying the structure and minimum content requirements established in N-STD-AS-0014 and supporting instructions specific to the procedure category [12].

N-STD-AS-0014 specifies the requirements for the structure, minimum content, and format of technical procedures. This standard also provides direction on additional requirements that are specific to various procedure categories. Technical procedures addressed under N-STD-AS-0014 include the following [12]:

- AIMs, which are prepared in accordance with N-INS-08120-10003, "Emergency Operating Procedure Format" [47];
- ARMs, which are prepared in accordance with N-INS-08120-10002, "Alarm Response Manuals" [48];
- Chemistry Control Procedures, which are prepared in accordance with N-INS-08120-10008, "Chemistry Control Procedure Format" [49];
- Chemistry Laboratory Procedures, which are prepared in accordance with N-INS-08120-10009, "Chemistry Laboratory Procedures Format" [50];
- Common Technical Procedures, which are prepared in accordance with N-INS-08120-10010, "Common Technical Procedures" [51];
- OMs, which are prepared in accordance with N-INS-08120-10000, "Operating Manuals" [29];
- OPMs, which are prepared in accordance with N-INS-08120-10005, "Operating Memos" [52];
- Operating Procedures, which are prepared in accordance with N-INS-08120-10006, "Operating Procedures Format" [53];
- Overall Unit Manuals, which are prepared in accordance with N-INS-08120-10001, "Overall Unit Manual Format" [54]; and
- SRSTs, which are prepared in accordance with N-INS-08120-10004, "Safety-Related System Tests and Operator Test Procedures" [55].

In addition to providing direction on specific requirements for the various categories of technical procedures, N-STD-AS-0014 also provides direction to the document author on how to determine the usage classification of the procedure. Specifically, Step 1.1.9 of N-STD-AS-0014 provides the following direction [12]:

> *Usage classification specifies the required frequency and method by which users shall refer to a technical procedure while performing an activity or task:*
>
> - *A technical procedure shall be assigned the appropriate usage classification based on risks associated from performing an activity from memory.*
> - *Activities may change usage classification within a single technical procedure. A change of usage classification within a technical procedure shall be clearly indicated.*

- *Criteria for the three usage classifications; continuous, reference and information, are described in N-STD-AS-0002 [20].*

Adherence to N-PROC-AS-0028 and N-STD-AS-0014 ensures that a consistent process is followed for classifying procedures in accordance with their significance to safety. This includes determining the type of procedure required, adherence to generic requirements for the development of all procedures, adherence to requirements specific to the type of procedure being prepared, and identification of the usage classification of the procedure based on risks associated with performing the procedure.

Conclusion:

The conclusion of this Review Task assessment is that governance is in place to ensure that procedures are categorized in accordance with their significance to safety. The intent of Review Task #10 is met and therefore Pickering NGS is compliant.

4.1.11 Review Task #11: Staff Involvement in Development of Procedures

Examine whether there is adequate involvement in the development of procedures by the staff who will use them.

Per Review Tasks #1 and #2, Sections 4.1.1 and 4.1.2 respectively, existing procedures are modified and new procedures are created in accordance with N-PROC-AS-0028 [13]. Frequently, these changes are initiated by TPARs that were filed based on feedback from staff participating in training activities (e.g., staff participating in simulator training). This demonstrates that staff who will use the procedures are involved in identifying the need to revise existing procedures or create new procedures.

In many cases, staff from Operations Support and Maintenance Support who prepare the procedures are drawn from the Operations and Maintenance crews who are responsible for executing the procedures. In addition, the review, validation, and verification stages of the procedure development process described in N-PROC-AS-0028 are used to obtain input from staff who will ultimately be using the procedure.

<u>Review</u>

Per Section 1.5 of N-PROC-AS-0028, the review process ensures that documents are correct, meet the intended function, and are usable by a qualified individual [13]. The document author identifies review requirements by completing N-FORM-10212 [43].

Staff identified as reviewers for a procedure complete a review of the procedure for acceptability within their jurisdiction. The document author addresses reviewer feedback and obtains concurrence from individual reviewers on the dispositions to their comments. Once reviewers are satisfied with the contents of the procedure, the procedure is circulated for validation.

Validation

Per Section 1.5 of N-PROC-AS-0028, the validation process ensures that documents are correct, meet the intended function, and are usable by a qualified individual. The method selected for technical procedure validation depends on various considerations such as the complexity of the document, availability of a suitable validation site, and the number of disciplines involved in the performance of the tasks. Validation methods include the following [13]:

- Field Validation: a validation method that requires tasks specified in the technical procedure be performed on actual plant equipment.
- Simulated Performance Validation: a validation method that requires tasks specified in the technical procedure be performed on simulators, models, mock-ups, or on shop equipment that is not considered to be plant equipment.
- Table-Top Discussion and Walk Through: a validation method that requires instructions in the technical procedure be talked through step-by-step followed by the steps being walked through in the normal work environment.

Irrespective of the validation method selected for a given procedure, the validation process requires involvement from staff who will be using the procedure, once it is approved. The document author addresses feedback from the validation process and obtains concurrence from individual validators on the dispositions to their comments. Once feedback from the validation process has been incorporated into the procedure, the procedure is circulated for verification.

Verification

Per Section 1.11 of N-PROC-AS-0028, the verification process ensures that documents are correct, meet the intended function, and are usable by a qualified individual [13].

The verifier of a procedure is a person, other than the document author, who is knowledgeable of the system or equipment to which the procedure applies, and qualified to at least the minimum level position necessary to perform the procedure or be considered a system expert. In addition, Section 1.12.2 of N-PROC-AS-0028 identifies requirements for certain Operations procedures to be verified by a Shift Manager or Control Room Shift Supervisor [13]. The document author addresses verifier feedback and obtains concurrence from the verifier on the dispositions to their comments. Once the verifier is satisfied with the contents of the procedure, the procedure is submitted for approval in accordance with N-PROC-AS-0028 [13].

Conclusion:

The conclusion of this Review Task assessment is that there is adequate involvement in the development of procedures by the staff who will use them. The intent of Review Task #11 is met and therefore Pickering NGS is compliant.

4.1.12 Review Task #12: Distribution and Control of Procedures

Evaluate the distribution process for the control, copying and removal of obsolete versions of procedures, so that only the last approved edition is used.

N-PROG-MP-0005, "Configuration Management", ensures that Pickering NGS is operated, maintained, and modified in accordance with the design and licensing basis for the station [56]. Compliance with this program ensures that the physical configuration of the plant matches configuration documents (e.g., procedures) for all states, including normal operation, upset, post-accident, and emergency conditions.

As discussed in Review Task #8, Section 4.1.8, there are various processes (e.g., receipt of new OPEX, changes to assumptions/limitations in safety analysis, or planned modifications to the plant), which may result in the need to revise existing procedures. If required, technical procedures are revised following the process documented in N-PROC-AS-0028. Per Section 1.14.1 of N-PROC-AS-0028, the document owner (or delegate) is responsible for ensuring the final approved technical procedure and procedure reference package is processed through Business Services[12] for issuance, distribution, and retention of the new procedure [13].

The issuance, distribution, and retention of new procedures is managed through OPG-PROC-0178, "Controlled Document Management" [57]. This procedure defines a process for managing the life cycle of controlled documents, including procedures, across OPG in order to:

- Ensure the latest applicable revision of a controlled document is identified and available for use;
- Minimize the risk of inadvertent use of obsolete and superseded documents; and
- Ensure approved Document Change Requests (DCRs) are linked to the applicable controlled documents, maintained, dispositioned, and available.

Per OPG-PROC-0178, prior to initiating work on revising an existing procedure, the document owner submits a request for the latest electronic version of the procedure to be checked out. This ensures that the last approved revision of the procedure is used as the starting point for implementing the required changes as part of the subsequent revision cycle. Following approval of the revised procedure, Business Services will issue the revised procedure in Asset Suite, so that it is available for use. As part of this process, the previous revision of the procedure is set to "Revised" in Asset Suite to mitigate the risk of inadvertent use of an obsolete or superseded procedure [57]. In the infrequent event that the revised procedure needs to be used while the old revision remains at "Issued" status in Asset Suite, Section 1.14.5 of N-PROC-AS-0028 specifies the mitigating actions that are implemented to ensure the latest revision of

[12] Business Services is now called the Information Management Execution Controlled Documents Unit.

the procedure is used in these situations [13]. Paper copies of revised procedures are distributed as required and are limited to essential and operationally significant locations (e.g., Main Control Room) [57].

The risk of staff using an obsolete version of a procedure is further mitigated through direction contained in N-STD-AS-0002 [20]. This standard provides direction to all staff on how to use and adhere to all governance including administrative and technical procedures. Specifically, Step 1.1.4 of N-STD-AS-0002 prompts staff to ensure the procedure they intend to use is the current in-use revision by obtaining it from a controlled copy storage location or confirming the revision in Asset Suite, in accordance with the direction per the usage classification [20].

Conclusion:

The conclusion of this Review Task is that there is a distribution process in place for the control, copying, and removal of obsolete versions of procedures, so that only the last approved edition is used. The intent of Review Task #12 is met and therefore Pickering NGS is compliant.

4.1.13 Review Task #13: Understanding and Acceptance of Procedures

Evaluate audits, self-assessments, safety performance and events to determine whether there is adequate understanding and acceptance of these procedures by managers and staff.

Assurance of adequate understanding and acceptance of procedures is established in part through the delivery of training to staff. N-PROG-TR-0005, "Training" [58], describes the training program for regular staff, contractors, temporary personnel, and other staff assigned work at OPG Nuclear. The training program provides the structure, processes, and tools for defining, developing, implementing, documenting, assessing, and improving the training required to ensure Nuclear staff have the appropriate knowledge, skill, and attitudes for safe and efficient plant operation.

The training program is complemented by the Human Performance program, which is documented in N-PROG-AS-0002 [18]. This program establishes a systematic framework for Human Performance management across OPG Nuclear to achieve higher levels of nuclear and industrial safety, higher unit reliability, and reduced operating costs through event-free operation. Ensuring that there is adequate understanding and acceptance of procedures by managers and staff is a key focus of the Human Performance program. The following aspects of the program are of particular relevance:

- N-PROC-OP-0005, "Pre-Job Briefing and Post-Job Debriefing" [19]
    - Effective pre-job briefs assist in the safe and efficient planning, preparation, and execution of plant activities that operate, maintain, or modify plant equipment.

- N-INS-09030.2-10000, "Event Free Challenge Process" [59]
    - Event free challenge meetings allow a task execution group to perform a final check on its state of readiness just prior to executing a task. Its use is intended for situations where the likelihood and consequence of error are high, to put appropriate defenses/barriers in place to ensure safe execution of work.
- N-STD-AS-0002, "Procedure Use and Adherence" [20]
    - This standard provides direction to all staff on how to use and adhere to all governance including administrative and technical procedures.
- N-INS-09030-10004, "Observation and Coaching" [60]
    - This instruction provides expectations for conducting workplace observations and coaching. It provides a means to reinforce standards of performance (e.g., procedure use and understanding) with a goal of achieving event-free plant operation.

Based on the above, whether there is adequate understanding of procedures by managers and staff can be determined by evaluating the effectiveness of N-PROG-TR-0005 [58] and N-PROG-AS-0002 [18]. Per the Pickering PSR2 Basis Document [1], effectiveness reviews (at Pickering NGS) of OPG programs used to demonstrate compliance with the PSR2 Assessment Basis are conducted for PSR2, using recent audit and self-assessment results. Audit and self-assessment results applicable to N-PROG-AS-0002 [18] are summarized in Section 4.3 and Appendix B of P-REP-03680-00014, "Pickering NGS PSR2 Safety Factor 10 Report: Organization, Management System, and Safety Culture" [26]. P-REP-03680-00014 [26] states that the review of audit and self-assessment findings associated with N-PROG-AS-0002 did not identify any gaps. Similarly, audit and self-assessment results applicable to N-PROG-TR-0005 [58] are summarized in Section 4.3 and Appendix B of P-REP-03680-00016, "Pickering NGS PSR2 Safety Factor 12 Report: Human Factors" [61]. P-REP-03680-00016 states that the review of audit and self-assessment findings associated with N-PROG-TR-0005 did not identify any gaps [61].

Conclusion:

The conclusion of this Review Task assessment is that there is an adequate understanding and acceptance of procedures by managers and staff. The intent of Review Task #13 is met and therefore Pickering NGS is compliant.

4.2 L/R/C/S Reviews

Per Section 2.2 of this report, a detailed compliance assessment for one L/R/C/S with content applicable to Safety Factor 11 is provided in Reference [6]. Associated findings applicable to Safety Factor 11 are summarized in Table 3 below.

Table 3: PSR2 L/R/C/S Compliance Assessment Results for Safety Factor 11

L/C/R/S Reviewed	PSR2 Compliance Assessment for Safety Factor 11
CSA N286-12, "Management System Requirements for Nuclear Facilities"	There are no PSR2 gaps for CSA N286-12. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with N286-12.

4.3 Audit and Self-Assessment Reviews

The OPG Nuclear Program reviewed for the Procedures Safety Factor is identified in Table 2, and details of the associated audit and self-assessment results for this N-PROG are provided in Appendix B.

In addition, P-REP-03680-00007, "Pickering NGS PSR2 Safety Factor 4 Report: Aging" [62], identified the following information related to the Conduct of Maintenance Program for consideration in this Safety Factor 11 report. Nuclear Oversight conducted a performance based audit, NO-2015-030 [63], of the Maintenance Program in December 2015 for Pickering NGS to determine whether requirements defined in governance are being effectively implemented. The audit concluded that the performance was not fully effective, identifying performance improvement opportunities in the areas of work planning, foreign material exclusion, use of performance improvement tools and staff practices. Four SCRs were initiated to address the findings through implementation of corrective actions. Three have been completed (P-2015-28890, P-2015-28884 and P-2015-28887). The remaining open SCR is related to improvements in foreign material exclusion practices. Associated corrective actions are expected to be completed in Q1 2017, with an effectiveness review in Q2 2017 (SCR P-2015-28880, AR# 28186936).

4.4 Additional Review Findings

As discussed in Section 3.4, the PSR2 Safety Factor 11 assessment also included a review of commitments previously made to the CNSC, open CNSC action items, and exemptions granted by the CNSC, as identified in the R04 Pickering LCH [4], to determine if there are any impacts associated with operation of the Pickering Units past 2020. The review also included identification and review of previously identified programmatic Darlington PSR1 gaps related to Safety Factor 11 to determine impacts associated with operation of the Pickering Units past 2020. This assessment did not identify any gaps for Safety Factor 11.

Findings from the review of previously identified PSR1 gaps in the Pickering Units 5-8 Continued Operations Plan [8] are provided in Reference [9]. Findings from the review of Fukushima Action Items are provided in Reference [10]. Results from the Continued Operations Plan and Fukushima Actions Items reviews will be considered in the Global Assessment process.

There were no PSR2 gaps identified in this Safety Factor 11 report that require discussion in other Safety Factor reports.

5.0 RESULTS AND CONCLUSIONS

OPG Governance, Programs, Policies, Procedures, Instructions and Guidelines related to Safety Factor 11 were reviewed for the thirteen PSR2 Review Tasks in Section 4.1 of this report and resulted in no Pickering PSR2 gaps. L/R/C/S and OPG Nuclear Program audit and self-assessment reviews for Safety Factor 11 were prepared per Sections 4.2 and 4.3, respectively, and resulted in no Pickering PSR2 gaps. Per Section 4.4, this report also included identification and review of previously identified programmatic Darlington PSR1 gaps related to Safety Factor 11 (to ascertain the implications of extending Pickering NGS operation beyond 2020), as well as a review of the R04 Pickering LCH [4] for any impacts of Pickering NGS operation beyond 2020 on: a) OPG commitments previously made to the CNSC, b) open CNSC action items, and c) exemptions granted by the CNSC (all related to Safety Factor 11), which resulted in no Pickering PSR2 gaps.

The review of Safety Factor 11 has confirmed that the Pickering NGS processes for managing, implementing and adhering to operating and working procedures and for maintaining compliance with operational limits and conditions and regulatory requirements are adequate and effective and ensure plant safety.

6.0 REFERENCES

[1] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[2] CNSC REGDOC-2.3.3, *Periodic Safety Reviews*, April 2015.

[3] IAEA Safety Guide No. SSG-25, *Periodic Safety Review for Nuclear Power Plants*, 2013.

[4] CNSC Report, LCH-PNGS-R004, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[5] OPG Report, P-REP-03680-00003 R000, *Pickering NGS Periodic Safety Review 2 (PSR2): Definition of Safety Factor Review Tasks*, May 2016.

[6] OPG Report, P-REP-03680-0586480 R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 9, 11, and 15*, to be issued.

[7] OPG Program, N-PROG-OP-0001 R008, *Nuclear Operations*, December 2015.

[8] OPG Plan, NK30-PLAN-00531-00001 R005, *Pickering 5-8 Continued Operations Plan*, December 2015.

[9] OPG Report, P-REP-03680-00024 R000, *Pickering 5-8 Continued Operations Plan Review in Support of PNGS Periodic Safety Review 2 (PSR2),* to be issued.

[10] OPG Report, P-REP-03680-00022 R000, *Fukushima Action Item Review in Support of PNGS Periodic Safety Review 2 (PSR2)*, to be issued.

[11] OPG Report, NK38-REP-03680-10185 R002, *Darlington NGS Integrated Implementation Plan (IIP)*, April 2015.

[12] OPG Standard, N-STD-AS-0014 R007, *Requirements for Technical Procedures,* November 2014.

[13] OPG Procedure, N-PROC-AS-0028 R017, *Development, Review and Approval of Technical Procedures,* July 2015.

[14] OPG Procedure, OPG-PROC-0001 R009, *Process Administrative Governance Documents,* April 2015.

[15] OPG Form, N-FORM-10014 R011, *Technical Procedure Action Request,* May 2016.

[16] OPG Instruction, OPG-INS-00700-0001 R000, *Document Change Request Data Administration,* March 2015.

[17] OPG Charter, N-CHAR-AS-0002 R018, *Nuclear Management System,* March 2015.

[18] OPG Program, N-PROG-AS-0002 R015, *Human Performance,* October 2014.

[19] OPG Procedure, N-PROC-OP-0005 R012, *Pre-Job Briefing and Post-Job Debriefing,* June 2013.

[20] OPG Standard, N-STD-AS-0002 R015, *Procedure Use and Adherence,* August 2015.

[21] OPG Procedure, N-PROC-RA-0023 R018, *Fleetwide Program Health and Performance Reporting,* August 2013.

[22] OPG Procedure, N-PROC-RA-0048 R017A, *Conducting Performance Based Audits and Assessments,* May 2015.

[23] OPG Procedure, N-PROC-RA-0097 R008, *Self-Assessment and Benchmarking,* December 2014.

[24] CSA Standard, CSA N286-05, *Management System for Nuclear Power Plants.*

[25] OPG Program, N-PROG-RA-0010 R013, *Independent Assessment,* April 2014.

[26] OPG Report, P-REP-03680-00014 R000, *Pickering NGS PSR2 Safety Factor 10 Report: Organization, Management System, and Safety Culture,* to be issued.

[27] OPG Procedure, N-PROC-RA-0022 R032, *Processing Station Condition Records,* November 2014.

[28] OPG Procedure, N-PROC-RA-0035 R018, *Operating Experience Process,* October 2014.

[29] OPG Instruction, N-INS-08120-10000 R004, *Operating Manuals,* May 2015.

[30] OPG Procedure, NA44-AIM-014-09013-01 R003, *PNGS-A and 018 Abnormal Incident Manual Index,* June 2008.

[31] OPG Procedure, NA44-AIM-014-09013-07 R010, *Critical Safety Parameter (CSP) Monitoring and Restoration Procedure,* April 2015.

[32] OPG Procedure, NK30-AIM-058-09013-07 R023, *Critical Safety Parameter Monitoring and Restoration,* July 2015.

[33] OPG Standard, N-STD-MP-0019 R001, *Beyond Design Basis Accident Management,* September 2014.

[34] OPG Form, N-FORM-10141 R007, *Writer's Guide Review Checklist,* April 2015.

[35] OPG Program, N-PROG-RA-0003 R010, *Corrective Action,* January 2015.

[36] OPG Report, P-REP-03680-00013 R000, *Pickering NGS PSR2 Safety Factor 9 Report: Use of Experience from Other Nuclear Power Plants and Research Findings*, to be issued.

[37] OPG Program, N-PROG-MP-0001 R014, *Engineering Change Control,* January 2015.

[38] OPG Process, N-PROC-MP-0090 R012, *Modification Process,* April 2015.

[39] OPG Program, N-PROG-MP-0014 R005, *Reactor Safety Program,* September 2015.

[40] OPG Procedure, N-PROC-MP-0086 R004, *Safety Analysis Basis and Safety Report Update,* December 2014.

[41] OPG Procedure, N-PROC-RA-0094 R006, *Discovery Issue Resolution Process,* June 2015.

[42] OPG Standard, N-STD-MP-0016 R002, *Safe Operating Envelope,* June 2012.

[43] OPG Form, N-FORM-10212 R022, *Review and Validation Screens,* June 2014.

[44] OPG Report, P-REP-03680-00009 R000, *Pickering NGS PSR2 Safety Factor 5 Report: Deterministic Safety Analysis,* to be issued.

[45] OPG Report, P-REP-03680-00010 R000, *Pickering NGS PSR2 Safety Factor 6 Report: Probabilistic Safety Assessment,* to be issued.

[46] OPG Manual, N-BDB-03600-00002 R000, *OPG Emergency Mitigating Equipment For Beyond Design Basis Accidents: Technical Basis Document,* October 2015.

[47] OPG Instruction, N-INS-08120-10003 R004, *Emergency Operating Procedure Format,* September 2014.

[48] OPG Instruction, N-INS-08120-10002 R005, *Alarm Response Manuals,* March 2015.

[49] OPG Instruction, N-INS-08120-10008 R002, *Chemistry Control Procedure Format,* April 2014.

[50] OPG Instruction, N-INS-08120-10009 R003, *Chemistry Laboratory Procedures Format,* March 2014.

[51] OPG Instruction, N-INS-08120-10010 R007, *Common Technical Procedures,* November 2013.

[52] OPG Instruction, N-INS-08120-10005 R006, *Operating Memos,* June 2014.

[53] OPG Instruction, N-INS-08120-10006 R002, *Operating Procedures Format,* January 2015.

[54] OPG Instruction, N-INS-08120-10001 R002, *Overall Unit Manual Format,* April 2015.

[55] OPG Instruction, N-INS-08120-10004 R003, *Safety-Related System Tests and Operator Test Procedures,* April 2011.

[56] OPG Program, N-PROG-MP-0005 R005, *Configuration Management,* June 2012.

[57] OPG Procedure, OPG-PROC-0178 R000, *Controlled Document Management,* October 2015.

[58] OPG Program, N-PROG-TR-0005 R016, *Training,* January 2016.

[59] OPG Instruction, N-INS-09030.2-10000 R002, *Event Free Challenge Process,* January 2016.

[60] OPG Instruction, N-INS-09030-10004 R000, *Observation and Coaching,* October 2014.

[61] OPG Report, P-REP-03680-00016 R000, *Pickering NGS PSR2 Safety Factor 12 Report: Human Factors,* to be issued.

[62] OPG Report, P-REP-03680-00007 R000, *Pickering NGS PSR2 Safety Factor 4 Report: Aging,* July 2016.

[63] OPG Nuclear Oversight Report, N-REP-01070-0576012 T06, *NO-2015-030 Conduct of Maintenance – Pickering,* December 2015.

Appendix A: Nomenclature

ADL	Affected Documents List
AIM	Abnormal Incident Manual
AOO	Anticipated Operational Occurrence
ARM	Alarm Response Manual
BDBA	Beyond Design Basis Accident
CANDU	CANada Deuterium Uranium
CNSC	Canadian Nuclear Safety Commission
COG	CANDU Owners Group
CSP	Critical Safety Parameter
DBA	Design Basis Accident
DCR	Document Change Request
DEC	Design Extension Condition
EC	Engineering Change
ECC	Engineering Change Control
EMEG	Emergency Mitigating Equipment Guideline
IAEA	International Atomic Energy Agency
ISR	Integrated Safety Review
NGS	Nuclear Generating Station
OM	Operating Manual
OPEX	Operating Experience
OPG	Ontario Power Generation
OPM	Operating Memo
PARTS	Pickering A Return to Service
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
PROL	Power Reactor Operating Licence
PSR	Periodic Safety Review
PSR1	Periodic Safety Review 1 (earlier OPG PSR work and other associated assessments)
PSR2	Periodic Safety Review 2 (subsequent PSR per CNSC REGDOC-2.3.3)
SA	Severe Accident

SAMG	Severe Accident Management Guidance
SCR	Station Condition Record
SDE	Safety and Design Envelope
SOE	Safe Operating Envelope
SRST	Safety-Related System Test
SSC	Structures, Systems and Components
TPAR	Technical Procedure Action Request

Appendix B: Audit and Self-Assessment Results

B.1 N-PROG-OP-0001, "Nuclear Operations"

The Nuclear Operations Program for Ontario Power Generation (OPG) Nuclear, encompasses a series of standards and procedures to ensure safety of the public, environment, plant personnel and plant equipment. The program establishes safe, uniform, and efficient operating practices and processes within Nuclear facilities, which provide nuclear professionals the ability to ensure facilities are operated in compliance with the Power Reactor Operating License, Operating Policies and Principles, and other applicable regulations and standards. The Nuclear Operations Program format is based on the operations-related sections of the World Association of Nuclear Operators (WANO) Performance Objectives and Criteria. These sections are:

- Foundations;
- Functional Areas; and
- Cross-Functional Areas.

The Governance and Services section completed a self-assessment in September 2015, BAS15-001424-SA [B.1.1], in order to assess the health of the Nuclear Operations Program governance framework, which is applicable for both Pickering and Darlington NGS. No findings/SCRs were generated from this self-assessment, however it was recommended to initiate a DCR in order to reference Canadian Standards Association (CSA) N286-12, "Management System Requirements for Nuclear Facilities" which has since been completed.

The Operations and Maintenance division completed a self-assessment in August 2015, P15-000134-SA [B.1.2], of the Common Services Department Conduct of Operations at Pickering NGS (e.g., review of Common Services operator behaviours as well as an evaluation of Supervising Nuclear Operators ability to provide effective task oversight). It was concluded that the operations team, operators, and supervising nuclear operators completed routines and tasks in accordance with OPG governance. No SCRs were generated as a result of this self-assessment.

Nuclear Oversight conducted a performance based audit which assessed the effectiveness of Pickering NGS' conduct of field operations against the requirements of N-PROG-OP-0001, "Nuclear Operations", in March 2015 (NO-2015-001) [B.1.3]. The audit concluded that the performance was not fully effective, identifying performance improvement opportunities in the area of monitoring and reinforcement of operator fundamentals for routine tasks. One SCR was initiated to address the above finding (SCR P-2015-05319), which required corrective actions to be implemented. All of the necessary corrective actions associated with this SCR have been completed and the underlying issues have been addressed.

References

[B.1.1] Self-Assessment Report, BAS15-001424-SA R1, *Program Assessment – N-PROG-OP-0001*, September 30, 2015.

[B.1.2] Self-Assessment report, P15-000134-SA, *Conduct of Operations Assessment of the Common Services Department Operators Behaviours,* August 24, 2015.

[B.1.3] Audit Report, N-REP-01070-0533384 (NO-2015-001), *Nuclear Oversight Audit Report OPGN NO-2015-001 Pickering Conduct of Operations,* March 5, 2015.

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REVISE AND RESUBMIT	
Signature	Date 16 Dec 2016
Name: Mike Ruffolo, Manager	
Dept: Pickering Engineering - Aging Management & Strategic Initiatives	
OPG Proprietary	
Doc No.: P-REP-03680-00016 Rev: 000	
This acceptance does not relieve the contractor from responsibility for errors or omissions or from any obligations or liability under this contract.	

Pickering NGS PSR2 Safety Factor 12 Report: Human Factors

PS112/RP/005 R01

December 16, 2016

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Revision Summary – For Amec Foster Wheeler Report PS112/RP/005

Rev	Date	Author	Comments
R00	July 29, 2016	A. Go, R. Jayasundera	Initial issue for OPG review and comment.
R01	December 16, 2016	K. Nicholson, R. Jayasundera	Incorporation of OPG comments.

EXECUTIVE SUMMARY

Ontario Power Generation (OPG) is performing a Periodic Safety Review (PSR) to support the possibility of continued operation of Pickering Nuclear Generating Station (NGS) beyond 2020. The PSR (referred to as "PSR2") is a subsequent PSR building on the review basis of earlier OPG Integrated Safety Reviews and other associated assessments. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

Part of PSR2 involves the preparation of Safety Factor reports for each of fifteen major topic areas. Safety Factor reports consist of:

- Assessing compliance against "Review Tasks" identified in the PSR2 Basis Document [1]. These Review Tasks are derived from Canadian Nuclear Safety Commission (CNSC) REGDOC-2.3.3, "Periodic Safety Reviews" [2] and International Atomic Energy Agency (IAEA) SSG-25, "Periodic Safety Review for Nuclear Power Plants" [3];

- Documenting assessments against applicable modern Laws, Regulations, Codes and Standards (L/R/C/Ss) as defined in Reference [1]; and

- Reviewing the effectiveness of applicable OPG Nuclear Programs that support the above assessments, through review of audit and self-assessment results.

The PSR2 review of Safety Factor 12, *Human Factors* is presented in this report. OPG Governance, Programs, Policies, Procedures, Instructions and Guidelines related to Safety Factor 12 were reviewed for the eight PSR2 Review Tasks specified in Section 4.1 of this report. L/R/C/S and OPG Nuclear Program effectiveness reviews for Safety Factor 12 were prepared per Sections 4.2 and 4.3, respectively. Per Section 4.4, the PSR2 assessment includes a review of previously identified PSR1 gaps related to Safety Factor 12 (to ascertain the implications of extending Pickering NGS operation beyond 2020), as well as a review of the R04 Pickering Licence Conditions Handbook [4] for any impacts of Pickering NGS operation beyond 2020 on: a) OPG commitments previously made to the CNSC, b) open CNSC action items, and c) exemptions granted by the CNSC (all related to Safety Factor 12).

The results of the review of Safety Factor 12 are discussed in Section 5.0. The review has confirmed that the various human factors that may affect the safe operation of Pickering NGS have been appropriately addressed. As discussed in Section 5.0, the review identified no Pickering PSR2 gaps.

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LIST OF TABLES AND FIGURES

1.0 INTRODUCTION

Ontario Power Generation (OPG) is performing a Periodic Safety Review (PSR) to support the possibility of continued operation of Pickering Nuclear Generating Station (NGS) beyond 2020.[1] A comprehensive Integrated Safety Review (ISR) was completed for Pickering Units 5 through 8 in 2009 in support of refurbishment and continued operation. Pickering Units 1,4 integrated safety assessments were also performed for Pickering A Return to Service (PARTS) in support of approval to restart Units 1 and 4. In addition to these Pickering-specific studies, the 2013 Darlington ISR performed extensive code and standard reviews that were updated in relation to the versions that were assessed in the 2009 Pickering B ISR.[2] These previous ISRs are considered to constitute the first PSR completed for Pickering (referred to as "PSR1"). The current PSR (referred to as "PSR2") is a subsequent PSR building on the basis of earlier OPG integrated safety assessments through review of the various studies, assessments and licence renewals performed since PSR1. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

PSR2 will support and complement the licence renewal application for Pickering NGS going forward. Fifteen Safety Factors will be assessed as part of the PSR. The purpose of Safety Factor reviews is to confirm that the design, condition and operation of Pickering Units 1,4 and 5-8 (as well as common systems) will support continued safe operation for the period of PSR2 by:

- Assessing compliance against "Review Tasks" identified in the PSR2 Basis Document [1], which were derived from Canadian Nuclear Safety Commission (CNSC) REGDOC-2.3.3 [2] and International Atomic Energy Agency (IAEA) SSG-25 [3];

- Documenting assessments against applicable modern Laws, Regulations, Codes and Standards (L/R/C/Ss) (as defined in Reference [1]); and

- Reviewing the effectiveness of applicable OPG Nuclear Programs that support the above assessments, through review of audit and self-assessment results.

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1 Currently, Pickering Units 5-8 are approved to operate to 247,000 Effective Full Power Hours. This operation limit is expected to be reached on some units in 2020. For the purposes of PSR2, OPG assumes operation of Pickering NGS for up to eight additional years, from 2020 until 2028. OPG will make a decision regarding the permanent shut down dates for the six reactors following the performance of a technical evaluation that will include PSR2, and will communicate it to the CNSC as required by the current Power Reactor Operating Licence.

2 Much of the compliance assessment and evaluation of Safety Factor health for the Darlington ISR is based on programs and practices that apply across OPG's nuclear operations. As a result, where Pickering is confirmed to follow the same nuclear programs and practices as were assessed for Darlington, the Darlington ISR programmatic conclusions are applicable to Pickering. As discussed in Section 1.0, an effectiveness review (effectiveness at Pickering NGS) of OPG programs used to demonstrate compliance with the PSR2 Assessment Basis will be conducted using recent audit and self-assessment results.

As outlined in IAEA SSG-25 [3], the objective of the review of Human Factors Safety Factor 12 is to: "evaluate the various human factors that may affect the safe operation of the nuclear power plant and to seek to identify improvements that are reasonable and practicable." REGDOC-2.3.3 [2] requires that: "The licensee shall conduct a PSR in accordance with this regulatory document for the period until the next PSR or, if applicable, until the end of commercial operation of the plant." For this Safety Factor 12 Report, the objective is to confirm that the various human factors that may affect the safe operation of Pickering NGS have been appropriately addressed. Per the Pickering PSR2 Basis Document [1], analysis of gaps and potential safety enhancements for Pickering NGS (including identification of improvements that are reasonable and practicable to implement) is addressed as part of the Global Assessment process. Preparation of a plan for the implementation of safety enhancements is addressed by the PSR2 Integrated Implementation Plan.

This report documents the results of the review of Safety Factor 12 for Pickering PSR2. The report is based on the OPG Governance, Programs, data, and material available up to January 15, 2016 which is the freeze date for PSR2.

2.0 SCOPE OF REVIEW

2.1 Review Task Assessments

The Pickering PSR2 Safety Factor 12 Review Tasks are defined in Reference [1]. Details of the derivation of these Review Tasks from CNSC REGDOC-2.3.3 [2] and IAEA SSG-25 [3] are shown in Reference [5]. The Safety Factor 12 Review Tasks are:

1) Confirm that there are procedures to ensure that a minimum number of qualified staff, appropriate to the operating state of the plant, is available at all times.

2) Confirm that adequate staff training facilities, training staff and training programs exist.

3) Confirm that the method of selecting staff for new positions and for promotions involves systematic and validated staff selection methods and a method for succession planning.

4) Confirm that there are appropriate programs for initial, refresher, and upgrade training. For operating staff, this should include the use of simulators.

5) Establish that there is training in safety culture, including for management staff, that includes work supervision practices and internal communication practices and expectations.

6) Confirm there are fitness for duty guidelines relating to hours of work, health and substance abuse.

7) Confirm that the human–machine interface is considered in the design of the control room and other workstations, that analysis of human information requirements and task workload is performed, and that there is linkage to the Probabilistic Safety Assessment, Deterministic Safety Analyses and Hazard Analysis. This review should include a discussion of how guidance such as U.S. NRC NUREG-0700 Revision 2, "Human-System Interface Design Review Guidelines" [6], and NUREG-0711 Revision 2, "Human Factors Engineering Program Review Model" [7], identified in CNSC REGDOC-2.5.2 are relevant to the design of Pickering Units 1,4 and Units 5-8. (Note: Review Task #7 is addressed in the Plant Design Safety Factor report.)

8) Confirm the style and clarity of procedures provides an appropriate level of detailed guidance for staff through a review of plant events identifying inadequate procedures as a contributing cause.

The methodology for the reviews is discussed in Section 3.1. Review Task findings are summarized in Section 4.1 of this Report.

2.2 L/R/C/S Reviews

The applicable Laws, Regulations, Codes and Standards relevant to the Human Factors Safety Factor are identified in Reference [1] and are listed in Table 1 below. Table 1 also identifies the modern version and date of each L/R/C/S to be considered, the Safety Factor(s) to which each document is applicable, and the type of review that will be completed in PSR2.

All of the Safety Factor 12 L/R/C/S reviews are incremental in nature. The definition of an Incremental Review is as follows:

- Incremental Review: For L/R/C/Ss that have been reviewed in PSR1 but have had revisions since the last review, a topical review will be performed of the changes.

The methodology for the reviews is discussed in Section 3.2. A detailed assessment for each L/R/C/S is provided in Appendix B of Reference [8]. Associated findings will be summarized in Section 4.2 of this report.

Table 1: L/R/C/Ss Reviewed for Human Factors Safety Factor 12

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
Additional L/R/C/Ss						
1	CNSC G-323	Ensuring the Presence of Sufficiently Qualified Staff at Class I Nuclear Facilities-Minimum Shift Complement	2007	10, 12	Incremental	G-323 addressed as part of Darlington ISR.
2	CNSC G-278	Human Factors Verification and Validation Plans	2003	1, 12	Incremental	G-278 addressed as part of Pickering B and Darlington ISRs.
3	CNSC G-276	Human Factors Engineering Program Plans	2003	1, 12	Incremental	G-276 addressed as part of Pickering B and Darlington ISRs.
4	CSA N290.12	Human Factors in Design for Nuclear Power Plants	N290.12-14	1, 12	Incremental [3]	N290.12 not addressed as part of Pickering B or Darlington ISRs. OPG has completed a gap analysis against mandatory requirements of N290.12.

[3] Per CNSC's request in P-CORR-03680-0607223, "Pickering PSR2 – Change to Review Type for CSA N290.12" [9], the Review Type for CSA N290.12-14 was changed from High Level to Incremental.

2.3 OPG Program Effectiveness Reviews

The OPG Nuclear Program (N-PROG) reviewed for Safety Factor 12 is listed in Table 2 below.[4] The methodology for the effectiveness reviews is discussed in Section 3.3. The assessment results of the N-PROG in Table 2 are provided in Appendix B, and findings are summarized in Section 4.3.

Table 2: OPG Program Reviewed for Safety Factor 12

Document Number	Document Title
N-PROG-TR-0005 [10]	Training

2.4 Additional Reviews

The PSR2 Safety Factor 12 Report includes a review of the R04 Pickering Licence Conditions Handbook (LCH) [4] for any impacts of Pickering NGS operation beyond 2020 on the following (all related to Safety Factor 12):

- OPG commitments previously made to the CNSC;
- Open CNSC action items; and
- Exemptions granted by the CNSC.

The PSR2 assessment includes identification and review of previously identified PSR1 gaps related to Safety Factor 12 to ascertain the implications of extending Pickering NGS operation beyond 2020. The methodology for these reviews is described in Section 3.4. Any PSR2 gaps identified as a result of programmatic Darlington PSR1 gaps related to Safety Factor 12 are discussed in Section 4.4 of this report. The review of Pickering PSR1 gaps previously identified in the Pickering Units 5-8 Continued Operations Plan (COP) [11] is provided in a separate PSR2 COP Review Report.

In addition, Fukushima Action Items (FAIs) were reviewed to identify implications of extending operation beyond 2020 (if any). This review is presented in a separate PSR2 FAI Review Report.

Any PSR2 gaps identified as a result of the Safety Factor 12 review which need to be addressed in other Safety Factor Reports are discussed in Section 4.4 of this report.

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4 The list of Nuclear Programs to be assessed for effectiveness for PSR2 was derived from review of current OPG Governance. Although there may be content in Nuclear Programs that is applicable to multiple Safety Factors, N-PROG reviews are only provided in one Safety Factor report and are not duplicated.

3.0 METHODOLOGY

The sub-sections below summarize the methodology used to assess Review Tasks, L/R/C/Ss, and Nuclear Program effectiveness for the Human Factors Safety Factor.

3.1 Review Tasks

As discussed earlier, the Safety Factor Review Tasks are derived from CNSC REGDOC-2.3.3 [2] and IAEA SSG-25 [3], taking into consideration the Review Tasks used in the Pickering B and Darlington ISRs (as derived in [5]).

For each Safety Factor 12 Review Task identified in Section 2.1, a confirmation of the existence of applicable OPG Governance, Programs, Policies and Procedures (as well as Instructions and Guidelines, as applicable) was performed. Compliance against Review Tasks is also assessed by reference to applicable Condition Assessments, safety analyses and operating experience, as required.

The Review Task assessments identify Compliances and Gaps as defined below:

- Compliance: Compliance indicates that either the safety requirement or the intent of the Review Task is met.
- Gap: A Gap indicates that the intent of the Review Task is not met.

3.2 L/R/C/S Reviews

The process to identify the modern L/R/C/Ss that are applicable to the PSR2 Assessment Basis involved first creating a broad list from multiple sources (potential candidate L/R/C/Ss) and then filtering it to identify those that are most significant and that are applicable to the PSR2 scope. The identification and selection criteria are detailed in Reference [1]. The result of the identification and selection process was a set of modern L/R/C/Ss that became part of the "PSR2 Assessment Basis".

PSR2 is focused on the extension of Pickering NGS operations beyond 2020, and will conduct reviews against a baseline of past PSR1 work. As a subsequent PSR, PSR2 focuses on changes in requirements, plant conditions, operating experience and new information. Since PSR2 is an update of previous ISRs, it incorporates reviews of L/R/C/Ss that have occurred as new versions have been issued. Since this assessment is a subsequent PSR, the focus is on identifying differences between what was previously assessed and what is now different within the current Pickering PSR2 Assessment Basis. In general, these differences relate to:

- More recent (new or revised) L/R/C/S versions than what was previously assessed as part of PSR1;[5]

- Safety significant differences between Pickering and Darlington, if the Darlington ISR is the basis for the earlier assessment;

- Implications of extending Pickering NGS operation beyond 2020; and

- Safety significant differences between Pickering Units 1,4 and Units 5-8.

As described in Reference [1], L/R/C/S review types are clause-by-clause, high level or incremental. Most of the L/R/C/Ss in the PSR2 Assessment Basis receive incremental reviews since PSR2 is an update of previous PSR1 assessments and clause-by-clause or high level reviews for the majority of the L/R/C/Ss in the PSR2 Assessment Basis have already been completed. Implementation plans (including gap analyses or code-over-code reviews) also exist for the latest editions of many L/R/C/Ss. As a result, incremental review is also used in circumstances where an L/R/C/S in the PSR2 Assessment Basis was not assessed in previous PSR1 reviews but an implementation plan currently exists for compliance.

The PSR2 incremental reviews in this report include an assessment of the intent of recent changes to the L/R/C/Ss on a topic or subject-matter basis where there is potential to impact nuclear safety. Incremental reviews provide:

- A summary of the purpose of the L/R/C/S;

- Pertinent background information about the current revision of the L/R/C/S that is being considered;

- Identification of which Safety Factor(s) are applicable to the current revision of the L/R/C/S;

- A description of which version(s) of the L/R/C/S were assessed for PSR1 (i.e., Darlington ISR (for programmatic content), Pickering B ISR and PARTS code reviews);

- Identification of whether the current version of the L/R/C/S is an update of a previous version of the L/R/C/S that was assessed in PSR1 (and if so, a description of the major changes in the latest revision is provided as discussed below);

[5] "New" refers to a code or standard that was not previously considered in the context of earlier assessments. "Revised" refers to an updated version of a code or standard that was previously considered in the context of earlier assessments. Where a document has a new number/type, but addresses the same topic from the same organization, it is a "revised", not "new", document (e.g., if a REGDOC replaces a CNSC G or RD document).

- An assessment of the applicability of PSR1 assessment findings (gaps and conclusions), including the implications of extending Pickering NGS operation beyond 2020 if any;

- An assessment of the applicability of assessment findings that address more recent (post-PSR1) editions of the L/R/C/S, including any implementation or transition plans that are already committed to by OPG; and

- Where PSR1 and post-PSR1 assessments are not sufficient to address changes in the latest edition of the L/R/C/S, an assessment of the changes from the previously assessed edition of the L/R/C/S (including identification of any safety significant PSR2 gaps which result).

High Level reviews provide the same information as above, where applicable, in a similar format. However, given that High Level L/R/C/Ss generally have not received past assessment during PSR1, the incremental review content is augmented by a high level, section-by-section assessment of the degree of conformance of Pickering NGS with the L/R/C/S (demonstrating, with supporting evidence, whether the intent of the requirements stipulated in the document are met).

There are currently no L/R/C/S clause-by-clause reviews identified in the PSR2 Assessment Basis.

The Safety Factor 12 L/R/C/S reviews identify Compliances and Gaps as defined below:[6]

- Compliance:
    - Where an Incremental review has been performed, Compliance indicates that the change in the safety requirement, per the topical review, is met.
    - Where a High Level review has been performed, Compliance indicates that the intent of the safety requirement is met. (Note: No High Level reviews were performed as part of Safety Factor 12.)
    - For Clause-by-Clause reviews of modern L/R/C/Ss, Compliance indicates that the safety requirement is met. (Note: No Clause-by-Clause reviews were performed as part of Safety Factor 12.)
- Gap:

[6] Safety Factor assessments for Review Tasks and L/R/C/S reviews make use of: a) OPG Governance, Programs, Policies and Procedures which support the assessment arguments, b) Commitments previously made to the CNSC, open CNSC action items, and exemptions granted by the CNSC (all related to the Safety Factor under review), as identified in the R04 Pickering LCH [4], c) Identification of previously identified Pickering-specific or programmatic PSR1 gaps related to the Safety Factor under review and the status of OPG's improvement plan(s) or other dispositions to address these, and d) Assessments and reviews performed since the PSR1 documents were completed.

- Where an Incremental review has been performed, a Gap indicates that the change in the safety requirement, per the topical review, is not met.

- Where a High Level review has been performed, a Gap indicates that the intent of the safety requirement is not met. (Note: No High Level reviews were performed as part of Safety Factor 12.)

- For Clause-by-Clause reviews of modern L/R/C/Ss, a Gap indicates that the safety requirement is not met. (Note: No Clause-by-Clause reviews were performed as part of Safety Factor 12.)

The reviews assume that use of the word:

- “Shall" is used in an L/R/C/S to express a requirement, i.e., a provision that the licensee is obliged to satisfy in order to comply with the standard;
- "Should" is used to express a recommendation or that which is advised but not required;
- "May" is used to express an option or that which is permissible within the limits of the standard; and
- "Can" is used to express possibility or capability.

3.3 OPG Program Effectiveness Reviews

As discussed earlier, effectiveness reviews (effectiveness at Pickering NGS) of OPG programs used to demonstrate compliance with the PSR2 Assessment Basis were conducted, using recent applicable audit and self-assessment results:

- OPG Nuclear Oversight independent performance-based Program audits (typically performed in 1 to 5 year cycles) and self-assessments. This includes review of associated Station Condition Records and Action Requests to determine the status of any resulting corrective actions; and
- CNSC “Type I” and “Type II” inspections of the effectiveness and performance of OPG programs, where discussed in OPG audits or self-assessments.

There are many audits and self-assessments that are performed to assess the effectiveness of important aspects of each program. A sample of audits and self-assessments has been summarized for each program in order to demonstrate that program effectiveness is being assessed on an ongoing basis. The focus of these reviews was on effectiveness of the programs at Pickering NGS, where specific information is available. Results from these audits and self-assessments will be considered in the Global Assessment process. It is noted that audits and self-assessments are, by their nature, self-critical and are used to drive excellence in performance. As a result, the broad review scope of program audits focuses on identifying improvement opportunities rather than presenting a balanced picture of program performance.

Program effectiveness is also monitored and addressed through the Fleetview Program Health and Performance Reporting process [12]. This process involves direct oversight by the Chief Nuclear Officer, and includes participation by the Nuclear Executive Committee members. Programs are reviewed, senior oversight is provided, and improvement plans are generated.

The list of Nuclear Programs to be assessed for each Safety Factor was derived from review of current OPG Governance, and has used the most recent version of these documents as of the PSR2 freeze date of January 15, 2016.

3.4 Additional Reviews

A review of the R04 Pickering LCH [4] was performed to determine if there are any impacts associated with operation of the Pickering Units past 2020 on the following (all related to Safety Factor 12):

- Commitments previously made to the CNSC;
- Open CNSC action items; and
- Exemptions granted by the CNSC.

The PSR2 assessment includes identification and review of previously identified Pickering-specific or programmatic PSR1 gaps related to Safety Factor 12 (as identified in the Darlington ISR Integrated Implementation Plan [13] and Pickering Units 5-8 Continued Operations Plan [11]) to ascertain the status of OPG's improvement plan(s) or other dispositions to address these and the implications of extending operation beyond 2020 (if any).[7]

Fukushima Action Items were reviewed to identify implications of extending operation beyond 2020 (if any). The methodology for this review is provided in a separate PSR2 FAI Review Report.

Any PSR2 gaps identified as a result of the Safety Factor 12 review which need to be addressed in other Safety Factor Reports are also discussed.

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[7] PSR2 includes consideration and confirmation that the findings of PSR1 remain valid, as applicable, for the operation period. This includes assessment of PSR1 conclusions against implications resulting from extended operation. In particular, Pickering PSR1 results are applicable to PSR2 if there was a PSR1 gap that is still open, or if a closed PSR1 gap could be affected by extended operation. If so these gaps are carried forward into PSR2 for consideration in the Global Assessment. (When references to PSR1 are made, the source document is identified and the relevant text from that source document is summarized in the context of PSR2.) With respect to the Darlington ISR, much of the evaluation of Safety Factor health is based on programs and practices that apply across OPG’s nuclear operations. As a result, Darlington ISR programmatic conclusions are applicable to Pickering PSR2 where Pickering is confirmed to follow the same nuclear programs and practices that were assessed for Darlington. Darlington PSR1 results are applicable to Pickering PSR2 if there are Darlington PSR1 gaps that are found to be relevant to Pickering PSR2.

4.0 REVIEW FINDINGS

4.1 Review Tasks

The sub-sections below provide an assessment of the adequacy of applicable OPG Governance, Programs, Policies and Procedures (as well as Instructions and Guidelines, as applicable) in demonstrating compliance against the Safety Factor 12 Review Tasks.

4.1.1 Review Task #1: Procedures to Ensure Minimum Number of Qualified Staff

Confirm that there are procedures to ensure that a minimum number of qualified staff, appropriate to the operating state of the plant, is available at all times.

P-INS-09100-00003, "Pickering Minimum Shift Complement" [14] is the procedure that defines the Minimum Shift Complement (MSC) staffing requirements to ensure safe conditions are maintained during normal operations along with the capability to be able to respond to all station emergencies. Per P-INS-09100-00003 [14], MSC is the minimum number of qualified workers who shall be present at all times to ensure the safe operation of the Pickering facility, respond to all credible events, and to ensure adequate emergency response capability for the most resource intensive conditions. P-INS-09100-00003 [14] is written to comply with CNSC guidelines G-323, "Ensuring the Presence of Sufficient Qualified Staff at Class I Nuclear Facilities – Minimum Staff Complement" and G-278, "Human Factors Verification and Validation Plans".

The MSC for the Operations Work Group (Shift Management Team and Operators), Emergency Response Team (Emergency Response Management, Maintenance, Chemistry, Stores), and the Emergency Response Organization (ERO) (Emergency Response Management, Shift Management, Operations, Maintenance, Fuel Handling, Plant and Resource Coordinators) is defined in Tables 1-1, 1-2, and 1-3 of P-INS-09100-00003 [14], respectively.

P-INS-09260-00008, "Duty Crew Minimum Complement Assurance" [15], defines the responsibilities and processes to ensure that the MSC according to P-INS-09100-00003 [14] is met, as well as the correct use and updating of the local area network based Minimum Complement Compliance Program (MCCP), which is used to ensure MSC is met. The instruction defines: actions required when below complement, duty crew accounting, absence reporting, "step-up" (i.e., designating a temporary (but qualified) change in work group or ERO assignment), Minimum Availability Requirement (MAR), emergency role qualification, and position assignments.

Per N-PROG-OP-0001, "Nuclear Operations [16]," the Director of Operations and Maintenance – Operations (DOM-O) is accountable for ensuring qualified competent staff are in place to implement requirements of the Nuclear Operations program.

N-INS-03490-10003, “Minimum Shift Complement Resources, Qualifications and Procedures Required for Responding to Resource Limiting Events” [17], provides instructions to ensure that procedures and qualifications linked to the MSC are maintained such that Pickering NGS remains compliant with the MSC. Table 1 of N-INS-03490-10003 [17] lists the Training and Qualification Documents (TQDs) associated with MSC staff qualification requirements.

Training, qualification and certification processes for control room and certain field positions ensure that the staff is competent to perform the functions assigned to them. The simulator is used extensively for initial training and qualification, as well as for refresher / requalification training. Per N-INS-08920-10002, “Simulator-based Initial Certification Examinations for Shift Personnel” [18], Simulator Exercise Guides are used as part of training for certified staff. Trainees are assessed per N-INS-09110-10059, “Simulator Performance Observation and Crew Critiques” [19].

N-TQD-503-00001, “Nuclear Emergency Response Organization Training and Qualification Description” [20], lists the qualification requirements for staff required to fill ERO roles in the event of an emergency. Appendix A of N-TQD-503-00001 [20] lists the Radiation Protection qualification requirements for each ERO position.

Conclusion:

The conclusion of this Review Task assessment is that OPG has programs and procedures in place to ensure that minimum staffing levels of qualified staff are available for all modes of operation. The intent of Review Task #1 is met and therefore Pickering NGS is compliant.

4.1.2 Review Task #2: Adequate Staff Training Programs and Resources

Confirm that adequate staff training facilities, training staff and training programs exist.

OPG Program N-PROG-TR-0005, “Training” [10], describes the Training Program for regular staff, contractors, temporary personnel and other staff assigned work at OPG Nuclear. The Training Program provides the structure, process and tools for defining, developing, implementing, documenting, assessing, and improving the training required for nuclear staff. The Training Program equips staff with the appropriate knowledge, skills, and attitude for safe and efficient plant operation. Training is also used to minimize the impact of plant operation on the environment, health and public safety, and to drive human performance improvements. The program describes structure and content of Nuclear Training governing documents. The structure of training program governance is presented in the figure below.

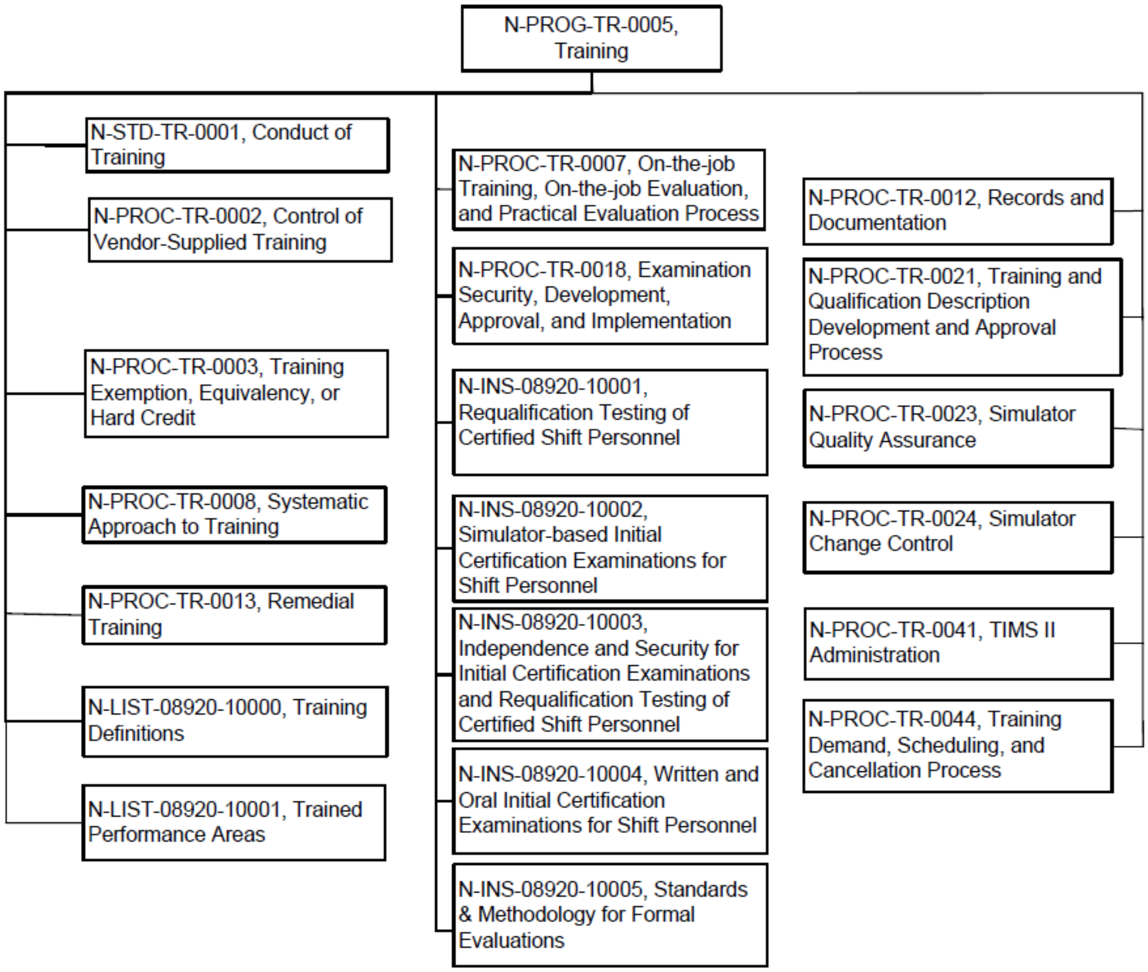


Figure 1: Nuclear Training Program Governing Documents (Per N-PROG-TR-0005 [10])

In general, Nuclear Training Program content originated from the following sources:

- CNSC objectives and criteria for regulatory evaluations of training, licence conditions, and applicable standards;
- World Association of Nuclear Operators;
- United States Department of Energy Training Handbooks and Guides to Good Practices;
- Institute of Nuclear Power Operations; and
- Training procedures used at other nuclear power generating utilities.

Training governing documents define the following:

- Formal training standards and processes;
- Systematic approach for analysis of potential training opportunities and design, development, delivery, evaluation and revision of quality training;
- Requirements for qualifying staff to meet performance expectations, improve job performance and operational efficiency;
- Qualification tracking and maintenance process; and
- Training material filing, and maintenance requirements.

To facilitate its training program, Pickering NGS makes use of various training facilities (both on-site and off-site), including the simulator, the Pickering Learning Centre, facilities at the University of Ontario Institute of Technology and conference centres.

OPG Procedure, N-PROC-TR-0021, "Training and Qualification Description Development and Approval Process" [21] provides requirements for developing, approving, revising, and implementing the TQDs and Qualification Guides (QGs).

TQDs associated with specific performance areas are listed in N-LIST-08920-10001, "Trained Performance Areas" [22]. The list indicates the:

- Organization responsible for supporting the associated training; and
- TQDs that form part of the Major Trained Performance Areas.

The Manager, Training, Planning and Design, is required to approve the TQDs included in N-LIST-08920-10001 [22].

OPG Procedure N-PROC-TR-0008," Systematic Approach to Training" [23], provides the process used to identify potential training changes, to confirm training requirements through training needs analysis, to specify specific training requirements through job and task analysis, and to design, develop, implement, and evaluate training. The procedure describes the following processes:

- Communication of a perceived training need to the Training organization;
- Analysis of a perceived training need;
- Design, development, implementation, and evaluation of training to support proper job performance and individual development;
- Linkage of training to conditions, scope, complexity, actions, and standards associated with job performance expectations; and
- Incorporation of operating experience into training.

The standard, N-STD-TR-0001, "Conduct of Training" [24], identifies specific accountabilities when implementing Training Programs. The standard provides general requirements for a Trainer and describes requirements and accountability for implementing Nuclear Training. The Trainer collectively refers to contract, temporary, Subject Matter Expert, or trainer who delivers nuclear training programs presented in the TQDs listed in N-LIST-08920-10001 [22]. Trainers should be qualified in accordance with N-TQD-602-00001, "Nuclear Trainer Training and Qualification Description" [25].

Training effectiveness, quality, and efficiency is optimized through prudent use of trainers, facilities, equipment, and training technologies to ensure the following:

- Facilities and training methods are appropriate for the type of learning involved and level of proficiency required;
- Training schedules are issued in a timely manner;
- Training is integrated with other Nuclear work programs to provide a complete overview of resource availability and allocation; and
- Personnel attend training at scheduled times.

N-PROC-TR-0002, "Control of Vendor-Supplied Training" [26], establishes requirements, processes, and accountabilities for control of training developed by Vendors and training delivered by Vendors within OPG Nuclear premises and externally. This procedure addresses the following:

- Vendor instructor capabilities and qualifications to instruct OPG Nuclear staff; and
- Requirements for review, verification, and acceptance of Vendor-supplied training prior to and after training delivery.

N-INS-08920-10001, "Requalification Testing of Certified Shift Personnel" [27], provides instruction for:

- Requirements for requalification tests that OPG Nuclear Certified Shift Personnel will have successfully completed when seeking renewal of certification, in accordance with Class I Nuclear Facilities Regulations;
- Processes that OPG Nuclear will follow in developing, conducting and grading written and simulator-based requalification tests that demonstrate that Certified Shift Personnel have retained the knowledge and skills required to work competently in their assigned positions; and
- Ensuring requalification tests are administered in a consistent manner and in accordance with requirements endorsed by the CNSC.

OPG employs an electronic management system, Training Information Management System, Version II (TIMS II) (N-PROC-TR-0041, "TIMS II Administration" [28]) to assist with administration of training support functions. This system ensures consistent, high quality training is efficiently and effectively delivered, documented, and archived as evidence of worker qualification status.

OPG Procedure, N-PROC-TR-0044, "Training Demand, Scheduling and Cancellation Process" [29], describes the process for planning and scheduling training using TIMS II to streamline scheduling functions. With respect to adequacy of training resources, including training facilities, this procedure requires that the Training Managers, Nuclear Programs and Training:

- Ensure training plans are consistent with the 5-year business plan;
- Ensure adequate delivery resources to support training schedules;
- Ensure Business Unit Line Management is informed of any shortfalls;
- Approve and communicate short-term modifications; and
- Develop a draft training schedule based on training demand, trainer availability and training facility availability.

The "Health of Training" reports for trained performance, described in N-PROG-TR-0005 [10], are produced quarterly and assess training quality and effectiveness based on the following objectives:

- Objective 1: Training for Performance Improvement;
- Objective 2: Management of Training Processes and Resources;
- Objective 3: Initial Training and Qualification;
- Objective 4: Continuing Training;
- Objective 5: Conduct of Training and Trainee Evaluation; and
- Objective 6: Training Effectiveness Evaluation.

OPG Instruction N-INS-08920-10017, "Training Committees" [30], provides training oversight requirements of the Nuclear Training Oversight Committee, Training Councils, Training Program Review Committees, and Curriculum Review Committees (CRC).

The "Health of Training" reports are discussed during quarterly Training Program Review Committee meetings for each major training performance area. These meetings are attended by senior management for the specific performance area and held with the following objectives (example of Engineering Performance Area):

- Define Initial and Continuing Training Programs;
- Provide oversight of the Training Program effectiveness and training efficiencies;
- Review and evaluate new training requests from the CRCs;
- Review the "Health of Training" report to ensure action plans address identified risks or weaknesses;
- Review and approve recommendations to Training and Line Management concerning training curricula, content, and schedules for initial and continuing training;
- Ensure that training program impact assessments adequately address changes from industrial guidelines, regulatory requirements, internal procedures and processes, plant modifications, organizational changes affecting roles and responsibilities, Operating Experience (OPEX), or emerging performance shortfalls;
- Ensure consistency between sites;
- Review impact of proposed TQD/QG changes to staff indicators and the need for a communication plan for significant changes;
- Review and approve changes to TQDs and changes to QGs that affect more than one CRC;
- Ensure line support is provided to review TQD/QGs per governance;
- Ensure sufficient number of qualified staff are in place to support safe operation and outage programs;
- Determine if supporting CRCs are required; and
- Oversee deliverables from CRCs.

Conclusion:

The conclusion of this Review Task assessment is that OPG has processes and oversight in place to confirm that adequate staff training facilities, training staff and training programs are available. The intent of Review Task #2 is met and therefore Pickering NGS is compliant.

4.1.3 Review Task #3: Staff Selection and Succession Management

Confirm that the method of selecting staff for new positions and for promotions involves systematic and validated staff selection methods and a method for succession planning.

Staff Selection

As mentioned in the addendum to the Darlington ISR (NK38-REP-03680-10080-ADD-001 [31]) the staff selection processes at OPG follow proven, validated, industry standard methods. These processes have been created and improved over many years of experience by Human Resources professionals in conjunction with reviews of industry methods. Interviews, reference and performance checks, as well as (in some instances where necessary) performance testing are used as part of the selection process. Below are a few examples of procedural documents for the hiring of specific staff:

- Maintenance Trades: N-INS-08930-10001 R000, "Hiring of Maintenance Trades";
- Operators: N-INS-08930-10002 R001, "Nuclear Operator Hiring Process"; and
- New Graduate Engineers: N-INS-08930-10003 R000, "New Graduate Engineer Hiring Process".

For management positions Stratum IV and above (nominally department managers and higher) role documents are in place, which outline managerial requirements for the role and identify key role accountabilities. The information in the role documents, N-MAN-08131-10000 (multiple sheets) [32], is used as a source of selection criteria for these management positions. Additionally, OPG has a leadership model which identifies expected behaviours of all employees at each level of the organization. These behaviours are also integrated into selection/training programs and for management staff, they are integrated into performance measurement and succession processes. Selection criteria for these roles may reference both governance and other OPG documentation, including: Job Documents (Job Evaluation System), Role Documents (Nuclear Governance), Job Family and Leadership Training Qualification Descriptions, and the OPG Leadership Model.

Selection of individuals to fill licenced positions is based on the guide, N-GUID-08930.12-10000, "Nuclear Operator Recruitment Guide" [33]. This document is classified as confidential and therefore further details from this guide are not provided here. The authorized training program applies for the selected individuals to fill the licenced positions.

Nuclear Operators are hired per N-INS-08930-10002, which describes the process for internal and external hiring of Nuclear Operators in Training (NOIT). Qualified applicants are invited to perform the standardized ability tests (which are developed by an external company) and subsequently attend candidate interviews. For

Maintenance Trades, staff are hired per N-INS-08930-10001, which describes the requirements for the sourcing and selection process for Electrical & Control Technicians, Mechanical Technicians, and both external and internal Apprentice positions. Similar to recruiting for NOIT, qualified candidates perform standardized testing and interviews.

At OPG, job descriptions identify role specific knowledge and skills. Individuals recruited to these roles bring the identified knowledge and skills and/or are provided training as documented in Training Qualification Descriptions (TQDs) and Qualification Guides (QGs). Prior to being assigned independent work, needed training is required and individuals must be deemed competent by their supervisor. The procedure N-PROC-TR-0021 [21], provides requirements for developing, approving, revising, and implementing the TQDs and QGs. TQDs identified in N-LIST-08920-10001 [22], and supporting QGs, reside within the N-PROC-TR-0021 [21] framework. TQDs and QGs document entry-level, initial and continuing training requirements.

The new hire engineering support training is intended to support the continued availability of trained and competent staff for technical positions within Engineering. N-TQD-403-00001, "Nuclear Engineering Support Personnel Training and Qualification Description" [34], provides the education background requirement for the new hire engineering support as engineering graduates or equivalent. Once recruited, the new hires are required to go through a structured training program (per N-PROC-TR-0008 [23]). The program is based on what a new graduate engineer needs to know to work at a nuclear power plant. There are initial and continuing training elements which apply to all Engineering Support Personnel as well as Duty Area elements which are specific to the different positions within Engineering. Completion of the qualifications required to perform work independently in the different engineering departments varies and could take a number of years.

Professional Engineer (P. Eng.) accreditation is stated as necessary for the specific activities identified in Appendix A of N-LIST-01300-10000, "Bounded Document Set" [35].

Training and qualification description documents are also available for Operations and Maintenance staff. New hires perform training per their training schedule. N-LIST-08920-10001 [22] identifies the list for trained performance jobs (i.e., performance areas in the training program), and supporting QGs reside within the N-PROC-TR-0021 [21] framework. These TQDs and QGs include but are not limited to:

- Authorized Nuclear Operator;
- Control Room Shift Supervisor;
- Fire protection;
- Health Physicist;
- Security;

- Shift Manager/Control Room Shift Supervisor;
- Trainer;
- Unit 0 Control Room Operator;
- Welder, welding operators, brazers, and examination personnel; and
- Maintenance.

Succession Management

The Succession Planning process at OPG is described on the OPG Powernet Human Resources page. As required by the Charter of the OPG Compensation, Leadership and Governance committee of the Board of Directors, OPG reports, at least annually, on key elements of OPG's workforce profile, talent recruitment and retention strategies, and succession plans for the President and CEO, Direct Reports to the CEO, and Nuclear senior management positions. This reporting is achieved through an annual enterprise-wide succession and talent management process that includes a talent review process, which assesses and calibrates individual performance and potential. This information is used as a source for succession planning processes, where identification of Ready Now, Ready Short and Ready Long Term successors are identified for key roles, and needed development actions are identified to close gaps to readiness. Leaders utilize a Succession Planning Toolkit (including reviews, individual development plans and other tools as required) to satisfy both the corporate requirements and to support them in local succession planning efforts. In the Nuclear Business Unit of OPG, an integrated, functionally based succession planning process is in place for key roles across Nuclear Operations and Nuclear Projects. This process is facilitated by Human Resources and succession activities are implemented by Nuclear Peer teams, including the Nuclear Executive Committee.

OPG also has a program titled ACCELERATE, which is designed to develop and accelerate the development of a pipeline of "Ready Now" employees for future leadership roles. The program is intended to address the development interests of a small portion of OPG's talent pool, specifically those who aspire to leadership roles two or more levels higher. More information on the program is available via OPG Powernet or in the OPG 2014 Sustainable Development Report found on the OPG public website [36].

Within the Engineering disciplines, N-GUID-00130-10002, "Guide for Engineering Knowledge Retention" [37] supplements the OPG Succession Planning process by providing a strategy for identifying any current risk areas for each Engineering Department, and the degree of effectiveness and possible improvements for retaining the necessary knowledge within the department through effective knowledge transfer.

Conclusion:

The conclusion of this Review Task assessment is that OPG has methods of selecting staff for new positions and promotions involving systematic and validated staff selection methods. OPG also has a systematic method in place for succession planning. The intent of Review Task #3 is met and therefore Pickering NGS is compliant.

4.1.4 Review Task #4: Initial, Refresher and Upgrade Training

Confirm that there are appropriate programs for initial, refresher, and upgrade training. For operating staff, this should include the use of simulators.

OPG Procedure N-PROC-TR-0021, "Training and Qualification Description Development and Approval Process" [21], provides requirements for developing, approving, revising, and implementing the TQDs and QGs. The TQDs and the QGs document entry-level, initial and continuing training requirements.

TQDs contain details of training stages and requirements for each stage as follows:

- Initial Training: introduce and develop job related knowledge, skills, and performance standards, preparing personnel to independently perform assigned duties and tasks; and

- Continuing Training: used to maintain and enhance knowledge and skills, and address areas such as plant equipment and procedure changes, infrequently used and difficult skills, knowledge and skills weaknesses, and lessons learned from operating experience. Continuing training may also provide refresher type training at a specified frequency, and may be required to maintain qualified status.

Continuing training is developed in accordance with N-INS-08920-10021, "Continuing and Requalification Training- Curriculum Development and Implementation Process" [38]. Annual continuing training requirements, including impact on qualification status, are presented in the TQD. These requirements include reference to any existing continuing training plan. The continuing training meets the needs of both refresher and upgrade training.

OPG instruction document N-INS-08920-10002, "Simulator-Based Initial Certification Examinations for Shift Personnel" [18], provides specific instructions on the application of IAEA- EG2, "Competency Assessments for Nuclear Industry Personnel" [39], for initial CNSC certification, which includes:

- Processes for planning, developing, conducting and grading simulator-based certification examinations;

- Examination follow-up activities including processes for dealing with passes, conditional passes, and appeal of examination results; and

- Requirements, criteria and guidelines for administering simulator-based initial certification examinations in an equitable and consistent manner.

As discussed in Section 4.1.2 for Review Task #2, OPG instruction document N-INS-08920-10001, "Requalification Testing of Certified Shift Personnel" [27], provides requirements for requalification tests that OPG Certified Shift Personnel must successfully complete when seeking renewal of certification. The instruction also identifies processes that OPG follows in developing, conducting and grading written and simulator-based requalification tests that demonstrate that Certified Shift Personnel have retained knowledge and skills required to work competently in their assigned positions.

Remedial Training (N-PROC-TR-0013 [40]) and re-take of training is required for trainees not meeting pass criteria for Training and Qualification programs for OPG Nuclear. Remedial training notification requirements for failures that occasionally occur during re-certification of authorized staff are identified in N-INS-08920-10001, "Requalification Testing of Certified Shift Personnel" [27].

As mentioned in the discussion of Review Task #2 (Section 4.1.2), OPG Procedure N-PROC-TR-0008, "Systematic Approach to Training" [23], provides the process used to identify potential training changes. Systematic Approach to Training (SAT) confirms training requirements through training needs analysis; it also designs, develops, implements, and evaluates training. The objective of SAT is to guide the development of performance based training to support job performance requirements, and individual development at OPG Nuclear.

The "Health of Training" reports are used to assess training quality and effectiveness and are reviewed at both the site and Nuclear executive levels. The governance for these reports, and the scope and objectives of the review, are described in the discussion of Review Task #2 (Section 4.1.2).

Conclusion:

The conclusion of this Review Task assessment is that OPG has programs for initial, refresher and upgrade training. OPG also has training programs for operating staff that use simulators, as well as simulator-based examinations. The intent of Review Task #4 is met and therefore Pickering NGS is compliant.

4.1.5 Review Task #5: Training in Safety Culture for Management Staff

Establish that there is training in safety culture, including for management staff, that includes work supervision practices and internal communication practices and expectations.

The Board of Directors and the Chief Nuclear Officer (CNO) take an active role in communicating the importance of safety. This is demonstrated through OPG's Nuclear Safety Policy, N-POL-0001 [41]. The policy states:

> *"Nuclear Safety shall be the overriding priority in all activities performed in support of OPG nuclear facilities. Nuclear Safety shall have clear priority over schedule, cost and production."*

N-CHAR-AS-0002, "Nuclear Management System" [42], gives authority to the nuclear safety processes and defines responsibilities. It specifies that the CNO is accountable for:

> *"The effectiveness of the overall Management System in ensuring our Nuclear facilities are operated and maintained using sound Nuclear safety and defense-in-depth practices to ensure radiological risks to workers, the public, and environment are as low as reasonably achievable, and in keeping with the Nuclear Safety Policy, and the best practices of the international Nuclear community."*

OPG Charter N-CHAR-AS-0002 [42], describes the nuclear quality program, while N-STD-AS-0020, "Nuclear Management Systems Organization" [43], outlines its implementation. Additional guidance is given in N-PROG-OP-0001, "Nuclear Operations" [16], N-PROG-MA-0004, "Conduct of Maintenance" [44], and N-PROG-MP-0007, "Conduct of Engineering" [45]. N-TQD-601-00001, "Leadership and Management Training and Qualification Description" [46], provides qualification requirements for supervisors and managers which includes "Safety Culture for Managers".

Managers ensure that tasks are executed as defined through N-PROG-AS-0002, "Human Performance" [47]. This program establishes a systematic framework for Human Performance management across OPG to achieve higher levels of nuclear and industrial safety, higher unit reliability, and reduced operating costs through event-free operation. This is accomplished through pre-job briefings, post job debriefings, self-checking programs, independent verification, communications, self-assessments, and an observation and coaching program.

OPG Standard N-STD-OP-0012, "Conservative Decision-Making" [48], derives its authority from the Human Performance program and specifically states in Section 1.2.4, "Safety shall remain the number one priority ahead of production or cost."

OPG Program N-PROG-TR-0005 [10] describes the training program for regular staff, contractors, temporary personnel, and other staff assigned to OPG Nuclear. The program also includes the structure and tools for developing and implementing the training required to ensure safe and efficient plant operation.

Overall nuclear safety is emphasized through various forums, ranging from specific reactor safety seminars to reviews of relevant OPEX.

Effective Pre-Job and Post-Job Briefs assist in the safe and efficient planning, preparation and execution of activities that directly or indirectly operate, maintain or modify an OPG generating facility. A formal process is defined in N-PROC-OP-0005, "Pre-Job Briefing and Post-Job Debriefing" [49].

Nuclear Safety is also emphasized through instillation of a questioning attitude in all staff. For tasks with novel content, additional requirements are specified in N-PROC-OP-0001, "Conduct of Infrequently Performed Tests or Evolutions" [50].

The following are a list of procedures that define management expectations in regards to safety:

- N-STD-OP-0036, "Operational Decision-Making", provides principles for effective operational decision making and a systematic approach for the application of these principles enabling operational decisions that support safe and reliable plant operation, both in the near and long term. Conservative decision-making is one of the event-free tools used during operational decision-making;

- The tools for rigorous and prudent approach are all documented in N-PROG-AS-0002, "Human Performance" [47], and in N-STD-AS-0002, "Procedure Use and Adherence" [51]; and

- Detailed guidance on the importance of precise communication and tools for implementation are described in N-STD-OP-0002, "Communications" [52].

Each level of supervision has a generic position qualification supplemented by specific qualifications necessary for the specific position. Expectations on supervisory aspects of a position, and training for those expectations are contained in leadership training courses.

The Leadership and Management Training Program qualification requirements are documented in N-TQD-601-00001, "Leadership and Management Training and Qualification Description" [46]. Appendix A of N-TQD-601-00001 outlines the program elements that each level of supervision must pass to be qualified in their position.

Embedded in each supervisor, manager, and director qualification is the requirement to complete training as follows:

- Safety Culture (all management levels), PEL 65556: Provides training to managers focusing on the accountability of managers to ensure there is a strong and healthy safety culture within their work group [54]. It looks at Safety Culture from three different perspectives: the individual, the manager and the organizational framework. The training is provided to managers to enable them to demonstrate the behaviours and accountabilities of a Nuclear Supervisor in accordance with N-POL-0001, "Nuclear Safety Policy" [41] (as mentioned in N-OVH-65556-00001 R013, "Nuclear Safety Culture for Managers" [55]). This training is a full day in length and is provided to participants in a classroom setting.

- First Level Management Program Parts 1 and Part 2 (all management levels): Both Part 1 and Part 2 courses are provided to all management levels in a classroom setting. Aspects of the training courses highlight OPG's values and behaviours, such as observing for traits of a healthy nuclear safety culture in OPG staff [56].

- Middle Management Program (Department Managers).

These courses contain training on the tools supervisors use to maintain a safe workplace and to reinforce the expected behaviours in the workforce that reflect a strong Safety Culture. For example, the event free tools are explicitly covered. In addition, specific methods of communicating messages from Senior Management to staff are included in management training. Communication methods covered through the training for management include items such as briefing notes from Managers, roll-outs of new or revised procedures and important items of general concern.

Additionally, supervisors and managers participate in the annual Leadership Continuing Training program [46]. The program has two components. The first is the Leadership Skills Continuing Training (QUAL 35544), which is an annual workshop that contains skill refresher training. There are a total of three workshops and one is completed each year; after the third year, the workshop cycles back to the Year 1 workshop. The second component of the annual Leadership Continuing Training program is Significant Operating Experience Report (SOER) 10-2 Leadership Continuing Training (QUAL 32104) [46]. A self-study Computer Assisted Learning (CAL) module is completed each year. Similarly to the leadership skills workshops, there are a total of three modules; one CAL is completed each year for three years and will cycle back and restart at the Year 1 CAL.

Managers may also attend the Nuclear Professional Development Seminar (NPDS) training or Senior Nuclear Plant Manager (SNPM) training, both of which have the following course objectives:

- Enhancing supervisors ability to: identify, analyze and solve leadership issues encountered in nuclear plants; sustain and strengthen job performance; and integrate lessons learned into plans to improve self, staff and the department; and

- Identifying the behaviours and attitudes that demonstrate valuing a positive safety culture.

Training is provided in safety culture traits, practices and expectations as well as the expectations and behaviours that support work supervision practices. This includes communication skills and expectations. However, safety culture is embedded in many processes and cuts across all functions and organizational levels. Safety culture is addressed more broadly in Safety Factor 10 Report; in this Review Task, the focus is on management training in safety culture since management are accountable for ensuring that all staff contribute to a positive safety culture.

Conclusion:

The conclusion of this Review Task assessment is that there is training in safety culture, including for management staff, that includes work supervision practices and internal communication practices and expectations. The intent of Review Task #5 is met and therefore Pickering NGS is compliant.

4.1.6 Review Task #6: Fitness for Duty Guidelines

> *Confirm there are fitness for duty guidelines relating to hours of work, health and substance abuse.*

Section 1.5.2 of N-CHAR-AS-0002, "Nuclear Management System" [42], identifies expectations with respect to Fitness for Duty. It requires that the Fitness for Duty expectations be communicated to all staff through "Human Resources Overview" training and adherence to the Corporate Safety Rules [57] (under Common Safety Rule 1.2).

All supervisors in OPG Nuclear are required to complete a training course on the Continuous Behaviour and Observation Program, N-CMT-62808-00001 [58]. This program trains supervisors to detect people not fit for duty, by developing awareness to recognize and respond to behaviours, including drug and alcohol abuse that may include a risk to the security, safety or health of employees, facilities and the public. It trains supervisors to be aware, through direct observation of changes in the behaviours of their employees, to assess the risk that is posed by these changes and to respond accordingly to the potential risk.

For all OPG Nuclear staff, Nuclear General Employee Training reinforces Fitness for Duty expectations.

With respect to individual employee awareness of fitness for duty, OPG's expectations are well understood. Health and Safety principles are outlined in the Business Code of Conduct and fitness for duty is documented in both the Corporate Safety Rules and the Nuclear Operations & Maintenance Handbook.

Limits to the hours of work for OPG Nuclear employees are identified in Appendix A of N-PROC-HR-0002, "Limits of Hours of Work" [59], which is compliant with the Employment Standards Act (ESA) and the expectations of the CNSC. N-PROC-HR-0002 [59] identifies the process for monitoring and controlling the hours of work for OPG Nuclear employees. OPG Nuclear is required to meet both the regulatory expectations of the CNSC, requirements of the ESA, Ontario and Collective Agreement provisions regarding hours of work. Through N-PROC-HR-0002 [59], the hours of work for OPG Nuclear employees are controlled, monitored, reported, and assessed for compliance to both the legislative requirements, as well as the CNSC expectations. The individual assigning work is made aware of the hours worked by the person receiving the assignment and identifies what limits are applicable to the employee. Ongoing monitoring of hours worked is required to ensure that limits are not exceeded.

Conclusion:

The conclusion of this Review Task assessment is that adequate guidance is in place relating to hours of work, health and substance abuse. The intent of Review Task #6 is met and therefore Pickering NGS is compliant.

4.1.7 Review Task #7: Design for Human-Machine Interfaces

Confirm that the human–machine interface is considered in the design of the control room and other workstations, that analysis of human information requirements and task workload is performed, and that there is linkage to the Probabilistic Safety Assessment, Deterministic Safety Analyses and Hazard Analysis. This review should include a discussion of how guidance such as U.S. NRC NUREG-0700 Revision 2, "Human-System Interface Design Review Guidelines" and NUREG-0711 Revision 2, "Human Factors Engineering Program Review Model" identified in CNSC REGDOC-2.5.2 are relevant to the design of Pickering Units 1,4 and Units 5-8.

Review Task #7 is addressed in the Pickering PSR2 Plant Design Safety Factor report.

4.1.8 Review Task #8: Style and Clarity of Procedures

Confirm the style and clarity of procedures provides an appropriate level of detailed guidance for staff through a review of plant events identifying inadequate procedures as a contributing cause.

OPG Standard N-STD-AS-0014, "Requirements for Technical Procedures" [60], specifies requirements for the structure, minimum content, and format of technical procedures. Compliance with this standard is enforced through the use of approved instructions and templates for preparation of procedures. This standard ensures that the technical procedures developed and used throughout OPG Nuclear facilities:

- Promote safe and efficient operation;
- Reflect industry best practice; and
- Document compliance with regulatory requirements.

This standard takes guidance from industry best practice in developing the instructions and requirements for preparation of technical procedures to minimize human errors. Section 4.3.2 of N-STD-AS-0014 [60] lists a number of developmental references including those that provide guidance on writing styles to address human factors. These are:

- American Institute of Chemical Engineers: Guidelines for Writing Effective Operating and Maintenance Procedures, New York, Center for Chemical Process Safety, 1996.
- ANSI/ANS 3.2 – 1994, Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants.
- Blake G, Bly RW: The Elements of Technical Writing, New York: Macmillan USA, 1993.

- Campbell JJ, Zimmerman C: Fundamentals of Procedure Writing, Columbia: GP Publishing, Inc., 1988.
- CSA-N286.5-95, Overall Quality Assurance Program Requirements for Nuclear Power Plants, Section 5.2, Maintenance Procedures, 1995.
- CSA-N286.5-95, Overall Quality Assurance Program Requirements for Nuclear Power Plants, Section 4.2.2 Operating Procedures, Subsection 4.2.2.2, 1995.
- CSA-N286.5-95, Overall Quality Assurance Program Requirements for Nuclear Power Plants, Section 4.2.4, Non-Routine and Emergency Operating Procedures, 1995.
- Haramundanis K: The Art of Technical Documentation, Boston: Digital Press, 1998.
- Parker RC: Looking Good in Print: A Guide to Basic Design for Desktop Publishing, Chapel Hill: Ventana Press, 1988.
- Sabin W, Millar W, Sine S, Strashok GW: The Gregg Reference Manual Fourth Canadian Edition, New York: McGraw-Hill Ryerson, 1995.
- Wieringa D, Moore C, Barnes V: Procedure Writing, Principle and Practices, Columbus: Battelle Press, 1998.

The standard provides detailed guidance for style and clarity based on inputs from these developmental documents. Based on these developmental documents, a set of detailed instructions (OPG-MAN-08100-0001, "Template Creation and Maintenance" [61]) has been developed. Once a procedure has been prepared, reviewed and approved, before it is filed in Asset Suite and made available for use, a complete final quality check of the document verifying that the minimum acceptance criteria per OPG-PROC-0178, "Controlled Document Management" [62], is performed.

Availability of the templates, which have been developed based on industry best practices, ensures style and clarity that enhance performance and simplify communication.

Achieving procedural clarity and an appropriate level of detailed guidance in procedures is also ensured through use of the procedure validation process and promotion of a healthy Technical Procedure Action Request (TPAR) process. All new procedures and major revisions to existing procedures are validated either in the simulator or in the field prior to use as per N-PROC-AS-0028 R018 [63]. Per Section 1.5 of N-PROC-AS-0028, the validation process ensures that documents are correct, meet the intended function, and are usable by a qualified individual. Staff are encouraged to provide input to procedures through the TPAR process. Enhancements and improvements identified while using a procedure (OPEX) are captured through submission of a TPAR to revise the procedure in accordance with N-PROC-AS-0028 [63].

In addition to the requirements for technical procedures, OPG Standard OPG-STD-0001, "Requirements for Administrative Guidance Documents" [64], specifies the requirements for administrative governance documents. The standard provides the criteria for selecting the correct document type for a particular use and the standard format for administrative governance documents (e.g., specifying the use of a template to ensure all necessary sections are included). The standard also references OPG Guide OPG-GUID-08130-0001, "Writing Guide for Administrative Governance Documents" [65], which provides recommendations for creating or revising Governance Documents.

To address the Review Task requirement of conducting a "review of plant events that identify inadequate procedures [procedural quality] as a contributing cause," a search through the Pickering Station Condition Record (SCR) database was conducted and a number of SCRs were identified that potentially pointed to clarity and style of procedures as a contributing factor in plant events. In all cases, these issues have since been addressed through evaluation and revision of the relevant procedures.

The Pickering SCR database was queried for procedural quality as a causal factor for events over the 5 year period of 2011 to 2015. Specifically, the following two causal factors were queried: Administrative Procedures and Technical Procedures. These causal factors are applied to identify adverse conditions associated with procedure creation and maintenance processes.

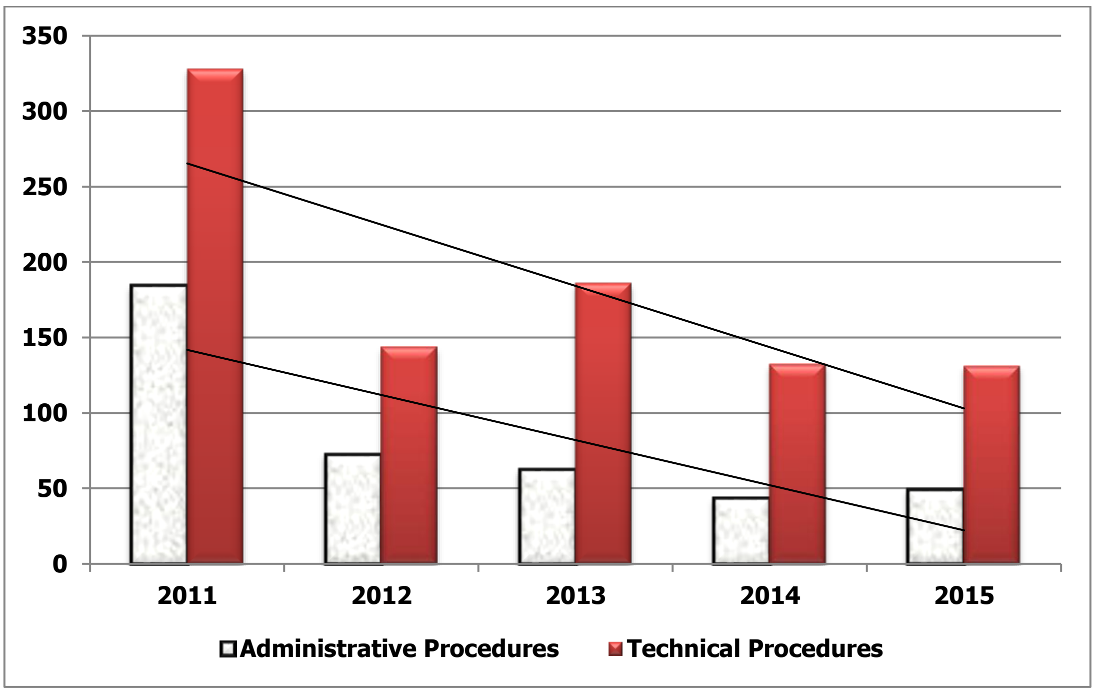
The search yielded a number of SCRs, which were screened to identify SCRs with conditions potentially caused directly or indirectly by clarity issues or lack of detail in procedures, based on the SCR description. Of note, the majority of the conditions in this subset of SCRs were addressed prior to a plant event occurring, through the use of event free tools and the Corrective Action process. As such, this subset was screened further to identify conditions resulting in plant events; this resulted in identification of only 13 SCRs with procedure style and clarity as a potential contributor to plant events (over the 5 year period). A summary of the events, grouped by event type, is presented in the table below.

Plant Event Type	**SCR**
Unexpected System / Component Response	P-2011-12569 P-2013-02471 P-2014-24071 P-2015-09903
Delay in the Completion of Work	P-2011-02619 P-2011-02678 P-2011-11931 P-2011-16400 P-2011-23006
Chemistry Out of Specification	P-2011-12104

Plant Event Type	SCR
Equipment Performance / Impact	P-2013-06430 P-2014-12737
Radiation Protection Performance	P-2011-09359

Most of these procedure issues were attributed to either a lack of sufficient detail, or inconsistencies between various reference documents and procedures. However, as noted earlier, in all cases these issues have since been addressed through evaluation and revision of the relevant procedures.

The results from the SCR database query for SCRs with causal factors relating to procedural quality over the 5 year period 2011 to 2015 are illustrated in the chart below. The trend over this time period demonstrates continuing improvement.



Pickering SCRs with Causal Factors Related to Procedural Quality

Conclusion:

A review of the Pickering SCR database was conducted to identify adverse station conditions potentially caused by procedural clarity and style issues over the past 5 years. Only 13 SCRs were identified where procedural style and clarity issues were potential contributing factors that led to plant events. For each of these SCRs, corrective actions were taken to remedy the situation and close any potential clarity gaps found within OPG procedures. The 5 year trend demonstrates continuing improvement in this area. Therefore, the conclusion of this Review Task assessment is that the style and clarity of procedures provides an appropriate level of detailed guidance for staff, as confirmed through a review of plant events identifying inadequate procedures [procedural quality] as a potential contributing factor. The intent of Review Task #8 is met and therefore Pickering NGS is compliant.

4.2 L/R/C/S Reviews

Per Section 2.2 of this report, detailed reviews for four L/R/C/Ss with content applicable to Safety Factor 12 are provided in Reference [6]. Associated findings applicable to Safety Factor 12 are summarized in Table 3 below.

Table 3: PSR2 L/R/C/S Review Results for Safety Factor 12

L/C/R/S Reviewed	**PSR2 L/R/C/S Review Results for Safety Factor 12**
CNSC G-323 (2007), "Ensuring the Presence of Sufficiently Qualified Staff at Class I Nuclear Facilities-Minimum Shift Complement"	There are no PSR2 gaps for CNSC G-323 (2007). Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CNSC G-323 (2007).
CNSC G-278 (2003), "Human Factors Verification and Validation Plans"	There are no PSR2 gaps for CNSC G-278 (2003). Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CNSC G-278 (2003).
CNSC G-276 (2003), "Human Factors Engineering Program Plans"	There are no PSR2 gaps for CNSC G-276 (2003). Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CNSC G-276 (2003).
CSA N290.12-14, "Human Factors in Design for Nuclear Power Plants"	There are no PSR2 gaps for CSA N290.12-14. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CSA N290.12-14.

4.3 OPG Program Effectiveness Reviews

The OPG Nuclear Program reviewed for Safety Factor 12 is identified in Table 2, and details of the associated effectiveness reviews for the N-PROG are provided in Appendix B.

4.4 Additional Review Findings

As discussed in Section 3.4, the PSR2 Safety Factor 12 assessment also included a review of commitments previously made to the CNSC, open CNSC action items, and exemptions granted by the CNSC, as identified in the R04 Pickering LCH [4], to determine if there are any impacts associated with operation of the Pickering Units past 2020. The review also included identification and review of previously identified programmatic Darlington PSR1 gaps related to Safety Factor 12 to determine impacts associated with operation of the Pickering Units past 2020. This assessment did not identify any gaps for Safety Factor 12.

Findings from the review of previously identified PSR1 gaps in the Pickering Units 5-8 Continued Operations Plan [11] are provided in a separate PSR2 COP Review Report. Findings from the review of Fukushima Action Items are provided in a separate PSR2 FAI Review Report. Results from the Continued Operations Plan and Fukushima Action Items reviews will be considered in the Global Assessment process.

There were no PSR2 gaps identified in this Safety Factor 12 report that require discussion in other Safety Factor reports.

5.0 RESULTS AND CONCLUSIONS

OPG Governance, Programs, Policies, Procedures, Instructions and Guidelines related to Safety Factor 12 were reviewed for the eight PSR2 Review Tasks in Section 4.1 of this report and resulted in no PSR2 gaps. L/R/C/S and OPG Nuclear Program effectiveness reviews for Safety Factor 12 were prepared per Sections 4.2 and 4.3, respectively, and resulted in no PSR2 gaps. Per Section 4.4, this report also included identification and review of previously identified programmatic Darlington PSR1 gaps related to Safety Factor 12 (to ascertain the implications of extending Pickering NGS operation beyond 2020), as well as a review of the R04 Pickering LCH [4] for any impacts of Pickering NGS operation beyond 2020 on a) OPG commitments previously made to the CNSC, b) open CNSC action items, and c) exemptions granted by the CNSC (all related to Safety Factor 12), which resulted in no PSR2 gaps.

The review of Safety Factor 12 has confirmed that the various human factors that may affect the safe operation of Pickering NGS have been appropriately addressed.

6.0 REFERENCES

[1] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[2] CNSC REGDOC-2.3.3, *Periodic Safety Reviews*, April 2015.

[3] IAEA Safety Guide No. SSG-25, *Periodic Safety Review for Nuclear Power Plants*, 2013.

[4] CNSC Report, LCH-PNGS-R004, *Pickering NGS: Licence Conditions Handbook*, December 23, 2015.

[5] OPG Report, P-REP-03680-00003 R000, *Pickering NGS Periodic Safety Review 2 (PSR2): Definition of Safety Factor Review Tasks*, May 30, 2016.

[6] U.S. NRC, NUREG-0700 R02, *Human-System Interface Design Review Guidelines*, May 2002.

[7] U.S. NRC, NUREG-0711 R02, *Human Factors Engineering Program Review Model*, February 2004.

[8] OPG Report, P-REP-03680-00021 R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 8, 10, 12, 13, and 14,* December 2016.

[9] OPG Correspondence, P-CORR-03680-0607223 R000, *Pickering PSR2 – Change to Review Type for CSA N290.12*, July 25, 2016.

[10] OPG Program, N-PROG-TR-0005 R016, *Training*, January 5, 2016.

[11] OPG Plan, NK30-PLAN-00531-00001 R005, *Pickering 5-8 Continued Operations Plan*, December 2015.

[12] OPG Program, N-PROC-RA-0023 R018, *Fleetview Program Health and Performance Reporting*, August 2013.

[13] OPG Report, NK38-REP-03680-10185 R002, *Darlington NGS Integrated Implementation Plan (IIP)*, April 30, 2015.

[14] OPG Instruction, P-INS-09100-00003 R009, *Pickering Minimum Shift Complement*, December 2014.

[15] OPG Instruction, P-INS-09260-00008 R007, *Duty Crew Minimum Complement Assurance*, February 2013.

[16] OPG Program, N-PROG-OP-0001 R008, *Nuclear Operations*, December 2015.

[17] OPG Instruction, N-INS-03490-10003 R000, *Minimum Shift Complement Resources, Qualifications and Procedures Required For Responding to Resource Limiting Events*, November 2013.

[18] OPG Instruction, N-INS-08920-10002 R006, *Simulator-based Initial Certification Examinations for Shift Personnel*, December 2013.

[19] OPG Instruction, N-INS-09110-10059 R004, *Simulator Performance Observation and Crew Critiques,* April 2015.

[20] OPG Training and Qualification Description, N-TQD-503-00001 R017, *Nuclear Emergency Response Organization Training and Qualification Description*, January 2016.

[21] OPG Procedure, N-PROC-TR-0021 R011, *Training and Qualification Description Development and Approval Process*, June 27, 2016.

[22] OPG List, N-LIST-08920-10001 R008, *Trained Performance Areas*, March 2, 2016.

[23] OPG Procedure, N-PROC-TR-0008 R020, *Systematic Approach to Training*, March 31, 2016.

[24] OPG Standard, N-STD-TR-0001 R018, *Conduct of Training*, May 29, 2012.

[25] OPG Training and Qualification Description, N-TQD-602-00001 R017, *Nuclear Trainer Training and Qualification Description*, April 1, 2015.

[26] OPG Procedure, N-PROC-TR-0002 R007, *Control of Vendor-Supplied Training*, November 4, 2013.

[27] OPG Instruction, N-INS-08920-10001 R004, *Requalification Testing of Certified Shift Personnel*, October 24, 2012.

[28] OPG Procedure, N-PROC-TR-0041 R011, *TIMS II Administration*, March 12, 2013.

[29] OPG Procedure, N-PROC-TR-0044 R006, *Training Demand Scheduling and Cancellation Process,* October 16, 2014.

[30] OPG Instruction, N-INS-08920-10017 R005, *Training Committees*, October 15, 2015.

[31] OPG Report, NK38-REP-03680-10080-ADD-001 R000, *Addendum to the Management Safety Factor Report for Darlington ISR,* April 2014.

[32] OPG Manual, N-MAN-08131-10000, *Approved Roles (multiple sheets)*, 2005-2015.

[33] OPG Guide, N-GUID-08930.12-10000 R000, *Nuclear Operators – Recruitment Guide*, April 17, 2007 (confidential).

[34] OPG Training and Qualification Description, N-TQD-403-00001 R011, *Nuclear Engineering Support Personnel Training and Qualification Description*, April 28, 2015.

[35] OPG List, N-LIST-01300-10000 R008, *Bounded Document Set*, November 17, 2014.

[36] OPG Report, http://www.opg.com/news-and-media/Reports/Sustainable_Development_Report_2014.pdf, *2014 Sustainable Development Report*, 2015.

[37] OPG Guide, N-GUID-00130-10002 R001, *Guide for Engineering Knowledge Retention*, October 8, 2010.

[38] OPG Instruction, N-INS-08920-10021 R000, *Continuing and Requalification Training - Curriculum Development and Implementation Process*, March 23, 2009.

[39] IAEA EG2, *Competency Assessments for Nuclear Industry Personnel*, April 2006.

[40] OPG Procedure, N-PROC-TR-0013 R008, *Remedial Training*, June 27, 2011.

[41] OPG Policy, N-POL-0001 R003, *Nuclear Safety Policy*, April 7, 2014.

[42] OPG Charter, N-CHAR-AS-0002 R018, *Nuclear Management System*, March 6, 2015.

[43] OPG Standard, N-STD-AS-0020 R014, *Nuclear Management Systems Organizations*, May 19, 2016.

[44] OPG Program, N-PROG-MA-0004 R011, *Conduct of Maintenance*, May 5, 2015.

[45] OPG Program, N-PROG-MP-0007 R012, *Conduct of Engineering*, October 26, 2012.

[46] OPG Training and Qualification Description, N-TQD-601-00001 R017, *Leadership and Management Training and Qualification Description*, May 4, 2015.

[47] OPG Program, N-PROG-AS-0002 R016, *Human Performance*, May 17, 2016.

[48] OPG Standard, N-STD-OP-0012 R004, *Conservative Decision-Making*, October 11, 2012.

[49] OPG Procedure, N-PROC-OP-0005 R012, *Pre-job Briefing And Post-job Debriefing*, June 6, 2013.

[50] OPG Procedure, N-PROC-OP-0001 R008, *Conduct of Infrequently Performed Tests or Evolutions*, May 27, 2016.

[51] OPG Standard, N-STD-AS-0002 R015, *Procedure Use and Adherence*, September 2, 2015.

[52] OPG Standard, N-STD-OP-0002 R003, *Communications*, June 16, 2014.

[53] OPG Procedure, N-PROC-AS-0077 R007, *Nuclear Safety Culture Assessment*, October 23, 2014.

[54] OPG Lesson Plan, N-LP-65556-00001 R006, *Safety Culture for Managers*, May 2013.

[55] OPG Overhead, N-OVH-65556-00001 R013, *Nuclear Safety Culture for Managers*, February 2014.

[56] OPG Overhead, N-OVH-71256-00007 R000, *First Level Manager Program – Part 1, Module 1 – Observation*, March 2016.

[57] OPG Document, Corporate Safety Rules, *Corporate Safety Rules*, April 15, 2014.

[58] OPG Training, N-CMT-62808-00001 R001, *Continuous Behaviour Observation Program (CBOP) Participants Materials - Workbook Components*, May 12, 2004.

[59] OPG Procedure, N-PROC-HR-0002 R004, *Limits of Hours of Work*, August 2, 2012.

[60] OPG Standard, N-STD-AS-0014 R007, *Requirements for Technical Procedures*, December 10, 2014.

[61] OPG Manual, OPG-MAN-08100-0001 R003, *Template Creation and Maintenance*, January 11, 2013.

[62] OPG Procedure, OPG-PROC-0178 R001, *Controlled Document Management*, March 14, 2016.

[63] OPG Procedure, N-PROC-AS-0028 R018, *Development, Review and Approval of Technical Procedures*, June 21, 2016.

[64] OPG Standard, OPG-STD-0001 R006, *Requirements for Administrative Governance Documents*, June 30, 2016.

[65] OPG Guide, OPG-GUID-08130-0001 R001, *Writing Guide for Administrative Governance Documents*, January 23, 2014.

Appendix A: Nomenclature

CAL	Computer Assisted Learning
CNO	Chief Nuclear Officer
CNSC	Canadian Nuclear Safety Commission
COP	Continued Operations Plan
CRC	Curriculum Review Committees
DOM-O	Director of Operations and Maintenance – Operations
ERO	Emergency Response Organisation
ESA	Employment Standards Act
FAI	Fukushima Action Item
IAEA	International Atomic Energy Agency
ISR	Integrated Safety Review
LCH	Licence Conditions Handbook
L/R/C/S	Laws, Regulations, Codes and Standards
MAR	Minimum Availability Requirement
MCCP	Minimum Complement Compliance Program
MSC	Minimum Shift Complement
N-PROG	Nuclear Program
NGS	Nuclear Generating Station
NPDS	Nuclear Professional Development Seminar
NOIT	Nuclear Operator In Training
OPEX	Operating Experience
OPG	Ontario Power Generation
PARTS	Pickering A Return to Service
P. Eng	Professional Engineer
PSR	Periodic Safety Review
PSR1	Periodic Safety Review 1 (earlier OPG PSR work and other associated assessments)
PSR2	Periodic Safety Review 2 (subsequent PSR per CNSC REGDOC-2.3.3)
QG	Qualification Guide
SAT	Systematic Approach to Training
SCR	Station Condition Record

SNPM	Senior Nuclear Plant Manager
SOER	Significant Operating Experience Report
TQD	Training Qualification Description
TPAR	Technical Procedure Action Request

Appendix B: OPG Program Effectiveness Review Results

B.1 N-PROG-TR-0005, "Training"

The Training Program is key to improving nuclear station safety, reliability, cost effectiveness and worker competence. The program is used to develop and maintain competent personnel to safely operate, maintain and improve plant performance, to minimize the impact of plant operation on the environment, health and public safety, and to drive human performance improvement.

In general, the Training Program content originated from the following sources:

- Canadian Nuclear Safety Commission objectives and criteria for regulatory evaluations of training, licence conditions and applicable standards;
- World Association of Nuclear Operators;
- United States Department of Energy Training Handbooks and Guides to Good Practices;
- Institute of Nuclear Power Operations; and
- Training procedures used at other nuclear power generating stations.

Nuclear Oversight conducted an audit in April 2012, NO-2012-0005 [B.1.1], of the Training Program for both Pickering and Darlington NGS. The objective of the audit was to confirm that the Training Programs at Ontario Power Generation (OPG) Nuclear are being effectively managed and are in compliance with governing documents. The audit concluded that there were performance improvement opportunities applicable to Pickering NGS in the areas of process inputs for training, sustainability of the long term staffing strategy for non-licensed operator training, some instances of delivery of training by staff prior to full completion of their Trainer qualification designation and governance associated with the Job Coach function.

Four SCRs were initiated to address the above findings (SCR N-2012-02030, N-2012-02027, N-2012-02028 and N-2012-02029), which required corrective actions to be implemented. These SCRs (and the associated Action Requests) have since been closed and the necessary corrective actions were completed to address the underlying issues.

The Nuclear Programs and Training division completed a self-assessment in July 2012, NO12-000415-SA [B.1.2], in order to assess the health of the governance framework for the Training Program for both Pickering and Darlington NGS. This involved a review of related SCRs, governance framework, Asset Suite and revision records. No findings/SCRs were generated as a result of this self-assessment.

References

[B.1.1] Nuclear Oversight Audit, N-REP-01070-0409230, *Audit OPGN NO-2012-005, Training Program,* April 20, 2012.

[B.1.2] Self-Assessment, *Program Management Assessment – N-PROG-TR-0005, Training,* NO15-000415-SA, July 31, 2012.

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ONTARIO POWER GENERATION	
ACCEPTED	✓
ACCEPTED AS NOTED	
REVISE AND RESUBMIT	
Signature	**Date** 14 Dec 2016.
Name: Mike Ruffolo, Manager	
Dept: Pickering Engineering - Aging Management & Strategic Initiatives	
OPG Proprietary	
Doc No.: P-REP-03680-00017	**Rev: 000**
This acceptance does not relieve the contractor from responsibility for errors or omissions or from any obligations or liability under this contract.	

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Pickering NGS PSR2 Safety Factor 13 Report: Emergency Planning

PS112/RP/017 R01

December 13, 2016

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Revision Summary – For Amec Foster Wheeler Report PS112/RP/017

Rev	Date	Author	Comments
R00	July 15, 2016	J. Morris, R. Jayasundera	Initial issue for OPG review and comment.
R01	December 13, 2016	J. Morris, R. Jayasundera	Updated report addressing OPG comments on R00.

EXECUTIVE SUMMARY

Ontario Power Generation (OPG) is performing a Periodic Safety Review (PSR) to support the possibility of continued operation of Pickering Nuclear Generating Station (NGS) beyond 2020. The PSR (referred to as "PSR2") is a subsequent PSR building on the review basis of earlier OPG Integrated Safety Reviews and other associated assessments. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

Part of PSR2 involves the preparation of Safety Factor reports for each of fifteen major topic areas. Safety Factor reports consist of:

- Assessing compliance against "Review Tasks" identified in the PSR2 Basis Document [1]. These Review Tasks are derived from Canadian Nuclear Safety Commission (CNSC) REGDOC-2.3.3, "Periodic Safety Reviews" [2] and International Atomic Energy Agency (IAEA) SSG-25, "Periodic Safety Review for Nuclear Power Plants" [3];

- Documenting assessments against applicable modern Laws, Regulations, Codes and Standards (L/R/C/Ss) as defined in Reference [1]; and

- Reviewing the effectiveness of applicable OPG Nuclear Programs that support the above assessments, through review of audit and self-assessment results.

The PSR2 review of Safety Factor 13, *Emergency Planning* is presented in this report. OPG Governance, Programs, Policies, Procedures, Instructions and Guidelines related to Safety Factor 13 were reviewed for the eleven PSR2 Review Tasks specified in Section 4.1 of this report. L/R/C/S and OPG Nuclear Program effectiveness reviews for Safety Factor 13 were prepared per Sections 4.2 and 4.3, respectively. Per Section 4.4, the PSR2 assessment includes a review of previously identified PSR1 gaps related to Safety Factor 13 (to ascertain the implications of extending Pickering NGS operation beyond 2020), as well as a review of the R04 Pickering Licence Conditions Handbook [4] for any impacts of Pickering NGS operation beyond 2020 on: a) OPG commitments previously made to the CNSC, b) open CNSC action items, and c) exemptions granted by the CNSC (all related to Safety Factor 13).

The results of the review of Safety Factor 13 are discussed in Section 5.0. The review has confirmed that OPG Nuclear has: a) adequate plans, staff, facilities and equipment in place for dealing with emergencies, and b) there are adequate arrangements in place for regular emergency training and exercises, and interaction and coordination with local and national authorities. The review identified two gaps that will need to be addressed further as part of the PSR2 Global Assessment process.

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LIST OF TABLES AND FIGURES

1.0 INTRODUCTION

Ontario Power Generation (OPG) is performing a Periodic Safety Review (PSR) to support the possibility of continued operation of Pickering Nuclear Generating Station (NGS) beyond 2020.[1] A comprehensive Integrated Safety Review (ISR) was completed for Pickering Units 5 through 8 in 2009 in support of refurbishment and continued operation. Pickering Units 1,4 integrated safety assessments were also performed for Pickering A Return to Service (PARTS) in support of approval to restart Units 1 and 4. In addition to these Pickering-specific studies, the 2013 Darlington ISR performed extensive code and standard reviews that were updated in relation to the versions that were assessed in the 2009 Pickering B ISR.[2] These previous ISRs are considered to constitute the first PSR completed for Pickering (referred to as "PSR1"). The current PSR (referred to as "PSR2") is a subsequent PSR building on the basis of earlier OPG integrated safety assessments through review of the various studies, assessments and licence renewals performed since PSR1. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

PSR2 will support and complement the licence renewal application for Pickering NGS going forward. Fifteen Safety Factors will be assessed as part of the PSR. The purpose of Safety Factor reviews is to confirm that the design, condition and operation of Pickering Units 1,4 and 5-8 (as well as common systems) will support continued safe operation for the period of PSR2 by:

- Assessing compliance against "Review Tasks" identified in the PSR2 Basis Document [1], which were derived from Canadian Nuclear Safety Commission (CNSC) REGDOC-2.3.3 [2] and International Atomic Energy Agency (IAEA) SSG-25 [3];

- Documenting assessments against applicable modern Laws, Regulations, Codes and Standards (L/R/C/Ss) (as defined in Reference [1]); and

- Reviewing the effectiveness of applicable OPG Nuclear Programs that support the above assessments, through review of audit and self-assessment results.

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1 Currently, Pickering Units 5-8 are approved to operate to 247,000 Effective Full Power Hours. This operation limit is expected to be reached on some units in 2020. For the purposes of PSR2, OPG assumes operation of Pickering NGS for up to eight additional years, from 2020 until 2028. OPG will make a decision regarding the permanent shut down dates for the six reactors following the performance of a technical evaluation that will include PSR2, and will communicate it to the CNSC as required by the current Power Reactor Operating Licence (PROL).

2 Much of the compliance assessment and evaluation of Safety Factor health for the Darlington ISR is based on programs and practices that apply across OPG's nuclear operations. As a result, where Pickering is confirmed to follow the same nuclear programs and practices as were assessed for Darlington, the Darlington ISR programmatic conclusions are applicable to Pickering. As discussed in Section 1.0, an effectiveness review (effectiveness at Pickering NGS) of OPG programs used to demonstrate compliance with the PSR2 Assessment Basis will be conducted using recent audit and self-assessment results.

As outlined in IAEA SSG-25 [3], the objective of the review of Emergency Planning Safety Factor 13 is to determine: "(a) whether the operating organization has in place adequate plans, staff, facilities and equipment for dealing with emergencies; and (b) whether the operating organization's arrangements have been adequately coordinated with the arrangements of local and national authorities and are regularly exercised." REGDOC-2.3.3 [2] requires that: "The licensee shall conduct a PSR in accordance with this regulatory document for the period until the next PSR or, if applicable, until the end of commercial operation of the plant."

This report documents the results of the review of Safety Factor 13 for Pickering PSR2. The report is based on the OPG Governance, Programs, data, and material available up to January 15, 2016 which is the freeze date for PSR2.

2.0 SCOPE OF REVIEW

2.1 Review Task Assessments

The Pickering PSR2 Safety Factor 13 Review Tasks are defined in Reference [1]. Details of the derivation of these Review Tasks from CNSC REGDOC-2.3.3 [2] and IAEA SSG-25 [3] are shown in Reference [5]. The Safety Factor 13 Review Tasks are:

1) Confirm the full range of accidents and radiation emergencies have been identified and studied.

2) Confirm the appropriate response and mitigation strategies have been developed and have taken account of major changes at site and around the site (industrial, commercial, residential development).

3) Confirm that the station organization includes dedicated Emergency Response personnel on duty at the plant at all times, to handle accidents and emergencies.

4) Assess the adequacy of the training program for emergency response personnel including training, emergency exercises and qualification records.

5) Confirm there is a process for notification of staff that will be brought in to assist in the management of the response in the longer term.

6) Determine that there is a classification of accidents to guide the type of response.

7) Confirm there is a mechanism for notifying and informing relevant off-site organizations such as the police, fire departments, hospitals, ambulance services, regulatory bodies, local authorities, government, public welfare authorities and the news media.

8) Confirm the availability of sufficient communications equipment at the plant and at the off-site Emergency Centre to permit effective communications with Emergency Response Teams, both on and off site.

9) Assess adequacy of the emergency response procedures and training and exercises for all site staff. Confirm that integrated and partial emergency exercises have been conducted to check satisfactory function of the emergency organization and its equipment.

10) Confirm the adequacy of on-site equipment and facilities for emergencies and offsite emergency facilities or locations, including walkdowns of relevant areas on and off the site.

11) Confirm development or existence of a program for Severe Accident Management.

The methodology for the reviews is discussed in Section 3.1. Review Task findings are summarized in Section 4.1 of this report.

2.2 L/R/C/S Reviews

The applicable Laws, Regulations, Codes and Standards relevant to the Emergency Planning Safety Factor are identified in Reference [1] and are listed in Table 1 below. Table 1 also identifies the modern version and date of each L/R/C/S to be considered, the Safety Factor(s) to which each document is applicable, and the type of review that will be completed in PSR2.

All of the Safety Factor 13 L/R/C/S reviews are high level or incremental in nature. The definitions of High Level Review and Incremental Review are as follows:

- High Level Review: New L/R/C/Ss not referenced in Pickering PROL 48.02/2018 but which are in the PSR2 Assessment Basis will be subject to a high level review. In a high level review, the degree of conformance with clauses or groups of clauses in the L/R/C/S is demonstrated by supporting evidence stating whether the intent of the requirements stipulated in the requirement document is met; and

- Incremental Review: For L/R/C/Ss that have been reviewed in PSR1 but have had revisions since the last review, a topical review will be performed of the changes.

The methodology for the reviews is discussed in Section 3.2. A detailed assessment for each L/R/C/S[3] is provided in Reference [6]. Associated findings are summarized in Section 4.2 of this report.

Table 1: L/R/C/Ss Reviewed for Emergency Planning Safety Factor 13

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
L/R/C/Ss Referenced in Pickering NGS PROL 48.02/2018						
1	CSA N293[3]	Fire Protection for Nuclear Power Plants	N293-12	1, 7, 13	Incremental	N293 addressed as part of Pickering B and Darlington ISRs, as well as PARTS code reviews.

[3] The PSR2 review of CSA N293-12 is in progress. As discussed in Section 4.2, gaps identified from this review will be applicable to the Plant Design Safety Factor, and hence, results are presented in the Pickering NGS PSR2 Plant Design Safety Factor Report.

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
2	CNSC REGDOC-2.10.1*	Nuclear Emergency Preparedness and Response	2014	13	Incremental	Transition plan in place and gap assessment has been performed by OPG.
Additional L/R/C/Ss						
3	CSA N1600	General Requirements for Nuclear Emergency Management Programs	N1600-14	13	High Level	Not referenced in PROL 48.02/2018. Not reviewed as part of Pickering B or Darlington ISRs.

* Superseding documents to those currently in Pickering NGS PROL 48.02/2018.

2.3 OPG Program Effectiveness Reviews

The OPG Nuclear Program (N-PROG) reviewed for Safety Factor 13 is listed in Table 2 below.[4] The methodology for the effectiveness reviews is discussed in Section 3.3. The assessment results of the N-PROG in Table 2 are provided in Appendix B, and findings are summarized in Section 4.3.

Table 2: OPG Programs Reviewed for Safety Factor 13

Document Number	Document Title
N-PROG-RA-0001 [7]	Consolidated Nuclear Emergency Plan

2.4 Additional Reviews

The PSR2 Safety Factor 13 report includes a review of the R04 Pickering Licence Conditions Handbook (LCH) [4] for any impacts of Pickering NGS operation beyond 2020 on the following (all related to Safety Factor 13):

- OPG commitments previously made to the CNSC;
- Open CNSC action items; and
- Exemptions granted by the CNSC.

[4] The list of Nuclear Programs to be assessed for effectiveness for PSR2 was derived from review of current OPG Governance. Although there may be content in Nuclear Programs that is applicable to multiple Safety Factors, N-PROG reviews are only provided in one Safety Factor report and are not duplicated.

The PSR2 assessment includes identification and review of previously identified PSR1 gaps related to Safety Factor 13 to ascertain the implications of extending Pickering NGS operation beyond 2020. The methodology for these reviews is described in Section 3.4. Any PSR2 gaps identified as a result of programmatic Darlington PSR1 gaps related to Safety Factor 13 are discussed in Section 4.4 of this report. The review of Pickering PSR1 gaps previously identified in the Pickering Units 5-8 Continued Operations Plan (COP) [8] is provided in a separate PSR2 COP Review Report.

In addition, Fukushima Action Items (FAIs) were reviewed to identify implications of extending operation beyond 2020 (if any). This review is presented in a separate PSR2 FAI Review Report.

Any PSR2 gaps identified as a result of the Safety Factor 13 review which need to be addressed in other Safety Factor reports are discussed in Section 4.4 of this report.

3.0 METHODOLOGY

The sub-sections below summarize the methodology used to assess Review Tasks, L/R/C/Ss, and Nuclear Program effectiveness for the Emergency Planning Safety Factor.

3.1 Review Tasks

As discussed earlier, the Safety Factor Review Tasks are derived from CNSC REGDOC-2.3.3 [2] and IAEA SSG-25 [3], taking into consideration the Review Tasks used in the Pickering B and Darlington ISRs (as derived in [5]).

For each Safety Factor 13 Review Task identified in Section 2.1, a confirmation of the existence of applicable OPG Governance, Programs, Policies and Procedures (as well as Instructions and Guidelines, as applicable) was performed. Compliance against Review Tasks is also assessed by reference to applicable Condition Assessments, safety analyses and operating experience, as required.

The Review Task assessments identify Compliances and Gaps as defined below:

- <u>Compliance</u>: Compliance indicates that either the safety requirement or the intent of the Review Task is met.

- <u>Gap</u>: A Gap indicates that the intent of the Review Task is not met.

3.2 L/R/C/S Reviews

The process to identify the modern L/R/C/Ss that are applicable to the PSR2 Assessment Basis involved first creating a broad list from multiple sources (potential candidate L/R/C/Ss) and then filtering it to identify those that are most significant and that are applicable to the PSR2 scope. The identification and selection criteria are detailed in Reference [1]. The result of the identification and selection process was a set of modern L/R/C/Ss that became part of the "PSR2 Assessment Basis".

PSR2 is focused on the extension of Pickering NGS operations beyond 2020, and will conduct reviews against a baseline of past PSR1 work. As a subsequent PSR, PSR2 focuses on changes in requirements, plant conditions, operating experience and new information. Since PSR2 is an update of previous ISRs, it incorporates reviews of L/R/C/Ss that have occurred as new versions have been issued. Since this assessment is a subsequent PSR, the focus is on identifying differences between what was previously assessed and what is now different within the current Pickering PSR2 Assessment Basis. In general, these differences relate to:

- More recent (new or revised) L/R/C/S versions than what was previously assessed as part of PSR1;[5]

- Safety significant differences between Pickering and Darlington, if the Darlington ISR is the basis for the earlier assessment;

- Implications of extending Pickering NGS operation beyond 2020; and

- Safety significant differences between Pickering Units 1,4 and Units 5-8.

As described in Reference [1], L/R/C/S review types are clause-by-clause, high level or incremental. Most of the L/R/C/Ss in the PSR2 Assessment Basis receive incremental reviews since PSR2 is an update of previous PSR1 assessments and clause-by-clause or high level reviews for the majority of the L/R/C/Ss in the PSR2 Assessment Basis have already been completed. Implementation plans (including gap analyses or code-over-code reviews) also exist for the latest editions of many L/R/C/Ss. As a result, incremental review is also used in circumstances where a L/R/C/S in the PSR2 Assessment Basis was not assessed in previous PSR1 reviews but an implementation plan currently exists for compliance.

The PSR2 incremental reviews in this report include an assessment of the intent of recent changes to the L/R/C/Ss on a topic or subject-matter basis where there is potential to impact nuclear safety. Incremental reviews provide:

- A summary of the purpose of the L/R/C/S;

- Pertinent background information about the current revision of the L/R/C/S that is being considered;

- Identification of which Safety Factor(s) are applicable to the current revision of the L/R/C/S;

- A description of which version(s) of the L/R/C/S were assessed for PSR1 (i.e., Darlington ISR (for programmatic content), Pickering B ISR and PARTS code reviews);

- Identification of whether the current version of the L/R/C/S is an update of a previous version of the L/R/C/S that was assessed in PSR1 (and if so, a description of the major changes in the latest revision is provided as discussed below);

[5] "New" refers to a code or standard that was not previously considered in the context of earlier assessments. "Revised" refers to an updated version of a code or standard that was previously considered in the context of earlier assessments. Where a document has a new number/type, but addresses the same topic from the same organization, it is a "revised", not "new", document (e.g., if a REGDOC replaces a CNSC G or RD document).

- An assessment of the applicability of PSR1 assessment findings (gaps and conclusions), including the implications of extending Pickering NGS operation beyond 2020 if any;

- An assessment of the applicability of assessment findings that address more recent (post-PSR1) editions of the L/R/C/S, including any implementation or transition plans that are already committed to by OPG; and

- Where PSR1 and post-PSR1 assessments are not sufficient to address changes in the latest edition of the L/R/C/S, an assessment of the changes from the previously assessed edition of the L/R/C/S (including identification of any safety significant PSR2 gaps which result).

High Level reviews provide the same information as above, where applicable, in a similar format. However, given that High Level L/R/C/Ss generally have not received past assessment during PSR1, the Incremental review content is augmented by a high level, section-by-section assessment of the degree of conformance of Pickering NGS with the L/R/C/S (demonstrating, with supporting evidence, whether the intent of the requirements stipulated in the document are met).

There are currently no L/R/C/S clause-by-clause reviews identified in the PSR2 Assessment Basis.

The Safety Factor 13 L/R/C/S reviews identify Compliances and Gaps as defined below:[6]

- <u>Compliance</u>:
    - Where an Incremental review has been performed, Compliance indicates that the change in the safety requirement, per the topical review, is met.
    - Where a High Level review has been performed, Compliance indicates that the intent of the safety requirement is met.
    - For Clause-by-Clause reviews of modern L/R/C/Ss, Compliance indicates that the safety requirement is met. (Note: No Clause-by-Clause reviews were performed as part of Safety Factor 13.)

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6 Safety Factor assessments for Review Tasks and L/R/C/S reviews make use of: a) OPG Governance, Programs, Policies and Procedures which support the assessment arguments, b) Commitments previously made to the CNSC, open CNSC action items, and exemptions granted by the CNSC (all related to the Safety Factor under review), as identified in the R04 Pickering LCH [4], c) Identification of previously identified Pickering-specific or programmatic PSR1 gaps related to the Safety Factor under review and the status of OPG's improvement plan(s) or other dispositions to address these, and d) Assessments and reviews performed since the PSR1 documents were completed.

- Gap:
    - Where an Incremental review has been performed, a Gap indicates that the change in the safety requirement, per the topical review, is not met.
    - Where a High Level review has been performed, a Gap indicates that the intent of the safety requirement is not met.
    - For Clause-by-Clause reviews of modern L/R/C/Ss, a Gap indicates that the safety requirement is not met. (Note: No Clause-by-Clause reviews were performed as part of Safety Factor 13.)

The reviews assume that use of the word:

- "Shall" is used in an L/R/C/S to express a requirement, i.e., a provision that the licensee is obliged to satisfy in order to comply with the standard;
- "Should" is used to express a recommendation or that which is advised but not required;
- "May" is used to express an option or that which is permissible within the limits of the standard; and
- "Can" is used to express possibility or capability.

3.3 OPG Program Effectiveness Reviews

As discussed earlier, effectiveness reviews (effectiveness at Pickering NGS) of OPG programs used to demonstrate compliance with the PSR2 Assessment Basis were conducted, using recent applicable audit and self-assessment results:

- OPG Nuclear Oversight independent performance-based Program audits (typically performed in 1 to 5 year cycles) and self-assessments. This includes review of associated Station Condition Records and Action Requests to determine the status of any resulting corrective actions; and
- CNSC "Type I" and "Type II" inspections of the effectiveness and performance of OPG programs, where discussed in OPG audits or self-assessments.

There are many audits and self-assessments that are performed to assess the effectiveness of important aspects of each program. A sample of audits and self-assessments has been summarized for each program in order to demonstrate that program effectiveness is being assessed on an ongoing basis. The focus of these reviews was on effectiveness of the programs at Pickering NGS, where specific information is available. Results from these audits and self-assessments will be considered in the Global Assessment process. It is noted that audits and self-assessments are, by their nature, self-critical and are used to drive excellence in

performance. As a result, the broad review scope of program audits focuses on identifying improvement opportunities rather than presenting a balanced picture of program performance.

Program effectiveness is also monitored and addressed through the Fleetview Program Health and Performance Reporting process [9]. This process involves direct oversight by the Chief Nuclear Officer, and includes participation by the Nuclear Executive Committee members. Programs are reviewed, senior oversight is provided, and improvement plans are generated.

The list of Nuclear Programs to be assessed for each Safety Factor was derived from review of current OPG Governance, and has used the most recent version of these documents as of the PSR2 freeze date of January 15, 2016.

3.4 Additional Reviews

A review of the R04 Pickering LCH [4] was also performed to determine if there are any impacts associated with operation of the Pickering Units past 2020 on the following (all related to Safety Factor 13):

- Commitments previously made to the CNSC;
- Open CNSC action items; and
- Exemptions granted by the CNSC.

The PSR2 assessment includes identification and review of previously identified Pickering-specific or programmatic PSR1 gaps related to Safety Factor 13 (as identified in the Darlington ISR Integrated Implementation Plan [10] and Pickering Units 5-8 Continued Operations Plan [8]) to ascertain the status of OPG's improvement plan(s) or other dispositions to address these and the implications of extending operation beyond 2020 (if any).[7]

[7] PSR2 includes consideration and confirmation that the findings of PSR1 remain valid, as applicable, for the operation period. This includes assessment of PSR1 conclusions against implications resulting from extended operation. In particular, Pickering PSR1 results are applicable to PSR2 if there was a PSR1 gap that is still open, or if a closed PSR1 gap could be affected by extended operation. If so these gaps are carried forward into PSR2 for consideration in the Global Assessment. When references to PSR1 are made, the source document is identified and the relevant text from that source document is summarized in the context of PSR2. With respect to the Darlington ISR, much of the evaluation of Safety Factor health is based on programs and practices that apply across OPG's nuclear operations. As a result, Darlington ISR programmatic conclusions are applicable to Pickering PSR2 where Pickering is confirmed to follow the same nuclear programs and practices that were assessed for Darlington. Darlington PSR1 results are applicable to Pickering PSR2 if there are Darlington PSR1 gaps that are found to be relevant to Pickering PSR2.

Fukushima Action Items were reviewed to identify implications of extending operation beyond 2020 (if any). The methodology for this review is provided in a separate PSR2 FAI Review Report.

Any PSR2 gaps identified as a result of the Safety Factor 13 review which need to be addressed in other Safety Factor reports are also discussed.

4.0 REVIEW FINDINGS

4.1 Review Tasks

The sub-sections below provide an assessment of the adequacy of applicable OPG Governance, Programs, Policies and Procedures (as well as Instructions and Guidelines, as applicable) in demonstrating compliance against the Safety Factor 13 Review Tasks.

4.1.1 Review Task #1: Range of Accidents and Radiation Emergencies

Confirm the full range of accidents and radiation emergencies have been identified and studied.

The Consolidated Nuclear Emergency Plan (CNEP) provides a written basis to document concepts, roles, and resources required by OPG Nuclear to implement and maintain its emergency response capability to protect the public, employees, and the environment in the event of a nuclear emergency [7]. The CNEP defines a nuclear emergency as an emergency which poses an actual or potential hazard to public health and property or the environment from ionizing radiation, whose source is a major nuclear installation.

The emergency response capability established in accordance with the CNEP must be sufficiently flexible to be used for a broad range of events both within and beyond the design basis. That is, the full range of accidents and radiation emergencies spans a wide range of events from an impairment of a plant system to the occurrence of a Beyond Design Basis Accident (BDBA) that progresses into a Severe Accident (SA).

The basis for emergency planning is described in Section 1.1 of the CNEP [7]. Developmental references for both on-site and off-site planning are listed in Section 4.2.2 of the CNEP [7]; this list represents the primary source from which the emergency plan is developed. Two of the documents listed in this section are the Pickering NGS 1-4 Station Safety Report, NA44-SR-01320-00002, "PN 1-4 Safety Report: Part 3 – Accident Analysis" [11] and the Pickering NGS 5-8 Station Safety Report, NK30-SR-01320-00003, "PN 5-8 Safety Report: Part 3 – Accident Analysis" [12].

The Pickering NGS 1-4 and Pickering NGS 5-8 Safety Reports, References [11] and [12] respectively, identify and study the full range of accidents and radiation emergencies that are a part of the station design basis. As part of Review Task #3 of the Deterministic Safety Analysis Safety Factor Report, an assessment is performed to determine if the postulated events, event sequences, and event combinations covered by the existing safety analysis are sufficient when compared against those for a modern Nuclear Power Plant (NPP) in accordance with the methodology in CNSC REGDOC-2.4.1, "Deterministic Safety Analysis" [13].

In addition to the analyzed events in the Safety Reports, the ongoing Operating Experience process described in N-PROC-RA-0035, "Operating Experience Process" [14] monitors events around the world to determine if there is any unforeseen event that may have applicability to Pickering NGS. If/when an event is deemed to be applicable, the emergency response process is reviewed to ensure the response to such an event would be adequate. Enhancements to the emergency response would be considered as a part of this process.

The CNEP [7] also recognizes the need to address beyond design basis events, including Design Extension Conditions and SAs. Requirements for OPG's approach to BDBA management are documented in N-STD-MP-0019, "Beyond Design Basis Accident Management" [15], with BDBA strategies implemented through Emergency Mitigating Equipment Guidelines (EMEGs) and Severe Accident Management Guidance (SAMG). A more detailed discussion of the EMEGs and SAMG is provided as part of Review Task #11 (Section 4.1.11).

Conclusion:

The conclusion of this Review Task assessment is that there is governance in place to ensure that the full range of accidents and radiation emergencies have been identified and studied. The intent of Review Task #1 is met and therefore Pickering NGS is compliant.

4.1.2 Review Task #2: Development of Response and Mitigation Strategies

Confirm the appropriate response and mitigation strategies have been developed and have taken account of major changes at site and around the site (industrial, commercial, residential development).

Response and Mitigation Strategies

OPG's emergency response and mitigation strategies are documented in Section 1.2.3 of the CNEP [7]. Response and mitigation strategies consist of a combination of on-site and off-site actions, as dictated by the nature of the emergency.

During the initial emergency phase, Main Control Room (MCR) staff perform the assessment of plant status and where possible, the identification of damage to plant equipment. Appropriate plant procedures are used and MCR staff initiate an immediate operations response to strive towards taking the plant to a safe and stable configuration. These procedures include normal and non-standard operating procedures contained in Operating Manuals and Abnormal Incidents Manual (AIM) procedures. Each of the AIM procedures has specific entry conditions and entry into an AIM procedure is potential entry into the regime of on-site Emergency Response Organization (ERO) implementing procedures. P-MAN-03490-00005, "Index – Emergency Response Manual 5" [16] contains a listing of implementing procedures for various ERO roles.

The MCR team utilizes resources of the on-site Emergency Operations Centre (EOC) to mobilize and deploy the necessary emergency teams. The EOC personnel may, if necessary, mobilize, brief, and deploy Emergency Response Teams to address specific emergencies, such as a fire, medical emergency, or a search and rescue operation. Appropriate off-site support groups (i.e., non-OPG resources) such as fire and/or ambulance may be activated to respond to the site. In addition, the EOC personnel may mobilize and deploy both in-plant and off-site radiation survey teams who shall provide additional data to assess the plant status. If the MCR emergency mitigation strategy requires repair of damaged equipment, then the EOC personnel will assemble, brief, and deploy the appropriate emergency repair teams. MCR staff will seek, as appropriate, consultative, technical, and resource assistance from the Site Management Centre (SMC) who will in turn seek, as appropriate, assistance from the Corporate Emergency Operations Facility (CEOF).

During the response phase, the MCR and SMC staff will, as a continuing action, evaluate the implementation of the emergency mitigation strategy and modify it as necessary.

As per Section 1.3.2 of the CNEP [7], agreements exist with local fire departments for on-site fire-fighting support. Arrangements and procedures also exist for local ambulance service and hospital support for casualties from nuclear sites. These agreements are also documented in Section 1.17.4 of N-PROG-RA-0012, "Fire Protection" [17]. Toronto Hospital Corporation, Western Division has been provincially designated and funded as the radiation trauma centre for Ontario. This includes the capability to deal with contaminated casualties, trauma, and acute radiation syndrome. Rouge Valley Ajax and Pickering Hospital is the primary local hospital designated to receive contaminated casualties from Pickering NGS.

In 2012, a Mutual Aid Agreement between Canada's five major nuclear operators was formalized [18]. The agreement outlines the type of emergency support that may be provided, and the processes involved if a nuclear operator suffers a major emergency and requires assistance. For example, if there is a need for additional Emergency Mitigating Equipment (EME), OPG can contact other Canadian nuclear facilities (i.e., Bruce Power, NB Power, or Canadian Nuclear Laboratories) to ascertain what equipment is available for support [19]. Specific to the use of EME, OPG also has the capability to share some of this equipment between Pickering and Darlington depending on actions required as part of the event response.

Arrangements for support and technical assistance exist within the CANDU Owners Group (COG) and the Institute of Nuclear Power Operations (INPO). INPO operates a 24-hour emergency assistance line and an Emergency Response Centre in Atlanta to provide support to member utilities. As required, INPO will coordinate with the Electric Power Research Institute as part of providing this support. INPO also issues an Emergency Resources Manual with listings of specialized services and equipment and contact phone numbers for all US Nuclear utilities. This manual is available in the CEOF and on-line. Provisions for emergency engineering and technical support are available from Amec Foster Wheeler, CANDU Energy Inc. and Atomic Energy of

Canada Ltd. Similar contracts are made available by COG for securing support from other Canadian nuclear utilities.

As per Section 1.3.3 of the CNEP [7], OPG shall assist the province and designated municipalities in their planning and preparedness for a nuclear emergency, and carry out and assist the province in conducting studies to enhance public safety during a nuclear emergency. As prescribed in the Provincial Nuclear Emergency Response Plan (PNERP) [20], the designated municipalities for Pickering NGS are Durham Region and the City of Toronto. The host municipality for Pickering is the City of Peterborough.

There are external off-site organizations defined in provincial and municipal nuclear emergency plans that OPG interfaces and interacts with during the course of an emergency response. These external organizations are as follows:

1) Office of the Fire Marshal and Emergency Management

    - Provincial Emergency Operations Centre (PEOC)

      The PEOC is located at the Ministry of Community Safety and Correctional Services, Office of the Fire Marshal and Emergency Management, in Toronto and is the provincial facility and organization that directs off-site emergency response operations. OPG provides call-in staff to fill an official liaison officer position in the PEOC Operations Section, some technical positions in the PEOC Scientific Section, and to support environmental radiation monitoring techniques. OPG provides and maintains software codes for dose projection, dedicated telecommunications links, and training and drills as requested to support the PEOC. During the response stage, site shift and management staff make emergency notification to the PEOC as appropriate. A regular flow of technical data and situation updates, including off-site survey data, is provided to the PEOC from the incident site.

2) Ministry of Health and Long Term Care

    In response to a nuclear emergency, the Ministry of Health and Long Term Care activates the Ministry Emergency Operations Centre and implements the Radiation Health Protection Plan in accordance with the PNERP. The Radiation Health Protection Plan is designed to guide health sector planning at both the provincial and local levels across Ontario in order to effectively respond to a nuclear emergency.

    The Ministry of Health and Long Term Care coordinates with the PEOC to provide advice and guidance regarding the implementation of precautionary and protective actions. This includes deciding when to administer potassium iodide (KI) pills for thyroid blocking and ensuring information on the source and type of radiation is communicated to health care facilities in a timely manner. OPG staff located in the PEOC may interact (as required) with Ministry of Health and Long Term Care staff as part of the event response.

3) Durham Region

- Regional EOC

  The Regional EOC is comprised of elected officials, regional department staff (e.g., roads and works, emergency planning, social services), school board staff, police, fire, and medical representatives. The primary responsibility of this organization is to implement the public protective action directives in the locally affected area to protect the public. Only the PEOC provides direction and information on off-site response to the Regional EOC.

  OPG provides a local site representative to fulfill the OPG liaison officer position at the Regional EOC. The liaison officer provides coordination with the site for resources and off-site response activities in the local area. The liaison officer provides interpretation of the significance of radioactivity measurements and technical background information to the regional staff. Official emergency communication with the Regional EOC is through the liaison officer and the SMC.

- Reception Centres and Emergency Worker Centres

  Reception Centres are established to provide, among other functions, a Monitoring and Decontamination Unit (MDU) for evacuees. These centres are under the control and direction of the Regional EOC. OPG Nuclear site organization and call-in personnel staff and equip the MDU function of these centres. MDU staff members are trained to carry out radiation monitoring and decontamination. OPG MDU supervisors maintain communication with the SMC to keep the ERO apprised of the MDU status and any need for further resources.

  Emergency Worker Centres are established during nuclear emergencies to monitor and control radiation exposure of external emergency workers who may be required to enter areas affected by radiation. Similar to Reception Centres, Emergency Worker Centres are under the authority of the Regional EOC. Emergency Worker Centres are staffed by OPG call-in personnel with equivalent qualifications as Reception Centre MDU staff. Communication is regularly maintained between the Emergency Worker Centre OPG supervisors and the SMC and Regional EOC.

4) City of Toronto

- Municipal EOC

  A Municipal EOC is equivalent to a Regional EOC with regards to its role in the emergency response. OPG provides the same level of support to the City of Toronto Municipal EOC as it does to the Durham Region Regional EOC.

- Reception Centres and Emergency Worker Centres

  Reception Centres and Emergency Worker Centres associated with the City of Toronto have the same functionality as corresponding centres for the Durham Region. OPG provides the same level of support to the City of Toronto Reception Centres and Emergency Worker Centres as it does for corresponding centres in the Durham Region.

5) City of Peterborough

   - Reception Centre

     The City of Peterborough Reception Centre has the same functionality as corresponding centres for the Durham Region and the City of Toronto. OPG provides the same level of support to the City of Peterborough Reception Centre as it does for corresponding centres in the Durham Region and the City of Toronto.

6) CNSC

   The CEOF interfaces with the CNSC EOC and its respective staff.

As noted in Section 1.3.4 of the CNEP [7], the PNERP requires Pickering NGS to procure adequate quantities of stable iodine tablets for their Primary Zone population. Other operational responsibilities related to stable iodine tablets are prescribed in the Ministry of Health and Long Term Care's Radiation Health Response Plan. In consultation with the designated municipalities, OPG Emergency Preparedness (EP) Department procures stable iodine tablets and maintains them within expiry dates. Distribution of iodine tablets is the responsibility of the Durham Region and the City of Toronto. Iodine tablets were pre-distributed to the Primary Zone population for Pickering NGS in 2015.

Designated Primary Zone municipalities are also required to establish and maintain a public alerting system in accordance with the PNERP. This requirement is discussed in further detail under Review Task #7.

Changes at Site and Around the Site

To take into account major changes at site and around the site, OPG Approved Roles Document, N-MAN-08131-10000 Sht: S4-0245, "Manager, Emergency Preparedness" [21] provides specific accountability to the Department Manager of EP to interface with the regulator and other external stakeholders as necessary to ensure that the relevant regulatory and licensing requirements are built into the EP program and stakeholder interfaces are effectively managed.

The Department Manager of EP is a member of the Durham Regional Emergency Management Coordinating Committee as well as the Provincial Nuclear Emergency Management Coordinating Committee. These committees meet on a regular basis and review issues that could impact emergency response, including local development which may modify/increase risks.

In addition, OPG's Operating Experience process monitors events around the world to determine if there is any unforeseen event that may have applicability to Pickering NGS. If/when an event is deemed to be applicable, the emergency response process is reviewed to ensure the event can be adequately dealt with.

Appropriateness of Response and Mitigation Strategies

Response and mitigation strategies are developed in accordance with the deterministic safety analysis and with insights from the probabilistic safety assessment. Thus, the appropriateness of these strategies is dependent on the adequacy of the existing safety analysis, which is evaluated in the following Safety Factor reports:

1) Deterministic Safety Analysis

2) Probabilistic Safety Assessment

3) Hazards Analysis

A review of the reports listed above did not identify any PSR2 gaps which are applicable to the Emergency Planning Safety Factor report.

The adequacy of the response and mitigation strategies that have been developed is demonstrated primarily through drills and exercises. On an annual basis, the EP Department assesses the ERO performance to the established objectives identified in N-INS-03490-10002, "Conduct of Emergency Preparedness Drills and Exercises" [22] and reviews all drill and exercise related Station Condition Records (SCRs) and Action Requests (ARs) to monitor status and ensure completeness. These assessments are documented in the EP Drill and Exercise Performance Objectives Reports. Deficiencies found during drills and exercises are documented using the SCR system in accordance with N-PROC-RA-0022, "Processing Station Condition Records" [23]. Corrective Action plans are developed and corrective actions are initiated and tracked to completion.

A review of the latest performance objectives report, N-REP-03490-10038, "2014 Emergency Preparedness Drill and Exercise Performance Objectives Report" [24] indicates that all corrective actions in 2014 were correctly dispositioned and being tracked to completion, if not already completed.

Conclusion:

The conclusion of this Review Task assessment is that response and mitigation strategies have been developed and there is governance in place to monitor the appropriateness of these strategies, accounting for major changes at site and around the site (industrial, commercial, residential development). The intent of Review Task #2 is met and therefore Pickering NGS is compliant.

4.1.3 Review Task #3: Emergency Response Personnel

> *Confirm that the station organization includes dedicated Emergency Response personnel on duty at the plant at all times, to handle accidents and emergencies.*

Sections 1.6.2 and 2.2.1 of the CNEP [7] specify the requirement for OPG to put an administrative control in place that ensures the shift minimum complement is maintained at all times. N-PROC-RA-0046, "Emergency Response Organization Staffing and Availability" [25] provides an organized method of selecting, staffing, and maintaining the ERO to satisfy the requirements specified in the CNEP. The ERO includes on-shift staff and on-call personnel that are activated to respond to the site and OPG Nuclear emergency response centres. The minimum positions required for the Shift ERO are identified in Figure 4 of the CNEP [7].

Further details of the staffing levels for the duty shift are provided in P-INS-09100-00003, "Pickering Minimum Shift Complement" [26] and P-INS-09260-00008, "Duty Crew Minimum Complement Assurance" [27]. These documents also define responsibilities and processes including confirming on a shift-to-shift basis that the Duty Crew ERO are fully qualified for their assigned roles, as identified in the Training Information Management System. This is confirmed using the Minimum Complement Coordination Program. Use of these tools and processes ensures that the shift minimum complement is met at all times at Pickering NGS.

Conclusion:

The conclusion of this Review Task assessment is that there are administrative controls in place to ensure that dedicated Emergency Response personnel are on duty at the plant at all times, to handle accidents and emergencies. The intent of Review Task #3 is met and therefore Pickering NGS is compliant.

4.1.4 Review Task #4: Adequacy of Emergency Response Training Program

> *Assess the adequacy of the training program for emergency response personnel including training, emergency exercises and qualification records.*

Training and Qualification Records

OPG's ERO training and qualification process is described in Section 1.6.3 of the CNEP [7]. Specifically, the CNEP refers to N-TQD-503-00001, "Nuclear Emergency Response Organization Training and Qualification" [28], which establishes the training and qualification requirements for individuals assigned to the ERO. It is based on the nuclear industry best practice, Systematic Approach to Training, which utilizes a task-based analysis to determine the training requirements and a performance-based analysis to verify effectiveness. The Training Information Management System is used to maintain the training program for OPG Nuclear and is the official database for documenting qualification records associated with personnel assigned to ERO positions.

As per Sections 1.1 and 1.2 of N-TQD-503-00001 [28], personnel assigned to ERO positions are selected based on skills and knowledge they already possess by way of their normal work assignments. Initial training is provided to ensure that ERO personnel have the knowledge and skills needed to independently perform tasks associated with the identified ERO position.

Section 1.5 of N-TQD-503-00001 [28] describes requirements related to continuing training. Continuing training is provided to maintain and enhance knowledge, skills and performance standards required to perform tasks of ERO positions. It also facilitates confirmation that incumbents still possess knowledge and skills required for correct execution of tasks associated with assigned emergency response roles. ERO personnel participate in at least one practical training component every 18 months.

Emergency Exercises

Section 1.6.5.2 of the CNEP [7] refers to N-PROC-RA-0045, "Emergency Preparedness Drills and Exercises" [29], which defines drill and exercise requirements, including activity and frequency requirements.

The EP Drill and Exercise Program provides a means of validating the effectiveness of OPG Nuclear's emergency response capability to ensure the safety of employees, the public, and the environment in the event of a nuclear emergency. The program serves the following purposes:

- Develop and maintain the skills of the ERO.
- Test the effectiveness of emergency plans and procedures, facilities and equipment, and training.
- Demonstrate the adequacy of plans and preparedness to respond to events ranging from minor to severe accidents.

The EP Drill and Exercise Program is comprised of Self Assessed Crew Practices (SACPs), Evaluated Drills, and Evaluated Exercises.

SACPs provide an opportunity for shift crews to develop and maintain their skills, and are planned and executed by the crews, under the direction of the Shift Manager (SM). A SACP should involve participation of:

- EOC including the SM and Emergency Shift Assistant (ESA)[8];
- In-Plant Survey Teams; and
- Off-Site Survey Teams.

SACPs are not identified in the annual drill and exercise schedule; they are scheduled by Operations in consultation with Work Control and are on the station work planning

[8] Per P-INS-03491-00002, the Shift Advisor Technical will perform the role of the ESA [33].

schedule. Minimum frequency requirements for SACPs are specified in N-PROC-RA-0045 [29].

OPG's drill and exercise program is conducted as per N-INS-03490-10002, "Conduct of Emergency Preparedness Drills and Exercises" [22], which describes the overall process, requirements, and guidance for the conduct of nuclear emergency response drills and exercises. It is based on INPO 14-0003 Emergency Drill and Exercise Guidelines [30], with appropriate modifications to reflect specific OPG and regulatory requirements.

As per Section 1.1.1 of N-INS-03490-10002 [22], a drill and exercise schedule is developed annually for each OPG Nuclear site. The approved schedule is provided to the Province, Durham Region, the CNSC, and the City of Toronto. The intent of sharing this schedule is to encourage participation of these key off-site agencies in the drills and exercises. The minimum frequency requirements for drills and exercises are identified in Table 1 of N-PROC-RA-0045 [29].

A drill is an evaluated demonstration by multiple participants of actions taken in response to a simulated emergency. A drill includes interactions between individuals or emergency response facilities that would be expected to occur in an emergency. Performance is evaluated against established objectives and demonstration criteria. A drill may be full-scale, involving testing of a full ERO, or it may be focused on one specific area. A focus-area drill is limited in scope and may involve simulation of facilities or responders. These include a single facility drill, tabletop drills, and mini drills.

Emergency exercises test the adequacy of EP programs and the implementation of emergency measures. This includes an evaluation of the adequacy of the procedures and training of the ERO to respond to an emergency. Emergency exercises simulate emergency events and conditions over a minimum of several hours, in order to test the integrated performance of the EP program.

Nuclear emergency response drills are the most frequently conducted evaluated drills, demonstrating the response capability to a simulated nuclear emergency as identified in the CNEP. Drill scenario development may include a multi-disciplinary team to ensure drill fidelity and realism. An evaluated drill may involve the participation of some of or the entire ERO, and may also include station staff, Provincial and Regional off-site authorities, and/or the CNSC.

Nuclear emergency response exercises involve a full ERO complement and most or all of the station's emergency response facilities. Exercises are conducted at each site once every three years and where practical include participation of Provincial and Regional off-site authorities and the CNSC (subject to their availability). Exercises involve a multi-disciplinary scenario development working group and are run from the station simulators to ensure drill fidelity and realism.

In order to support the successful execution of drills and exercises, a team is established comprised of controllers and evaluators. Controllers are responsible for establishing drill conditions, controlling participant actions, and in certain cases coaching participants. Evaluators are responsible for evaluating participant performance against scenario specific objectives and criteria, and assessing the adequacy of plans, procedures, facilities, and equipment.

Adequacy of Training Program for Emergency Response Personnel

The adequacy of the training program for emergency response personnel is primarily demonstrated through drills and exercises, and supplemented by self-assessments. On an annual basis, the EP Department assesses the ERO performance to the established objectives identified in N-INS-03490-10002, "Conduct of Emergency Preparedness Drills and Exercises" [22] and reviews all drill and exercise related SCRs and ARs to monitor status and ensure completeness. These assessments are documented in the EP Drill and Exercise Performance Objectives Reports. Deficiencies found during drills and exercises are documented using the SCR system in accordance with N-PROC-RA-0022, "Processing Station Condition Records" [23]. Corrective Action plans are developed and corrective actions are initiated and tracked to completion.

A review of the latest performance objectives report, N-REP-03490-10038, "2014 Emergency Preparedness Drill and Exercise Performance Objectives Report" [24] indicates that all corrective actions in 2014 were correctly dispositioned and being tracked to completion, if not already complete.

Conclusion:

There is adequate governance in place to support the execution of relevant activities including training, emergency exercises, and qualification records that are required as part of an adequate training program for emergency response personnel. The intent of Review Task #4 is met and therefore Pickering NGS is compliant.

4.1.5 Review Task #5: Process for Notification of Staff

Confirm there is a process for notification of staff that will be brought in to assist in the management of the response in the longer term.

The ERO is activated by the ESA who is directed by the SM as per P-INS-03491-00001, "Incident Shift Manager" [31]. The process to activate the ERO is outlined in Appendix C of P-INS-03491-00002, "Incident Emergency Shift Assistant" [33].

The ERO activation is initiated using the ERO notification system (MIR3 Intelligent Notification System), which is an automated electronic system. Should the ERO notification system fail, a manual call-out is conducted by SMC responders as specified in Section 2.0 of N-MAN-03491-10000 [34]. The ERO contact information is maintained current in N-MAN-03491-10000 [34].

Staff that requires this sort of notification is referred to as the augmented OPG ERO and is required to assist in the management and the response in the longer term. This is referenced accordingly in Sections 1.2.2 and 2.2 of the CNEP [7]. In accordance with the ERO duty schedule, on-duty ERO staff is divided into three duty teams identified by a colour: Red Team, Green Team, and Blue Team. Each duty team is "on-duty" for a period of two consecutive weeks in a six-week cycle. The duty team that is "on-duty" at the time of the notification would initially respond to the event. However, in the event of a widespread loss of communications or off-site power in Ontario, it is expected that all SMC and CEOF ERO staff (including staff not on duty and in the on-call pool) will immediately report to their facility if it is safe to do so.

Depending on the nature of the event, the ERO may be active for an extended period of time such that it is necessary to relieve staff who was initially called in as part of the initial response. As per Section 2.6 of N-GUID-03491-50010, "Emergency Response Organization Expectations" [35], on receipt of a Real Event notification of abnormal incident or higher, all responders are expected to respond to the call-out (i.e., all on-call staff would respond to the notification, with only staff from the "on-duty" colour team assembling at the appropriate ERO facility), and remain available and fit for duty in order to ensure staffing for subsequent shifts if necessary. Logistics related to co-ordinating a shift change for the ERO, as required, are the responsibility of the Resource Deployment Manager. N-INS-03491-10022, "Resource Deployment Manager" [36], provides direction for the activities required by the Resource Deployment Manager to co-ordinate the shift change, including the identification and notification of ERO staff who will be providing relief.

Depending on the severity of the event, it may be necessary to activate the Crisis Management and Communications Centre (CMCC). The CMCC is an off-site facility whose primary function is to provide corporate leadership and executive level decisions related to any incident requiring CMCC activation until the emergency is mitigated and a recovery strategy is in place. The decision to activate the CMCC is made by the Chief Nuclear Officer in consultation with the Emergency Recovery Director who is located in the CEOF. Once the decision has been made to activate the CMCC, Security personnel will initiate the CMCC notification process in accordance with OPG-PROC-0028, "Crisis Management & Communications Centre Procedure" [37].

Conclusion:

The conclusion of this Review Task assessment is that there exists a process for the notification of staff that will be brought in to assist in the management of the response in the longer term. The intent of Review Task #5 is met and therefore Pickering NGS is compliant.

4.1.6 Review Task #6: Classification of Accidents

Determine that there is a classification of accidents to guide the type of response.

Emergency assessment and classification of accidents are performed as described in Section 1.2.2 of the CNEP [7]. Event classification is performed to determine the extent of the on-site response mobilization and the required notification. The process of notifying and informing government agencies is also outlined in Section 1.2.2 of the CNEP.

As specified in Section 1.2.2.1 of the CNEP [7], the first action is the identification, classification and, if appropriate, declaration of a station emergency. A station emergency is defined as a sudden unexpected occurrence of unusual radiological conditions with the potential for accidental exposure to staff or the public exceeding regulatory limits. A station emergency can also be declared for a non-radiological event requiring protection of on-site personnel and activation of the ERO to deal with the event. Entry into the declaration of a station emergency will arise from a diagnosis in response to a reactor or plant process upset or abnormal condition. Once it is recognized that a threat to plant staff or the public exists, on-site mobilization of the ERO is required. The station emergency tone would be activated to inform staff to assemble and account and to initiate immediate mobilization of the shift ERO, and if appropriate, ERO augmentation.

The Shift Manager is responsible for the initial classification of the event and the required response, as described in Section 1.3 of P-INS-03491-00001, "Incident Shift Manager" [31]. The event is classified based on Appendix A, "Emergency Classification, Categorization, and Notification" of this instruction.

Once the process of emergency classification has been completed, a second process is undertaken to determine the Provincial notification category. The assessment process for off-site categorization is governed by the criteria set by the PNERP [20]. This is translated procedurally into an off-site notification criteria matrix, which has specific entry conditions for each category of notification.

The Shift Manager is also responsible for determining the provincial notification category, as described in Section 1.4 of P-INS-03491-00001, "Incident Shift Manager" [31]. The Provincial notification category is determined based on Appendix B, "Notification Criteria Matrix" of this instruction. Specifically, an event is categorized as a Reportable Event, Abnormal Incident, On-Site Emergency, or General Emergency.

In addition to the initial classification of the event and identification of the Provincial notification category, a provisional International Nuclear Event Scale (INES) rating is prepared by the OPG INES Officer under the direction of the CEOF Technical Support Director as per N-INS-03491-10001, "CEOF Technical Support Director" [32]. Once the provisional INES rating has been approved by the appropriate ERO staff it is provided to the CNSC. The provisional INES rating is under CNSC jurisdiction and is approved and released to external stakeholders by the CNSC INES National Officer.

Conclusion:

The conclusion of this Review Task assessment is that there exists a process to classify accidents to guide the type of response. The intent of Review Task #6 is met and therefore Pickering NGS is compliant.

4.1.7 Review Task #7: Notification of Off-Site Organizations

Confirm there is a mechanism for notifying and informing relevant off-site organizations such as the police, fire departments, hospitals, ambulance services, regulatory bodies, local authorities, government, public welfare authorities and the news media.

Section 1.2.2.5 of the CNEP [7] identifies off-site emergency notifications to external agencies (provincial, municipal, CNSC) as one of the three important notification actions that need to occur in order to ensure an effective emergency response takes place. As specified in Section 1.2.2.2 of the CNEP [7], once the process of emergency classification has been completed, a second process is undertaken to determine the Provincial notification category. The assessment process for off-site categorization is governed by the criteria set by the PNERP. This is translated procedurally into an off-site notification criteria matrix, which has specific entry conditions for each category of notification. Appendix A, "Provincial Notification Categories" of the CNEP [7] contains generic definitions of PNERP notification categories. Appendix B, "Notification Criteria Matrix" of the CNEP [7] contains the Notification Criteria Matrix.

Site-specific derivatives of this matrix are included in site procedures to simplify the determination of notification category (e.g., process failures identified in the plant's AIMs that have been pre-categorized into the Provincial notification categories).

Off-site emergency actions are made following categorization. The time requirement for the plant to notify the Provincial contact point is within 15 minutes after the event has been categorized. This is the provincially set target for notification performance, and a regulatory requirement. The information identified in the official notification form should be confirmed as part of the notification process. The designated municipal or regional contact points also receive the same emergency notification shortly after the Province. Regulatory notification to the CNSC is made after the off-site agency and ERO notifications have been completed. CNSC notification target time is within 15 minutes of ERO activation.

P-INS-03491-00001 [31] provides direction for the SM to respond to an event in accordance with the requirements contained in the CNEP. Step 1.4 of P-INS-03491-00001 [31] prompts the SM to refer to Appendix B, "Notification Criteria Matrix" to determine the applicable PNERP category for the event. Off-site notifications are then initiated per Step 1.5 of P-INS-03491-00001 [31], which prompts the ESA to initiate notifications per P-INS-03491-00002 [33].

The ESA completes off-site notifications within 15 minutes of event categorization per Step 1.3 of P-INS-03491-00002 [33], which includes the Province of Ontario, Durham Region, and the City of Toronto. Per Step 1.5 of P-INS-03491-00002 [33], the ESA will notify the CNSC within 15 minutes of ERO activation. Once notified, these external organizations ensure that other stakeholder groups under their jurisdiction (e.g., ambulance services, public welfare authorities, etc.) receive the appropriate notifications.

OPG has a responsibility to communicate with the public, media, stakeholders and employees during nuclear emergencies as identified in Section 1.2.5 of the CNEP [7]. To facilitate this, OPG has a plan in place and procedures that govern crisis communications response. OPG also supports the Province and municipalities who provide coordinated communications under the jurisdiction of the PNERP.

The CEOF is established as the primary emergency response operational interface for external agencies and authorities (Province's PEOC and CNSC's EOC). This ensures communication demands on the site response organization are kept to a minimum, allowing a focused site response on incident mitigation and nuclear safety. The CEOF executes its mandate by [7]:

- Managing the overall OPG Nuclear response.
- Mobilizing and coordinating corporate resources.
- Ensuring Provincial and external requirements are met.
- Supporting the CMCC response.
    - As discussed in Section 4.1.5, the primary function of the CMCC is to provide corporate leadership, strategic direction, and executive level decisions to any and all incidents requiring CMCC support until the emergency is mitigated and a long-term recovery strategy is in place. The CMCC is an off-site facility that is staffed by 10 representatives from the various OPG Business Units [37].

N-STD-AS-0010, "Nuclear Crisis Communications Standard" [38] governs public communications on behalf of OPG Nuclear in the event of a nuclear emergency and ensures consistent and coordinated public information response. Crisis communications response is implemented by OPG's Corporate Relations and Communications with support from the SMC and/or the CMCC, or with assistance and guidance from ERO personnel.

There are various ERO roles that support communications with key external stakeholders as part of the event response, as outlined below [7]:

1) Municipal EOC/Regional EOC Liaison Officer

   OPG provides a local site representative at each of the activated Municipal EOCs and Regional EOCs to fulfill the OPG liaison officer position. The liaison officer provides coordination with the site for resources and off-site response activities in the local area. Official emergency communications with Municipal EOC/Regional EOC are through the liaison officer and the SMC.

2) Reception Centres and Emergency Worker Centre Supervisors

   OPG Nuclear site organization and call-in personnel staff and equip the MDU function of the Reception Centres. OPG MDU supervisors communicate with the

SMC to keep the ERO apprised of the MDU status and any need for future resources.

Emergency Worker Centres are staffed by OPG call-in personnel with equivalent qualifications as Reception Centre MDU staff. Communication is regularly maintained between the Emergency Worker Centre OPG supervisors and the SMC and Regional EOC.

3) PEOC – OPG Representative

The OPG representative in the PEOC Operations Section may initially report to the CEOF to receive briefing on the details of the incident. Upon activation of the PEOC, this representative will proceed to the PEOC to act as the official OPG representative and liaison.

Activities of the representative at the PEOC include the following:

- Relaying protective action decisions to the CEOF;
- Communicating provincial requests for services and resources; and
- Identifying operational response issues and potential discrepancies for OPG resolution.

4) Emergency Response Manager

The Emergency Response Manager is located in the SMC and has authority over the site ERO once command and control is assumed. Primary duties of the Emergency Response Manager include:

- Ensuring communication of any upgrades of initial notification categories to provincial and municipal contact points during the first four hours or until such time as formally indicated by the provincial authority.
- Ensuring provision of key decision support information (emergency data transmittal) from the site to the provincial authority.
- Providing review and concurrence of media releases and employee bulletins.
- Any other elements identified for the site by the PNERP (e.g., notification of site evacuation, notification of venting, provision of off-site radiological monitoring, staffing commitments at off-site centres).

5) Emergency Recovery Director

The Emergency Recovery Director is located in the CEOF and has authority to oversee the site response and manage the overall OPG nuclear emergency response. The Emergency Recovery Director has principal accountabilities for communications and issue response with the following:

- CNSC – Director General – Directorate of Power Regulation; and
- PEOC – Commander – Command Section.

OPG also supports the Province and municipalities who provide coordinated emergency communications under the jurisdiction of the PNERP. Per Section 1.3.5 of the CNEP [7], OPG provides resources and assistance to the designated Primary Zone municipalities to enable them to establish and maintain a public alerting system as required by the PNERP.

In addition, the PNERP identifies the requirement for a public education program for the area surrounding the nuclear facilities. The program is coordinated by the Office of the Fire Marshal and Emergency Management and includes consultation with the appropriate stakeholders and advisors, including the designated municipalities and nuclear installations. Section 1.3.6 of the CNEP [7] states that the program message shall be consistent with the PNERP and provide adequate information to recipients to enable them to effectively protect themselves during a nuclear emergency. OPG responsibilities are also identified in Section 1.3.6 of the CNEP [7].

Conclusion:

The conclusion of this Review Task assessment is that there are mechanisms in place for notifying and informing relevant off-site organizations such as the police, fire departments, hospitals, ambulance services, regulatory bodies, local authorities, governments, public welfare authorities, and the news media. The intent of Review Task #7 is met and therefore Pickering NGS is compliant.

4.1.8 Review Task #8: Availability of Communications Equipment

Confirm the availability of sufficient communications equipment at the plant and at the off-site Emergency Centre to permit effective communications with Emergency Response Teams, both on and off site.

The available communications equipment within OPG Nuclear's emergency facilities is described in Section 1.5 of the CNEP [7]. Emergency facilities are equipped with the necessary voice communications equipment, including back-up, and other equipment that includes fax machines, personal computers, status boards, area radiation monitoring equipment, radiation survey kits, off-site monitoring vehicles and meteorological monitoring data readout equipment, as appropriate to the facility. Other support facilities have phone communications equipment, including back-up fax machines and radios as appropriate. All OPG Nuclear emergency facilities have the capability to access WebEOC, which is a web-based information management software used by the ERO that allows real-time information posting and multidirectional communication over an Internet connection. WebEOC is provided on a dedicated server backed up by a server maintained physically separate from the main production server [39].

The Pickering site has a variety of emergency communications equipment, including back-up. The Station Public Telephone System is the primary telephone system. Satellite phones are available in the Pickering 1,4 and Pickering 5-8 Shift Manager Offices, the SMC, and the CEOF in the event that all other telecommunication methods become unavailable. Fax machines equipped with Station Public Telephone System and trunk lines are available. The Pickering site has an emergency radio communications system with dedicated frequencies. On-site and off-site field teams are equipped with cell phones and/or portable radios. Base radio stations are available at a number of on-site locations such as the MCR. Off-site field team vehicles are equipped with mobile communication and a back-up system. While the equipment described above is considered to be sufficient, OPG has a Telecommunications Enhancement Project in progress to equip facilities with additional equipment to further enhance telecommunications [40]. The fact that this project remains in progress is not considered to be a PSR2 gap as the objective of the project is to enhance existing telecommunication capabilities as opposed to addressing a deficiency.

OPG emergency response facilities are linked to the PEOC, Municipal EOC, and Regional EOC through landline phones and other systems to allow information transfer. OPG has also established reliable contingency communications systems (e.g., Nuclear Emergency Telephone System).

P-MAN-03490-00002, "Pickering ERO Equipment and Facility Manual" [41] provides requirements and direction for Pickering ERO facilities configuration management and maintenance to ensure readiness, and to provide contingency actions, in accordance with the CNEP. It provides a description of the SMC, EOC, and SM offices including facility set-up, documentation, communications and other equipment and processes dedicated to or specifically set up for the Pickering ERO. N-MAN-03490-10000, "CEOF Equipment and Facility Manual" [42], provides the requirements and direction for CEOF configuration management and maintenance to ensure readiness, and to provide contingency actions, in accordance with the CNEP. OPG-PROC-0028, "Crisis Management & Communications Centre Procedure" [37], provides the framework for the CMCC during the response to incidents that require CMCC activation. This procedure also provides requirements and direction for CMCC configuration management and maintenance to ensure readiness, and to provide contingency actions, in accordance with the CNEP.

N-PROC-RA-0040, "Maintenance and Testing of Emergency Preparedness Facilities and Equipment" [43] defines the process used to monitor, periodically test and maintain the emergency response facilities and equipment to ensure operability 24 hours a day, 7 days a week. This includes testing and facility walk-through frequencies and covers the different types of equipment in the facilities, such as faxes, computers, radiation instruments, communication, meteorological and data transmitting equipment. A complete list of EP facility and inventory check documents is provided in N-LIST-

03490-10028, “Emergency Preparedness Facility and Inventory Check Documents” [44], which includes Pickering, Darlington, and off-site facilities.

The availability of communications equipment within OPG Nuclear’s emergency facilities is managed under the Equipment Important to Emergency Response (EITER) program. EITER includes systems, structures, and components, as well as essential tools and equipment, necessary to implement the CNEP. The overall management process for EITER is documented in N-PROC-RA-0133, “Management of Equipment Important to Emergency Response” [45].

EITER is listed in the following documents according to its location:

- N-INS-03491-10025, “Unavailability of Equipment Important to Emergency Response – Off-Site” [46];

- D-INS-03491-10000, “Unavailability of Equipment Important to Emergency Response – Darlington” [47];

- P-INS-03491-00050, “Unavailability of Equipment Important to Emergency Response – Pickering” [48].

These instructions also contain the required actions (including compensatory measures) that are to be taken if EITER becomes unavailable. The adequacy of the EITER program is discussed in Review Task #10 of this report.

Conclusion:

The conclusion of this Review Task assessment is that there is governance in place to ensure the availability of sufficient communications equipment at the plant and at the off-site Emergency Centres to permit effective communications with Emergency Response Teams, both on and off site. The intent of Review Task #8 is met and therefore Pickering NGS is compliant.

4.1.9 Review Task #9: Adequacy of Emergency Response Procedures

Assess adequacy of the emergency response procedures and training and exercises for all site staff. Confirm that integrated and partial emergency exercises have been conducted to check satisfactory function of the emergency organization and its equipment.

Procedures

The adequacy of emergency response procedures is demonstrated through drills and exercises. The adequacy of the drills and exercises program is discussed in Review Task #4 of this report.

Section 1.6.1 of the CNEP [7] describes the documentation and governance that has been established to implement the CNEP. All EP documents and their control and periodic review are governed by OPG-PROC-0001, "Process Administrative Governance Documents" [49], and N-PROG-AS-0006, "Records and Document Control" [50]. P-MAN-03490-00005, "Index – Emergency Response Manual 5" [16] lists the suite of procedures and instructions for each of the ERO specific positions.

Training

The adequacy of the training program for emergency response personnel is discussed in Review Task #4 of this report.

OPG's ERO training and qualification is described in Section 1.6.3 of the CNEP [7]. The CNEP refers to N-TQD-503-00001, "Nuclear Emergency Response Organization Training and Qualification" [28], which establishes the training and qualification requirements for individuals assigned to the ERO. It is based on the nuclear industry best practice, Systematic Approach to Training, which utilizes a task-based analysis to determine the training requirements and a performance-based analysis to verify effectiveness. In addition, all site staff receive the Nuclear General Employee Training in accordance with N-TQD-501-00001, "Nuclear General Employee Training and Qualification Description" [51]. The Nuclear General Employee Training is delivered to all employees requiring unescorted access to the nuclear facility so they are knowledgeable in the emergency response actions required of them.

Exercises and Drills

The adequacy of the drills and exercises program for site staff is discussed in Review Task #4 of this report.

Section 1.6.5.2 of the CNEP [7] refers to N-PROC-RA-0045, "Emergency Preparedness Drills and Exercises" [29], which defines drill and exercise requirements, including activity and frequency requirements. On an annual basis, the EP Department assesses the ERO performance in drills and exercises to the established objectives identified in N-INS-03490-10002, "Conduct of Emergency Preparedness Drills and Exercises" [22] and reviews all drill and exercise related SCRs and ARs to monitor status and ensure completeness. These assessments are documented in the EP Drill and Exercise Performance Objectives Reports. Deficiencies found during drills and exercises are documented using the SCR system in accordance with N-PROC-RA-0022, "Processing Station Condition Records" [23]. Corrective Action plans are developed and corrective actions are initiated and tracked to completion.

OPG has conducted several integrated and partial emergency exercises to check satisfactory function of the emergency organization and its equipment. The results of each exercise are documented in an after-exercise report that discusses the response performance for various portions of the ERO and identifies strengths, good practices, observations, opportunities for improvement, and findings specific to each aspect of the response. A selection of relevant reports for Pickering NGS is provided below:

- P-REP-03490-00053, Emergency Preparedness Evaluated Drill Report – Pickering Station Emergency Drill (Jan 23 2013) [52];
- P-REP-03490-00052, Emergency Preparedness – Pickering Emergency Mitigating Equipment (EME) Drill Report (February 6 2013) [53];
- P-REP-03490-00054, Emergency Preparedness Evaluated Drill Report – Pickering Station Emergency Drill (July 3 2013) [54];
- P-REP-03490-00055, Emergency Preparedness – Pickering Nuclear DBA/SAMG Report (September 23 2013) [55];
- P-REP-03490-00056, Emergency Preparedness – Pickering Nuclear DBA/SAMG Report (November 1 2013) [56];
- N-REP-03491.23-10041, Emergency Preparedness Exercise Report – Pickering Station Emergency Exercise 15-Oct-2014 [57]; and
- N-REP-03491.23-10042, Emergency Preparedness Exercise Report – Pickering Station Emergency Exercise – 25Nov2015 [58].

Conclusion:

The conclusion of this Review Task assessment is that there is governance in place to ensure the adequacy of the emergency response procedures and training and exercises for all site staff. It has also been confirmed that integrated and partial emergency exercises have been conducted at Pickering NGS to check satisfactory function of the emergency organization and its equipment. The intent of Review Task #9 is met and therefore Pickering NGS is compliant.

4.1.10 Review Task #10: Adequacy of On-Site Equipment and Facilities

Confirm the adequacy of on-site equipment and facilities for emergencies and offsite emergency facilities or locations, including walkdowns of relevant areas on and off the site.

A range of equipment and facilities, located both on-site and off-site, are used to support the overall emergency response. P-MAN-03490-00002, "Pickering ERO Equipment and Facility Manual" [41] provides requirements and direction for Pickering ERO facilities configuration and management to ensure readiness, and to provide contingency actions, in accordance with the CNEP. It provides a description of the SMC, EOC, and SM offices including facility set-up, documentation, communications and other equipment and processes dedicated to or specifically set up for the Pickering ERO. N-MAN-03490-10000, "CEOF Equipment and Facility Manual" [42], provides the requirements and direction for CEOF configuration management and maintenance to ensure readiness, and to provide contingency actions, in accordance with the CNEP.

The availability of the necessary equipment and facilities to implement the CNEP is managed through the EITER program. Section 1.6.6 of the CNEP [7] refers to N-PROC-RA-0133, "Management of Equipment Important to Emergency Response" [45], which provides the overall management process for EITER. This procedure provides a framework to assure that when EITER is removed from service or is in a degraded condition, the correct restoration priority is assigned, compensatory measures are implemented and the equipment is promptly restored to a functional condition. The intent of this procedure is to ensure readiness of equipment and facilities required to implement the emergency plan. The Station Alignment Meeting package includes a Risk Management matrix, which identifies EITER issues. This information is reviewed daily and any EITER issues are flagged to the station operating staff and the station senior management team.

EITER includes systems, structures, and components, as well as essential tools and equipment, necessary to implement the emergency plan as described in the CNEP. EITER also includes EME as it is not managed under any other existing program to identify unavailability, restoration priority, or compensatory actions. EITER is listed in the following documents according to its location:

- N-INS-03491-10025, "Unavailability of Equipment Important to Emergency Response – Off-Site" [46];
- D-INS-03491-10000, "Unavailability of Equipment Important to Emergency Response – Darlington" [47];
- P-INS-03491-00050, "Unavailability of Equipment Important to Emergency Response – Pickering" [48].

These instructions also contain the required actions (including compensatory measures) that are to be taken in the event that EITER becomes unavailable.

It is important to note that the EITER program excludes equipment that is already managed by existing programs. Specific examples are listed below [45]:

1) Personal Protective Equipment (PPE): This equipment is available at Pickering NGS and is managed by plant programs and personal protection training. Further, non-standardized PPE is scenario-dependent and hence will be accessed and used as-required by personnel. This is governed by OPG-PROC-0010, "Health and Safety Management Program" and related governance.

2) Radiation Protection Equipment (Personal): Personal Radiation Protection equipment (e.g., personal dosimetry, radiation clothing, and decontamination supplies) is managed by Radiation Protection procedures and programs under governing document N-PROG-RA-0013, "Radiation Protection". It is readily available at multiple sites and can be easily deployed as required.

3) Fire Protection Equipment: This equipment is managed under N-PROG-RA-0012, "Fire Protection". This program meets the intent of the EITER program; i.e., the Fire Protection equipment that provides an Emergency Response function is

maintained, inspected and tested; impairments are managed by a comprehensive process including compensatory actions.

4) First Aid Equipment: This equipment is managed under N-PROG-RA-0012, "Fire Protection" and implementing standard N-STD-RA-0028, "First Aid". This standard identifies processes, overall requirements, and staff accountabilities to ensure effective First Aid is established and maintained.

5) SAMG Instrumentation and Equipment: The SAMG response uses plant instrumentation from systems that are managed by existing processes so these are not included. EME is included in the EITER program.

6) Supplies including fuel and other consumables: There are pre-established processes for the stocking and resupply of provisions required for emergency management including fuel supplies. Therefore, consumables are not managed as EITER.

Appendix A, "Guidance on Developing Equipment Important to Emergency Response List" of N-PROC-RA-0133 [45], provides guidance that can be used to determine whether new or additional equipment should be added to the EITER program.

N-PROC-RA-0040, "Maintenance and Testing of Emergency Preparedness Facilities and Equipment" [43], outlines the process by which emergency facilities and equipment are inspected, inventoried, operationally checked and tested. This procedure applies only to equipment and facilities used in a radiological emergency that is consistent with the CNEP.

As per Section 1.2 of N-PROC-RA-0040, a list of inventory and equipment checks including frequency is defined in N-LIST-03490-10028, "Emergency Preparedness Facility Inventory and Check Documents" [44]. This list summarizes the various implementing documents that are used to perform maintenance and testing activities for Pickering, Darlington, and off-site equipment and facilities. Walkdowns for individual facilities are performed per the corresponding frequency identified in N-LIST-03490-10028 to confirm adequate equipment is available.

Activities related to the testing and maintenance of EME are documented in N-INS-03600-10002, "Beyond Design Basis Emergency Mitigating Equipment Testing and Maintenance Process" [59]. This document establishes the instructions for testing and maintenance of EME and associated connection points to station systems credited to support BDBA mitigation. It ensures fleet wide consistency and rigour of the process of testing, maintenance, monitoring, and reporting in support of BDBA management. Appendix A, "EME Test and Maintenance" of N-INS-03600-10002 [59], summarizes the various testing and maintenance activities that are performed for EME. N-BDB-03600-00001, "Emergency Mitigating Equipment Inventory" [60] contains a complete listing of EME inventory. N-BDB-03600-00002, "OPG Emergency Mitigating Equipment for Beyond Design Basis Accidents: Technical Basis Document" [61] documents the technical basis for the EME that has been procured by OPG.

The adequacy of EITER is primarily demonstrated through the use of drills and exercises. As per N-PROC-RA-0045, "Emergency Preparedness Drills and Exercises" [29], all major elements of the emergency plan are tested every five years, with a full scale integrated exercise once every three years, per site. Table 1 of N-PROC-RA-0045 specifies the minimum drill and exercise frequency requirements.

A review of the latest performance objectives report, N-REP-03490-10038, "2014 Emergency Preparedness Drill and Exercise Performance Objectives Report" [24] indicates that all corrective actions in 2014 were correctly dispositioned and most of the actions had been completed. Similar findings were obtained from N-REP-03490-10034, "2013 Emergency Preparedness Drill and Exercise Performance Objectives Report" [62].

Pickering recently completed an exercise in November 2015 involving a multi-unit severe accident. The scope of the exercise was developed to demonstrate OPG's response capability to an event which has progressed from a Design Basis Accident (DBA), into a BDBA involving a multi-unit event, with subsequent event progression into a multi-unit severe accident requiring deployment of EME. The results of this exercise have been documented in N-REP-03491.23-10042 R000, "Emergency Preparedness Exercise Report – Pickering Station Emergency Exercise – 25Nov2015" [58]. The majority of ERO instructions utilized and actions taken in response to the simulated events were deemed to be acceptable. There were a number of opportunities for improvement identified, along with corresponding corrective actions. These enhancements do not represent safety significant findings and are therefore not a PSR2 gap.

Conclusion:

The conclusion of this Review Task assessment is that there is governance in place to ensure the adequacy of on-site equipment and facilities for emergencies and offsite emergency facilities, including walkdowns of relevant areas on and off the site. The intent of Review Task #10 is met and therefore Pickering NGS is compliant.

4.1.11 Review Task #11: Severe Accident Management Program

Confirm development or existence of a program for Severe Accident Management.

As per Section 1.2.3.3 of the CNEP [7], N-PROG-MP-0014 "Reactor Safety Program" provides the framework for major aspects of safe operation. These include safety analysis basis and analysis of record, safe operating envelope and beyond design basis accident management.

One element of the Reactor Safety program relates to the management of BDBAs and SAs. In this context, a BDBA refers to a relatively low frequency event sequence that is not included in the plant design basis (due to the low frequency of occurrence) and is not necessarily bounded by the analyses of the station design basis. If the consequences of such events are significant core degradation, these BDBAs are referred to as SAs.

OPG's Severe Accident Management (SAM) Program has been implemented through N-STD-MP-0019, "Beyond Design Basis Accident Management" [15] and is supported in its execution through SAMG and ERO organization procedures. N-STD-MP-0019 was developed using guidance from CNSC G-306, "Severe Accident Management Programs for Nuclear Reactors" [63] and CNSC REGDOC 2.3.2, "Accident Management: Severe Accident Management Programs for Nuclear Reactors" [64].

SAMG is a set of written guidance to implement strategies should a BDBA progress to a SA. The physical processes that govern SA phenomena are complex and, consequently, SAMG cannot be made highly dependent on detailed analyses because of limited understanding of certain SA phenomena and other uncertainties associated with SA causes and progression. However, reasonable strategies for coping with SA progression can be identified and developed using "state of the art" reviews, Probabilistic Safety Assessments, and insights on accident behaviours from accident analyses. SAMG allows flexibility in application and the SAMG document set is referred to as "guidance". SAMG documentation has been approved for use under the NA44-SAM-09013-10000 document series for Pickering 1-4 and the NK30-SAM-09013-10000 document series for Pickering 5-8.

Post-Fukushima, the CNSC assigned a series of Fukushima Action Items to OPG. A subset of these actions related to enhancing the capability of the SAMG program at Pickering NGS. All Fukushima Action Items have been closed for OPG and the CNSC has concluded that OPG has strengthened reactor defence-in-depth and enhanced its emergency response at Pickering NGS in response to lessons learned from the Fukushima nuclear accident [65].

Conclusion:

The conclusion of this Review Task assessment is that there is a Severe Accident Management Program at Pickering NGS. The intent of Review Task #11 is met and therefore Pickering NGS is compliant.

4.2 L/R/C/S Reviews

As per Section 2.2 of this report, detailed reviews for three L/R/C/Ss with content applicable to Safety Factor 13 are provided in Reference [6]. Associated findings applicable to Safety Factor 13 are summarized in Table 3 below.

Table 3: PSR2 L/R/C/S Review Results for Safety Factor 13

L/C/R/S Reviewed	PSR2 L/R/C/S Review Results for Safety Factor 13
CSA N293-12, "Fire Protection for Nuclear Power Plants"	Gaps identified from the PSR2 incremental review of N293-12 will be applicable to the Plant Design Safety Factor. Results are presented in the Pickering NGS PSR2 Plant Design Safety Factor Report.

L/C/R/S Reviewed	PSR2 L/R/C/S Review Results for Safety Factor 13
CNSC REGDOC-2.10.1 (2014), "Nuclear Emergency Preparedness and Response"	There is one PSR2 REGDOC-2.10.1 (2014) gap which relates to Safety Factor 13 (Emergency Planning): 1. OPG has completed a gap analysis for transition to REGDOC-2.10.1 and has developed an action plan to achieve compliance. The transition plan that OPG has committed in order to bring Darlington into compliance with REGDOC-2.10.1 applies across the nuclear fleet and will also bring Pickering into compliance. Updating OPG governance to ensure that the Pickering Evacuation Time Estimate study is maintained and to define how the Potassium Iodide (KI) pill program will be sustained is in progress. As these two actions are not yet complete, this is identified as a PSR2 gap (**Pickering PSR2 Gap SF13-1**).
CSA N1600-14, "General Requirements for Nuclear Emergency Management Programs"	There are no PSR2 gaps for CSA N1600-14. Per the definition of Compliance for a High Level review, Pickering has a PSR2 Compliance associated with CSA N1600-14.

4.3 OPG Program Effectiveness Reviews

The OPG Nuclear Program reviewed for Safety Factor 13 is identified in Table 2, and details of the associated effectiveness reviews for the N-PROG are provided in Appendix B.

4.4 Additional Review Findings

As discussed in Section 3.4, the PSR2 Safety Factor 13 assessment also included a review of commitments previously made to the CNSC, open CNSC action items, and exemptions granted by the CNSC, as identified in the R04 Pickering LCH [4], to determine if there are any impacts associated with operation of the Pickering Units past 2020. The review also included identification and review of previously identified programmatic Darlington PSR1 gaps related to Safety Factor 13 to determine impacts associated with operation of the Pickering Units past 2020.

A review of the Darlington Integrated Improvement Plan (IIP) [10] for gaps that may need to be reassessed in the context of Pickering PSR2 for operation past 2020, identified the following:

- **Gap SF13-2:** Darlington Gap IIP-OI 046 was identified to assess the Emergency Response Projection (ERP) code for potential enhancements to address multi-unit Beyond Design Basis Event scenarios. The action assigned to this gap (AR 28175339, TCD Q1 2017) is also applicable for Pickering NGS and is therefore a gap for Pickering PSR2. ERP is used to assist in decision making for off-site emergency response actions. Specifically, ERP is used to evaluate potential off-site consequences and predict the timing of containment re-pressurization.

Findings from the review of previously identified PSR1 gaps in the Pickering Units 5-8 Continued Operations Plan [8] are provided in a separate PSR2 COP Review Report. Findings from the review of Fukushima Action Items are provided in a separate PSR2 FAI Review Report. Results from the Continued Operations Plan and Fukushima Action Items reviews will be considered in the Global Assessment process.

There were no PSR2 gaps identified in this Safety Factor 13 report that require discussion in other Safety Factor reports.

5.0 RESULTS AND CONCLUSIONS

OPG Governance, Programs, Policies, Procedures, Instructions and Guidelines related to Safety Factor 13 were reviewed for the eleven PSR2 Review Tasks in Section 4.1 of this report and resulted in no Pickering PSR2 gaps. L/R/C/S and OPG Nuclear Program effectiveness reviews for Safety Factor 13 were prepared per Sections 4.2 and 4.3, respectively, and resulted in PSR2 Gap SF13-1 below. Per Section 4.2, the PSR2 review of CSA N293-12 is in progress. However, gaps identified in this review will be applicable to the Plant Design Safety Factor, and hence, results are presented in the Pickering NGS PSR2 Plant Design Safety Factor Report. Per Section 4.4, this report also included identification and review of previously identified programmatic Darlington PSR1 gaps related to Safety Factor 13 (to ascertain the implications of extending Pickering NGS operation beyond 2020), as well as a review of the R04 Pickering LCH [4] for any impacts of Pickering NGS operation beyond 2020 on a) OPG commitments previously made to the CNSC, b) open CNSC action items, and c) exemptions granted by the CNSC (all related to Safety Factor 13), which resulted in PSR2 Gap SF13-2.

The two gaps identified that will need to be addressed as part of Pickering PSR2 are:

- **Gap SF13-1**: OPG has completed a gap analysis for transition to CNSC REGDOC-2.10.1 and has developed an action plan to achieve compliance. The transition plan that OPG has committed in order to bring Darlington into compliance with REGDOC-2.10.1 applies across the nuclear fleet and will also bring Pickering into compliance. Updating OPG governance to ensure that the Pickering Evacuation Time Estimate study is maintained and to define how the Potassium Iodide (KI) pill program will be sustained is in progress. As these two actions are not yet complete, this is identified as a PSR2 gap.

- **Gap SF13-2**: Darlington Gap IIP-OI 046 was identified to assess the ERP code for potential enhancements to address multi-unit Beyond Design Basis Event scenarios. The action assigned to this gap (AR 28175339, TCD Q1 2017) is also applicable for Pickering NGS and is therefore a gap for Pickering PSR2.

The review of Safety Factor 13 has confirmed that OPG Nuclear has: a) adequate plans, staff, facilities and equipment in place for dealing with emergencies, and b) there are adequate arrangements in place for regular emergency training and exercises, and interaction and coordination with local and national authorities.

6.0 REFERENCES

[1] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[2] CNSC REGDOC-2.3.3, *Periodic Safety Reviews*, April 2015.

[3] IAEA Safety Guide No. SSG-25, *Periodic Safety Review for Nuclear Power Plants*, March 2013.

[4] CNSC Report, LCH-PNGS-R004, *Pickering NGS: Licence Conditions Handbook*, December 23, 2015.

[5] OPG Report, P-REP-03680-00003 R00, *Pickering NGS PSR2: Definition of Safety Factor Review Tasks*, May 30, 2016.

[6] OPG Report, P-REP-03680-00021 R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 8, 10, 12, 13, and 14,* December 2016.

[7] OPG Nuclear Program, N-PROG-RA-0001 R014, *Consolidated Nuclear Emergency Plan*, May, 2015.

[8] OPG Plan, NK30-PLAN-00531-00001 R005, *Pickering 5-8 Continued Operations Plan*, December 2015.

[9] OPG Procedure, N-PROC-RA-0023 R018, *Fleetview Program Health and Performance Reporting*, August 2013.

[10] OPG Report, NK38-REP-03680-10185 R002, *Darlington NGS Integrated Implementation Plan (IIP)*, April 30, 2015.

[11] OPG Report, NA44-SR-01320-00002 R004, *PN 1-4 Safety Report: Part 3 – Accident Analysis,* October 2013.

[12] OPG Report, NK30-SR-01320-00003 R004, *PN 5-8 Safety Report: Part 3 – Accident Analysis,* October 2014.

[13] CNSC REGDOC-2.4.1, *Deterministic Safety Analysis*, May 2014.

[14] OPG Procedure, N-PROC-RA-0035 R018, *Operating Experience Process,* October 2014.

[15] OPG Standard, N-STD-MP-0019 R001, *Beyond Design Basis Accident Management,* September 2014.

[16] OPG Manual, P-MAN-03490-00005 R015, *Index – Emergency Response Manual 5,* September 2015.

[17] OPG Program, N-PROG-RA-0012 R011, *Fire Protection,* July 2015.

[18] OPG Agreement, N-LEGL-03490-0413370, *Mutual Aid Agreement for Nuclear Emergency Support,* November 2012.

[19] OPG Memorandum, N-CORR-09013-0542905, *Beyond Design Basis Event Response – Emergency Mitigating Equipment (EME) – Equipment Available from Other Canadian Nuclear Facilities,* June 2015.

[20] Province of Ontario Nuclear Emergency Plan, *Provincial Nuclear Emergency Response Plan Master Plan*, 2009.

[21] OPG Manual, N-MAN-08131-10000 Sht: S4-0245 R002, *Manager, Emergency Preparedness,* November 2014.

[22] OPG Instruction, N-INS-03490-10002 R005, *Conduct of Emergency Preparedness Drills and Exercises,* December 2014.

[23] OPG Procedure, N-PROC-RA-0022 R032, *Processing Station Condition Records,* November 2014.

[24] OPG Report, N-REP-03490-10038 R000, *2014 Emergency Preparedness Drill and Exercise Performance Objectives Report,* August 2015.

[25] OPG Procedure, N-PROC-RA-0046 R008, *Emergency Response Organization Staffing and Availability,* April 2015.

[26] OPG Instruction, P-INS-09100-00003 R009, *Pickering Minimum Shift Complement,* December 2014.

[27] OPG Instruction, P-INS-09260-00008 R007, *Duty Crew Minimum Complement Assurance,* February 2013.

[28] OPG Description, N-TQD-503-00001 R017, *Nuclear Emergency Response Organization Training and Qualification Description,* December 2015.

[29] OPG Procedure, N-PROC-RA-0045 R010, *Emergency Preparedness Drills and Exercises,* December 2015.

[30] INPO, INPO 14-003, *Emergency Drill and Exercise Guideline*, 2014.

[31] OPG Instruction, P-INS-03491-00001 R018A, *Incident Shift Manager,* February 2015.

[32] OPG Instruction, N-INS-03491-10001 R008, *CEOF Technical Support Director,* May 2015.

[33] OPG Instruction, P-INS-03491-00002 R016, *Incident Emergency Shift Assistant,* November 2014.

[34] OPG Manual, N-MAN-03491-10000 R012, *Emergency Telephone Directory,* October 2015.

[35] OPG Guide, N-GUID-03491-50010 R002, *Emergency Response Organization Expectations,* July 2015.

[36] OPG Instruction, N-INS-03491-10022 R000, *Resource Deployment Manager,* May 2014.

[37] OPG Procedure, OPG-PROC-0028 R005, *Crisis Management & Communications Centre (CMCC) Procedure,* February 2015.

[38] OPG Standard, N-STD-AS-0010 R003, *Nuclear Crisis Communications Standard,* January 2011.

[39] OPG Guideline, N-GUID-03491-10000 R000, *WebEOC User's Guide,* June 2009.

[40] OPG Correspondence, N-CORR-00531-06984 R000, *Pickering NGS and Darlington NGS – OPG Specific Action Items Related to Closed Fukushima Action Items – Progress Update No. 5,* November 2015.

[41] OPG Manual, P-MAN-03490-00002 R004, *Pickering ERO Equipment and Facility Manual,* March 2015.

[42] OPG Manual, N-MAN-03490-10000 R002, *CEOF Equipment and Facility Manual,* February 2015.

[43] OPG Procedure, N-PROC-RA-0040 R006, *Maintenance and Testing of Emergency Preparedness Facilities and Equipment,* April 2015.

[44] OPG List, N-LIST-03490-10028 R000, *Emergency Preparedness Facility and Inventory Check Documents,* April 2015.

[45] OPG Procedure, N-PROC-RA-0133 R000, *Management of Equipment Important to Emergency Response,* December 2014.

[46] OPG Instruction, N-INS-03491-10025 R000, *Unavailability of Equipment Important to Emergency Response – Off-Site,* December 2014.

[47] OPG Instruction, D-INS-03491-10000 R002, *Unavailability of Equipment Important to Emergency Response – Darlington,* April 2016.

[48] OPG Instruction, P-INS-03491-00050 R002, *Unavailability of Equipment Important to Emergency Response – Pickering,* November 2015.

[49] OPG Procedure, OPG-PROC-0001 R009, *Process Administrative Governance Documents,* April 2015.

[50] OPG Program, N-PROG-AS-0006 R011, *Records and Document Control,* October 2014.

[51] OPG Description, N-TQD-501-00001 R013, *Nuclear General Employee Training and Qualification Description,* September 2014.

[52] OPG Report, P-REP-03490-00053 R000, *Emergency Preparedness Evaluated Drill Report – Pickering Station Emergency Drill 23 Jan 2013,* April 2014.

[53] OPG Report, P-REP-03490-00052 R001, *Emergency Preparedness – Pickering Emergency Mitigating Equipment (EME) Drill Report, February 6, 2013,* April 2013.

[54] OPG Report, P-REP-03490-00054 R000, *Emergency Preparedness Evaluated Drill Report – Pickering Station Emergency Drill 03 July 2013,* July 2013.

[55] OPG Report, P-REP-03490-00055 R000, *Emergency Preparedness – Pickering Nuclear DBA/SAMG Drill Report, September 23, 2013,* November 2013.

[56] OPG Report, P-REP-03490-00056 R001, *Emergency Preparedness – Pickering Nuclear DBA/SAMG Drill Report, November 1, 2013,* March 2014.

[57] OPG Report, N-REP-03491.23-10041 R000, *Emergency Preparedness Exercise Report – Pickering Station Emergency Exercise – 15 Oct 2014,* January 2015.

[58] OPG Report, N-REP-03491.23-10042 R000, *Emergency Preparedness Exercise Report – Pickering Station Emergency Exercise – 25Nov2015,* April 2016.

[59] OPG Instruction, N-INS-03600-10002 R000, *Beyond Design Basis Emergency Mitigating Equipment Testing and Maintenance Process,* March 2015.

[60] OPG Manual, N-BDB-03600-00001 R000, *Emergency Mitigating Equipment Inventory,* November 2015.

[61] OPG Manual, N-BDB-03600-00002 R000, *OPG Emergency Mitigating Equipment for Beyond Design Basis Accidents: Technical Basis Document,* October 2015.

[62] OPG Report, N-REP-03490-10034 R000, *2013 Emergency Preparedness Drill and Exercise Performance Objectives Report,* March 2014.

[63] CNSC Regulatory Guide G-306, *Severe Accident Management Programs for Nuclear Reactors,* May 2006.

[64] CNSC Regulatory Document REGDOC 2.3.2, *Accident Management: Severe Accident Management Programs for Nuclear Reactors,* September 2013.

[65] OPG Correspondence, N-CORR-00531-06906, *OPG Progress Report No. 7 on CNSC Action Plan – Fukushima Action Items,* November 2015.

Appendix A: Nomenclature

AIM	Abnormal Incidents Manual
AR	Action Request
BDBA	Beyond Design Basis Accident
CANDU	CANada Deuterium Uranium
CEOF	Corporate Emergency Operations Facility
CMCC	Crisis Management and Communications Centre
CNEP	Consolidated Nuclear Emergency Plan
CNSC	Canadian Nuclear Safety Commission
COG	CANDU Owners Group
COP	Continued Operations Plan
CSA	Canadian Standards Association
DBA	Design Basis Accident
EITER	Equipment Important to Emergency Response
EME	Emergency Mitigating Equipment
EMEG	Emergency Mitigating Equipment Guideline
EOC	Emergency Operations Centre
EP	Emergency Preparedness
ERO	Emergency Response Organization
ERP	Emergency Response Projection
ESA	Emergency Shift Assistant
FAI	Fukushima Action Item
IAEA	International Atomic Energy Agency
IIP	Integrated Implementation Plan
INES	International Nuclear Event Scale
INPO	Institute of Nuclear Power Operations
ISR	Integrated Safety Review
LCH	Licence Conditions Handbook
MCR	Main Control Room
MDU	Monitoring and Decontamination Unit
NGS	Nuclear Generating Station
NPP	Nuclear Power Plant

N-PROG	Nuclear Program
OPG	Ontario Power Generation
PARTS	Pickering A Return to Service
PEOC	Provincial Emergency Operations Centre
PNERP	Provincial Nuclear Emergency Response Plan
PPE	Personal Protective Equipment
PROL	Power Reactor Operating Licence
PSR	Periodic Safety Review
PSR1	Periodic Safety Review 1 (earlier OPG PSR work and other associated assessments)
PSR2	Periodic Safety Review 2 (subsequent PSR per REGDOC-2.3.3)
SA	Severe Accident
SACP	Self-Assessed Crew Practice
SAMG	Severe Accident Management Guidance
SCR	Station Condition Record
SM	Shift Manager
SMC	Site Management Centre

Appendix B: OPG Program Effectiveness Review Results

B.1 N-PROG-RA-0001, "Consolidated Nuclear Emergency Plan"

The purpose of the Consolidated Nuclear Emergency Plan (CNEP) is to provide a written basis to document the concepts, roles, and resources required by Ontario Power Generation (OPG) Nuclear to implement and maintain its emergency response capability to protect the public, employees, and the environment in the event of a nuclear emergency. It provides a framework for interaction with external authorities and defines OPG commitments under the Provincial Nuclear Emergency Response Plan (PNERP). The CNEP applies to both the Pickering and Darlington Nuclear facilities.

The CNEP deals with emergency situations that may endanger the safety of on-site staff, the environment, and the public. It also represents a basis for controlling changes and modifications to the OPG Nuclear Emergency Preparedness (EP) program. Provisions of the CNEP apply to any associated potential threat of release of radioactive material (for example, the need for off-site notification, situation updates and confirmation of any radioactive releases). Liquid emission response and transportation of radioactive material emergency response are governed by separate plans.

The Emergency Management department completed a self-assessment, NO13-000443-SA [B.1.1], in November 2013 in order to assess compliance with N-PROG-RA-0001, "Consolidated Nuclear Emergency Plan", including a review of all agreements with offsite authorities. No findings were generated, however a series of recommendations were generated to ensure that the Provincial Nuclear Emergency Response Plan requirements are properly documented. The recommendations were captured under SCRs N-2013-21141, N-2013-21131 and AR 28161879, which have all been completed.

The Emergency Preparedness department completed a self-assessment, NO15-001449-SA [B.1.2], in February 2016 in order to review the requirements of REGDOC-2.10.1, "Nuclear Emergency Preparedness and Response" against N-PROG-RA-0001, to confirm compliance, identify any gaps, and recommend actions to close any identified gaps. The self-assessment was applicable to both Pickering and Darlington NGS and concluded that the CNEP met the majority of the REGDOC-2.10.1 requirements. However, there are additional REGDOC-2.10.1 requirements which need to be incorporated into OPG governance (e.g., an update of the CNEP to reference the planning basis, a listing of agreements with applicable offsite agencies to be referenced in the CNEP, and a validation process to demonstrate compliance with the CNEP and procedures), which may include developing new processes and consulting with the province and industry partners. AR# 28184526 was initiated, which required corrective actions to be implemented in order to address any gaps between the requirements of REGDOC-2.10.1 and N-PROG-RA-0001. This AR is expected to be completed by Q4 2017.

Nuclear Oversight conducted a performance based audit of the Emergency Preparedness program (note, N-PROG-RA-0001 is the program level document for Emergency Preparedness) in April 2013, NO-2013-030 [B.1.3], in order to confirm that the program is being effectively managed and is in compliance with regulatory and OPG governance requirements for both Pickering and Darlington NGS. The audit identified performance improvement opportunities applicable to Pickering NGS related to alignment with standards, readiness of Emergency Preparedness equipment and facilities, and alignment of CNEP with Provincial Implementing Plans.

Three SCRs were initiated to address the above findings (SCRs N-2013-01608, N-2013-01609 and N-2013-01610), which required corrective actions to be implemented. These SCRs have since been closed and the necessary corrective actions were completed to address the underlying issues.

References

[B.1.1] NO13-000443-SA, *Self-Assessment Report – EP Program (N-PROG-RA-0001) including a review of all agreements with Off-Site Authorities,* November 15, 2013.

[B.1.2] NO15-001449-SA, *Self-Assessment Report - CNSC REGDOC 2.10.1 vs N-PROG-RA-0001, CNEP Review, gap identification and transition plan*, February 23, 2016.

[B.1.3] NO-2013-030 (N-REP-01070-0435168), *Audit OPGN NO-2013-030 – Emergency Preparedness*, April 5, 2013.

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REVISE AND RESUBMIT	
Signature	Date: 13 Dec 2016
Name: Mike Ruffolo, Manager	
Dept: Pickering Engineering - Aging Management & Strategic Initiatives	
OPG Proprietary	
Doc No.: P-REP-03680-00018	Rev: 000
This acceptance does not relieve the contractor from responsibility for errors or omissions or from any obligations or liability under this contract.	

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Pickering NGS PSR2 Safety Factor 14 Report: Radiological Impact on the Environment

PS112/RP/013 R01

December 12, 2016

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Revision Summary – For Amec Foster Wheeler Report PS112/RP/013

Rev	Date	Author	Comments
R00	July 18, 2016	M. Bale, R. Jayasundera	Initial issue for OPG review and comment.
R01	December 12, 2016	M. Bale, R. Jayasundera	Updated report addressing OPG comments on R00 Report.

EXECUTIVE SUMMARY

Ontario Power Generation (OPG) is performing a Periodic Safety Review (PSR) to support the possibility of continued operation of Pickering Nuclear Generating Station (NGS) beyond 2020. The PSR (referred to as "PSR2") is a subsequent PSR building on the review basis of earlier OPG Integrated Safety Reviews and other associated assessments. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

Part of PSR2 involves the preparation of Safety Factor reports for each of fifteen major topic areas. Safety Factor reports consist of:

- Assessing compliance against "Review Tasks" identified in the PSR2 Basis Document [1]. These Review Tasks are derived from Canadian Nuclear Safety Commission (CNSC) REGDOC-2.3.3, "Periodic Safety Reviews" [2] and International Atomic Energy Agency (IAEA) SSG-25, "Periodic Safety Review for Nuclear Power Plants" [3];

- Documenting assessments against applicable modern Laws, Regulations, Codes and Standards (L/R/C/Ss) as defined in Reference [1]; and

- Reviewing the effectiveness of applicable OPG Nuclear Programs that support the above assessments, through review of audit and self-assessment results.

The PSR2 review of Safety Factor 14, *Radiological Impact on the Environment* is presented in this report. OPG Governance, Programs, Policies, Procedures, Instructions and Guidelines related to Safety Factor 14 were reviewed for the seven PSR2 Review Tasks in Section 4.1 of this report. L/R/C/S and OPG Nuclear Program effectiveness reviews for Safety Factor 14 were prepared per Sections 4.2 and 4.3, respectively. Per Section 4.4, the PSR2 assessment includes a review of previously identified PSR1 gaps related to Safety Factor 14 (to ascertain the implications of extending Pickering NGS operation beyond 2020), as well as a review of the R04 Pickering Licence Conditions Handbook [4] for any impacts of Pickering NGS operation beyond 2020 on: a) OPG commitments previously made to the CNSC, b) open CNSC action items, and c) exemptions granted by the CNSC (all related to Safety Factor 14).

The results of the review of Safety Factor 14 are discussed in Section 5.0 of this report. The review has confirmed that Pickering NGS has an adequate and effective program for monitoring the radiological impact of the plant on the environment, which ensures that emissions are properly controlled and are as low as reasonably achievable. As discussed in Section 5.0, the review identified no Pickering PSR2 gaps.

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1.0 INTRODUCTION

Ontario Power Generation (OPG) is performing a Periodic Safety Review (PSR) to support the possibility of continued operation of Pickering Nuclear Generating Station (NGS) beyond 2020.[1] A comprehensive Integrated Safety Review (ISR) was completed for Pickering Units 5 through 8 in 2009 in support of refurbishment and continued operation. Pickering Units 1,4 integrated safety assessments were also performed for Pickering A Return to Service (PARTS) in support of approval to restart Units 1 and 4. In addition to these Pickering-specific studies, the 2013 Darlington ISR performed extensive code and standard reviews that were updated in relation to the versions that were assessed in the 2009 Pickering B ISR.[2] These previous ISRs are considered to constitute the first PSR completed for Pickering (referred to as "PSR1"). The current PSR (referred to as "PSR2") is a subsequent PSR building on the basis of earlier OPG integrated safety assessments through review of the various studies, assessments and licence renewals performed since PSR1. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

PSR2 will support and complement the licence renewal application for Pickering NGS going forward. Fifteen Safety Factors will be assessed as part of the PSR. The purpose of Safety Factor reviews is to confirm that the design, condition and operation of Pickering Units 1,4 and 5-8 (as well as common systems) will support continued safe operation for the period of PSR2 by:

- Assessing compliance against "Review Tasks" identified in the PSR2 Basis Document [1], which were derived from Canadian Nuclear Safety Commission (CNSC) REGDOC-2.3.3 [2] and International Atomic Energy Agency (IAEA) SSG-25 [3];

- Documenting assessments against applicable modern Laws, Regulations, Codes and Standards (L/R/C/Ss) (as defined in Reference [1]); and

- Reviewing the effectiveness of applicable OPG Nuclear Programs that support the above assessments, through review of audit and self-assessment results.

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1 Currently, Pickering Units 5-8 are approved to operate to 247,000 Effective Full Power Hours. This operation limit is expected to be reached on some units in 2020. For the purposes of PSR2, OPG assumes operation of Pickering NGS for up to eight additional years, from 2020 until 2028. OPG will make a decision regarding the permanent shut down dates for the six reactors following the performance of a technical evaluation that will include PSR2, and will communicate it to the CNSC as required by the current Power Reactor Operating Licence (PROL).

2 Much of the compliance assessment and evaluation of Safety Factor health for the Darlington ISR is based on programs and practices that apply across OPG's nuclear operations. As a result, where Pickering is confirmed to follow the same nuclear programs and practices as were assessed for Darlington, the Darlington ISR programmatic conclusions are applicable to Pickering. As discussed in Section 1.0, an effectiveness review (effectiveness at Pickering NGS) of OPG programs used to demonstrate compliance with the PSR2 Assessment Basis will be conducted using recent audit and self-assessment results.

As outlined in IAEA SSG-25 [3], the objective of the review of Safety Factor 14 is to: "determine whether the operating organization has an adequate and effective programme for monitoring the radiological impact of the plant on the environment, which ensures that emissions are properly controlled and are as low as reasonably achievable." REGDOC-2.3.3 [2] requires that: "The licensee shall conduct a PSR in accordance with this regulatory document for the period until the next PSR or, if applicable, until the end of commercial operation of the plant."

This report documents the results of the review of Safety Factor 14 for Pickering PSR2. The report is based on the OPG Governance, Programs, data, and material available up to January 15, 2016 which is the freeze date for PSR2.

2.0 SCOPE OF REVIEW

2.1 Review Task Assessments

The Pickering PSR2 Safety Factor 14 Review Tasks are defined in Reference [1]. Details of the derivation of these Review Tasks from CNSC REGDOC-2.3.3 and IAEA SSG-25 are shown in Reference [5]. The Safety Factor 14 Review Tasks are:

1) Confirm there are procedures in place to ensure that permitted release limits of radiological substances are not exceeded and, if they are, that appropriate corrective action is taken to minimize the possibility of limits being exceeded in the future.

2) Confirm records of radiological effluent release are maintained in accordance with regulatory requirements.

3) Confirm that a program exists to define the requirements for alarm systems to respond to unplanned effluent releases from on-site facilities.

4) Confirm the environmental data recorded by the station is published and is available on request to the general public.

5) Review the environmental data recorded by the station and compare with the values measured before the plant was put into operation.

6) Confirm there is a process to address changes in the use of land external to the site with respect to the impact on public safety from facility releases.

7) Confirm that the monitoring program is appropriate and sufficiently comprehensive. In particular, confirm that the radiological impact of the plant on the environment is not significant compared with that due to other sources of radiation.

The methodology for the reviews is discussed in Section 3.1. Review Task findings are summarized in Section 4.1 of this report.

2.2 L/R/C/S Reviews

The applicable Laws, Regulations, Codes and Standards relevant to the Radiological Impact on the Environment Safety Factor are identified in Reference [1] and are listed in Table 1 below. Table 1 also identifies the modern version and date of each L/R/C/S to be considered, the Safety Factor(s) to which each document is applicable, and the type of review that will be completed in PSR2.

All of the Safety Factor 14 L/R/C/S reviews are high level or incremental in nature. The definitions of High Level Review and Incremental Review are as follows:

- High Level: New L/R/C/Ss not referenced in Pickering PROL 48.02/2018 but which are in the PSR2 Assessment Basis will be subject to a high level review. In a high level review, the degree of conformance with clauses or groups of clauses in the L/R/C/S is demonstrated by supporting evidence stating whether the intent of the requirements stipulated in the requirement document is met; and,

- Incremental Review: For L/R/C/Ss that have been reviewed in PSR1 but have had revisions since the last review, a topical review will be performed of the changes.

The methodology for the reviews is discussed in Section 3.2. A detailed assessment for each L/R/C/S is provided in References [6] and [7]. Associated findings are summarized in Section 4.2 of this report.

Table 1: L/R/C/Ss Reviewed for Radiological Impact on the Environment Safety Factor 14

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
L/R/C/Ss Referenced in Pickering NGS PROL 48.02/2018						
1	CSA[3] N288.1	Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities	N288.1-14	8, 14	Incremental	N288.1 addressed as part of Pickering B and Darlington ISRs.
2	CSA N288.4	Environmental Monitoring Program at Class I Nuclear Facilities and Uranium Mines and Mills	N288.4-10	8, 14	Incremental	N288.4 addressed as part of Pickering B and Darlington ISRs.
3	CNSC REGDOC-2.9.1*	Environmental Protection Policies, Programs and Procedures	2013	8, 14	Incremental	REGDOC-2.9.1 addressed as part of Darlington ISR. S-296 also addressed as part of Pickering B and Darlington ISRs.
Additional L/R/C/Ss						
4	CNSC G-228	Developing and Using Action Levels	2001	8, 14, 15	Incremental	G-228 addressed as part of Pickering B and Darlington ISRs.

[3] CSA – Canadian Standards Association

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
5	CSA N288.6	Environment Risk Assessments at Class I Nuclear Facilities and Uranium Mines and Mills	N288.6-12	8, 14	Incremental [4]	N288.6 not addressed as part of Pickering B or Darlington ISRs. Implementation Plan and clause-by-clause review have been prepared for Pickering Environmental Monitoring Program compliance with N288.6.
6	CSA N288.5	Effluent Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills Facilities	N288.5-11	8, 14	Incremental [4]	N288.5 not addressed as part of Pickering B or Darlington ISRs. OPG has performed a gap analysis and completed all actions in the implementation plan to satisfy mandatory requirements of N288.5.
7	CSA N288.3.4	Performance Testing of Nuclear Air-Cleaning Systems at Nuclear Facilities	N288.3.4-13	8, 14	Incremental [4]	N288.3.4 addressed as part of Darlington ISR, but not addressed as part of Pickering B ISR. OPG has completed a gap analysis and is developing an implementation plan to satisfy mandatory requirements of N288.3.4.
8	CSA N288.7	Groundwater Protection Programs at Class I Nuclear Facilities and Uranium Mines and Mills	N288.7-15	14	High Level	First edition of N288.7 issued in 2015. Not addressed as part of Pickering B or Darlington ISRs. OPG is developing a gap analysis and implementation plan to satisfy mandatory requirements of N288.7.

* Superseding documents to those currently in Pickering NGS PROL 48.02/2018.

[4] Per Section 3.2.2 of the R02 PSR2 Basis Document [1]: "Table D1 identifies the review type to be applied to each of the Laws, Regulations, Codes and Standards in the PSR2 Assessment Basis. Following further assessment of past work, the review type of a listed modern Law, Regulation, Code or Standard may be changed from Clause-by-Clause or High Level to Incremental." Past assessments of CSA N288.3.4, N288.5 and N288.6 were reviewed and implementation plans with gap assessments were identified. As a result, the Review Type for these three L/R/C/Ss was changed from High Level to Incremental since "… implementation plans exist for many of the codes and standards not addressed in PSR1 and therefore an incremental review will be applied to these documents" [1].

2.3 OPG Program Effectiveness Reviews

The OPG Programs (N-PROGs or OPG-PROGs) reviewed for Safety Factor 14 are listed in Table 2 below.[5] The methodology for the effectiveness reviews is discussed in Section 3.3. The assessment results of each of the N-PROGs or OPG-PROGs in Table 2 are provided in Appendix B, and findings are summarized in Section 4.3.

Table 2: OPG Programs Reviewed for Safety Factor 14

Document Number	Document Title
N-PROG-OP-0006 [8]	Environmental Management
OPG-PROG-0005 [9]	Environmental Management System

2.4 Additional Reviews

The PSR2 Safety Factor 14 Report includes a review of the R04 Pickering Licence Conditions Handbook (LCH) [4] for any impacts of Pickering NGS operation beyond 2020 on the following (all related to Safety Factor 14):

- OPG commitments previously made to the CNSC;
- Open CNSC action items; and
- Exemptions granted by the CNSC.

The PSR2 assessment includes identification and review of previously identified PSR1 gaps related to Safety Factor 14 to ascertain the implications of extending Pickering NGS operation beyond 2020. The methodology for these reviews is described in Section 3.4. Any PSR2 gaps identified as a result of programmatic Darlington PSR1 gaps related to Safety Factor 14 are discussed in Section 4.4 of this report. The review of Pickering PSR1 gaps previously identified in the Pickering Units 5-8 Continued Operations Plan (COP) [10] is provided in a separate PSR2 COP Review Report.

In addition, Fukushima Action Items (FAIs) were reviewed to identify implications of extending operation beyond 2020 (if any). This review is presented in a separate PSR2 FAI Review Report.

Any PSR2 gaps identified as a result of the Safety Factor 14 review which need to be addressed in other Safety Factor Reports are discussed in Section 4.4 of this report.

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[5] The list of Nuclear Programs to be assessed for effectiveness for PSR2 was derived from review of current OPG Governance. Although there may be content in Nuclear Programs that is applicable to multiple Safety Factors, N-PROG reviews are only provided in one Safety Factor report and are not duplicated.

3.0 METHODOLOGY

The sub-sections below summarize the methodology used to assess Review Tasks, L/R/C/Ss, and Nuclear Program effectiveness for the Radiological Impact on the Environment Safety Factor.

3.1 Review Tasks

As discussed earlier, the Safety Factor Review Tasks are derived from CNSC REGDOC-2.3.3 [2] and IAEA SSG-25 [3], taking into consideration the Review Tasks used in the Pickering B and Darlington ISRs (as derived in [5]).

For each Safety Factor 14 Review Task identified in Section 2.1, a confirmation of the existence of applicable OPG Governance, Programs, Policies and Procedures (as well as Instructions and Guidelines, as applicable) was performed. Compliance against Review Tasks is also assessed by reference to applicable Condition Assessments, safety analyses and operating experience, as required.

The Review Task assessments identify Compliances and Gaps as defined below:

- Compliance: Compliance indicates that either the safety requirement or the intent of the Review Task is met.
- Gap: A Gap indicates that the intent of the Review Task is not met.

3.2 L/R/C/S Reviews

The process to identify the modern L/R/C/Ss that are applicable to the PSR2 Assessment Basis involved first creating a broad list from multiple sources (potential candidate L/R/C/Ss) and then filtering it to identify those that are most significant and that are applicable to the PSR2 scope. The identification and selection criteria are detailed in Reference [1]. The result of the identification and selection process was a set of modern L/R/C/Ss that became part of the "PSR2 Assessment Basis".

PSR2 is focused on the extension of Pickering NGS operations beyond 2020, and will conduct reviews against a baseline of past PSR1 work. As a subsequent PSR, PSR2 focuses on changes in requirements, plant conditions, operating experience and new information. Since PSR2 is an update of previous ISRs, it incorporates reviews of L/R/C/Ss that have occurred as new versions have been issued. Since this assessment is a subsequent PSR, the focus is on identifying differences between what was previously assessed and what is now different within the current Pickering PSR2 Assessment Basis. In general, these differences relate to:

- More recent (new or revised) L/R/C/S versions than what was previously assessed as part of PSR1;[6]
- Safety significant differences between Pickering and Darlington, if the Darlington ISR is the basis for the earlier assessment;
- Implications of extending Pickering NGS operation beyond 2020; and
- Safety significant differences between Pickering Units 1,4 and Units 5-8.

As described in Reference [1], L/R/C/S review types are clause-by-clause, high level or incremental. Most of the L/R/C/Ss in the PSR2 Assessment Basis receive incremental reviews since PSR2 is an update of previous PSR1 assessments and clause-by-clause or high level reviews for the majority of the L/R/C/Ss in the PSR2 Assessment Basis have already been completed. Implementation plans (including gap analyses or code-over-code reviews) also exist for the latest editions of many L/R/C/Ss. As a result, incremental review is also used in circumstances where a L/R/C/S in the PSR2 Assessment Basis was not assessed in previous PSR1 reviews but an implementation plan currently exists for compliance.

The PSR2 incremental reviews in this Report include an assessment of the intent of recent changes to the L/R/C/Ss on a topic or subject-matter basis where there is potential to impact nuclear safety. Incremental reviews provide:

- A summary of the purpose of the L/R/C/S;
- Pertinent background information about the current revision of the L/R/C/S that is being considered;
- Identification of which Safety Factor(s) are applicable to the current revision of the L/R/C/S;
- A description of which version(s) of the L/R/C/S were assessed for PSR1 (i.e., Darlington ISR (for programmatic content), Pickering B ISR and PARTS code reviews);
- Identification of whether the current version of the L/R/C/S is an update of a previous version of the L/R/C/S that was assessed in PSR1 (and if so, a description of the major changes in the latest revision is provided as discussed below);

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6 "New" refers to a code or standard that was not previously considered in the context of earlier assessments. "Revised" refers to an updated version of a code or standard that was previously considered in the context of earlier assessments. Where a document has a new number/type, but addresses the same topic from the same organization, it is a "revised", not "new", document (e.g., if a REGDOC replaces a CNSC G or RD document).

- An assessment of the applicability of PSR1 assessment findings (gaps and conclusions), including the implications of extending Pickering NGS operation beyond 2020 if any;

- An assessment of the applicability of assessment findings that address more recent (post-PSR1) editions of the L/R/C/S, including any implementation or transition plans that are already committed to by OPG; and

- Where PSR1 and post-PSR1 assessments are not sufficient to address changes in the latest edition of the L/R/C/S, an assessment of the changes from the previously assessed edition of the L/R/C/S (including identification of any safety significant PSR2 gaps which result).

High Level reviews provide the same information as above, where applicable, in a similar format. However, given that High Level L/R/C/Ss generally have not received past assessment during PSR1, the Incremental review content is augmented by a high level, section-by-section assessment of the degree of conformance of Pickering NGS with the L/R/C/S (demonstrating, with supporting evidence, whether the intent of the requirements stipulated in the document are met).

There are currently no L/R/C/S clause-by-clause reviews identified in the PSR2 Assessment Basis.

The Safety Factor 14 L/R/C/S reviews identify Compliances and Gaps as defined below:[7]

- <u>Compliance</u>:
    - Where an Incremental review has been performed, Compliance indicates that the change in the safety requirement, per the topical review, is met.
    - Where a High Level review has been performed, Compliance indicates that the intent of the safety requirement is met.
    - For Clause-by-Clause reviews of modern L/R/C/Ss, Compliance indicates that the safety requirement is met. (Note: No Clause-by-Clause reviews were performed as part of Safety Factor 14.)

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[7] Safety Factor assessments for Review Tasks and L/R/C/S reviews make use of: a) OPG Governance, Programs, Policies and Procedures which support the assessment arguments, b) Commitments previously made to the CNSC, open CNSC action items, and exemptions granted by the CNSC (all related to the Safety Factor under review), as identified in the R04 Pickering LCH [4], c) Identification of previously identified Pickering-specific or programmatic PSR1 gaps related to the Safety Factor under review and the status of OPG's improvement plan(s) or other dispositions to address these, and d) Assessments and reviews performed since the PSR1 documents were completed.

- Gap:
    - Where an Incremental review has been performed, a Gap indicates that the change in the safety requirement, per the topical review, is not met.
    - Where a High Level review has been performed, a Gap indicates that the intent of the safety requirement is not met.
    - For Clause-by-Clause reviews of modern L/R/C/Ss, a Gap indicates that the safety requirement is not met. (Note: No Clause-by-Clause reviews were performed as part of Safety Factor 14.)

The reviews assume that use of the word:

- "Shall" is used in an L/R/C/S to express a requirement, i.e., a provision that the licensee is obliged to satisfy in order to comply with the standard;
- "Should" is used to express a recommendation or that which is advised but not required;
- "May" is used to express an option or that which is permissible within the limits of the standard; and
- "Can" is used to express possibility or capability.

3.3 OPG Program Effectiveness Reviews

As discussed earlier, effectiveness reviews (effectiveness at Pickering NGS) of OPG programs used to demonstrate compliance with the PSR2 Assessment Basis were conducted, using recent applicable audit and self-assessment results:

- OPG Nuclear Oversight independent performance-based Program audits (typically performed in 1 to 5 year cycles) and self-assessments. This includes review of associated Station Condition Records and Action Requests to determine the status of any resulting corrective actions; and
- CNSC "Type I" and "Type II" inspections of the effectiveness and performance of OPG programs, where discussed in OPG audits or self-assessments.

There are many audits and self-assessments that are performed to assess the effectiveness of important aspects of each program. A sample of audits and self-assessments has been summarized for each program in order to demonstrate that program effectiveness is being assessed on an ongoing basis. The focus of these reviews was on effectiveness of the programs at Pickering NGS, where specific information is available. Results from these audits and self-assessments will be considered in the Global Assessment process. It is noted that audits and self-assessments are, by their nature, self-critical and are used to drive excellence in performance. As a result, the broad review scope of program audits focuses on

identifying improvement opportunities rather than presenting a balanced picture of program performance.

Program effectiveness is also monitored and addressed through the Fleetview Program Health and Performance Reporting process [11]. This process involves direct oversight by the Chief Nuclear Officer, and includes participation by the Nuclear Executive Committee members. Programs are reviewed, senior oversight is provided, and improvement plans are generated.

The list of Nuclear Programs to be assessed for each Safety Factor was derived from review of current OPG Governance, and has used the most recent version of these documents as of the PSR2 freeze date of January 15, 2016.

3.4 Additional Reviews

A review of the R04 Pickering LCH [4] was also performed to determine if there are any impacts associated with operation of the Pickering Units past 2020 on the following (all related to Safety Factor 14):

- Commitments previously made to the CNSC;
- Open CNSC action items; and
- Exemptions granted by the CNSC.

The PSR2 assessment includes identification and review of previously identified Pickering-specific or programmatic PSR1 gaps related to Safety Factor 14 (as identified in the Darlington ISR Integrated Implementation Plan [12] and Pickering Units 5-8 Continued Operations Plan [10]) to ascertain the status of OPG's improvement plan(s) or other dispositions to address these and the implications of extending operation beyond 2020 (if any).[8]

Fukushima Action Items were reviewed to identify implications of extending operation beyond 2020 (if any). The methodology for this review is provided in a separate PSR2 FAI Review Report.

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[8] PSR2 includes consideration and confirmation that the findings of PSR1 remain valid, as applicable, for the operation period. This includes assessment of PSR1 conclusions against implications resulting from extended operation. In particular, Pickering PSR1 results are applicable to PSR2 if there was a PSR1 gap that is still open, or if a closed PSR1 gap could be affected by extended operation. If so these gaps are carried forward into PSR2 for consideration in the Global Assessment. (When references to PSR1 are made, the source document is identified and the relevant text from that source document is summarized in the context of PSR2.) With respect to the Darlington ISR, much of the evaluation of Safety Factor health is based on programs and practices that apply across OPG's nuclear operations. As a result, Darlington ISR programmatic conclusions are applicable to Pickering PSR2 where Pickering is confirmed to follow the same nuclear programs and practices that were assessed for Darlington. Darlington PSR1 results are applicable to Pickering PSR2 if there are Darlington PSR1 gaps that are found to be relevant to Pickering PSR2.

Any PSR2 gaps identified as a result of the Safety Factor 14 review which need to be addressed in other Safety Factor Reports are also discussed.

4.0 REVIEW FINDINGS

4.1 Review Tasks

The sub-sections below provide an assessment of the adequacy of applicable OPG Governance, Programs, Policies and Procedures (as well as Instructions and Guidelines, as applicable) in demonstrating compliance against the Safety Factor 14 Review Tasks. The Review Tasks focus on radiological impacts only. While OPG programs also address non-radiological impacts, they are not within the scope of this Safety Factor.

4.1.1 Review Task #1: Permitted Release Limits

Confirm there are procedures in place to ensure that permitted release limits of radiological substances are not exceeded and, if they are, that appropriate corrective action is taken to minimize the possibility of limits being exceeded in the future.

4.1.1.1 Procedures to Ensure Permitted Release Limits are not Exceeded

P-MAN-03480-00001, "Environmental Emissions Control" [13] provides the Chemistry Laboratory, Operators, Shift Managers and Environment Compliance Section with a summary of the limits and actions required for the control of environmental emissions at Pickering NGS. This document includes routine environmental monitoring of radiological, chemical (non-radiological) and thermal emissions.

Ensuring Radiological CNSC Emission Limits are not Exceeded

Through the Nuclear Safety and Control Act (NSCA) [14], the CNSC stipulates the requirements in Section 10.1 of the Pickering NGS Operating Licence (PROL-48.02/2018, "Nuclear Power Reactor Operating Licence: Pickering Nuclear Generating Station") [15] for the licensee to control and monitor releases of nuclear substances from the nuclear facility. One way in which the releases are controlled and monitored is to ensure that the releases shall not exceed the limits identified in OPG Reports NA44-REP-03482-00001, "Derived Release Limits and Environmental Action Levels for Pickering Nuclear Generating Station A" [16] and NK30-REP-03482-00001, "Derived Release Limits and Environmental Action Levels for Pickering Nuclear Generating Station B" [17].

Section 1.2.4 of N-PROG-OP-0006, "Environmental Management" [8] describes the effluent control program, which establishes controls and limits associated with radiological releases and which ensures releases are only emitted through approved and monitored pathways. Aspects of this program include maintaining radiological emissions to the environment from OPG facilities below the applicable regulatory emission limits (Derived Release Limits (DRLs)), and As Low As Reasonably Achievable (ALARA), taking social and economic factors into account, as described in N-STD-OP-

0042, “Controlling Radiation Exposure of the Public and the Environment to as Low as Reasonably Achievable” [18]. This is achieved through elements such as:

- Management control over work practices
- Personnel qualification and training
- Control of public exposure to radiation, and
- Planning for unusual situations.

OPG also uses Internal Investigation Levels (IILs) to help keep radionuclide levels in effluents low and to provide early warning before reaching an Action Level. Actions taken when IILs or Action Levels are exceeded are discussed in Section 4.1.1.2.

N-STD-OP-0031, “Monitoring of Nuclear and Hazardous Substances in Effluents” [19], establishes minimum standards for the monitoring of nuclear and hazardous substances in airborne and waterborne effluents for Nuclear facilities operating under normal and abnormal operating conditions. This Standard describes the:

- Authority and hierarchical structure of effluent monitoring related documentation
- Fundamental principles and objectives of effluent monitoring
- Conditions that determine when monitoring is required
- Requirements for effluent monitoring programs and equipment.

In addition, N-PROC-MA-0002, “Work Planning” [20], requires that the environmental risk is assessed during work planning. The Station Alignment Meeting package includes a Risk Management matrix, which identifies environmental risk issues. This matrix is reviewed daily, and any changes in environmental risk status are flagged to the station operating staff and the station senior management team.

4.1.1.2 Corrective Action to Minimize the Possibility of Limits Being Exceeded in the Future

Radiological Hazardous Substance Release Non-Compliances

The framework for the control of radioactive effluents includes IILs, which are administrative targets meant to help keep radionuclide levels in effluents low and provide an early warning before reaching an Action Level. In accordance with P-REP-03480-00009, “Internal Investigation Levels and Normal Operating Levels for Radionuclide Releases in Airborne and Liquid Effluents from Pickering Nuclear” [21], exceeding an IIL does not require reporting to the CNSC, as it is an internal limit. When an IIL is exceeded, a Station Condition Record (SCR), in accordance with N-PROC-RA-0022, “Processing Station Condition Records” [22], is raised and, if required, a Corrective Action Plan (CAP) is prepared and executed in accordance with N-PROG-RA-0003, “Corrective Action” [23].

An Action Level is a specific dose of radiation that, if reached, may indicate that corrective actions are required associated with some part of the Radiation Protection Program. There are two types of Action Levels: one for each monitored radionuclide group release category, and one for the combined dose to the public from all radionuclide releases from all facilities on the site.

Section 16.0 of NA44-REP-03482-00001 [16], Section 16.0 of NK30-REP-03482-00001 [17], and Section 14.0 of P-REP-03482-00001, "Derived Release Limits and Environmental Action Levels for Pickering Nuclear Sewage Effluent" [24], define the two types of Action Levels:

- Action Levels for individual radionuclide release groups are designated according to the group represented, i.e., the airborne tritium, noble gas, particulate, iodine or carbon-14 Action Levels and the waterborne HTO, gross beta-gamma, or carbon-14 Action Levels.

- The Combined Dose Action Level represents a control on the total dose impact resulting from the release of all radionuclides on a site (airborne and waterborne). The total dose is not a measurable parameter. Total dose is conservatively calculated from all radionuclide release rate measurements by using DRLs as conversion factors.

In accordance with Section 2.8 of N-INS-03480-10008, "Instruction to Establish Environmental Action Levels for Nuclear Stations" [25], when it is discovered that a radionuclide release group Action Level or a site combined dose Action Level is reached, the following actions are required to take place:

- Notification of Canadian Nuclear Safety Commission
  Notification requirements are detailed in N-PROC-RA-0020, "Preliminary Event Notifications" [26].

- Determination of Cause for Reaching the Action Level
  An SCR is initiated, and the prescribed actions will be followed as outlined in the CNSC Radiation Protection Regulations. An investigation is conducted to determine the cause for reaching the Action Level and a CAP is developed in accordance with N-PROC-RA-0022 [22] and N-PROG-RA-0003 [23]. An SCR would also be initiated if an Action Level was approached.

- Preparation of Action Level Report
  N-PROC-RA-0005, "Written Reporting to Regulatory Agencies" [27], describes the process to be followed when submitting an Action Level Report to the CNSC.

Environmental staff are required to review the Action Levels periodically and when the station DRLs are reviewed, and to review the IILs annually at a minimum, to determine if they need to be revised. The review is based on trends in normal operating levels from quarterly and annual station performance data. As a result of the reviews, any revisions to IILs are documented in site level documents. Any significant changes to operating conditions, radionuclide release monitoring methods or equipment, or other circumstances may result in an earlier review of these levels, with subsequent changes if required. Action Levels are updated based on N-INS-03480-10008 [25]. IILs are documented in P-REP-03480-00009 [21] and updated as per Section 1.7 of N-STD-OP-0031 [19].

Environmental staff are required to perform an ALARA review every five years. ALARA is described in N-STD-OP-0042 [18], which provides specific directions for conducting a station-level review to evaluate the effectiveness of operations in keeping radiation exposures of the public and the environment to ALARA.

<u>Reporting Non-Compliance with Regulatory Release Limits</u>

Section 1.3.3 of N-PROG-OP-0006 [8] identifies OPG initiatives that support reporting requirements in accordance with CNSC regulations.

N-PROC-RA-0005 [27] describes the process nuclear staff follow to submit written event reports regarding a reportable event (in accordance with N-PROC-RA-0020 [26]) required or requested by a regulatory agency, including the CNSC.

N-PROC-RA-0047, "Communications with the Canadian Nuclear Safety Commission" [28], specifies the planning, review, approval, and records required for communications with the CNSC.

4.1.1.3 Conclusion

The assessment of this Review Task confirms that OPG has procedures in place to ensure that permitted release limits of radiological hazardous substances are not exceeded, and if they are, that appropriate mitigating and corrective actions are taken to minimize the possibility of limits being exceeded in the future. The intent of Review Task #1 is met and therefore Pickering NGS is compliant.

4.1.2 Review Task #2: Maintenance of Effluent Release Records

Confirm records of radiological effluent release are maintained in accordance with regulatory requirements.

4.1.2.1 Radiological Effluent Release Records

The CNSC through the NSCA [14] and Section 14 of SOR/2000-204, "Class 1 Nuclear Facilities Regulations" [29], stipulates the requirement in Section 10.1 of the Pickering NGS Operating Licence [15] for the licensee to control and monitor effluents for the release of nuclear substances.

OPG's environmental compliance is described in Section 1.4 of N-PROG-OP-0006 [8]. Section 1.2 of N-PROG-OP-0006 [8] discusses the programs to manage its environmental aspects, and refers to N-STD-OP-0031 [19], which establishes minimum standards for the monitoring and recording of radioactivity in airborne and liquid effluents from OPG facilities operating under normal and abnormal operating conditions.

Section 5.0 of N-STD-OP-0031 [19] states that the results of the effluent monitoring, which include the quantity and concentration of nuclear substances dispersed into the environment, and evaluation of effluent characteristics, shall be kept as permanent records in OPG's record management system.

Station Condition Records

As given in Section 1.4.3 of N-PROG-OP-0006 [8], in the event of non-compliances or potential adverse conditions (which may include unplanned releases and testing and monitoring deficiencies), OPG will record the event using the SCR process in accordance with N-PROC-RA-0022 [22]. In addition, OPG initiates an SCR for an exceedance of an IIL or Action Level for radiological substance releases in accordance with P-REP-03480-00009 [21] and N-INS-03480-10008 [25]. If required, an SCR would be followed up by a CAP.

4.1.2.2 Maintaining Effluent Release Records

Appendix A of N-LIST-00500-10000, "Routine Environment Regulatory Reports/Correspondence" [30] identifies the various regulatory reports required by the CNSC that are generated, including the frequency, applicable file numbers, and the responsible facility.

Section 1.3.5 of N-PROG-OP-0006 [8] describes the OPG programs which facilitate the maintenance of these radiological effluent release records. N-PROG-OP-0006 [8] refers to N-PROG-AS-0006, "Records and Document Control" (superseded by OPG-PROG-0001, "Information Management" [31], see Document Change Request (DCR) 132951 "N-PROG-AS-0006 has been superseded by OPG-PROG-0001 – Update References" [32] for identification of the change to the document), which establishes a series of standards and procedures for management of Nuclear records and documents throughout their life cycle, regardless of media. The program lays out requirements for managed systems of all activities related to records and documents.

Section 4.2 of N-PROC-RA-0022 [22] describes that electronic and paper records of SCRs will be retained as permanent records in accordance with applicable codes and standards.

4.1.2.3 Conclusion

The assessment of this Review Task confirms that OPG has programs in place to ensure records of radiological effluent release are maintained in accordance with regulatory requirements. The intent of Review Task #2 is met and therefore Pickering NGS is compliant.

4.1.3 Review Task #3: Requirements for Alarm Systems

Confirm that a program exists to define the requirements for alarm systems to respond to unplanned effluent releases from on-site facilities.

Section 1.2.4 of N-PROG-OP-0006 [8], specifies that N-STD-OP-0031 [19] establishes minimum standards for the monitoring of radioactivity in airborne and liquid effluents in facilities operating under normal and abnormal operating conditions.

Section 2.4 of N-STD-OP-0031 [19] specifies the requirements for alarm systems. This standard states that alarms may be either physical devices or procedures which are required to alert appropriate staff of monitor/sampler malfunction, loss of sample/test results, or emissions in excess of a set point or target as specified below:

- Performance monitoring

  1) Monitor/sampler inoperative; e.g., loss of power

  2) Sample/test results unavailable; e.g., loss of sample or failure of sample collection system

  3) Cumulative emissions above set point or target

- Control monitoring

  1) Monitoring inoperative; e.g., loss of power

  2) Sample/test results unavailable; e.g., loss of sample or sample collection system failure

  3) Emissions above set point or target

Specific monitoring requirements are described in Sections 2.5 to 2.8 of N-STD-OP-0031 [19], respectively for:

- Airborne Effluents

- Waterborne Effluents, and

- Forebay Influent and Condenser Cooling Water or Outfall Streams.

As per N-STD-OP-0031 [19], OPG facilities are required to have an Emission Monitoring Plan that documents the site's Emission Monitoring Program. P-PLAN-03480-00001, "Pickering Nuclear Radioactive and Hazardous Emissions Monitoring Plan" [33], demonstrates that a comprehensive program for monitoring and controlling effluent releases is in place.

In addition, a summary of the limits and actions required for the control of an environmental release from Pickering NGS is provided in P-MAN-03480-00001 [13].

4.1.3.1 Conclusion

The assessment of this Review Task confirms OPG has programs to define the requirements for alarm systems to respond to unplanned effluent releases from on-site facilities. The intent of Review Task #3 is met and therefore Pickering NGS is compliant.

4.1.4 Review Task #4: Station Environmental Data Publication

Confirm the environmental data recorded by the station is published and is available on request to the general public.

4.1.4.1 Published Radiological Environmental Data

Section 1.3.3 of N-PROG-OP-0006 [8] references N-LIST-00500-10000 [30], which documents the routine regulatory correspondence generated by the Environmental Reporting, Environmental Services and Pickering Operations Support Departments and other departments issuing environmental reports as required. The regulatory agencies involved are the Ontario Ministry of the Environment and Climate Change (MOECC), Statistics Canada, Environment Canada, and the CNSC.

Although not a regulatory requirement, OPG publishes radiological environmental data recorded by the station in the Radiological Environmental Monitoring Program (REMP) and Environmental Emissions data reports, which are available to the public through OPG's corporate website [34].

4.1.4.2 Responding to Public and Non-Regulatory Stakeholder Requests

OPG program N-PROG-OP-0006 [8] makes reference to OPG standard N-STD-AS-0013, "Nuclear Public Information and Disclosure" [35]. This standard documents the process for provision of information to stakeholders and the public regarding activities and operations, including environmental management, as well as the receipt, documentation and response to concerns, complaints, and inquiries received from stakeholders and the public.

Section 1.1.1 of N-STD-AS-0013 [35] states that:

- OPG Nuclear shall develop, maintain and implement an annual public information work plan that supports the commitments of the Public Information Disclosure and Transparency Protocol and is conducted in accordance with the ethical principles of integrity, excellence and citizenship as outlined in the OPG Code of Business Conduct. Plans will be developed taking into consideration:
    - The type of facility and activities being regulated.

- The risks to public health, safety and security, and the environment posed by the facility or activity.

- The level of public interest or concern.

- Communications with stakeholders and the public are conducted in a planned manner by, or in consultation with, appropriate Stakeholder Relations staff.

- Information should be communicated on an ongoing and timely basis, and should be respective of both the public's perception of risk and the level of public interest of station operations, activities, and anticipated effects on the environment and the health and safety of persons.

- Public communications shall be informative, timely, accurate and material information will be disclosed in accordance with applicable legal and regulatory requirements.

Section 1.2 of N-STD-AS-0013 [35] states that:

- Methods of communication include the preferred use of modern electronic means such as internet and social media, where possible, but also include multiple communication vehicles to enhance public understanding and ensure effective reach of all appropriate target audiences. Methods of communication may include the following:

  - Public meetings and briefings
  - Stakeholder meetings and briefings
  - Community engagement and information sharing and consultation
  - Newsletters
  - Information brochures, videos and fact sheets on operations and activities
  - Nuclear divisional and company performance reports and quarterly emissions data reports
  - Posting and communicating reports and regulatory information relating to health, safety and environment
  - Presentations
  - Paid advertising
  - Public tours (limited)
  - Electronic communications and notices

- Community event participation
- Stakeholder notifications
- Media releases
- Public access to information and face-to-face contact
- Websites and social media.

Section 1.7 of N-STD-AS-0013 [35] states that:

- Corporate Stakeholder Relations staff shall receive, document and respond to concerns, complaints, and irregular inquiries, including those related to impacts by the facility on the environment, from stakeholders and the public.

N-STD-AS-0013 [35] also gives direction on public information strategies and products, public information disclosure and transparency period, regulatory and public disclosure of significant events and reports, trained and qualified stakeholder relations staff, public and media opinion, and communication standards.

4.1.4.3 Conclusion

The assessment of this Review Task confirms that OPG has governance in place for ensuring that the environmental data recorded by the station is published and is available on request to the general public. The intent of Review Task #4 is met and therefore Pickering NGS is compliant.

4.1.5 Review Task #5: Pickering Environmental Data Measurements

Review the environmental data recorded by the station and compare with the values measured before the plant was put into operation.

OPG does not have a program in place to compare current and pre-operational environmental data. The lack of a program to compare current and pre-operational environmental data falls under Gap #357 in the 2007 Pickering B ISR [36]. This gap was evaluated in NK30-CORR-00770-0201414, "Pickering B Integrated Safety Review Gap Evaluations: Environment" [37], where it was concluded that no further action was recommended to address this discrepancy. This is therefore not a PSR2 gap.

Initial environmental effects studies (HSD-TS-90-5, "Pickering NGS-B: Analysis of Pre- and Post-operational Environmental Radiological Data" [38]) were performed over a ten-year period that covered the pre-operational phase (1979-1981), commissioning phase (1982-1985), and the initial operational phase (1986-1988) of Pickering Units 5-8. The results of these studies contributed to the development of the Environmental Monitoring Program (EMP) in effect at Pickering 1&4 and 5-8. Presently, OPG maintains an off-site EMP [39] (REMP until 2013), which includes Pickering NGS, in accordance with N-PROC-OP-0025, "Management of the Environmental Monitoring Programs" [40] and P-REP-03443-00003, "Detailed Design of Environmental

Monitoring Program for Pickering Nuclear Generating Station" [41], as required to meet the environmental protection requirements specified in the Nuclear Power Reactor Operating Licence [15]. The EMP requires that environmental samples obtained by the station be compared to background radiation levels recorded at control monitoring locations in an environment free from the influence of OPG Nuclear facilities. Therefore, for analysis, measurements taken at control monitoring locations serve as baseline pre-operational radiological data. The annual EMP reports include long-term trend plots for select radiological data, including pre-operational data. The 2005 EMP report identified tritium data dating to 1975. The comparison of environmental data from Pickering NGS to environmental data taken at background locations provides an indication of the effects that Pickering NGS operation could have on the environment, and therefore serves the same purpose as a comparison to values measured prior to Pickering NGS operation. The process of comparing environmental data to background levels will not be affected by Pickering NGS operation beyond 2020. Therefore, this is not a PSR2 gap.

4.1.5.1 Conclusion

The information discussed above is an update to the argument used to determine no further action was required to address the fact that OPG does not have a program in place to compare current and pre-operational environmental data. The documentation on which the argument is based has not fundamentally changed since NK30-CORR-00770-0201414 [37] was authored. Therefore, the conclusions of the gap evaluation have not changed and no further action is required as a result of this PSR2 assessment. This resolved PSR1 Pickering ISR issue is therefore not a PSR2 gap. The intent of Review Task #5 is met and therefore Pickering NGS is compliant.

4.1.6 Review Task #6: Use of Land External to Pickering NGS Site

Confirm there is a process to address changes in the use of land external to the site with respect to the impact on public safety from facility releases.

The Pickering NGS Operating Licence [15] states in Licence Condition 1.4 that, the licensee shall control the use and occupation of lands situated within the exclusion zone described in NA44-SR-01320-00001, "Pickering A Safety Report" [42], and NK30-SR-01320-00001, "Pickering B Safety Report – Part 1" [43].

4.1.6.1 Pickering Safety Reports

Land use surrounding the plant is discussed in NA44-SR-01320-00001 [42], and NK30-SR-01320-00001 [43]. Part 1, Section 2 of each of the Pickering Safety Reports discusses the site location and access, surrounding populations, land use, local agriculture, industries, fishing, recreation, and transportation. Per NA44-SR-01320-00001 [42] and NK30-SR-01320-00001 [43], and in accordance with the Pickering NGS LCH [4], it is a requirement to update the Pickering Safety Reports periodically (at least every 5 years as per CNSC REGDOC 3.1.1, "Reporting Requirements for Nuclear

Power Plants” [44]) and subsequently submit them to the CNSC as reference documents.

4.1.6.2 Environmental Monitoring Program

Changes in the use of land external to the site with respect to the impact on public safety from facility releases are addressed in OPG Procedure, N-PROC-OP-0025 [40].

N-PROC-OP-0025 [40] provides direction and accountabilities for design, implementation, and operation of EMPs at OPG nuclear facilities. As stated in Section 1.2 of N-PROC-OP-0025 [40], the EMP shall consider the applicability of the following objectives:

a) Assess the level of risk on human health and safety, and the potential biological effects in the environment of the contaminants and physical stressors of concern arising from the facility;

b) Demonstrate compliance with limits on the concentration and/or intensity of contaminants and physical stressors in the environment or their effect on the environment;

c) To check, independently of effluent monitoring, on the effectiveness of containment and effluent control, and provide public assurance of the effectiveness of containment and effluent control;

d) Verify the predictions made by the Environmental Risk Assessment (ERA), refine models used in the ERA, or reduce the uncertainty in the predictions made by the ERA.

The program design is also structured to satisfy the other EMP objectives as listed in Section 1.2 of N-PROC-OP-0025 [40]. Section 1.5.2 of N-PROC-OP-0025 [40] outlines the design framework in support of radiological dose calculations, which consists of the following design elements:

- Provincial background data
- Site specific survey
- Pathway analysis
- Identification of the contaminants (radionuclides) and pathways of importance.

Site Specific Survey

Site specific surveys allow Environment Operations Support (EOS) to identify the various potential critical groups around each nuclear site and are used for development of the EMPs and site DRLs, and for calculating public collective dose. Site specific survey instructions are documented in N-INS-03481-10000, “Instruction for Performing a Site Specific Survey for Ontario Power Generation Nuclear Sites” [45].

Specific responsibilities of EOS related to Site Specific Surveys as outlined in N-PROC-OP-0025 [40] are as follows:

a) EOS shall conduct a site specific survey for Pickering NGS and its surrounding environment to gather local characteristics, such as:

- Population distribution
- Produce distribution
- Land use patterns by the public around the facility (e.g., farming, industrial areas, well water usage, and recreational usage)
- Dietary patterns (e.g., sources of drinking water, fraction of food intake obtained from local sources)

b) EOS should review the site specific surveys every five years to assess the validity of data and determine if updated surveys are required.

c) EOS should update the site specific surveys when significant changes affecting local characteristics occur, such as land use patterns.

d) EOS should document the results of the updated site specific surveys in individual reports for Pickering and Darlington.

Pathway Analysis

Pathway analysis identifies the significance of each environmental transport pathway for radiological contaminants released by station emissions. Doses shall be calculated by radionuclide and pathway of exposure for each potential critical group using the projected facility emissions for the next five years. Specific responsibilities of EOS related to Pathway Analysis as outlined in N-PROC-OP-0025 [40] are as follows:

a) EOS should conduct a pathway analysis for Pickering NGS every five years. The results of the most recent pathway analyses are documented in the program design review report for Pickering, P-REP-03443-00004, "Pickering Environmental Monitoring Program (EMP) Review" [46].

b) EOS should use the following parameters and models, as applicable, when calculating doses for pathways analyses:

- Pickering NGS Site DRL model, as described in the most recent DRL reports;
- Most recent site-specific survey results;
- Station operational changes and projected five year emissions;
- Previous three to five years of meteorological data;

- Central food intake rates as per CSA N288.1-08, "Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities" [47], and N-INS-03443-00001, "Methodology for Data Analysis and Public Dose Determination for the Environmental Monitoring Program" [48].

Identification of Contaminants and Pathways of Importance

a) Guidelines for determination of significant contaminants and pathways are outlined in Section 7.5 of CSA N288.4-10, "Environmental Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills [49] and N-INS-03443-00001 [48].

b) Pathways, radionuclides, and contaminants of concern identified as significant contributors to receptor exposure shall form the basis of the EMP sampling plan. Guidance for determining which sampling locations should be included in the EMPs is provided in Section 7.6 of CSA N288.4-10 [49].

4.1.6.3 Conclusion

The assessment of this Review Task confirms OPG has a process to address changes in the use of land external to the site with respect to the impact on public safety from facility releases. The intent of Review Task #6 is met and therefore Pickering NGS is compliant.

4.1.7 Review Task #7: Monitoring Program Comprehensiveness

Confirm that the monitoring program is appropriate and sufficiently comprehensive. In particular, confirm that the radiological impact of the plant on the environment is not significant compared with that due to other sources of radiation.

4.1.7.1 Monitoring Programs and Assessments

A number of programs and assessments are in place in order to assure adequate provision for the protection of the environment and human health and safety.

Environmental Management System

OPG-PROG-0005, "Environmental Management System" [9] outlines the Environmental Management System (EMS) requirements to implement the requirements of OPG-POL-0021, "Ontario Power Generation Environmental Policy" [50], including monitoring and measurement requirements. As stated in Section 1.3.3 of OPG-PROG-0005 [9], OPG monitors and measures performance against objectives and targets established in business planning through quarterly reporting requirements. Benchmarking is an important tool that is used to assist in setting performance objectives and targets, against which environmental performance is measured. Section 1.5.1 of OPG-PROG-0005 [9] states that, where appropriate, procedures include that monitoring and measurement equipment is calibrated or verified to be functioning properly and that records are retained.

The EMS is registered under the International Organization for Standardization (ISO) 14001 Standard, "Environmental Management Systems – Requirements with Guidance for Use" [51], and thus must be continually improved in accordance with this Standard.

N-PROG-OP-0006 [8] describes the requirements of the OPG EMS that provide the framework for environmental protection within OPG Nuclear, and continual improvement of environmental performance. This program ensures nuclear activities are conducted such that adverse environmental effects are prevented or mitigated.

Section 1.4.1 of N-PROG-OP-0006 [8] states that OPG maintains procedures to regularly measure and monitor its environmental performance. Measurements and observations allow OPG to track progress on meeting objectives and targets, demonstrate compliance with legal and other requirements, and provide data to evaluate operational controls. OPG also maintains procedures to ensure that information collected is valid, including N-PROC-MA-0069, "Control and Calibration of Measuring and Test Equipment" [52], which establishes the process for control and calibration of Measuring and Test Equipment.

Environmental Monitoring Program

N-PROC-OP-0025 [40] provides direction and accountabilities for design, implementation, and operation of EMPs at OPG's nuclear facilities. These programs are required to:

1. Assess potential effects on humans and the environment by nuclear substances.

2. Demonstrate and confirm that radiation doses to members of the public resulting from the operation of OPG nuclear facilities remain below the annual legal limit specified in the current Radiation Protection Regulations under the NSCA [14].

N-PROC-OP-0025 [40] also outlines quality management practices for the operation of the EMP. More information on the contents of N-PROC-OP-0025 [40], including EMP objectives and design elements, are given above in Section 4.1.6.2 of this Safety Factor report.

P-REP-03443-00003 [41] presents the detailed design of the EMP for Pickering NGS, following CSA N288.4-10 [49]. The EMP design addresses site specific objectives covering the aspects of regulatory requirements, risk assessment results, confirmation of effluent control, areas of regulatory interest, stakeholder commitments, environmental model confirmation, and due diligence. The EMP design also includes detailed monitoring design, requirements, and objectives related to Pickering NGS operations. ERA follow-up monitoring is also included.

Section 4.2.1 of P-REP-03443-00003 [41] discusses that public radiation dose should be calculated annually to demonstrate compliance with public dose limits and to demonstrate compliance with the requirement to operate at ALARA.

Presently, Pickering maintains an off-site EMP [39] in accordance with N-PROC-OP-0025 [40] and P-REP-03443-00003 [41], as required to meet the environmental protection requirements specified in the Nuclear Power Reactor Operating Licence [15]. As discussed in Section 1.5.2.1 of N-PROC-OP-0025 [40], radioactivity in the environment is measured near Pickering as well as at provincial background locations to determine the radiological impact on the public resulting from the operation of the station as part of the EMP.

Effluent Monitoring

Section 1.2.4 of N-PROG-OP-0006 [8] states that radiological emissions from OPG Nuclear facilities shall not exceed the DRLs specified in station Nuclear Power Reactor Operating Licences issued by the CNSC.

Section 1.2.4 of N-PROG-OP-0006 [8] makes reference to N-STD-OP-0031 [19], which establishes minimum requirements for the monitoring of nuclear substances in airborne and waterborne effluents from OPG nuclear facilities operating under normal and abnormal operating conditions. This document takes authority from N-PROG-OP-0006 [8].

Section 2.2 of N-STD-OP-0031 [19] discusses effluent monitoring criteria:

- Types of monitoring include performance monitoring and control monitoring.
- The effluent monitoring program of nuclear substances follows a risk-based approach based on the Maximum Probable Emission Rate. The requirement for performance monitoring, control monitoring, and emission reporting is determined based on the ratio of the Maximum Probable Emission Rate to the DRL for each radionuclide or radionuclide group.

Monitoring requirements and requirements for alarm systems for effluents given in N-STD-OP-0031 [19] are discussed above in Section 4.1.3 of this Safety Factor report.

As per N-STD-OP-0031 [19], OPG facilities are required to have an Emission Monitoring Plan that documents the site's Emission Monitoring Program. P-PLAN-03480-00001 [33] outlines the Pickering-specific monitoring requirements for the radiological airborne and liquid effluent pathways in Sections 1 and 2, respectively. This plan also examines compliance of the monitoring program with CSA N288.5-11, "Effluent Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills" [53], and therefore demonstrates that a comprehensive program for monitoring effluent releases is in place.

Groundwater Monitoring

Section 1.2.4 of N-PROG-OP-0006 [8] makes reference to N-PROC-OP-0044, "Contaminated Lands and Groundwater Management" [54], which outlines directions and accountabilities for identifying, assessing, and managing contaminated lands within OPG nuclear facilities.

Section 1.2 of N-PROC-OP-0044 [54] includes direction for establishing and managing a Nuclear groundwater monitoring program, including program design, sampling and analysis, quality assurance and quality control, data interpretation, and well inspection and maintenance. The groundwater monitoring program monitors the on-site groundwater quality changes over time.

4.1.7.2 Pickering Radiological Impact on the Environment

Dose to Humans

Environmental samples are collected as part of the EMP to support the public dose calculations. Sample radionuclide concentrations are compared against background concentrations.

As discussed in Section 1.5.2.1 of N-PROC-OP-0025 [40] and Section B.1.4.2.1 of CSA N288.4-10 [49], as part of the EMP, measured data are used together with station emissions data to determine the dose received by members of the public living near the station, known as potential critical groups. The highest estimated potential critical group dose establishes the official public dose for the site.

The dose[9] to the public resulting from Pickering station operations continues to be a very small percentage of the annual legal limit of 1,000 μSv/y. The critical group dose for 2014 was 1.2 μSv [39], which is on the order of 0.1% of the legal limit and is virtually unchanged from the 2013 dose of 1.1 μSv.

The Pickering NGS dose to the critical group for 2014 [39] was equivalent to 0.1% of the estimated background dose to humans around Pickering NGS site of 1,400 μSv/y from naturally-occurring and anthropogenic radiation. The critical group dose has remained at approximately 0.1% of the estimated background dose since 2009 [39].

Dose to Non-human Biota

Dose to non-human biota is not included as part of the EMP. Ecological Risk Assessments, which involve determining the radiological dose to non-human biota, are included as part of the ERA process. As per CSA N288.6-12, "Environmental Risk Assessments at Class I Nuclear Facilities and Uranium Mines and Mills" [55], the ERA is

[9] In some contexts in the CSA standards and OPG documents, the term "dose" is used interchangeably with "dose rate". The intended meaning of the term is apparent by reference to the units, with "dose/time period" indicating dose rates.

to be reviewed every five years or when major changes occur. This requirement is reflected in Section 1.5.1 of N-PROC-OP-0025 [40].

P-REP-07010-10012, "Environmental Risk Assessment for Pickering Nuclear" [56], was performed in 2014. As part of this assessment, an assessment of the radiological dose received by terrestrial and aquatic biota from air, surface water, and soil was performed. The limiting radionuclides used to represent all radionuclides in this assessment were those emitting gross beta/gamma, including H-3, Cs-134, Cs-137, Co-60, and Ar-41.

Representative doses to biota did not exceed radiological dose benchmarks given in CSA N288.6-12 [55], except for the maximum doses to the Earthworm (8.5 mGy/d) and Red-winged Blackbird (6.35 mGy/d) received on the Pickering NGS site [56]. However, these doses are driven by the maximum tritium concentrations in on-site soil close to the reactor buildings and therefore they are not representative of the dose received by the entire on-site population of Earthworms or by the more mobile Red-winged Blackbird. Off-site biota would also not be exposed to these levels. The mean doses received by these receptors are more representative: 2.37 mGy/d for Earthworms and 1.77 mGy/d for the Red-winged Blackbird [56]. These doses are below the CSA N288.6-12 radiological dose benchmarks [55]. These mean doses represent the highest potential doses received by non-human biota.

A background dose around the Pickering NGS site applicable to terrestrial and aquatic non-human biota was not available in OPG documentation. However, the doses to indicator species in P-REP-07010-10012 [56] encompass both background dose and dose from Pickering NGS site and do not exceed CSA N288.6-12 [55] dose benchmarks. These benchmarks provide a guideline for acceptable dose to non-human biota and hence, the radiological doses from Pickering NGS do not have a significant or adverse effect on non-human biota. This is therefore not a PSR2 gap.

<u>Radionuclide Levels in Environmental and Other Media</u>

In N-REP-03443-10014 [39], concentrations of radionuclides in environmental and other media around the Pickering NGS site, including air, fruits and vegetables, milk, and fish, were found to be higher than levels measured at background monitoring stations, as was expected [39]. Radionuclide levels in drinking water have remained below the OPG target of 100 Bq/L. Emissions have remained at a very small fraction of the DRLs [39].

Although the concentrations measured around Pickering site are higher than background levels, the emissions of radionuclides to the environment from Pickering NGS have remained at a very small fraction of the CSA N288.1-compliant DRLs ([39], [47]), and Pickering NGS meets the terms of its operating licence [15]. The Environmental Risk Assessment is reviewed every five years to ensure up-to-date assessment of the impact of Pickering NGS in terms of risk to biological receptors.

The CNSC has implemented its Independent Environmental Monitoring Program (IEMP) to verify the areas around nuclear facilities are safe. Under this program, the

CNSC measured radionuclide levels in environmental and other media around Pickering site in 2014 and 2015. Media sampled include air particulate, lake water, soil and sediment, grass and wild vegetation, and foodstuffs. Based on the IEMP, the CNSC has confirmed that there are no health impacts around the Pickering NGS facility. The measured values for all samples and for all reported radionuclides represent a fraction of the CNSC reference levels [57].

This is therefore not a PSR2 gap.

4.1.7.3 Conclusion

The assessment of this Review Task confirms OPG has appropriate and comprehensive monitoring programs in place for Pickering NGS.

Review Task #7 states that the radiological impact of the plant on the environment should not be significant compared with that due to other sources of radiation. This was confirmed for human exposure to radiation.

Doses to non-human biota were found to be less than radiological dose benchmarks in the most recent ERA [56] and therefore do not represent an adverse effect to non-human biota. Emissions of radionuclides to the environment from Pickering have remained at a very small fraction of the DRLs, and therefore Pickering NGS meets the terms of its operating licence [15]. The intent of Review Task #7 is met and therefore Pickering NGS is compliant.

4.2 L/R/C/S Reviews

As per Section 2.2 of this report, detailed reviews for eight L/R/C/Ss with content applicable to Safety Factor 14 are provided in References [6] and [7]. Associated findings applicable to Safety Factor 14 are summarized in Table 3 below.

Table 3: PSR2 L/R/C/S Review Results for Safety Factor 14

L/C/R/S Reviewed	PSR2 L/R/C/S Review Results for Safety Factor 14
CSA N288.1-14, "Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities"	There are no PSR2 gaps for CSA N288.1-14. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CSA N288.1-14.
CSA N288.4-10, "Environmental Monitoring Program at Class I Nuclear Facilities and Uranium Mines and Mills"	There are no PSR2 gaps for CSA N288.4-10. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CSA N288.4-10.

L/C/R/S Reviewed	PSR2 L/R/C/S Review Results for Safety Factor 14
CNSC REGDOC-2.9.1 (2013), "Environmental Protection Policies, Programs and Procedures"	There are no PSR2 gaps for CNSC REGDOC-2.9.1 (2013). Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CNSC REGDOC-2.9.1 (2013).
CNSC G-228, "Developing and Using Action Levels"	There are no PSR2 gaps for CNSC G-228 (2001). Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CNSC G-228 (2001).
CSA N288.6-12, "Environment Risk Assessments at Class I Nuclear Facilities and Uranium Mines and Mills"	There are no PSR2 gaps for CSA N288.6-12. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CSA N288.6-12.
CSA N288.5-11, "Effluent Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills Facilities"	There are no PSR2 gaps for CSA N288.5-11. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CSA N288.5-11.
CSA N288.3.4-13, "Performance Testing of Nuclear Air-Cleaning Systems at Nuclear Facilities"	There are no PSR2 gaps for CSA N288.3.4-13. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CSA N288.3.4-13.
CSA N288.7-15, "Groundwater Protection Programs at Class I Nuclear Facilities and Uranium Mines and Mills"	There are no PSR2 gaps for CSA N288.7-15. Per the definition of Compliance for a High Level review, Pickering has a PSR2 Compliance associated with CSA N288.7-15.

4.3 OPG Program Effectiveness Reviews

The OPG Nuclear Programs reviewed for Safety Factor 14 are identified in Table 2 and details of the associated effectiveness reviews for each of the N-PROGs or OPG-PROGs are provided in Appendix B.

4.4 Additional Review Findings

As discussed in Section 3.4, the PSR2 Safety Factor 14 assessment also included a review of commitments previously made to the CNSC, open CNSC action items, and exemptions granted by the CNSC, as identified in the R04 Pickering LCH [4], to determine if there are any impacts associated with operation of the Pickering Units past 2020. The review also included identification and review of previously identified programmatic Darlington PSR1 gaps related to Safety Factor 14 to determine impacts associated with operation of the Pickering Units past 2020. This assessment did not identify any gaps for Safety Factor 14.

Findings from the review of previously identified PSR1 gaps in the Pickering Units 5-8 Continued Operations Plan [10] are provided in a separate PSR2 COP Review Report. Findings from the review of Fukushima Action Items are provided in a separate PSR2 FAI Review Report. Results from the Continued Operations Plan and Fukushima Action Items reviews will be considered in the Global Assessment process.

There were no PSR2 gaps identified in this Safety Factor 14 Report that require discussion in other Safety Factor Reports.

5.0 RESULTS AND CONCLUSIONS

OPG Governance, Programs, Policies, Procedures, Instructions and Guidelines related to Safety Factor 14 were reviewed for the seven PSR2 Review Tasks in Section 4.1 of this report and resulted in no Pickering PSR2 gaps. L/R/C/S and OPG Nuclear Program effectiveness reviews for Safety Factor 14 were prepared per Sections 4.2 and 4.3, respectively, and resulted in no PSR2 gaps. Per Section 4.4, this report also included identification and review of previously identified programmatic Darlington PSR1 gaps related to Safety Factor 14 (to ascertain the implications of extending Pickering NGS operation beyond 2020), as well as a review of the R04 Pickering LCH [4] for any impacts of Pickering NGS operation beyond 2020 on a) OPG commitments previously made to the CNSC, b) open CNSC action items, and c) exemptions granted by the CNSC (all related to Safety Factor 14), which resulted in no PSR2 gaps.

The review of Safety Factor 14 has confirmed that Pickering NGS has an adequate and effective program for monitoring the radiological impact of the plant on the environment, which ensures that emissions are properly controlled and are as low as reasonably achievable.

6.0 REFERENCES

[1] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[2] CNSC REGDOC-2.3.3, *Periodic Safety Reviews*, April 2015.

[3] IAEA Specific Safety Guide No. SSG-25, *Periodic Safety Review for Nuclear Power Plants*, 2013.

[4] CNSC Report, LCH-PNGS-R004, *Pickering NGS: Licence Conditions Handbook*, December 22, 2015.

[5] OPG Report, P-REP-03680-00003 R000, *Pickering NGS Periodic Safety Review 2 (PSR2): Definition of Safety Factor Review Tasks*, May 30, 2016.

[6] OPG Report, P-REP-03680-0586480 R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 9, 11, and 15*, September 2016.

[7] OPG Report, P-REP-03680-00021 R000, *Pickering NGS PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 8, 10, 12, 13, and 14*, December 2016.

[8] OPG Program, N-PROG-OP-0006 R018, *Environmental Management*, April 21, 2015.

[9] OPG Program, OPG-PROG-0005 R004, *Environmental Management System*, October 24, 2014.

[10] OPG Plan, NK30-PLAN-00531-00001 R005, *Pickering 5-8 Continued Operations Plan*, December 2015.

[11] OPG Procedure, N-PROC-RA-0023 R018, *Fleetview Program Health and Performance Reporting*, August 2013.

[12] OPG Report, NK38-REP-03680-10185 R002, *Darlington NGS Integrated Implementation Plan (IIP)*, April 30, 2015.

[13] OPG Manual, P-MAN-03480-00001 R018, *Environmental Emisisons Control*, November 12, 2014.

[14] Government of Canada Consolidation, *Nuclear Safety and Control Act*, S.C., 1997, c. 9.

[15] CNSC, PROL-48.02/2018 Licence, *Nuclear Power Reactor Operating Licence: Pickering Nuclear Generating Station*, December 18, 2015.

[16] OPG Report, NA44-REP-03482-00001 R002, *Derived Release Limits and Environmental Action Levels for Pickering Nuclear Generating Station A,* January 28, 2011.

[17] OPG Report, NK30-REP-03482-00001 R002, *Derived Release Limits and Environmental Action Levels for Pickering Nuclear Generating Station B*, January 28, 2011.

[18] OPG Standard, N-STD-OP-0042 R003, *Controlling Radiation Exposure to the Public and the Environment to As Low As Reasonably Achievable*, August 2014.

[19] OPG Standard, N-STD-OP-0031 R006, *Monitoring of Nuclear and Hazardous Substance in Effluents*, October 2014.

[20] OPG Procedure, N-PROC-MA-0002 R028, *Work Planning*, April 2014.

[21] OPG Report, P-REP-03480-00009 R000, *Internal Investigation Levels and Normal Operating Levels for Radionuclide Releases in Airborne and Liquid Effluents from Pickering Nuclear*, March 2006.

[22] OPG Procedure, N-PROC-RA-0022 R032, *Processing Station Condition Records*, November 2014.

[23] OPG Program, N-PROG-RA-0003 R010, *Corrective Action*, January 2015.

[24] OPG Report, P-REP-03482-00001 R001, *Derived Release Limits and Environmental Action Levels for Pickering Nuclear Sewage Effluent*, July 2011.

[25] OPG Instruction, N-INS-03480-10008 R001, *Instructions to Establish Environmental Action Levels for Nuclear Stations*, July 2012.

[26] OPG Procedure, N-PROC-RA-0020 R018B, *Preliminary Event Notification*, December 2014.

[27] OPG Procedure, N-PROC-RA-0005 R015, *Written Reporting to Regulatory Agencies*, June 2015.

[28] OPG Procedure, N-PROC-RA-0047 R014, *Communications with the Canadian Nuclear Safety Commission*, April 2015.

[29] Government of Canada Department of Justice Regulation, SOR/2000-204, *Class 1 Nuclear Facilities Regulations*.

[30] OPG List, N-LIST-00500-10000 R004, *Routine Environment Regulatory Reports/Correspondence*, September 2014.

[31] OPG Program, OPG-PROG-0001 R009, *Information Management*, September 2015.

[32] OPG Engineering Change, DCR 132951 R003, *N-PROG-AS-0006 has been superseded by OPG-PROG-0001 – Update References*, July 2016.

[33] OPG Plan, P-PLAN-03480-00001 R008, *Pickering Nuclear Radioactive and Hazardous Emissions Monitoring Plan,* November 2015.

[34] OPG, *Reports*, 2016. [Online]. http://www.opg.com/news-and-media/Pages/reports.aspx. Accessed: June 2016.

[35] OPG Standard, N-STD-AS-0013 R007, *Nuclear Public Information and Disclosure*, February 2015.

[36] OPG Report, NK30-REP-03680-00010 R000, *Pickering NGS-B Integrated Safety Review – Environment*, May 2007.

[37] OPG Memorandum, NK30-CORR-00770-0201414, *Pickering B Integrated Safety Review Gap Evaluations: Environment*, September 2007.

[38] Ontario Hydro Technical Safety Standards, HSD-TS-90-5, *Pickering NGS-B: Analysis of Pre- and Post-operational Environmental Radiological Data*, July 1990. OPG document NK30-REF-07000-{54927}.

[39] OPG Report, N-REP-03443-10014 R000, *2014 Results of Environmental Monitoring Programs*, April 2015.

[40] OPG Procedure, N-PROC-OP-0025 R011, *Management of the Environmental Monitoring Programs,* January 8, 2016.

[41] OPG Report, P-REP-03443-00003 R000, *Detailed Design of Environmental Monitoring Program for Pickering Nuclear Generating Station*, September 2012.

[42] OPG Report, NA44-SR-01320-00001 R015, *Pickering A Safety Report*, July 2012.

[43] OPG Report, NK30-SR-01320-00001 R004, *Pickering B Safety Report – Part 1*, October 2010.

[44] CNSC, REGDOC 3.1.1, *Reporting Requirements for Nuclear Power Plants*, May 2014.

[45] OPG Instruction, N-INS-03481-10000 R000, *Instruction for Performing a Site-Specific Survey for Ontario Power Generation Nuclear Sites*, September 2009.

[46] OPG Report, P-REP-03443-00004 R000, *Pickering Environmental Monitoring Program (EMP) Review*, December 2015.

[47] Canadian Standards Association, CSA N288.1-08, *Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities*, September 2008.

[48] OPG Instruction, N-INS-03443-00001 R001, *Methodology for Data Analysis and Public Dose Determination for the Environmental Monitoring Program*, January 2016.

[49] Canadian Standards Association, CSA N288.4-10, *Environmental Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills*, May 2010.

[50] OPG Policy, OPG-POL-0021 R005, *Ontario Power Generation Environmental Policy*, November 2015.

[51] CAN/CSA-ISO 14001:2004, *Environmental Management Systems – Requirements with Guidance for Use*.

[52] OPG Procedure, N-PROC-MA-0069 R017B, *Control and Calibration of Measuring and Test Equipment*, January 2016.

[53] Canadian Standards Association, CSA N288.5-11, *Effluent Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills*, 2011.

[54] OPG Procedure, N-PROC-OP-0044 R003, *Contaminated Lands and Groundwater Management*, June 2014.

[55] Canadian Standards Association, CSA N288.6-12, *Environmental Risk Assessments at Class I Nuclear Facilities and Uranium Mines and Mills*, June 2012.

[56] OPG Report, P-REP-07010-10012 R000, *Environmental Risk Assessment Report for Pickering Nuclear*, January 2014.

[57] Canadian Nuclear Safety Commission, Independent Environmental Monitoring Program: Pickering Nuclear Generating Station, 2016. [Online]. http://nuclearsafety.gc.ca/eng/resources/maps-of-nuclear-facilities/iemp/pickering.cfm. Accessed: August 2016.

Appendix A: Nomenclature

ALARA	As Low As Reasonably Achievable
CAP	Corrective Action Plan
CNSC	Canadian Nuclear Safety Commission
COP	Continued Operations Plan
CPAD	Corrective/Preventive Action Database
CSA	Canadian Standards Association
DCR	Document Change Request
DRL	Derived Release Limit
EMP	Environmental Monitoring Program
EMS	Environmental Management System
EOS	Environment Operations Support
ERA	Environmental Risk Assessment
FAI	Fukushima Action Item
IAEA	International Atomic Energy Agency
IEMP	Independent Environmental Monitoring Program
IILs	Internal Investigation Levels
ISO	International Organization for Standardization
ISR	Integrated Safety Review
LCH	Licence Conditions Handbook
L/R/C/S	Laws, Regulations, Codes and Standards
MOECC	Ministry of the Environment and Climate Change
NGS	Nuclear Generating Station
NSCA	Nuclear Safety and Control Act
OPG	Ontario Power Generation
PARTS	Pickering A Return to Service
PCB	Polychlorinated Biphenyl
PROL	Power Reactor Operating Licence
PSR	Periodic Safety Review
PSR1	Periodic Safety Review 1 (earlier OPG PSR work and other associated assessments)
PSR2	Periodic Safety Review 2 (subsequent PSR per REGDOC-2.3.3)

REMP	Radiological Environmental Monitoring Program
SCR	Station Condition Record

Appendix B: OPG Program Effectiveness Review Results

B.1 N-PROG-OP-0006, "Environmental Management"

The Environmental Management Program describes both the requirements of the OPG EMS that provide the framework for environmental protection within OPG Nuclear, as well as Management's approach to ensure compliance with applicable environmental requirements, and conformance with the requirements of the ISO 14001 standard. The Environmental Management Program ensures OPG Nuclear activities are conducted such that adverse environmental effects are prevented or mitigated.

Nuclear Oversight conducted a performance based audit in June 2015, NO-2015-014, Environmental Management [B.1.1], to assess OPG Nuclear compliance with the Environmental Management Program (N-PROG-OP-0006), the ISO 14001 standard and to identify any performance improvement opportunities. The audit identified a number of opportunities for improvement; however, these were associated with conventional emissions and other non-nuclear issues, which are not applicable to this Safety Factor Report.

The Nuclear Programs department completed a self-assessment in July 2012, NO12-000421-SA [B.1.2], in order to assess the health of the Environmental Management governance framework, which is applicable to both Pickering and Darlington NGS. This involved a review of related SCRs, governance framework, Asset Suite and revision records. No actions were generated as a result of the self-assessment.

The Information Management department, Governance and Services section, completed a self-assessment in December 2014, BAS14-001300-SA [B.1.3], in order to assess compliance with N-PROG-OP-0006, Environmental Management. No findings/SCRs were generated from this self-assessment, however it was recommended to initiate a DCR in order to reference certain clauses of CSA N286-12, Management System Requirements for Nuclear Facilities [B.1.4], which has since been completed.

References

[B.1.1] OPG Nuclear Oversight Report, N-REP-01070-0550195 T06 (NO-2015-014), *NO-2015-014 Environmental Management,* June 26, 2015.

[B.1.2] OPG Self-Assessment Report, NO12-000421-SA, *Program Management Assessment – N-PROG-OP-0006, Environmental Management,* July 18, 2012.

[B.1.3] OPG Self-Assessment Report, BAS14-001300-SA, *Program Assessment of N-PROG-OP-0006, Environmental Management*, December 15, 2014.

[B.1.4] Canadian Standards Association, CSA N286-12, *Management System Requirements for Nuclear Facilities*, 2012.

B.2 OPG-PROG-0005, "Environmental Management System"

Ontario Power Generation (OPG) has developed an Environmental Management System (EMS) to implement the requirements of OPG-POL-0021, Environmental Policy [B.2.1]. The EMS and the employees that implement it are directed by OPG-POL-0033, OPG Business Model [B.2.2] and bounded by the model's Integrated Control Framework. The EMS builds on the OPG Business Model by providing specific direction on how the Environmental Policy is implemented.

OPG's Environmental Policy includes a commitment to register the EMS under the ISO 14001 Standard. Accordingly, OPG is committed to establish, implement, maintain and continually improve the EMS in accordance with the International Organization for Standardization (ISO) 14001 Standard and use the EMS as a means of describing how the ISO 14001 Standard requirements are fulfilled.

The Chemistry and Environment department completed a self-assessment in February 2014, P14-000425-SA [B.2.3], [B.2.1], in order to evaluate readiness for an EMS Nuclear Oversight audit for Pickering NGS. The scope of the self-assessment included a complete review of the effectiveness of the Pickering NGS EMS and how it conforms to planned arrangements for environmental management including requirements of ISO 14001. It was concluded that the areas of scope identified by the auditor prior to the preparation of the self-assessment were found to be properly addressed and no additional risk areas were identified. Hence, no findings/SCRs were generated.

Nuclear Oversight conducted a performance based audit for selected aspects of the EMS in May 2014, NO-2014-018 [B.2.4], for both Pickering and Darlington NGS. The audit identified nuclear-related performance improvement opportunities applicable to Pickering NGS in the areas of environmental governance implementation and OPG Environmental Aspects database effectiveness (note, 2 additional findings were generated which are associated with conventional emissions and are not applicable to this Safety Factor Report).

Two SCRs were initiated to address the findings (SCRs N-2014-15683 and N-2014-15686) which required corrective actions to be implemented. These SCRs have since been completed and the necessary corrective actions were completed to address the underlying issues.

References

[B.2.1] OPG Policy, OPG-POL-0021 R006, *Environmental Policy,* February 11, 2016.

[B.2.2] OPG Policy, OPG-POL-0033 R00, *OPG Business Model,* July 21, 2015.

[B.2.3] OPG Self-Assessment Report, *Evaluation of Pickering Environmental Management System (EMS) Nuclear Oversight March 2014 Audit Readiness*, P14-000425-SA, February 20, 2014.

[B.2.4] OPG Nuclear Oversight Report, N-REP-01070-0499510 T06 (NO-2014-018), *OPGN Environmental Management and Compliance*, May 15, 2014.

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REVISE AND RESUBMIT	
K Brammeier **Signature**	13 Apr 17 **Date**
EM **Name: Mike Ruffolo, Manager**	
Dept: Pickering Engineering - Aging Management & Strategic Initiatives	
OPG Proprietary	
Doc No.: P-REP-03680-00019 **Rev: 001**	
This acceptance does not relieve the contractor from responsibility for errors or omissions or from any obligations or liability under this contract.	

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Pickering NGS PSR2 Safety Factor 15 Report: Radiation Protection

PS112/RP/018 R04

April 11, 2017

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Revision Summary – For Amec Foster Wheeler Report PS112/RP/018

Rev	Date	Author	Comments
R00	July 29, 2016	B. McLean, R. Jayasundera, J. Cheng	Initial issue for OPG review and comment.
R01	September 23, 2016	B. McLean, R. Jayasundera, J. Cheng	Updated report addressing OPG comments on R00 Report.
R02	October 20, 2016	B. McLean, R. Jayasundera, J. Cheng	Updated Audit and Self-Assessment Reviews and Additional Reviews. This revision was issued as R000 of OPG document P-REP-03680-00019.
R03	February 22, 2017	R. Jayasundera, D. Moule	Updated Sections 2, 3, 4, and Appendix B to provide a level of detail which would be provided in an equivalent PSR1 Safety Factor Review, as Radiation Protection was not previously addressed as a stand alone review in PSR1.
R04	April 11, 2017	R. Jayasundera, D. Moule	Updated to address OPG comments on R03 report. This revision will be issued as R001 of OPG document P-REP-03680-00019.

EXECUTIVE SUMMARY

Ontario Power Generation (OPG) is performing a Periodic Safety Review (PSR) to support the possibility of continued operation of Pickering Nuclear Generating Station (NGS) beyond 2020. The PSR (referred to as "PSR2") is a subsequent PSR building on the review basis of earlier OPG Integrated Safety Reviews and other associated assessments. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

Part of PSR2 involves the preparation of Safety Factor reports for each of fifteen major topic areas. Safety Factor reports consist of:

- Assessing compliance against "Review Tasks" identified in the PSR2 Basis Document [1]. These Review Tasks are derived from Canadian Nuclear Safety Commission (CNSC) REGDOC-2.3.3, "Periodic Safety Reviews" [2] and International Atomic Energy Agency (IAEA) SSG-25, "Periodic Safety Review for Nuclear Power Plants" [3];[1]

- Documenting assessments against applicable modern Laws, Regulations, Codes and Standards (L/R/C/Ss) as defined in Reference [1]; and

- Reviewing the effectiveness of applicable OPG Nuclear Programs that support the above assessments, through review of audit and self-assessment results.

The PSR2 review of Safety Factor 15, *Radiation Protection*, is presented in this report. OPG Governance, Programs, Policies, Procedures, Instructions and Guidelines related to Safety Factor 15 were reviewed for the five PSR2 Review Tasks specified in Section 4.1 of this report. These Review Tasks were derived from CNSC REGDOC-2.3.3 and IAEA SSG-25, and address the broad aspects of Radiation Protection, including reactor design, Radiation Protection equipment and instrumentation, Radiation Protection during nuclear emergencies, improvements of Radiation Protection based on Operating Experience, and ALARA. L/R/C/S and OPG Nuclear Program effectiveness reviews for Safety Factor 15 were prepared per Sections 4.2 and 4.3, respectively. Per Section 4.4, the PSR2 assessment includes a review of previously identified PSR1 gaps related to Radiation Protection (to ascertain the implications of extending Pickering NGS operation beyond 2020), as well as a review of the R04 Pickering Licence Conditions Handbook [4] for any impacts of Pickering NGS operation beyond 2020 on: a) OPG commitments previously made to the CNSC, b) open CNSC action items, and c) exemptions granted by the CNSC (all related to Safety Factor 15).

The results of the review of Safety Factor 15 are discussed in Section 5.0 of this report. The review has confirmed that Radiation Protection has been adequately accounted for in the design

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[1] Although Radiation Protection related Review Tasks were addressed in various PSR1 Safety Factors, treatment of Radiation Protection as an independent, stand-alone Safety Factor is a new requirement per CNSC REGDOC-2.3.3 [2] (REGDOC-2.3.3 encompasses all of the PSR Safety Factors recommended by the IAEA in SSG-25 and expands upon it by adding Safety Factor 15, "Radiation Protection"). As a result, the level of detail provided for the Review Tasks in this PSR2 Safety Factor 15 report is equivalent to that which would be provided in a PSR1 Safety Factor review. In addition, RP related requirements will continue to be addressed under other Safety Factors, as required.

and operation of Pickering NGS, that Radiation Protection provisions (including design and equipment) provide adequate protection of persons from the harmful effects of radiation, and that contamination and radiation exposures and doses to persons are monitored and controlled and maintained As Low As Reasonably Achievable (ALARA). As discussed in Section 5.0, the review identified no Pickering PSR2 gaps.

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1.0 INTRODUCTION

Ontario Power Generation (OPG) is performing a Periodic Safety Review (PSR) to support the possibility of continued operation of Pickering Nuclear Generating Station (NGS) beyond 2020.[2] A comprehensive Integrated Safety Review (ISR) was completed for Pickering Units 5-8 in 2009 in support of refurbishment and continued operation. Pickering Units 1,4 integrated safety assessments were also performed for Pickering A Return to Service (PARTS) in support of approval to restart Units 1 and 4. In addition to these Pickering-specific studies, the 2013 Darlington ISR performed extensive code and standard reviews that were updated in relation to the versions that were assessed in the 2009 Pickering B ISR.[3] These previous ISRs are considered to constitute the first PSR completed for Pickering (referred to as "PSR1"). The current PSR (referred to as "PSR2") is a subsequent PSR building on the basis of earlier OPG integrated safety assessments through review of the various studies, assessments and licence renewals performed since PSR1. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

PSR2 will support and complement the licence renewal application for Pickering NGS going forward. Fifteen Safety Factors will be assessed as part of the PSR. The purpose of Safety Factor reviews is to confirm that the design, condition and operation of Pickering Units 1,4 and 5-8 (as well as common systems) will support continued safe operation for the period of PSR2 by:

- Assessing compliance against "Review Tasks" identified in the PSR2 Basis Document [1], which were derived from Canadian Nuclear Safety Commission (CNSC) REGDOC-2.3.3 [2] and International Atomic Energy Agency (IAEA) SSG-25 [3];
- Documenting assessments against applicable modern Laws, Regulations, Codes and Standards (L/R/C/Ss) (as defined in Reference [1]); and
- Reviewing the effectiveness of applicable OPG Nuclear Programs that support the above assessments, primarily through review of audit and self-assessment results.

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2 Currently, Pickering Units 5-8 are approved to operate to 247,000 Effective Full Power Hours. This operation limit is expected to be reached on some units in 2020. For the purposes of PSR2, OPG assumes operation of Pickering NGS for up to eight additional years, from 2020 until 2028. OPG will make a decision regarding the permanent shut down dates for the six reactors following the performance of a technical evaluation that will include PSR2, and will communicate it to the CNSC as required by the current Power Reactor Operating Licence (PROL).

3 Much of the compliance assessment and evaluation of Safety Factor health for the Darlington ISR is based on programs and practices that apply across OPG's nuclear operations. As a result, where Pickering is confirmed to follow the same nuclear programs and practices as were assessed for Darlington, the Darlington ISR programmatic conclusions are applicable to Pickering. An effectiveness review (effectiveness at Pickering NGS) of OPG programs used to demonstrate compliance with the PSR2 Assessment Basis is also conducted in PSR2.

Although Radiation Protection (RP) related Review Tasks were addressed in a number of PSR1 Safety Factor reviews (e.g., Plant Design, Safety Performance), treatment of RP as an independent, stand-alone Safety Factor is a new requirement per CNSC REGDOC-2.3.3 [2] (REGDOC-2.3.3 encompasses all of the PSR Safety Factors recommended by the IAEA in SSG-25 and expands upon it by adding Safety Factor 15, "Radiation Protection"). As a result, the level of detail provided for the Review Tasks in this PSR2 Safety Factor 15 report is equivalent to that which would be provided in a PSR1 Safety Factor review. In addition, RP related requirements continue to be addressed under other Safety Factors, as required.

As discussed in REGDOC-2.3.3:

- *"SSG-25 describes 14 Safety Factors that have been selected on the basis of international experience and are intended to cover all factors important to NPP [Nuclear Power Plant] safety. The scope, tasks and methodologies of these 14 Safety Factors are considered to meet the CNSC's expectations for corresponding Safety Factors 1–14 listed above. The CNSC has included an additional Safety Factor on radiation protection; the licensee should refer to Appendix A for guidance on the scope and tasks for the review of this Safety Factor."*

REGDOC-2.3.3 also requires that: "The licensee shall conduct a PSR in accordance with this regulatory document for the period until the next PSR or, if applicable, until the end of commercial operation of the plant." Consistent with REGDOC-2.3.3 [2] Appendix A, the objective of the review of Radiation Protection Safety Factor 15 is to confirm that:

- RP has been adequately accounted for in the design and operation of Pickering NGS;

- RP provisions (including design and equipment) provide adequate protection of persons from the harmful effects of radiation; and

- Contamination and radiation exposures and doses to persons are monitored and controlled, and maintained As Low As Reasonably Achievable (ALARA).

This report documents the results of the review of Safety Factor 15 for Pickering PSR2. The review covers all elements of Radiation Protection as applicable to Pickering NGS and as required under REGDOC-2.3.3. The report is based primarily on the OPG Governance, Programs, data and material available up to January 15, 2016, which is the freeze date for PSR2.

2.0 SCOPE OF REVIEW

2.1 Review Task Assessments

The Pickering PSR2 Safety Factor 15 Review Tasks are defined in Reference [1]. Details of the derivation of the Review Tasks from CNSC REGDOC-2.3.3 [2] and IAEA SSG-25 [3] are shown in Reference [5]. The Safety Factor 15 Review Tasks are:

1) Confirm the adequacy of the reactor design features for Radiation Protection.

2) Confirm the adequacy of the Radiation Protection equipment and instrumentation for radiation monitoring.

3) Confirm that adequate provisions are in place to address Radiation Protection of the public and workers during nuclear emergencies.

4) Confirm that the Radiation Protection provisions have been improved as the result of external operating experience.

5) The review will demonstrate that the ALARA principle has been incorporated in any modifications of the reactor design and operational programs and arrangements.

Appendix A (Sections A.3.1 to A.3.4) of REGDOC-2.3.3 [2] is used in the assessment of each Safety Factor 15 Review Task in order to ensure alignment with CNSC guidance relating to the RP Safety Factor. As discussed earlier, although RP related Review Tasks were addressed in various PSR1 Safety Factors, treatment of Radiation Protection as an independent, stand-alone Safety Factor is a new requirement per CNSC REGDOC-2.3.3. Compliance against RP Review Tasks is assessed by reference to applicable OPG programs and procedures, Pickering NGS specific design information, Condition Assessments, safety analyses, operating experience, and additional national and international guidance, to support the RP Review Task responses.

The methodology for the reviews is discussed in Section 3.1. Review Task findings are summarized in Section 4.1 of this Report.

2.2 L/R/C/S Reviews

The applicable Laws, Regulations, Codes and Standards relevant to the RP Safety Factor are identified in Reference [1] and are listed in Table 1 below. Table 1 also identifies the modern version and date of each L/R/C/S to be considered, the Safety Factor(s) to which each document is applicable, and the type of review that will be completed in PSR2.

All of the Safety Factor 15 L/R/C/S reviews are high level or incremental in nature. The definitions of High Level Review and Incremental Review are as follows:

- High Level Review: New L/R/C/Ss not referenced in PROL 48.02/2018 but which are in the PSR2 Assessment Basis will be subject to a high level review. In a high level review, the degree of conformance with clauses or groups of clauses in the L/R/C/S is demonstrated by supporting evidence stating whether the intent of the requirements stipulated in the requirement document is met; and

- Incremental Review: For L/R/C/Ss that have been reviewed in PSR1 but have had revisions since the last review, a topical review will be performed of the changes.

OPG programs and procedures, PSR1 and post-PSR1 assessments, and where applicable, Pickering NGS specific design information (including applicable design requirements), are used to support the L/R/C/S review responses.

The methodology for the reviews is discussed in Section 3.2. A detailed assessment for each L/R/C/S is provided in Reference [6]. Associated findings are summarized in Section 4.2 of this report.

Table 1: L/R/C/Ss Reviewed for Radiation Protection Safety Factor 15

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
1	CNSC G-129	Keeping Radiation Exposures and Doses "As Low As Reasonably Achievable (ALARA)"	2004	8, 15	Incremental	G-129 addressed as part of Pickering B and Darlington ISRs.
2	CNSC G-228	Developing and Using Action Levels	2001	8, 14, 15	Incremental	G-228 addressed as part of Pickering B and Darlington ISRs.
3	SOR/2000-202	The General Nuclear Safety and Control Regulations	Amended in June 2015	10, 15	Incremental	SOR/2000-202 addressed as part of Pickering B and Darlington ISRs.
4	SOR/2000-203	The Radiation Protection Regulations	Amended in June 2015	8, 15	Incremental	SOR/2000-203 addressed as part of Pickering B and Darlington ISRs.
5	CNSC REGDOC-2.2.3	Personnel Certification: Radiation Safety Officers	2014	15	High Level[4]	REGDOC-2.2.3 not addressed as part of Pickering B or Darlington ISRs.

4 There is no Class II nuclear facility or prescribed equipment within the bounds encompassed by the Pickering NGS PROL and there is no requirement to have a Radiation Safety Officer. Hence, a high level review of REGDOC-2.2.3 is not required because REGDOC-2.2.3 is not applicable to Pickering NGS.

2.3 OPG Program Effectiveness Reviews

The OPG Nuclear Program (N-PROG) reviewed for Safety Factor 15 is listed in Table 2 below.[5] The methodology for the effectiveness reviews is discussed in Section 3.3. The assessment results of the N-PROG in Table 2 are provided in Appendix B, and findings are summarized in Section 4.3.

Table 2: OPG Program Reviewed for Safety Factor 15

Document Number	Document Title
N-PROG-RA-0013 [7]	Radiation Protection

In addition to feedback on program effectiveness from audits conducted by Nuclear Oversight, and self-assessments completed by OPG work groups, OPG acquires additional feedback from Station Condition Records, Pre and Post-Job Briefing feedback, RP dose records, and reviews performed through the industry.

2.4 Additional Reviews

The PSR2 Safety Factor 15 Report includes a review of the R04 Pickering Licence Conditions Handbook (LCH) [4] for any impacts of Pickering NGS operation beyond 2020 on the following (all related to Safety Factor 15):

- OPG commitments previously made to the CNSC;
- Open CNSC action items; and
- Exemptions granted by the CNSC.

The PSR2 assessment includes identification and review of previously identified PSR1 gaps related to Radiation Protection to ascertain the implications of extending Pickering NGS operation beyond 2020. The methodology for these reviews is described in Section 3.4. Any PSR2 gaps identified as a result of programmatic Darlington PSR1 gaps related to Radiation Protection are discussed in Section 4.4 of this report. The review of Pickering PSR1 gaps previously identified in the Pickering Units 5-8 Continued Operations Plan [8] is provided in Reference [9].

In addition, Fukushima Action Items were reviewed to identify implications of extending operation beyond 2020 (if any). This review is presented in Reference [10].

Any PSR2 gaps identified as a result of the Safety Factor 15 review that are relevant to other Safety Factors are discussed in Section 4.4 of this report.

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[5] The list of Nuclear Programs to be assessed for effectiveness for PSR2 was derived from review of current OPG Governance. Although there may be content in Nuclear Programs that is applicable to multiple Safety Factors, N-PROG reviews are only provided in one Safety Factor report and are not duplicated.

3.0 METHODOLOGY

The sub-sections below summarize the methodology used to assess Review Tasks, L/R/C/Ss, and Nuclear Program effectiveness for the Radiation Protection Safety Factor.

3.1 Review Tasks

The Safety Factor Review Tasks are derived from CNSC REGDOC-2.3.3 [2] and IAEA SSG-25 [3], taking into consideration the Review Tasks used in the Pickering B and Darlington ISRs (as derived in [5]). Appendix A of REGDOC-2.3.3 [2] is used in the assessment of each Safety Factor 15 Review Task in order to ensure alignment with CNSC guidance relating to the RP Safety Factor. As discussed earlier, although RP related Review Tasks were addressed in a number of Safety Factors in PSR1, treatment of Radiation Protection as an independent, stand-alone Safety Factor is a new requirement per REGDOC-2.3.3. As a result, the SF15 RP Review Tasks are new for PSR2 and Review Task assessments are written with a level of detail commensurate with a PSR1 review.

For each Safety Factor 15 Review Task identified in Section 2.1, applicable OPG Governance, Programs, Policies and Procedures (as well as Instructions and Guidelines, as applicable) were assessed for compliance. Compliance against Review Tasks is also assessed by reference to applicable Pickering NGS specific design information, Condition Assessments, safety analyses, operating experience, and additional national and international guidance to support the RP Review Tasks responses.

The Review Task assessments identify Compliances and Gaps as defined below:

- Compliance: Compliance indicates that either the safety requirement or the intent of the Review Task is met.

- Gap: A Gap indicates that the intent of the Review Task is not met.

3.2 L/R/C/S Reviews

The process to identify the modern L/R/C/Ss that are applicable to the PSR2 Assessment Basis involved first creating a broad list from multiple sources (potential candidate L/R/C/Ss) and then filtering it to identify those that are most significant and that are applicable to the PSR2 scope. The identification and selection criteria are detailed in Reference [1]. The result of the identification and selection process was a set of modern L/R/C/Ss that became part of the "PSR2 Assessment Basis".

PSR2 is focused on the extension of Pickering NGS operations beyond 2020, and will conduct reviews against a baseline of past PSR1 work. As a subsequent PSR, PSR2 focuses on changes in requirements, plant conditions, operating experience and new information. Since PSR2 is an update of previous ISRs, it incorporates reviews of L/R/C/Ss that have occurred as new versions have been issued. Since this assessment is a subsequent PSR, the focus is on identifying differences between what was previously assessed and what is now different within the current Pickering PSR2 Assessment Basis. In general, these differences relate to:

- More recent (new or revised) L/R/C/S versions than what was previously assessed as part of PSR1;[6]
- Safety significant differences between Pickering and Darlington, if the Darlington ISR is the basis for the earlier assessment;
- Implications of extending Pickering NGS operation beyond 2020; and
- Safety significant differences between Pickering Units 1,4 and Units 5-8.

As described in Reference [1], L/R/C/S review types are clause-by-clause, high level or incremental. Most of the L/R/C/Ss in the PSR2 Assessment Basis receive incremental reviews since PSR2 is an update of previous PSR1 assessments and clause-by-clause or high level reviews for the majority of the L/R/C/Ss in the PSR2 Assessment Basis have already been completed. Implementation plans (including gap analyses or code-over-code reviews) also exist for the latest editions of many L/R/C/Ss. As a result, incremental review is also used in circumstances where a L/R/C/S in the PSR2 Assessment Basis was not assessed in previous PSR1 reviews but an implementation plan currently exists for compliance.

The PSR2 incremental reviews in this Report include an assessment of the intent of recent changes to the L/R/C/Ss on a topic or subject-matter basis where there is potential to impact nuclear safety. Incremental reviews provide:

- A summary of the purpose of the L/R/C/S;
- Pertinent background information about the current revision of the L/R/C/S that is being considered;
- Identification of which Safety Factor(s) are applicable to the current revision of the L/R/C/S;
- A description of which version(s) of the L/R/C/S were assessed for PSR1 (i.e., Darlington ISR (for programmatic content), Pickering B ISR and PARTS code reviews);
- Identification of whether the current version of the L/R/C/S is an update of a previous version of the L/R/C/S that was assessed in PSR1 (and if so, a

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[6] "New" refers to a code or standard that was not previously considered in the context of earlier assessments. "Revised" refers to an updated version of a code or standard that was previously considered in the context of earlier assessments. Where a document has a new number/type, but addresses the same topic from the same organization, it is a "revised", not "new", document (e.g., if a REGDOC replaces a CNSC G or RD document).

description of the major changes in the latest revision is provided as discussed below);

- An assessment of the applicability of PSR1 assessment findings (gaps and conclusions), including the implications of extending Pickering NGS operation beyond 2020 if any;

- An assessment of the applicability of assessment findings that address more recent (post-PSR1) editions of the L/R/C/S, including any implementation or transition plans that are already committed to by OPG; and

- Where PSR1 and post-PSR1 assessments are not sufficient to address changes in the latest edition of the L/R/C/S, an assessment of the changes from the previously assessed edition of the L/R/C/S (including identification of any safety significant PSR2 gaps which result).

High Level reviews provide the same information as above, where applicable, in a similar format. However, given that High Level L/R/C/Ss generally have not received past assessment during PSR1, the Incremental review content is augmented by a high level, section-by-section assessment of the degree of conformance of Pickering NGS with the L/R/C/S (demonstrating, with supporting evidence, whether the intent of the requirements stipulated in the document are met). There are currently no L/R/C/S clause-by-clause reviews identified in the PSR2 Assessment Basis.

L/R/C/S reviews make use of the following information, where applicable: a) OPG Governance, Programs, Policies and Procedures, b) Pickering NGS specific design information (including applicable design requirements), c) Commitments previously made to the CNSC, open CNSC action items, and exemptions granted by the CNSC related to the Safety Factor under review, as identified in the R04 Pickering LCH [4], d) Identification of previously identified Pickering-specific or applicable Darlington PSR1 gaps related to the Safety Factor under review and the status of OPG's improvement plan(s) or other dispositions to address these, and e) Assessments and reviews performed since the PSR1 documents were completed.

The Safety Factor 15 L/R/C/S reviews identify Compliances and Gaps as defined below:

- <u>Compliance</u>:

    - Where an Incremental review has been performed, Compliance indicates that the change in the safety requirement, per the topical review, is met.

    - Where a High Level review has been performed, Compliance indicates that the intent of the safety requirement is met.

    - For Clause-by-Clause reviews of modern L/R/C/Ss, Compliance indicates that the safety requirement is met. (Note: No Clause-by-Clause reviews were performed as part of Safety Factor 15.)

- Gap:

    - Where an Incremental review has been performed, a Gap indicates that the change in the safety requirement, per the topical review, is not met.

    - Where a High Level review has been performed, a Gap indicates that the intent of the safety requirement is not met.

    - For Clause-by-Clause reviews of modern L/R/C/Ss, a Gap indicates that the safety requirement is not met. (Note: No Clause-by-Clause reviews were performed as part of Safety Factor 15.)

The reviews assume that use of the word:

- "Shall" is used in an L/R/C/S to express a requirement, i.e., a provision that the licensee is obliged to satisfy in order to comply with the standard;

- "Should" is used to express a recommendation or that which is advised but not required;

- "May" is used to express an option or that which is permissible within the limits of the standard; and

- "Can" is used to express possibility or capability.

3.3 OPG Program Effectiveness Reviews

As discussed earlier, effectiveness reviews (effectiveness at Pickering NGS) of OPG programs used to demonstrate compliance with the PSR2 Assessment Basis were conducted, primarily using recent applicable audit and self-assessment results:

- OPG Nuclear Oversight independent performance-based Program audits (typically performed in 1 to 5 year cycles) and self-assessments. This includes review of associated Station Condition Records and Action Requests to determine the status of any resulting corrective actions; and

- CNSC "Type I" and "Type II" inspections of the effectiveness and performance of OPG programs, where discussed in OPG audits or self-assessments.

There are many audits and self-assessments that are performed to assess the effectiveness of important aspects of each program. A sample of audits and self-assessments has been summarized for each program in order to demonstrate that program effectiveness is being assessed on an ongoing basis. The focus of these reviews was on effectiveness of the programs at Pickering NGS, where specific information is available. Results from these audits and self-assessments will be considered in the Global Assessment process. It is noted that audits and self-assessments are, by their nature, self-critical and are used to drive excellence in performance. As a result, the broad review scope of program audits focuses on

identifying improvement opportunities rather than presenting a balanced picture of program performance.

Program effectiveness is also monitored and addressed through the Fleetview Program Health and Performance Reporting process [11]. This process involves direct oversight by the Chief Nuclear Officer, and includes participation by the Nuclear Executive Committee members. Programs are reviewed, senior oversight is provided, and improvement plans are generated.

In addition to regular feedback on program effectiveness from audits conducted by OPG Nuclear Oversight, and self-assessments completed by OPG work groups, OPG acquires additional feedback from Station Condition Records, Pre and Post-Job Briefing feedback, RP dose records, and reviews performed through industry.

The list of Nuclear Programs to be assessed for each Safety Factor was derived from a review of current OPG Governance.

3.4 Additional Reviews

A review of the R04 Pickering LCH [4] was performed to determine if there are any impacts associated with operation of the Pickering Units past 2020 on the following (all related to Safety Factor 15):

- Commitments previously made to the CNSC;
- Open CNSC action items; and
- Exemptions granted by the CNSC.

The PSR2 assessment includes identification and review of previously identified Pickering-specific or programmatic PSR1 gaps related to Radiation Protection (as identified in the Darlington ISR Integrated Implementation Plan [12] and Pickering Units 5-8 Continued Operations Plan [8]) to ascertain the status of OPG's improvement plan(s) or other dispositions to address these and the implications of extending Pickering NGS operation beyond 2020 (if any).[7] The methodology and review results

[7] PSR2 includes consideration and confirmation that the findings of PSR1 remain valid, as applicable, for the operation period. This includes assessment of PSR1 conclusions against implications resulting from extended operation. In particular, Pickering PSR1 results are applicable to PSR2 if there was a PSR1 gap that is still open, or if a closed PSR1 gap could be affected by extended operation. If so these gaps are carried forward into PSR2 for consideration in the Global Assessment. (When references to PSR1 are made, the source document is identified and the relevant text from that source document is summarized in the context of PSR2.) With respect to the Darlington ISR, much of the evaluation of Safety Factor health is based on programs and practices that apply across OPG's nuclear operations. As a result, Darlington ISR programmatic conclusions are applicable to Pickering PSR2 where Pickering is confirmed to follow the same nuclear programs and practices that were assessed for Darlington. Darlington PSR1 results are applicable to Pickering PSR2 if there are Darlington PSR1 gaps that are found to be relevant to Pickering PSR2.

associated with the Pickering PSR1 gaps previously identified in the Pickering Units 5-8 Continued Operations Plan [8] are provided in Reference [9].

Fukushima Action Items were reviewed to identify implications of extending operation beyond 2020 (if any). The methodology for this review is provided in Reference [10].

Any PSR2 gaps identified as a result of the Safety Factor 15 review that are relevant to other Safety Factors are also discussed.

4.0 REVIEW FINDINGS

4.1 Review Tasks

The sub-sections below provide an assessment of the adequacy of applicable OPG Governance, Programs, Policies and Procedures (as well as Instructions and Guidelines, as applicable) in demonstrating compliance against the Safety Factor 15 Review Tasks. Appendix A of REGDOC-2.3.3 [2] is used in the assessment of each Safety Factor 15 Review Task in order to ensure alignment with CNSC guidance relating to the RP Safety Factor.

4.1.1 Review Task #1: Reactor Design for Radiation Protection

Confirm the adequacy of the reactor design features for Radiation Protection.

REGDOC-2.3.3 [2], Appendix A.3.1, *Review of the reactor design features for radiation protection*, elaborates on this Review Task by stating the following:

"The review should identify all sources of radiation and radiation exposure pathways, with an evaluation of radiation doses that could be received by workers at the facility with consideration of contained and fixed sources, and potential sources of airborne radioactive material. The review should demonstrate that the ALARA principle has been incorporated in the reactor design and operational programs and arrangements, in order to minimize the number and locations of radiation sources and the radiation fields associated with them.

The review should determine that the design and layout of the reactor facility meets CNSC regulatory requirements and expectations for reactor facilities in the area of RP (e.g., REGDOC-2.5.2, Design of Reactor Facilities: Nuclear Power Plants and RD/GD-369, Licence Application Guide: Licence to Construct a Nuclear Power Plant). The review should include RP principles, and how they are incorporated into the reactor design and are of a sufficient depth to demonstrate the following:

- *Suitable provisions have been made in the design and layout of the reactor facility to keep occupational radiation dose below regulatory limits and ALARA, including:*
  - *Classification of areas (zoning) and access control*
  - *Aging of all materials and obsolescence of technology that could impair the radiological safety functions of SSCs [Structures, Systems, and Components]*
  - *Radiological hazard control*
  - *Decontamination of personnel, equipment and structures*
  - *Radiological monitoring (in-plant)*
- *SSCs have been adequately designed so that radiation exposures during all activities are optimized and justified.*

The following subsections address the requirements of this Review Task, as well as Appendix A.3.1 of REGDOC-2.3.3 [2], as noted above. Note that ALARA principles (including an evaluation of the radiation doses that could be received by workers at PNGS) are discussed in this Review Task at a high level and more detailed information is provided in Section 4.1.5 of this report.

RP Program Background

N-PROG-RA-0013, "Radiation Protection" [7], implements a series of standards and procedures for the conduct of activities within Pickering NGS, in order to achieve the following objectives:

1. Controlling occupational and public radiation exposure:

    - Keeping individual doses below regulatory limits;
    - Avoiding unplanned exposures;
    - Keeping individual risk from lifetime radiation exposure to an acceptable level; and
    - Keeping collective doses As Low As Reasonably Achievable (ALARA), social and economic factors taken into account.

2. Preventing the uncontrolled release of contamination or radioactive materials from the nuclear sites through the movement of people and materials.

3. Demonstrating the achievement of 1) and 2) through monitoring.

Figure 1 provides an illustration of the OPGN RP program governance structure [7].

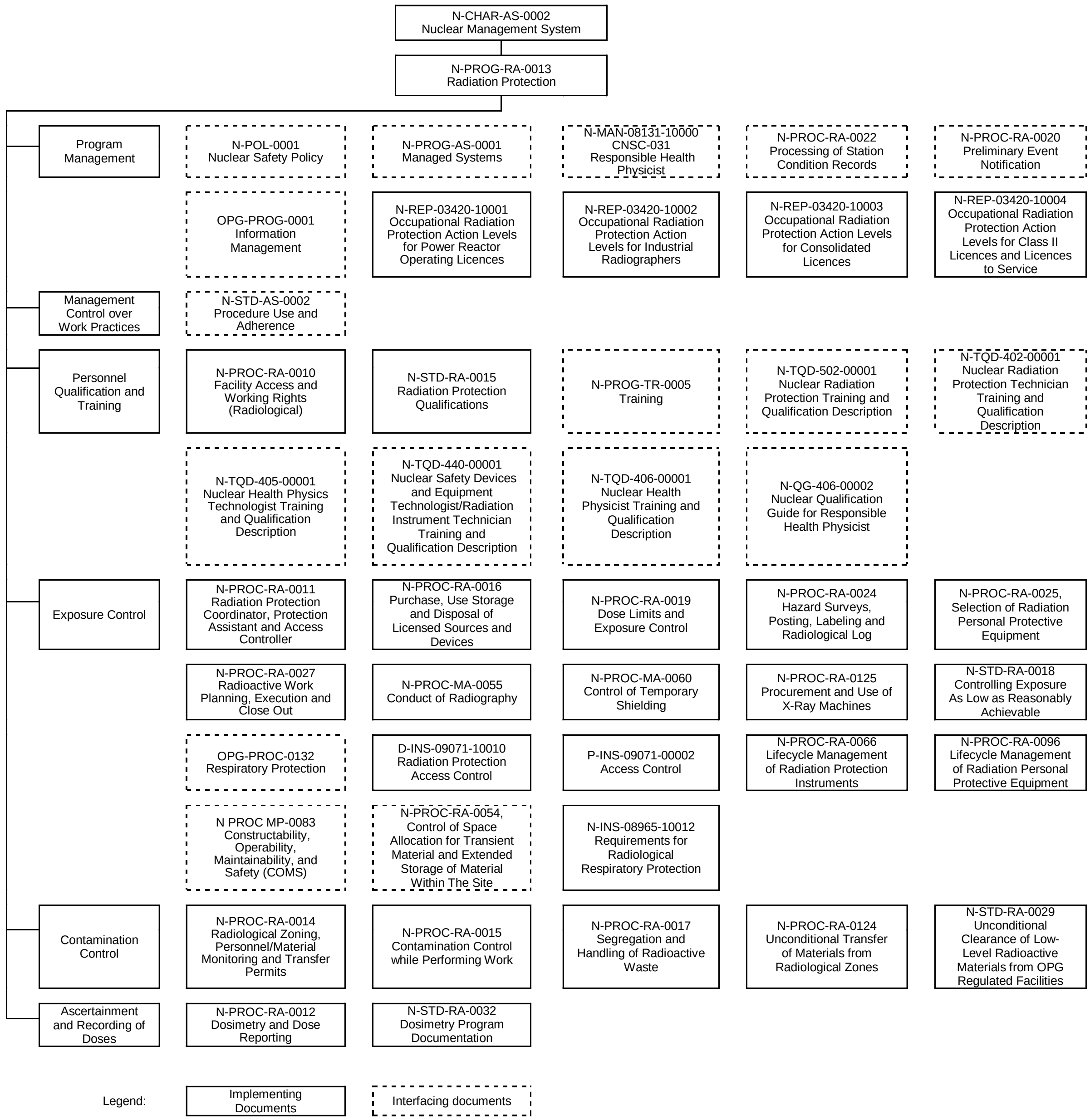


Figure 1 - Outline of Radiation Protection Governance

Per Section 1.5.3 of N-PROG-RA-0013 [7], the initial design of the station was created such that the layout and operation of facility structures, systems, components (SSCs) and processes were consistent with the established RP guidelines and contribute to maintaining occupational radiation exposures ALARA. Per Section 12 of the Pickering 1,4 and 5-8 Safety Reports [13], [14], during the design and early operation of Pickering NGS, a design improvement program for the purpose of occupational radiation exposure reduction during operation and maintenance was undertaken, based on operating experience (OPEX) at Nuclear Power Demonstration (NPD) and Douglas Point. The work program emphasized the importance of system chemistry, equipment maintainability and shielding. In particular, the purification system designs for the Heat Transport System (HTS), the End Shield Cooling System and the Irradiated Fuel Bay were upgraded. Also, HTS components had low cobalt steel alloys specified to reduce the production of Cobalt-60 by activation. Bearing components made of Stellite (which was found to be a significant source of the dissolved cobalt in the HTS system), were removed and replaced with components made of low cobalt alloys.

The current Pickering NGS design includes specific features to ensure radiation exposures during all activities are optimized and justified. This is achieved through the use of radiological zones, area radiation monitoring equipment and the use of shielding to control radiation exposures. The design also includes SSCs which limit the release of radioactivity to the environment, and therefore limit public dose. For example, in areas subject to D_2O leakage, the Reactor Buildings at Pickering NGS have Vapour Recovery Systems [15] to maintain dry atmospheres, which results in the removal of tritium from the Reactor Building atmospheres. The Filtered Air Discharge System (FADS) [16] is a dedicated safety system that can be used in accident situations to keep Containment sub-atmospheric following the depletion of the vacuum reserve in the Vacuum Building, preventing the unfiltered release of radionuclides into the environment that could occur if Containment pressure were above atmospheric pressure. In addition, when needed, the Negative Pressure Containment System at Pickering NGS automatically isolates the Containment volume to limit the release of radioactivity beyond the Containment boundary (known as Box-up) [17].

RP Design for Defence in Depth

As outlined in Part 2 of the Pickering 1,4 and 5-8 Safety Reports [13], [14], there are a number of barriers between radioactive materials and the general public. The barriers in place to prevent radioactivity from escaping to the environment include:

- The UO_2 fuel pellets, which bind the majority of radioactive fission products within a solid matrix;
- The fuel sheath, which contains the fission products not retained in the fuel matrix;
- The Heat Transport System boundary, which contains any leakage from the fuel sheath;
- The Containment structure, which contains any release from the Heat Transport System; and
- The exclusion zone surrounding the facility, which provides for dilution of any release from Containment.

The first three barriers prevent radioactive release accidents. So long as they are intact, very little radioactive material will escape into Containment. Containment and the exclusion zone come into play to mitigate doses when all of the first three barriers are breached (e.g., following a loss of coolant accident with fuel failures).

RP is an important element in all aspects of the design for Defence in Depth. Five levels of Defence in Depth are defined in the IAEA document INSAG-10, "Defence in Depth in Nuclear Safety" [18] (as well as REGDOC-2.5.2 [19]). The 5 levels of Defence in Depth at Pickering NGS can be summarized as follows:

- Level 1 includes the design provisions for protecting staff and the public from radiation during normal operation.
- Level 2 and 3 includes the provisions to support response to anticipated operating occurrences and Design Basis accidents to limit the effect of these events on workers, the public, and the environment.
- Level 4 includes the Beyond Design Basis aspects of the response to mitigate the radiological impact on workers, the public and the environment.
- Level 5 includes the mitigating off-site response including the provisions for Radiation Protection aspects of the off-site Emergency Response.

Note that detailed information regarding Pickering NGS alignment against these five levels is provided in Review Task 4 of Pickering NGS PSR2 Safety Factor 1 Report, "Plant Design" [20].

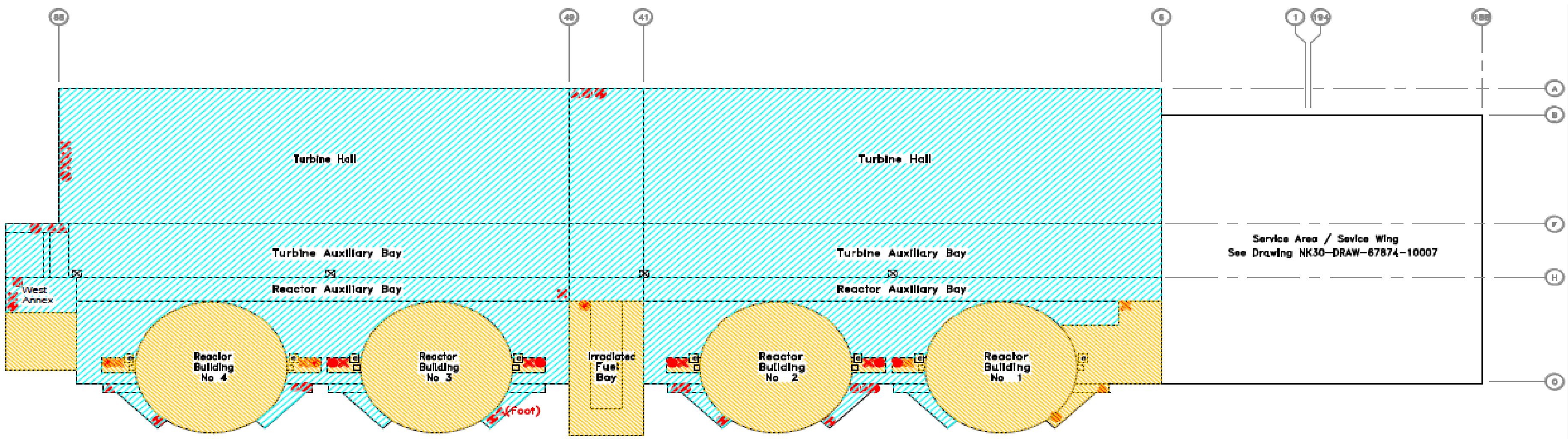
Radiological Zoning and Access Control

The protected area (inside the inner security fence) of Pickering NGS is divided into radiological zones in order to facilitate contamination control. Per Sections 12.3.4 of the Pickering 1,4 and 5-8 Safety Reports [13], [14], a description of the radiological zones is as follows:

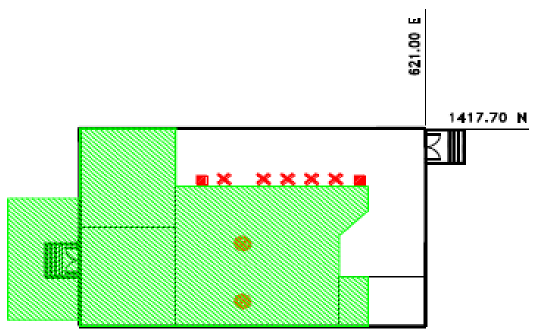
- Zone 1 – This zone contains no radioactive equipment and is normally free of contamination. It includes the administration building, the bridge between the administration and service buildings, the main lobby at the end of the bridge, the lunchroom, and the service wing S-701 office area.
- Zone 2 – This zone contains a minimal amount of radioactive equipment and normally should not have fixed or loose contamination present. This zone includes the Turbine Hall, Turbine Auxiliary Bay, parts of the Reactor Auxiliary Bay, Control Room, non-active shops, stores, showers, locker room, laundry facilities in the service wing, and the service wing extension offices.
- Zone 3 – This zone contains primarily equipment that is associated with radioactive systems. It includes the Reactor Buildings, sections of the Reactor Auxiliary Bay, decontamination centre, active overhaul and fuelling machine maintenance areas, heavy water upgrading building, Irradiated Fuel Bay, and parts of the waste management area in the service wing basement and the east annex and parts of the west annex.

- Unzoned Areas - These are areas within the protected area, but outside the powerhouse, Vacuum Building, or other buildings that are not Zone 1. It includes the roofs of buildings (below the 294’ elevation). Like Zone 1, it normally contains no radioactive sources. It differs from Zone 1 in that radioactive materials may be moved through the area provided that they are adequately shielded and contained. Eating and smoking are prohibited in an unzoned area except in designated areas.

Per Figures 151, 152, 153 and Figure 12-3, 12-4 and 12-5 of the Pickering 1,4 and 5-8 Safety Reports respectively [13], [14], the radiological zoning arrangement diagrams are shown in Figure 2 to Figure 7:



GROUND FLOOR
254'–0"



WEST ACCESS GUARDHOUSE
1"=10'–0"

LEGEND

FIELD CHANGE ROOM
ELEVATOR
STAIRS
HAND AND FOOT MONITOR
SMALL ARTICLE MONITOR
WHOLE BODY MONITOR
PORTABLE HAND–HELD MONITOR
PORTAL GAMMA MONITOR
ZONE 1
ZONE 2
ZONE 3

Figure 2 - Pickering 1-4 254' Elevation Zoning Arrangement

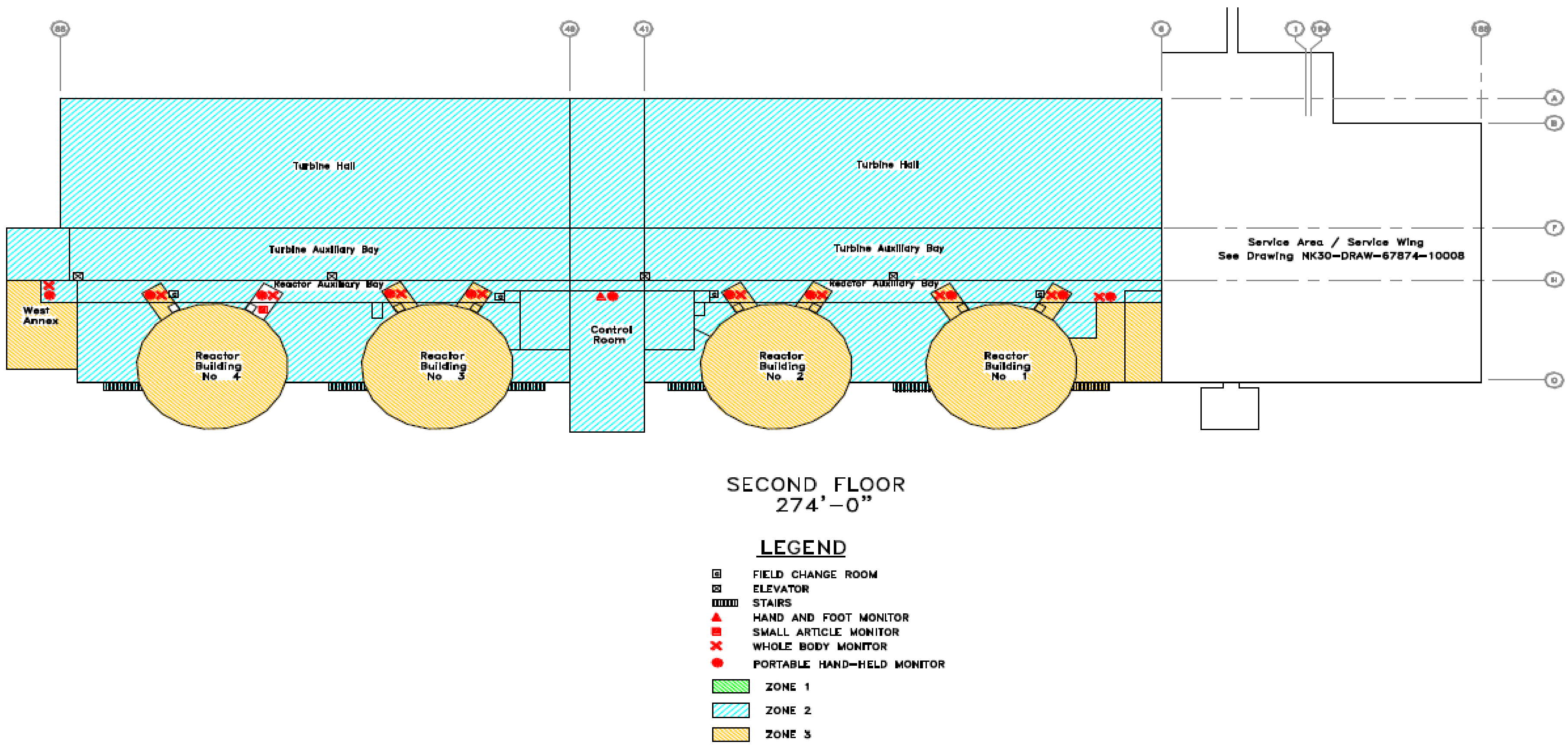


Figure 3 - Pickering 1-4 274′ Elevation Zoning Arrangement

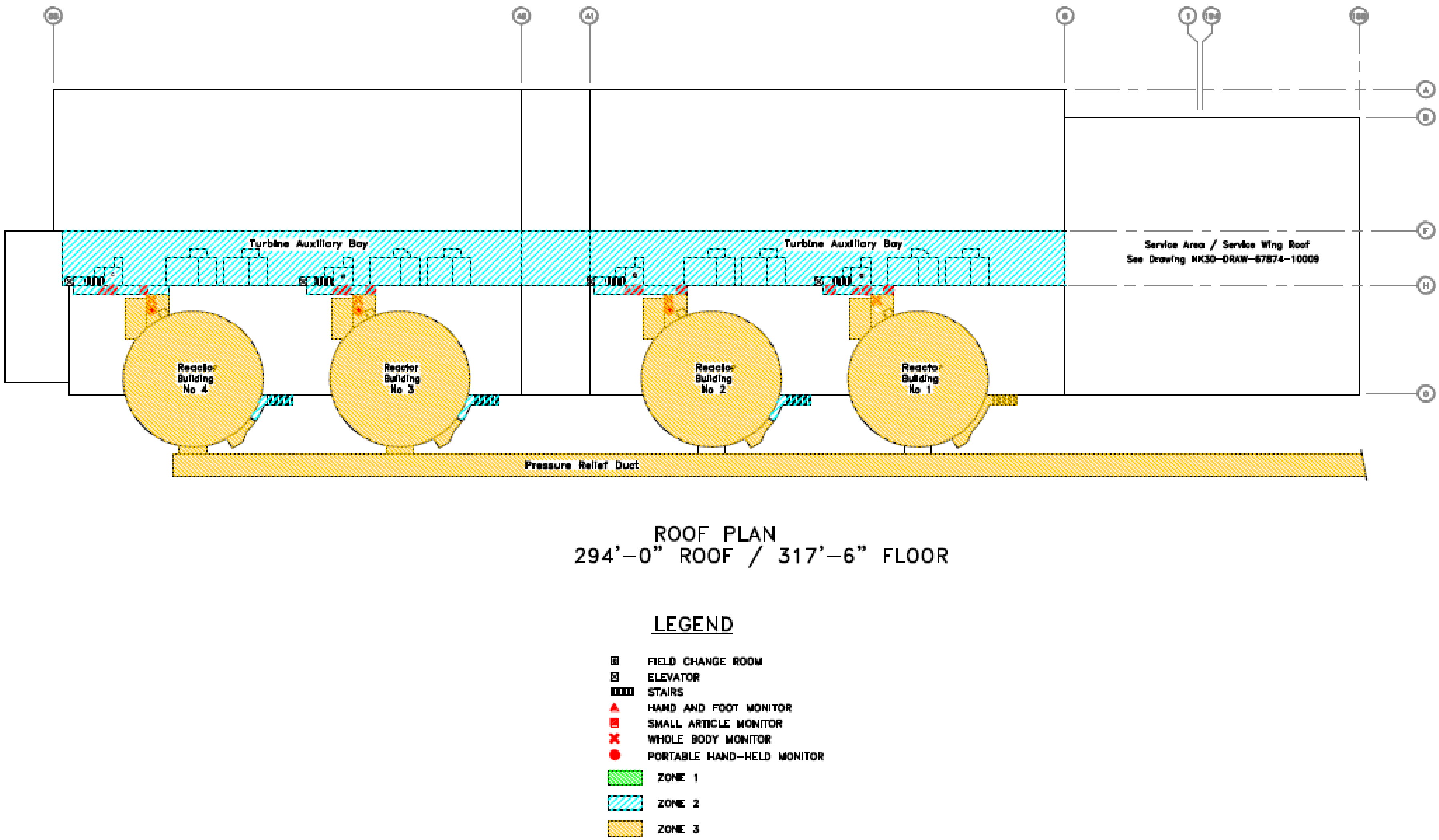


Figure 4 - Pickering 1-4 294′/317′ Elevation Zoning Arrangement

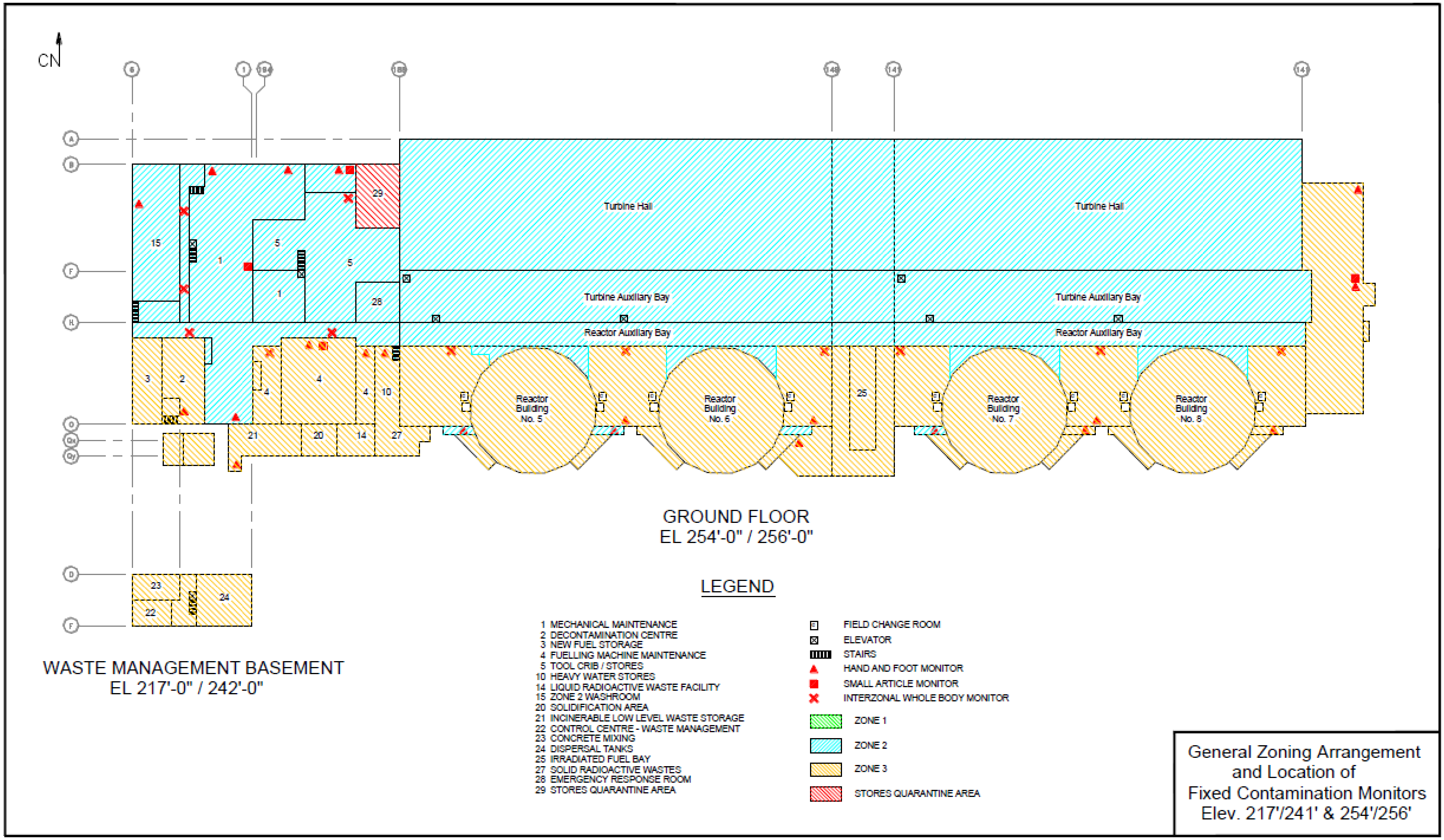


Figure 5 - Pickering 5-8 217'/241' & 254'/256' Elevation Zoning Arrangement

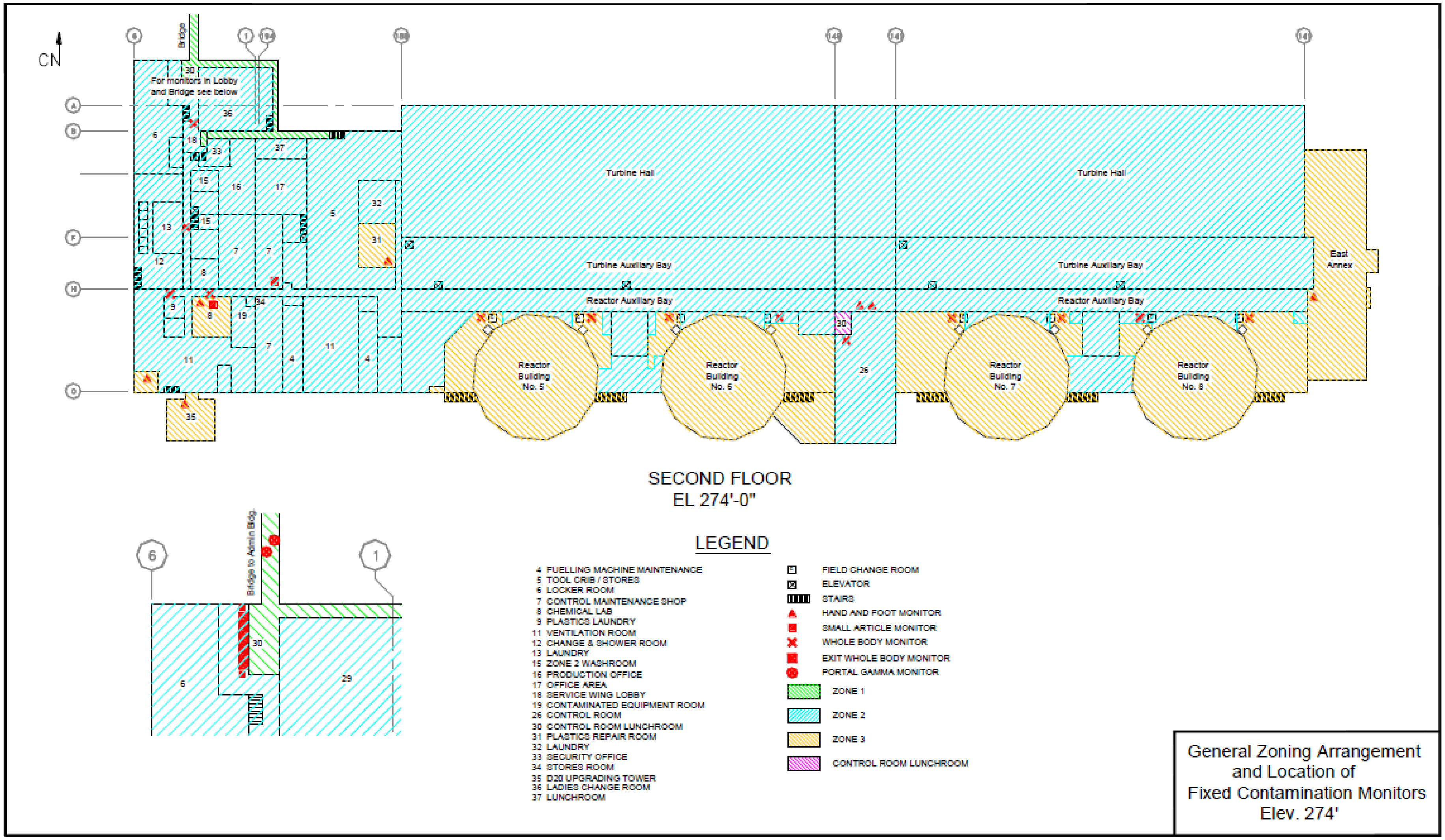


Figure 6 - Pickering 5-8 274' Elevation Zoning Arrangement

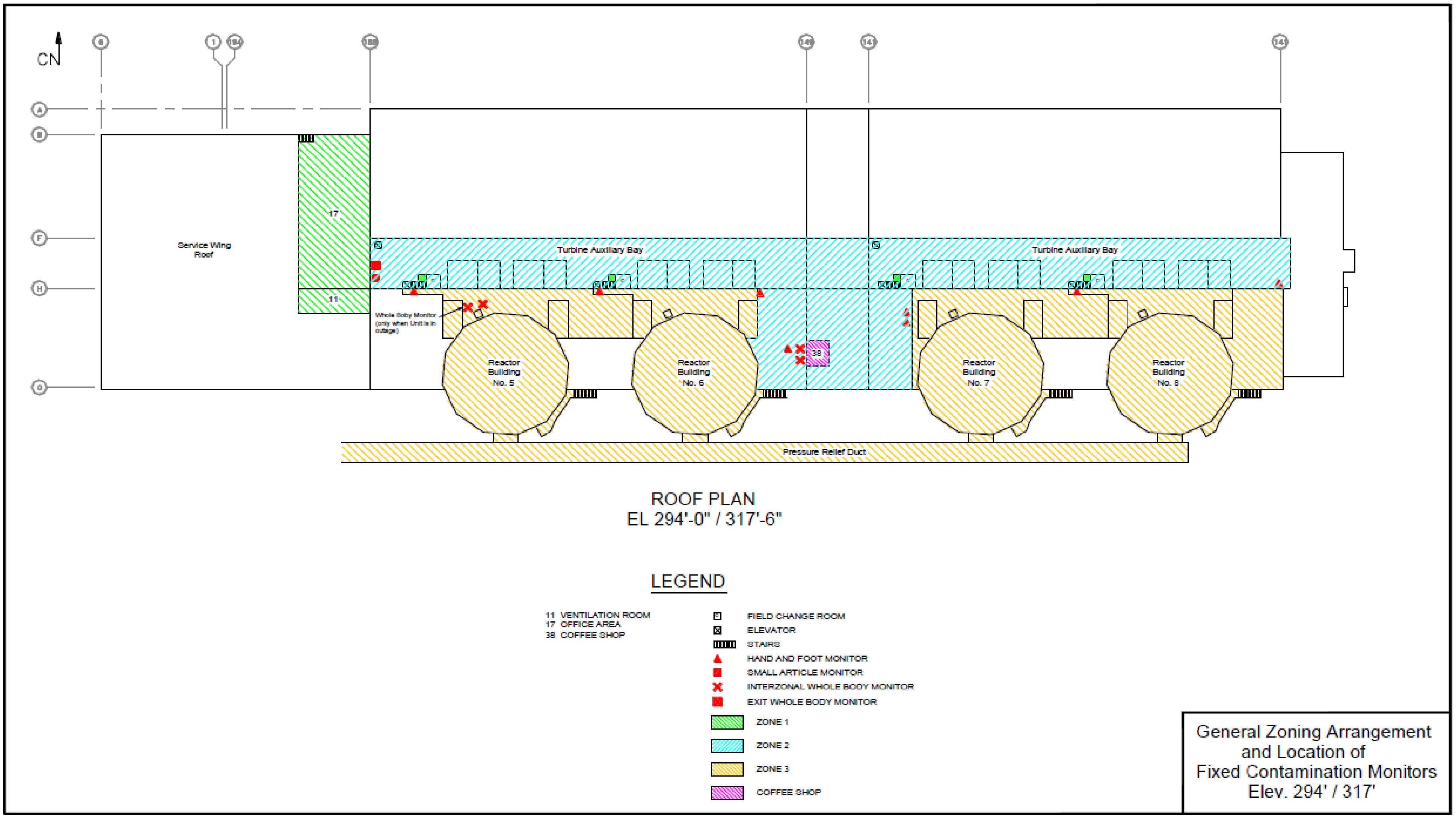


Figure 7 - Pickering 5-8 294′/317′ Elevation Zoning Arrangement

Per Section 12.3.3 of the Pickering 1,4 and 5-8 Safety Reports [13], [14], the Pickering NGS buildings are laid out to assist the management of all sources of radioactivity for the protection of station staff and the public. Operational procedures restrict access to the Reactor Building to qualified staff and to those under escort by qualified staff. Also, permanent signs and procedures warn and instruct staff of any possible danger from radiation. The sources of radiation which give rise to high radiation fields include the reactor (e.g. the fuel and other fixed sources created by activation), the HTS, the moderator system, and the fuelling machines. The source intensity of each is dependent on operating conditions. The Access Control System prevents unauthorized access by staff to areas that contain these high radiation sources. There are two states of reactor operation which require different degrees of access control:

- State 1: Reactor operation below 0.1% full power, which is a power level at which the radiation fields are acceptable for controlled access. During this type of operation, access control doors to areas directly influenced by reactor power may be kept open. Entry to the reactor, HTS, and moderator system areas is by normal procedural controls when the radiation levels are those associated with reactor operation below 0.1% full power.

- State 2: Reactor operation above 0.1% full power is a condition that normally requires all doors that give access to the controlled areas to be locked. Entry to those areas during this type of operation is controlled by specific access control procedures.

The doors to these Access Control Areas are opened with keys that are kept in transfer locks in the Main Control Room. These transfer locks are a part of the key accounting system. A red light on the access control panel is lit whenever a key is removed from its transfer lock in either state of reactor operation. When reactor power is greater than 0.1% and a key is removed from the key accounting system, an alarm advises that the key to an access controlled room has been removed from its transfer lock. Procedures for accessing areas impacted by station operations (including fuelling activities) for Pickering NGS are implemented via P-INS-09071-00002, "Access Control" [21], which defines the Radiation Protection requirements that all individuals must follow when entering Access Control Areas.

<u>Aging of Materials and Obsolescence of Technology that could impair Radiological Safety Functions of SSCs</u>

Maintaining RP design provisions throughout the operating life of Pickering NGS relies on various operations and maintenance programs that proactively account for equipment aging and obsolescence to ensure continued operation within the station's Design Basis and maintenance of doses ALARA.

Aging of SSCs can have RP implications in terms of reduced effectiveness of safety-related systems (e.g., containment), buildup of contamination within systems and components leading to increased dose rates, radiation or corrosive damage to containment systems leading to spills and/or airborne and surface contamination, and degradation of shielding.

Equipment aging is addressed through the Integrated Aging Management (IAM) Program, as specified in N-PROG-MP-0008, "Integrated Aging Management" [22]. The objective of the IAM program is to ensure the condition of critical Nuclear Power Plant (NPP) equipment is understood and that required activities are in place to assure the health of these components and systems while the plant ages. This is accomplished by establishing an integrated set of programs and activities that ensure the performance requirements of all critical station equipment are met on an ongoing basis.

The IAM Program also requires the preparation of Life Cycle Management Plans for Major Components (N-PROG-MA-0025 [23]), and scoping, screening and preparation of required Condition Assessments, to support the monitoring and mitigation of aging related degradation mechanisms. Action plans from the Condition Assessments are managed through the Work Management process and reported as appropriate in System and Component Health Reports. The IAM Program is supported by the Equipment Reliability Program (N-PROG-MA-0026 [24]), which includes a range of procedures and processes to ensure ongoing high levels of reliable performance of components important to nuclear safety, production, and environmental protection at the station.

Surveillance is carried out in accordance with the Component and Equipment Surveillance program (N-PROG-MA-0017 [25]), which describes the elements of a focused surveillance monitoring process, including inspection, maintenance, certification and testing, and is supported by the System Performance Monitoring procedure (N-PROC-MA-0024 [26]). The surveillance monitoring process covers the full range of RP design provisions commensurate with the importance of the equipment to nuclear safety (e.g. Periodic Inspection Programs for the concrete containment as prescribed under applicable CSA standards (CSA N287.7, "In-service examination and testing requirements for containment structures for CANDU nuclear power plants"), and calibration and functionality checks for radiation monitors). Hence, station SSCs supporting RP are routinely monitored, tested and maintained to ensure that any aging effects are not preventing the required level of performance from being sustained.

The IAM program also ensures that changes impacting the condition of plant systems due to aging are monitored and managed such that doses are maintained ALARA over the life of the station. Examples of this include the following:

- The Calandria Vault, Vault Structure Cooling and Shield Tank concrete walls and slabs are inspected for potential deterioration that could lead to reduction in shielding and hence increased dose rates or loss of containment. Any damage discovered as part of the surveillance monitoring process is repaired immediately to ensure shielding and containment integrity.

- Isolation valves are used to close off parts of systems to allow worker access for maintenance activities. These can degrade over time through corrosion, erosion or mechanical fatigue, leading to leakage of radioactive material. This can result in increased doses to workers from external exposure to gamma emitting radionuclides, skin or clothing contamination, and/or inhalation of airborne contamination from evaporation. These valves are tested regularly as part of the safety related system test program, and replaced if not functioning to design requirements.

The overall station SSC surveillance, maintenance and testing programs are discussed in detail in Pickering NGS PSR2 Safety Factor 2 Report, "Actual Condition of Structures, Systems, and Components Important to Safety" [27]. A discussion on the lifecycle management of Radiation Personal Protective Equipment and Radiation Protection instruments at Pickering NGS is provided in Section 4.1.2 of this Safety Factor 15 report.

In order to proactively identify obsolescence issues for SSCs supporting RP before they are encountered through response to equipment failure or other emergent circumstances, N-STD-MA-0024, "Obsolescence Management" [28], defines and implements a sustaining program to manage both proactive and reactive obsolescence issues associated with critical equipment and components. The activities interface with equipment reliability and life-cycle management strategies designed to sustain continued safe and reliable plant operation. The Obsolescence Management program provides direction on managing obsolescence issues pertaining to RP and other critical equipment and components, as defined in N-PROC-MA-0077, "Critical Equipment Identification and P-Categorization" [29]. The replacement of any obsolete or aging SSCs with an impact on RP must comply with the requirements of the Radiation Protection Program and N-STD-MA-0024 [28], such that the replacement item must have demonstrated equivalency and perform the intended design function. Hence, this prevents obsolescence of RP related SSCs from having any adverse impacts on maintaining radiological exposures ALARA.

Note that an evaluation of aging aspects affecting SSCs important to safety, is provided in PSR2 Safety Factor 4 Report, "Aging" [30].

<u>Radiological Hazard Control</u>

Radiation exposure pathways can be through either external exposure (irradiation by sources outside the body from identifiable sources) or internal exposure (irradiation by sources inside the body originating from airborne, surface and liquid contamination). In terms of protection from external radiation, shielding was designed to align with the dose rate targets specified in Table 64 of the Pickering 1,4 and in Table 12-1 of the Pickering 5-8 Safety Reports [13], [14]. The categories of shielding are as follows:

- Primary shield – shielding that attenuates radiation from the reactor.
- Secondary shield – shielding that attenuates radiation from the HT coolant.
- Auxiliary shield – shielding that attenuates radiation from auxiliary systems, such as moderator and fuelling machines.
- Special shield – any shielding not categorized above.

In terms of airborne contamination, per Section 12.3.4.3 and 12.3.6 of the Pickering 1,4 and 5-8 Safety Reports [13], [14], within Zones 2 and 3, airborne contamination is controlled by the adjustment of ventilation flows. Ventilation arrangements are designed such that transfer of atmosphere between different areas, due to a pressure differential, travels from the potentially less to the potentially more contaminated area. Exhaust ducts are provided with damper equipped local exhaust inlets, which can be connected to plastic tent-like structures over and around equipment that is contaminated, to give special ventilation during maintenance. The Breathing Air System supplies flexible hose connection stations throughout the plant where the use of ventilated plastic suits may be required.

The main airborne activity of concern for Pickering NGS is tritium. Release of tritium from the station is kept within limits by containment of heavy water within the HTS and moderator systems where the tritium is produced, by removal of heavy water vapour in the Reactor Building atmosphere (e.g. using Vapour Recovery Dryers), by recovery of any leakage from the heavy water circuits, and by good housekeeping in heavy water management areas. Per Section 11.4.7 of the Darlington Safety Report (Part 2) [31], the Darlington Tritium Removal Facility (TRF) reduces and maintains low tritium levels in OPGN heavy water inventories, including the Pickering NGS Moderator and HTS. By reducing the tritium concentrations in these systems, potential tritium exposures are reduced. The TRF extracts, concentrates, immobilizes and stores the tritium as a metal tritide inside a container vessel. Pickering NGS heavy water is transported to the TRF and returned following tritium extraction. The containers of extracted tritium are stored at Darlington NGS.

Per Section 1.6 of N-PROG-RA-0013 [7], a contamination control area is an area set up with barriers and warning signs to prevent inadvertent access, contain loose or liquid contamination, and prevent spread of contamination. The need to establish contamination control areas is identified during work planning or when a discovered contamination hazard cannot be removed immediately. Contamination control areas may be rubber areas or other containment systems that are clearly marked and bounded.

<u>Decontamination of Personnel, Equipment and Structures</u>

As outlined in the Pickering 1,4 and 5-8 Decontamination System design manuals [32], [33], decontamination facilities are located in various parts of the service wing of Pickering NGS as follows:

- The Active Maintenance Shop provides for the safe disassembly and steam cleaning/manual washing of large machine parts/assemblies and the storage of clean (but still active) parts.

- The Decontamination Centre receives small items from the Reactor Buildings or the service wing and also provides steam cleaning, soaking and manual scrubbing facilities plus shielding and/or ventilated storage for contaminated items which have been decontaminated but still display some fixed activity.

- Laundry facilities are provided for washing and drying items as applicable.

If contamination is detected on material during any monitoring activity, the contaminated surfaces or materials are contained or decontaminated in accordance with N-PROC-RA-0015, "Contamination Control While Performing Work" [34]. This procedure specifies work practices, measures, and techniques used to control radioactive contamination while working with contaminated materials to minimize the spread of contamination to people, equipment, and between work locations.

The focus of N-PROC-RA-0014, "Radiological Zoning, Personnel/Material Monitoring and Transfer Permits" [35] is to prevent the spread of radioactive contamination. The procedure also addresses the required response should contamination be detected on personnel.

Per References [32] and [33], for decontaminating very large pieces of equipment, buidling floors and walls (e.g. structures), manual scrubbing with the use of strong detergents and possibly weak acid solutions may be used. Reduction of exposure to personnel using these methods is accomplished through portable shielding, ventilation exhausts, protective clothing and air masks.

Radiological Monitoring (In-Plant)

Per Section 12.3.2 of the Pickering 1,4 and 5-8 Safety Reports [13], [14], fixed area radiation monitoring is provided to detect the occurrence of radiation hazards and to warn personnel of high radiation fields. An alarm is generated in the Main Control Room on high radiation level and high rate of increase of radiation level. The alarms also annunciate in the monitored area. These monitors are primarily in Access Control Areas. Staff that may be working in the area are warned by a horn and a rotating flasher located near the alarming detector. The flashers and horns continue to operate for the duration of the alarm condition.

Per Section 12.3.4.2 and 12.3.5.1 of the Pickering 1,4 and 5-8 Safety Reports [13], [14], contamination monitors are located throughout the station at inter-zonal boundaries and other points of importance to prevent the spread of contamination by movement of staff and equipment. The monitors are considered semi-portable and may be relocated on either a temporary or a permanent basis to reflect changes in contamination control requirements (e.g. as may be caused by changes in the pattern of work within the station).

Further details on RP equipment and instrumentation for radiation and contamination monitoring are provided in Section 4.1.2.

Design of Plant SSCs for Radiation Protection

SSCs have been designed such that radiation exposure during all activities is minimized and justified, and aligns with CNSC requirements and expectations. For example, CNSC requirements for the design and layout of the reactor facility in the area of RP are outlined in Section 8.13 of CNSC REGDOC-2.5.2 [19], which states:

"The design and layout of the plant shall make suitable provision to minimize exposure and contamination from all sources. This shall include the adequate design of SSCs to:

1. *control access to the plant*
2. *minimize exposure during maintenance and inspection*
3. *provide shielding from direct and scattered radiation*
4. *provide ventilation and filtering to control airborne radioactive materials*
5. *limit the activation of corrosion products by proper specification of materials*
6. *minimize the spread of active material*
7. *monitor radiation levels*
8. *provide suitable decontamination facilities"*

A description of the Pickering NGS RP design provisions has been provided in the preceding subsections. The provisions meet the intent of REGDOC-2.5.2 requirements as follows:

1. Control access to the plant (refer to Radiological Zoning and Access Control subsection);
2. Minimize exposure during maintenance and inspection (refer to Radiological Hazard Control subsection);
3. Provide shielding from direct and scattered radiation (refer to Radiological Hazard Control subsection);
4. Provide ventilation and filtering to control airborne radioactive materials (refer to Radiological Hazard Control subsection);
5. Limit the activation of corrosion products by proper specification of materials (refer to RP Program Background subsection);
6. Minimize the spread of active material (refer to Radiological Hazard Control and Decontamination of Personnel, Equipment and Structures subsections);
7. Monitor radiation levels (refer to Radiological Monitoring subsection); and
8. Provide suitable decontamination facilities (refer to Decontamination of Personnel, Equipment and Structures subsection).

CNSC RD/GD-369, "Licence to Construct a Nuclear Power Plant" [36], provides guidance in terms of the information that should be submitted in support of an application for a licence to construct a NPP. Section 11 of RD/GD-369 [36] outlines the RP related information to be submitted as part of the application. A description of the Pickering NGS RP design provisions and programs is provided in this Safety Factor 15 report. These RP provisions and programs meet the intent of the RD/GD-369 requirements. Alignment with the RD/GD-369 requirements is based on the information provided in this report as follows:

- Application of the ALARA principle (refer to Section 4.1.5);
- Radiation Sources (refer to Radiological Zoning and Access Control subsection in Section 4.1.1);
- Design features for Radiation Protection (refer to Section 4.1.1);
- Radiation Monitoring (refer to Radiological Monitoring subsection in Section 4.1.1); and
- Radiation Protection program (refer to RP Program Background subsection in Section 4.1.1).

In an effort to maintain radiation exposure ALARA, Pickering NGS has continually made improvements to station design, zoning, shielding and RP practices in response to Operating Experience (OPEX) (both internal and external). For Design Basis considerations, the most significant RP improvements arose in response to the accident at Three Mile Island in 1979. Response to this event included the preparation of detailed reviews of plant design and operation following an accident with fuel failures (Pickering NGS reviews are contained in References [37] and [38] respectively). These reviews initiated a number of significant plant improvements including:

- Airlock solid seals (at prescribed locations);
- Shielding around the Pickering 1-4 helium storage tank and Pickering 5-8 Emergency Coolant Injection System piping;
- Improved isolation doors and water collection for the Vacuum Building basement; and
- Improved post-accident isolation of Containment penetrations for D2O Addition/Transfer, Moderator Cover Gas, Reactor Building Vapour Recovery and Leakage Collection.

For Beyond Design Basis Accident (BDBA) considerations, the most significant OPEX is from the 2011 accident at Fukushima Daiichi. Response to this event included the introduction of new portable Emergency Mitigating Equipment (EME) (N-BDB-03600-00002, "OPG Emergency Mitigating Equipment For Beyond Design Basis Accidents: Technical Basis Document" [39]) supported by a detailed Pickering NGS post-accident habitability assessment [40]. The intent of this assessment was to confirm accessibility for existing plant facilities (e.g., Main Control Rooms) and to ensure that new BDBA mitigation provisions (e.g., BDBA monitoring and EME cooling provisions) would be accessible.

Conclusion:

The conclusion of this Review Task assessment is that reactor design features are adequate for Radiation Protection. The intent of Review Task #1 is met and therefore Pickering NGS is compliant.

4.1.2 Review Task #2: Radiation Protection Equipment and Instrumentation

> *Confirm the adequacy of the Radiation Protection equipment, and instrumentation for radiation monitoring.*

REGDOC-2.3.3 [2], Appendix A.3.2, *Review of radiation protection equipment and instrumentation for radiation monitoring*, elaborates on this Review Task by stating the following:

> *"The review of RP equipment and instrumentation for radiation monitoring should demonstrate adequate provisions for monitoring all significant radiation sources, in all activities throughout the lifetime of the reactor facility. These should cover operational states and accident conditions and, as practicable, beyond-design-basis accidents, including severe accidents. The review of the physical condition of RP instrumentation and equipment should be confirmed by walk downs where practicable to verify continued utility and functionality."*

The following subsections address the requirements of this Review Task, as well as Appendix A.3.2 of REGDOC-2.3.3 [2], as noted above.

Radiation Protection Equipment

N-PROG-RA-0013 [7] is the program document that identifies the various procedures and methods for selecting and using equipment that will control occupational dose, preventing uncontrolled release of contamination or radioactive materials, and keeping collective doses ALARA.

N-PROC-RA-0096, "Lifecycle Management of Radiation Personal Protective Equipment" [41] documents the established process followed at OPGN for the lifecycle management of Radiological Personal Protective Equipment (RPPE). This includes the process for the acquisition of approved RPPE, adding new equipment to the approved list and measures for control, maintenance, inspection, and walkdowns of existing supplies. N-EL-03425.01-10000 "Radiation Personal Protective Equipment Approved for Purchase" [42], is the document that tracks the approved RPPE, including radiation area clothing, respiratory protection, and welding apparel.

N-PROC-RA-0025, "Selection of Radiation Personal Protective Equipment" [43] outlines the procedure for the selection of appropriate RPPE, in order to prevent personal contamination and internal uptake and limit the spread of contamination at OPGN facilities. It provides details on the protective clothing and respiratory protection required for anticipated hazards and planned activities. Table 1 of N-PROC-RA-0025 outlines the protective clothing and respiratory protection, which can include basic coveralls, plastic suit ensembles, anti-contamination ensembles, overshoes, rubber gloves, particulate respirators, Ram's horn and hood, and radioiodine respirators [43].

N-PROC-RA-0015, "Contamination Control While Performing Work" [34] specifies the work practices, measures, techniques, and equipment used to control radioactive contamination and minimize the spread of contamination while working. It outlines the requirements for the establishment of contamination control areas, such as:

- Catch Containment (stand-alone),
- Self-Administered Rubber Area,
- Rubber Area,
- Rubber Change Area,
- Contaminated Area, and
- Permanent Rubber Area.

Appendix B of N-PROC-RA-0015 [34] provides the guidelines for decontamination of tools, including use of detergents and cleansers, foaming aerosol cleaners such as Rad Con, and fume hoods. Appendix C provides the guidelines for the decontamination of floor areas, including use of mops, wipes, HEPA filtered vacuum cleaners, and discusses cleaning techniques to systematically remove contamination. Appendix E outlines the guidelines for use of catch containment devices.

OPG-PROC-0132, "Respiratory Protection" [44] establishes the requirements for the selection, care and use of respiratory protection to protect workers against workplace hazardous atmospheres. It includes requirements for the quality of compressed breathing air and compressed breathing air systems.

N-PROC-MA-0060, "Control of Temporary Shielding" [45] outlines processes and controls for requesting, evaluating, approving, installing and removing temporary shielding to protect personnel from radiation, in accordance with the ALARA principle.

<u>Instrumentation for Radiation Monitoring</u>

N-PROC-RA-0066, "Lifecycle Management of Radiation Protection Instruments" [46], establishes procedures for:

- The acquisition of approved RP instrumentation;
- Adding new RP instrumentation to the approved list;
- The maintenance, inspection and calibration of RP instruments;
- Program monitoring, including the creation of performance indicators, as per N-INS-03425.41-10002, "Performance Indicators For Radiation Instruments" [47], and the initiation of corrective actions and SCRs to document adverse trends;

- The evaluation, performance and acceptance testing of new RP instrumentation; and
- The commissioning of fixed RP instruments.

Only approved RP instruments are purchased, as listed in N-EL-03425.42-10000, "List of Radiation Protection Instrumentation Approved for Purchase in OPGN" [48] or N-EL-03425.42-10001, "List of RP Instrumentation for Specialized Use in OPG Nuclear" [49].

N-PROC-RA-0012, "Dosimetry and Dose Reporting" [50] specifies criteria and methods for use of radiation dosimetry and dose control devices. It outlines the required dosimetry needed for entering different radiological zones of the plant, performing radioactive work, or for potential exposure to specific radiation hazards (such as C-14, radioiodine, airborne alpha, tritium oxide, and other airborne particulates). It also outlines proper usage for thermoluminescent dosimeters, electronic personal dosimeters, personal and air samplers. Remote monitoring of dosimetry for staff performing radioactive work can also be performed by RP staff using the Audio-Visual Teledosimetry System. N-INS-03428-10000, "Audio-Visual Teledosimetry System (AVTS) Instructions" [51] outlines the capabilities and instructions for use of the system. These capabilities include video monitoring, live dosimetry monitoring and communication with staff.

N-PROC-RA-0024, "Hazard Surveys, Posting, Labeling, and Radiological Log" [52], describes the requirements for surveying radiation hazards, posting, labeling, and recording hazard details to ensure conditions are tracked and communicated to other workers. Appendix A of N-PROC-RA-0024 [52] outlines the routine survey requirements for all radiological zones, eating and drinking areas, continuous air monitors, tritium air monitors, and workplace monitoring. Appendix B lists the operating instructions for measuring gamma radiation, surface contamination, tritium and airborne contamination, beta and neutron radiation, and for fixed RP instruments. This includes instructions for hand held instrumentation such as gamma survey meters, surface contamination survey meters, passive samplers, tritium survey meters, beta/gamma radiation meters, neutron meters, as well as semi-portable particulate monitors.

Fixed contamination monitoring, using equipment such as hand and foot monitors, small article monitors, fixed survey meters, and whole body contamination monitors, is performed using the directions specified in N-INS-09071-10004, "Use of Fixed Radiation Protection Instruments" [53]. At the point of exit from Zone 2 to Zone 1, sensitive whole body contamination monitors and portal monitors are used to monitor for contamination.

Tritium monitoring is done using various portable instruments or samplers depending on the location, application and sensitivity required [13], [14]. Gaseous diffusion units are normally used to determine tritium concentrations in most areas.

Particulate-in-air monitors can also be set up to monitor and alarm at work sites. In addition to the higher energy emission from more common radionuclides potentially present at the facility, these monitors are sensitive to low energy beta radiation from particulate carbon-14 [13],[14].

Per Section 12.3.2 of the Pickering 1,4 and 5-8 Safety Reports [13], [14], fixed area radiation monitoring is provided by Fixed Area Gamma Monitors[8] (FAGMs) and Semi Portable Area Gamma Monitors to detect the occurrence of radiation hazards and to warn personnel of high radiation fields. As noted in Review Task 1 (Section 4.1.1) in this report, an alarm is generated in the Main Control Room by the FAGMs on high radiation level and high rate of increase of radiation level and a local alarm is sounded (horn and rotating flasher) near the alarming detector. If necessary, area gamma monitoring can be supplemented with Portable Area Gamma Monitors (such as Thermo RMS3 HP270 Probes and Bot AR600 with 713 Probes).

Per Section 12.3.4.2 and 12.3.5.1 of the Pickering 1,4 and 5-8 Safety Reports [13], [14], contamination monitors are located throughout the station at inter-zonal boundaries and other points of importance to prevent the spread of contamination by movement of staff and equipment. The monitors are discussed in additional detail in Review Task 1 (Section 4.1.1) in this report.

Additional dose monitoring capability is provided by the Automated Near Boundary Gamma Monitoring System [54] and the Automated Source Term Gamma Monitoring System [55]. These two systems measure gamma dose rate at various points around the Pickering site. The primary functions supported by these systems are categorization of events and prediction of offsite dose during nuclear emergencies. However, both systems can be used to measure dose rate during normal operation, as they provide continuous data and can be viewed from computers on the Pickering local area network.

Communications equipment is an important part of the RP program. In addition to the headset communications which form part of the AVTS [51], communications between RP personnel and workers in the plant can be performed using the Pickering NGS telephone system, hand held radios, the Public Address System or the Emergency Communications System.

RP Equipment for Emergency Response

RP equipment and instrumentation available for radiation monitoring for nuclear emergencies (including Design Basis, and Beyond Design Basis Accidents) are outlined in N-PROG-RA-0001, "Consolidated Nuclear Emergency Plan" [56]. RPPE is issued to Emergency Response staff during an emergency, as follows:

> *In the event of a nuclear emergency, as appropriate, on-site emergency responders are provided with personal dosimeters and radiation monitoring equipment. Emergency response facilities are equipped with area radiation monitors and airborne samplers. All emergency teams mobilized by the*

[8] A discussion of recent issues related to calibration of FAGMs is included in Appendix B.

Emergency Operation Centre are briefed, prior to deployment, on radiation levels, dosimetry, assigned radiation exposure limits and protective clothing. Airborne hazard breathing protection is specified. KI tablets are available for issue if necessary. Site-specific emergency response implementing procedures contain emergency dose limits, details for radiological protection, and criteria for specifying radiation protection equipment. Personnel are also provided the necessary communication equipment prior to deployment.

The equipment needed for Emergency Response, communications systems, radiation monitoring capabilities, portable Emergency Response RP equipment, and radiation personnel protective equipment, is tracked using N-PROC-RA-0133, "Management of Equipment Important to Emergency Response (EITER)" [57]. This program identifies which equipment is required for the Emergency Response Organization to perform their roles, categorizes the equipment based on importance, redundancy, and whether it is required for Design Basis, or Beyond Design Basis Accidents, and outlines the availability requirements and mitigating actions to be taken should equipment be unavailable. The list of Pickering EITER is identified in P-INS-03491-00050 [58].

Additional Radiation Protection equipment would be available to OPG and Pickering NGS during a nuclear emergency as part of a Mutual Aid Agreement between Bruce Power, OPG, Atomic Energy of Canada Limited, Hydro-Quebec, and New Brunswick Power, as outlined in N-LEGL-03490-0413370 [59]. Annex 1 of the agreement outlines the equipment and expertise that can be supplied by each of the members to the agreement in the event of a nuclear emergency. This includes mobile radiation assessment, radiological survey equipment, shielding structures, dosimetry equipment such as TLDs, as well as health physics personnel to assist.

Additional Radiation Protection of the public during nuclear emergencies is achieved through the design of the plant. These provisions are detailed in Task 1 (Section 4.1.1) in this report.

There are plant systems which are designed to measure or limit radiation releases from the station. The Fixed Gaseous Process Radioactive Effluent Monitoring System (also known as stack monitoring), is used to monitor radioactive emissions from the Pickering NGS ventilation exhaust [60]. The Liquid Effluent Sampling and Monitoring System is used to monitor the liquid effluent stream to Lake Ontario [61].

<u>Calibration and Maintenance Requirements</u>

Section 20 of SOR/2000-207, "Nuclear Substances and Radiation Devices Regulations" [62], requires that all radiation survey meters be calibrated within the 12 month period preceding use. N-INS-09071-10009, "Requirements for the Calibration and Maintenance of Radiation Protection" [63] outlines the OPG maintenance and calibration requirements for RP instruments, which is compliant with Section 20 of the Regulation. N-INS-09071-10009 [63] makes a distinction between portable RP equipment (e.g. handheld survey meters, personal dosimeters, etc.), and fixed and semi portable contamination monitors. Portable RP equipment calibration and testing procedures are also outlined in N-INS-09071-10009, along with the requirements for the frequency of calibration and source checking, which allows for a functional check of the equipment.

The alarm set points and minimum source check frequency for fixed and semi portable monitors are outlined in Appendix A of N-INS-09071-10009. For FAGMs and Semi Portable Area Gamma Monitors, the responsibility for these systems lies with the Operations and Maintenance organizations since these are engineered systems [62]. Walkdowns, calibration, testing, and maintenance of these systems falls under the plant maintenance program, outlined in N-PROG-MA-0004, "Conduct of Maintenance" [64]. System performance monitoring of engineered systems is conducted per N-PROC-MA-0024 [26]. System and equipment walkdowns are conducted by system engineering staff and routine walkdowns are conducted by Operations and RP staff. The performance of these fixed systems is tracked using OPG's quarterly performance indicator reports, as outlined in N-INS-03425.41-10002 [47].

Any real or potential deficiencies identified in calibration or maintenance of RP equipment are filed as SCRs, which are categorized, evaluated and processed in accordance with N-PROC-RA-0022, "Processing Station Condition Records" [65]. Regular audits and self-assessments (which review these SCRs) are undertaken to determine the effectiveness of RP provisions at Pickering NGS. Note that a discussion of SCR trends is included in Appendix B of this document.

Conclusion:

The conclusion of this Review Task assessment is that the RP equipment and instrumentation for radiation monitoring at Pickering NGS are adequate. The intent of Review Task #2 is met and therefore Pickering NGS is compliant.

4.1.3 Review Task #3: Radiation Protection of the Public and Workers During Nuclear Emergencies

Confirm that adequate provisions are in place to address Radiation Protection of the public and workers during nuclear emergencies.

REGDOC-2.3.3 [2], Appendix A.3.3, *Review of radiation protection aspects for nuclear emergencies*, elaborates on this Review Task by stating the following:

> *"The review of RP aspects for nuclear emergencies should demonstrate the effectiveness of RP measures during a nuclear emergency. These measures may be significantly impacted by facility configuration and controls; or for example, the review should consider access controls, habitability controls, communications systems, adequate radiation monitoring capabilities, portable emergency response RP equipment, and radiation personnel protective equipment."*

Provisions to respond to nuclear emergencies constitute primarily Defence in Depth Levels 4 and 5 [17] to mitigate the radiological impact on workers, the public, the environment, and off-site response (the Defence in Depth levels were discussed further in Section 4.1.1). The following subsections address the requirements of this Review Task, as well as Appendix A.3.3 of REGDOC-2.3.3 [2], as noted above.

Nuclear Emergency Planning

A detailed community response to nuclear emergencies is outlined by the "Provincial Nuclear Emergency Response Plan" (PNERP) [66] maintained by the Province of Ontario. Among other things, the PNERP outlines the guidelines concerning provincial issuance of operational directives for various measures implemented in the unlikely event of a nuclear emergency. These include protective actions such as instructions for sheltering and/or evacuation, or operational measures such as ground or aerial monitoring and other measures for the management of off-site consequences and the support and coordination of offsite organizations. For many measures, including sheltering, evacuation, administration of thyroid blocking agents and restrictions placed on the consumption of affected foods and water, the plan further outlines Protective Action Levels, specified in terms of projected radiation doses, as both a lower level (below which the protective measure would not normally be justified) and an upper level (above which the protective measure shall be implemented, unless implementation clearly entails greater risks for the people involved). This graded approach ensures response commensurate with the emergency level.

N-PROG-RA-0001, "Consolidated Nuclear Emergency Plan" [56], defines OPG commitments under the Provincial Nuclear Emergency Response Plan and provides a framework for OPG's interaction with external authorities. More specifically, N-PROG-RA-0001 [56] documents the concepts, roles, and resources required to implement and maintain an Emergency Response capability to protect the public, employees, and the environment in the event of a nuclear emergency. As outlined in the Consolidated Nuclear Emergency Plan, OPG maintains an Emergency Response Organization that would lead the on-site and support the off-site response and recovery during Design Basis and Beyond Design Basis Accidents. Radiation Protection support is provided to the Emergency Response Organization by the following staff [56]:

- The Health Physics Manager at the Site Management Centre is responsible for the following tasks during a nuclear emergency:
    - Collection and transmission of source term and off-site survey data to the Province.
    - Redeployment of Off-Site Survey Teams at Provincial request.
    - On-site exposure control and management, contamination control strategies, and on-site protective action recommendations.
    - On-site dose consequence assessment.
    - Guidance on any radiological casualties, and radiological or hospital support.
    - Radiation Protection services.

- Approval of exposure permits as needed.
- Radiological habitability of Site Management Centre.

The Health Physics Manager has numerous call-in staff for health physics, environmental, and radiation control support, including technical and field staff.

- The Health Physics Director at the Corporate Emergency Operations Facility is responsible for off-site radiological consequence assessment, review and assessment of provincial off-site protective action decisions, and assessment of on-site protective action recommendations. Their counterpart in the Site Management Centre is the Health Physics Manager.

- The In-Plant Survey Team is responsible for hazard surveys, as directed by the In-Plant Co-ordinator, and basic plant damage or abnormal condition recognition.

- The Off-Site Survey Team conduct off-site radiation monitoring and sampling.

- Emergency Response Projection Operators run the Emergency Response Projection computer program to provide off-site impact assessments. They also provide dose projection analysis and explanation in support of the task objectives of the Health Physics Director and Technical Support Director.

- Station staff also support the response at emergency off-site centres that are established as required under the PNERP (discussed further below).

Training and qualification of these Radiation Protection related Emergency Response Organization roles is outlined in N-TQD-503-00001, "Nuclear Emergency Response Organization Training and Qualification Description" [67]. Each Emergency Response Organization member must also participate in Continuing Training as prescribed in station procedures. A summary report of each Emergency Preparedness drill/exercise is prepared. The Radiation Protection Department reviews these reports, and incorporates OPEX and lessons applicable to RP in the quarterly RP Department Performance Improvement Report [68]. Note that additional details on the training and qualification of RP staff are captured in Section 4.1.5.

OPGN procedures for responding to Nuclear Emergencies include Abnormal Incident Manuals, Severe Accident Management Guides, and Emergency Mitigating Equipment Guides. These procedures are outlined further in the PSR2 Safety Factor 13 Report on Emergency Planning [69]. For Design Basis, and Beyond Design Basis Accidents, Radiation Protection elements of Emergency Response are primarily the responsibility of the Health Physics Manager, as discussed above.

As described in Reference [59], additional Radiation Protection equipment, expertise, materials, and other support would be available to OPG and Pickering NGS in the event of a nuclear emergency as part of a Mutual Aid Agreement between Bruce Power, OPG, Atomic Energy of Canada Limited, Hydro-Quebec, and New Brunswick Power.

RP for the Public During Nuclear Emergencies

OPG's role in Radiation Protection of the public during a nuclear emergency is outlined in the Consolidated Nuclear Emergency Plan [56]. This includes the use of various assessment techniques to determine the extent of the on-site radiation impact and to predict the off-site radiation consequence to the public (e.g. radiological source term measurements, meteorological analysis, off-site dose projections, dose assessment verification using off-site survey results and core, fuel and system status assessment techniques). It also includes provisions for public alerting and education, communications to external stakeholders, setup and provision of monitoring and decontamination services at Reception Centres, and the maintenance of an adequate supply of iodine thyroid blocking agents. As detailed in Reference [70], additional proactive efforts, including the pre-distribution of iodine tablets to all residences and businesses within the 10km primary zone of Pickering NGS and an intensive public education campaign on what to do in the very unlikely event of a nuclear emergency, have also been completed by OPG with support from the Region of Durham and City of Toronto. Other operational responsibilities regarding thyroid blocking agents (stocking, distribution and administration) are prescribed by the province's Ministry of Health and Long Term Care as detailed in the PNERP [66].

As discussed in Section 4.1.1 of this report, protection of the public is also achieved through the Defence-in-Depth design measures including station equipment (reactor control and cooling systems) and physical design barriers (Containment), and Emergency Response equipment. The overall response is assisted by the following plant systems, which minimize the release of radioactive material or provide monitoring of radioactive releases to assist with radiation dose control:

- Gaseous Process Radioactive Effluent Monitoring System (stack monitoring),
- Liquid Effluent Sampling and Monitoring System,
- Vapour Recovery System,
- Filtered Air Discharge System, and
- Containment Box Up.

RP for Workers During Nuclear Emergencies

The Radiation Protection of workers during nuclear emergencies is addressed through adherence to OPG's processes and procedures, including the Consolidated Nuclear Emergency Plan [56]. The Consolidated Nuclear Emergency Plan [56] outlines specific provisions for the protection of workers during nuclear emergencies. These include provisions for:

- Increased monitoring and decontamination including the staffing of Emergency Worker Centres and Monitoring and Decontamination Units to provide radiation exposure control and dosimetry for off-site emergency workers;

- Site evacuation of all non-essential staff (instructions for evacuation of non-essential staff are documented in P-INS-03491-00032, "Relocation Dismissal, Evacuation" [71]);

- Issuance of thyroid blocking agent to on-site personnel and OPG emergency workers; and

- In the extremely unlikely event the need arises, provisions for appropriate medical treatment facilities.

RP equipment used for Emergency Response, including radiation monitoring capabilities, portable Emergency Response RP equipment, and radiation personnel protective equipment was discussed in Section 4.1.2 of this report. Communication equipment included as part of Pickering NGS EITER [58] includes radios, site telephone system access for the Emergency Operations Centre, Site Management Centre, and the Shift Managers office, Nuclear Emergency Telephone System, as well as the seismically qualified Emergency Communications System. In addition, the Emergency Telecommunications Enhancement Project is underway as reported in [72], to provide additional emergency communications enhancements in the event of a Beyond Design Basis Accident.

Access and habitability controls for Pickering NGS would be unchanged during nuclear emergencies, and are detailed in P-INS-09071-00002, "Access Controls" [21].

N-PROC-RA-0019, "Dose Limits and Exposure Control" [73] specifies requirements to manage dose within Exposure Control Limits and Administrative Dose Limits to control any worker's dose below CNSC regulatory limits. This document also lists limitations on work when workers are placed on removal, where further radiation exposure of the worker needs to be temporarily limited. Appendix B of N-PROC-RA-0019 [73] outlines emergency circumstances under which these exposures can be exceeded.

N-PROC-RA-0027, "Radioactive Work Planning, Execution and Close Out" [74], outlines the criteria to be used for planning and executing all radioactive work. It states that:

> *In the event the station Shift Manager (SM) declares an emergency, the Radiation Protection (RP) procedures are to be followed to the extent possible/practical, however some discretion is available to the SM to deviate from this procedure to take actions expeditiously to protect workers, the public or the environment. In such an event, the SM shall ensure:*
>
> 1. *ALARA principles continue to be applied when planning emergency actions.*
> 2. *The emergency plan ALARA considerations, including the planned dose estimates, are documented in writing.*
> 3. *A Station Condition Record (SCR) is filed after the event.*

4. *In all cases where this SM discretion is exercised, Radiation Protection Department shall conduct a post-event review to identify lessons learned.*

In the event of a nuclear emergency, briefings for workers would be performed using N-FORM-11073 [75], "Emergency Task Briefing", which is used to identify hazards, PPE, and backout limits for the tasks. Additional guidance for BDBEs is provided by N-GUID-09013-10003 "Radiation Protection – Nuclear Emergency Responders Guide for Beyond Design Basis Events" [76], which includes dosimetry requirements, RPPE, survey equipment and exposure control.

Plant design provisions to minimize worker dose (which are also applicable to nuclear emergencies) are discussed in Section 4.1.1. Specific RP features of the Pickering NGS design include improved post-accident isolation of Containment penetrations for D_2O Addition/Transfer, Moderator Cover Gas, Reactor Building Vapour Recovery and Leakage Collection (which were added in response to the Three Mile Island accident). There has also been an assessment of existing plant facilities (e.g., Main Control Rooms) to ensure that new BDBA mitigation provisions (e.g., BDBA monitoring and Emergency Mitigating Equipment cooling provisions) would be accessible. OPG References [37] and [38] provide additional information explaining how habitability requirements (including Radiation Protection requirements) were addressed when BDBA accident mitigation improvements were made.

Emergency Response for accidents involving radioactive sources other than the reactor core are addressed in two documents. N-INS-04162-10003, "Emergency Response" [77] provides guidelines for emergency situations involving radiography sources and exposure devices. These guidelines are used to develop event-specific response plans appropriate for the conditions present during an emergency. N-STD-RA-0036, "Radioactive Materials, Transportation Emergency Response Plan" [78] identifies the responsibilities of OPG with respect to the response strategy and concepts to enable an effective response to a transportation incident involving an OPG shipment of radioactive material. The plan also identifies the liaison and potential interface with external Emergency Response organizations.

Conclusion:

The conclusion of this Review Task assessment is that adequate provisions are in place to address Radiation Protection of both the public and workers during nuclear emergencies. The intent of Review Task #3 is met and therefore Pickering NGS is compliant.

4.1.4 Review Task #4: Improvement of Radiation Protection Based on Operating Experience

> *Confirm that the Radiation Protection provisions have been improved as the result of external operating experience.*

REGDOC-2.3.3 [2], Appendix A.3.4, *Review of radiation protection related to operating experience*, elaborates on this Review Task by stating the following:

> *"The review of RP-related operating experience (OPEX) should identify OPEX reports from other reactor facilities and relevant national and international experience and research findings. The review should verify that this information has been properly considered in the routine evaluation of OPEX and research developments and that appropriate action has been taken. The review of OPEX should seek to identify good practices and lessons learned elsewhere, and to take advantage of improved knowledge derived from research, in the area of RP."*

OPG ensures Radiation Protection is managed in a manner consistent with regulatory requirements and international standards. This includes identifying Radiation Protection related Operating Experience from other reactor facilities and relevant national and international experience and research findings. OPEX is incorporated into the RP program through Section 1.2.3 of N-PROG-RA-0013 [7], which states that the design and execution of the RP program is subject to ongoing monitoring through mechanisms including, but not limited to:

- External assessments performed by groups such as the CNSC or the World Association of Nuclear Operators (WANO);
- Reviews of industry operating experience;
- Benchmarking of OPG practices with the rest of the nuclear industry; and
- Information obtained from the CANDU Owners Group (COG) and research and development programs.

Events or conditions identified through these mechanisms that indicate real or potential deficiencies are filed as SCRs. At Pickering NGS, the process for documenting external OPEX as SCRs is conducted in accordance with N-PROC-RA-0035, "Operating Experience Process" [79]. This document states that OPEX SCRs for WANO Significant Operating Experience Reports (SOERs), WANO Significant Event Report (SERs), and Level 1 and 2 Institute of Nuclear Power Operations (INPO) Event Report (IERs) be initiated within 5 working days of their communication. SCRs are categorized, given a significance rating, and where warranted, evaluated for corrective actions to be taken to address deficiencies. SCRs are processed in accordance with N-PROC-RA-0022, "Processing Station Condition Records" [65].

The CANDU Owner's Group (COG) CANDU Radiation Protection Peer Group (RPPG), of which OPG is a member, was established to provide industry Radiation Protection management personnel with a forum to identify and address generic issues related to Radiation Protection within CANDU nuclear power plants and related facilities and organizations. The working group provides an environment for promoting awareness of issues and for developing common response strategies. COG also maintains a database of OPEX gathered from the CANDU reactors around the world, as well as major industry groups such as WANO, INPO, Electric Power Research Institute (EPRI), etc. This OPEX is screened at weekly meetings, which are attended by OPG staff. COG also co-ordinates CANDU specific research and development as well as joint projects between OPG and other utilities.

The Radiation Protection department creates a quarterly Performance Improvement report [68]. The reports document industry Operating Experience, such as INPO and WANO event reports, as well as OPEX from within OPG itself. RP related SCRs generated within the quarter are reviewed to determine what OPEX or lessons learned are applicable. Outstanding actions related to SCRs are also listed. The reports also track external evaluations of RP adequacy and effectiveness, such as assessments from the IAEA, WANO Peer Reviews, and other industry assessments that provide Operating Experience and industry best practices to OPG. The opportunities for improvement in these assessments are noted and described, and subsequent action items are listed and tracked for completion.

In addition to the response to the major OPEX events discussed earlier in this report (e.g., Fukushima/Three Mile Island), some examples of good practices and lessons learned from elsewhere that OPG has identified using the SCR process and the COG OPEX screening process include:

- P-2015-02695, "OPEX SCR - INPO ICES #313040 Shepherd Calibrator Interlock Failure". OPG received OPEX from INPO identifying a possible defect in the Shepherd Model 89 box calibrators employed at Pickering for the calibration of fixed radiation detection instruments. This led to the practice of opening the device housing to inspect the solenoids during routine inspection.

- N-2010-00632, "OPEX- Alpha Contamination found During Refurbishing at Bruce Power (Bruce SCR 28184910, BC 04533)". A COG screening meeting identified this OPEX from Bruce Power. The HEPA vacuum system and tooling used at Bruce A was insufficient to prevent higher than expected alpha contamination during work on the heat transport piping during refurbishment of Unit 2. As a result of this OPEX, OPG reviewed and revised the radioactive work planning process for alpha hazards, monitoring, engineering controls, protective equipment and administrative controls.

- N-2009-07252, "OPEX-SCR: V.C. Summer - WBC Did Not Find I-125 Seeds, INPO OE30151 - COG Record 45921, 14DEC09". COG identified applicable OPEX from the V.C. Summer Station where a whole body counter did not detect that a contractor receiving radiation treatment had I-125 seeds implanted in his body, which triggered an exit monitor when the contractor attempted to leave the station. As a result of this, OPG reassessed procedures to specify that sodium iodide detectors with limited shielding could be used in similar situations to locate, identify and assess the radioactivity.

- As an example of improved Radiation Protection provisions from external OPEX, Section 7.2 of P-CORR-00531-03719, "Application for Renewal of Pickering Nuclear Generating Station Power Reactor Operating Licence" [80] notes reductions in dose due to improvements in the management of scaffolding tasks based on industry OPEX.
- A COG Radiation Protection Workshop was held in Toronto in August 2016. Lessons learned and observations are documented in the Benchmarking Report P16-001981-SA, titled "2016 CANDU Owners Group Radiation Protection Workshop Benchmarking Report".

When making engineering changes, the Radiation Protection program N-PROG-RA-0013 [7] mandates that "engineers maintain or improve upon designs that reduce occupational exposures". RP staff review engineering changes to provide input for achieving these goals in accordance with N-PROC-MP-0083, "Constructability, Operability, Maintainability, and Safety (COMS)" [81]. External OPEX is included in these reviews by means of N-PROC-MP-0090, "Modification Process" [82], which mandates that appropriate OPEX be used to understand technical issues and provide the best possible input for making operational decisions during the detailed design phase of a Design Engineering Change. This is, in part, documented in Section 3.1 of each modification's N-FORM-10959 "Design Scoping Checklist" [83], which specifies the following OPEX reviews:

- SCRs in accordance with N-PROC-RA-0022 [65];
- Outside industry experience to decide on vendors for service or materials;
- WANO, EPRI, INPO and COG web-sites for relevant information; and
- An as-needed discussion with the OPEX Single Point Of Contact.

Additional information on how OPG receives, assesses, incorporates, and implements OPEX and research findings was included as part of the PSR2 Safety Factor 9 Report, "Use of Experience from Other Nuclear Power Plants and Research Findings" [84].

Conclusion:

The conclusion of this Review Task assessment is that Radiation Protection provisions have been improved as the result of external operating experience. The intent of Review Task #4 is met and therefore Pickering NGS is compliant.

4.1.5 Review Task #5: ALARA Principle in Reactor Design and Operational Programs

The review will demonstrate that the ALARA principle has been incorporated in any modifications of the reactor design and operational programs and arrangements.

Per REGDDOC-2.3.3 [2], Appendix A.3.1, *Review of the reactor design features for radiation protection,* demonstration of the incorporation of the ALARA principle in the reactor design and operational programs and arrangements is required. Section 4.1.1 of this report, identifies ALARA principles at Pickering NGS at a high level, while the following subsections provide more detailed information regarding the application of ALARA in reactor design and operational programs.

<u>ALARA Principle for Design Modifications</u>

N-PROG-RA-0013 [7] mandates that when making engineering changes, engineers maintain or improve upon designs that reduce occupational exposures throughout the lifecycle of the facility. At Pickering NGS, this process is implemented through N-STD-RA-0018, "Controlling Exposure as Low as Reasonably Achievable" [85]. Specifically, Section 1.6.2 "Plant Modifications" states that:

> *(a) ALARA principles are to be applied to any changes to facility design.*
>
> *(b) Station Engineering ensures proposed changes to radiological systems are reviewed by Facility RP Department in accordance with N-PROG-MP-0001, Engineering Change Control, and N-PROC-MP-0083, Constructability, Operability, Maintainability, and Safety (COMS).*
>
> *(c) Facility RP Department shall review all proposed changes to facility design which may affect radiation exposure. Even though the proposed change may not explicitly involve radiation exposure, an ALARA review may uncover additional improvements, which may lead to dose reduction.*

RP staff are required to review engineering changes to provide input for achieving the goal of reducing occupational exposure in accordance with N-PROC-MP-0083, "Constructability, Operability, Maintainability, and Safety (COMS)" [81]. N-PROC-MP-0083 [81] mandates a graded approach to address Radiation Protection / ALARA aspects depending on the potential radiological risk of a modification as identified for the particular modification on N-FORM-10958, "Modification Outline" [86].

The control of public and environmental exposure to radiation in accordance with the ALARA principle is similarly documented in N-STD-OP-0042, "Controlling Radiation Exposure to the Public and the Environment to As Low As Reasonably Achievable" [87], which mandates that each engineering modification with the potential to impact the environment be reviewed by Environment Operations Support using N-FORM-

10422, “Environmental Impact Worksheet” [88], as per N-PROC-MP-0090, “Modification Process” [82].

Additional ALARA provisions may be applied as appropriate during maintenance or inspection activities e.g. through the use of temporary radiation shielding. This would be prescribed during the planning of radioactive work. N-PROC-MA-0060, “Control of Temporary Shielding” [45] outlines the processes and controls for requesting, evaluating, approving, installing and removing temporary shielding.

Note that in addition to engineering changes which impact radiation exposure, RP staff review changes to the use of space in radiological zones in accordance with N-PROC-RA-0054, “Control of Space Allocation for Transient Material and Extended Storage of Material within the Site” [89]. This procedure prescribes the administrative requirements regarding control of space allocation, transient materials, extended storage of material, and re-locatable structures.

<u>ALARA for Operational Programs and Arrangements</u>

The RP program as described in N-PROG-RA-0013 [7], has an important objective in keeping collective doses ALARA. The design and execution of the RP program is subject to ongoing monitoring through mechanisms including, but not limited to:

- Worker identified issues or opportunities to improve in the design or implementation of the RP program;
- RP program self-assessments;
- Independent audits;
- Review of exceptional dosimetry and dose control device measurement results;
- External assessments performed by groups such as the CNSC or WANO;
- Reviews of industry operating experience;
- Benchmarking of OPG practices with the rest of the nuclear industry; and
- Review of COG and other research and development programs.

To ensure radiation doses that could be received by workers at Pickering NGS are ALARA, N-PROG-RA-0013 [7] states that individual worker doses, including those for contractors and visitors, are managed to Exposure Control Levels (ECLs) that are below administrative control levels that are in turn below the regulatory limits. The ECLs may only be increased with line manager’s approval up to the Administrative Dose Limits (ADLs). Increasing allowable doses above ADLs requires a higher level of management approval. N-PROC-RA-0019, “Dose Limits and Exposure Control” [73] specifies requirements to manage dose within ECLs and ADLs to control any worker’s dose below CNSC regulatory limits and lists limitations on work when workers are placed on removal, where further radiation exposure of the worker needs to be temporarily limited. Appendix B of N-PROC-RA-0019 [73] outlines emergency circumstances under which these exposures can be exceeded.

Collective dose performance targets for each facility are established annually by station management and take into account the reductions achievable through the application of ALARA techniques. As work is planned in more detail, collective dose projections are reviewed and actions taken to ensure dose is ALARA. N-STD-RA-0018, "Controlling Exposure as Low as Reasonably Achievable" [85] describes elements of a managed system for use at OPG Nuclear facilities to keep and demonstrate that occupational collective dose is ALARA, social and economic factors taken into account. Specifically, this standard:

- Requires ALARA committees be created to ensure high level participation of senior management;
- Requires estimating collective dose for general facility operations and outages;
- Requires continuous monitoring of the effectiveness of ALARA measures; and
- Defines responsibilities for setting performance objectives.

N-PROC-RA-0027, "Radioactive Work Planning, Execution and Close Out" [74], specifies the processes that are to be used for planning and executing all radioactive work to ensure doses are kept ALARA and unplanned exposures are avoided. For example, all radioactive work must be executed in accordance with conditions and constraints specified on an ALARA Section approved Radiation Exposure Permit (REP), appropriate for the task being performed. If the REP identifies that Electronic Personal Dosimeter dose rate alarms are anticipated, methods for limiting the number and duration of alarms are incorporated into the task evaluation. Also, per Section 1.4.2 of Reference [85], a radiation hazard information system is readily available to facilitate personnel job planning, job safety analysis, pre-job briefing, dose estimating and dose rate trending. The system consists of a database containing both current and historical radiological data and has the ability to search for information by location. Data provided by the database include results from measured external dose rates, airborne radiation hazards and surface contamination. This system is one element of the Radiation Information System, which is a suite of computer applications containing personnel dose and qualification status, dose management information, radiation hazard information, software to issue Electronic Personal Dosimeters and a tool to prepare REPs.

Per N-PROC-MA-0013, "Planned Outage Management" [90], Radiation Protection plans are prepared and in place to support outage scope and scheduled activities. Based on the projected source term and approved outage work scope along with the detailed ALARA reviews completed, RP staff is required to establish the final dose estimates for the outage. These estimates provide the basis to specifically challenge opportunities to reduce the dose impact of specific work efforts or projects.

N-STD-OP-0042, “Controlling Radiation Exposure to the Public and the Environment to As Low As Reasonably Achievable” [87], establishes requirements for OPGN facilities to keep radiological exposures and doses to the public and the environment ALARA (PE-ALARA). The PE-ALARA process is an ALARA approach that is factored into aspects of the design, operation and maintenance of the station, such that minimizing the release of radioactivity to the environment is considered. Timely identification and mitigation of increases in emissions are the cornerstone of the PE-ALARA process. Note that PSR2 Safety Factor 14 Report, “Radiological Impact on the Environment” [91], provides an evaluation of the program for monitoring the radiological impact of the plant on the environment, which ensures the emissions are properly controlled and are ALARA.

Furthermore, as a condition of the Dosimetry Service Licence, OPG prepares Annual Compliance Reports to assess the regulatory and program requirements to measure and record occupational exposure. The Annual Compliance Reports submitted to the CNSC in the last three years [92], [93], [94] (which also include independent reviews of SCRs) did not identify any dosimetry licence violations or unplanned RP-related event reports related to Pickering NGS.

<u>Application of ALARA</u>

Initiatives to drive improvements to the ALARA program are documented in the Site ALARA Strategy, which is updated as warranted to reflect the results of self-assessments, benchmarking, corrective action plans and industry best practices. The comprehensive strategy includes the following key aspects to improve performance [80].

- Source Term Characterization and Mitigation:

  Comprehensive surveys are performed during planned outages and during operation to characterize the radiological condition and source term of each unit, enabling the development of current ALARA plans and effective dose reduction initiatives for jobs. If radiological ‘hot spots’ are identified, ALARA staff determine the benefit of removal of the hot spot versus other dose reduction methods. If warranted, the removal of the hot spot is managed via the station work management program.

- Source Term Reduction:

  Industry best practices are implemented to reduce source term and accordingly the dose to workers. For example, a reduction in process fluid filtration pore size will improve the removal of insoluble radionuclides, which in turn will reduce general dose rates and hot spots. Pickering maintains an aggressive strategy to reduce internal dose to workers through heavy water de-tritiation. Heavy water with low tritium concentration is transferred from storage to all running units during operation. During planned maintenance outages, bulk Moderator water transfers are executed, decreasing the tritium concentrations. Other internal dose reduction initiatives include better

Vapour Recovery Dryer maintenance, more effective use of supplemental Vapour Recovery Dryers, and improvements in leak management. As a result, dose to workers performing maintenance is effectively managed.

- Innovative Shielding:

  The application of temporary shielding has proven effective in reducing dose rates where the source could not be eliminated. Innovative uses of shielding, including custom designed shielding (e.g. in-boiler head applications and improved reactor face applications) and enhancements to support structures to decrease installation time (e.g. for boiler drains), have been implemented by the Radiation Protection Department.

  As part of the long range ALARA Strategy, a reactor face shielding cabinet has been developed for use in outages, which will considerably reduce dose to inspection personnel.

- Remote Monitoring and Instrumentation for Dose Control:

  Remote monitoring and teledosimetry is a key component of the ALARA program. The installation of remote monitoring equipment has improved radioactive work planning and reduced dose to workers. Remotely operated cameras have been used to perform visual inspections and monitoring of inaccessible areas in support of plant Operations and Maintenance activities. Remotely operated robotic equipment has been used very successfully to significantly reduce dose to workers in elevated dose-rate environments. This includes the performance of detailed radiation surveys; removal, manipulation and shielding of high dose rate components; and non-destructive examinations in 'hard to access' locations.

Additional improvements are also noted in OPG Correspondence P-CORR-03680-0612567 [95] as follows:

- Establishing dose goals for radioactive work to improve individual and station dose performance;

- Use of robotics to perform tasks in radioactive work areas, reducing radiation exposure and therefore dose to workers;

- Use of dynamic learning activities to provide workers an opportunity to practice Radiation Protection fundamentals in a simulated radioactive work environment using remotely controlled radiofrequency technology;

- Implementation of remote reading radiation detection instrumentation and real time data transmission to facilitate improved job planning and awareness of current radiological conditions;

- Implementation of a gamma ray imaging spectrometer to perform enhanced radiation surveys and to identify areas with elevated dose rates, enabling more effective shielding to reduce dose to workers; and

- Improved Vapour Recovery Dryer performance.

Training Provisions and Qualifications

Per N-PROG-RA-0013 [7], all personnel working at a nuclear site are assigned an RP qualification level based on successful completion of training, which is maintained through periodic retraining and testing. Maintenance of the qualification is also contingent on ongoing demonstrated ability to perform appropriately at the qualification level. Training is in sufficient detail that workers can carry out their obligations as specified in the CNSC regulations.

The requirements for achieving and maintaining qualification levels are stated in N-STD-RA-0015, "Radiation Protection Qualifications" [96] and N-TQD-502-00001, "Nuclear Radiation Protection Training and Qualification Description". As outlined in References [96] and [97], the RP training and qualification process is a five level system as follows:

- Red – Assigned to workers who do not have any formal RP training or whose previous qualification has expired or been terminated.

- Orange 1 – Basic RP training that permits unescorted access to all zones, but not to radioactive work areas unless under the protection of a person holding a Yellow or Green qualification.

- Orange 2 – Builds on Orange 1 qualification and provides an additional working right of using the contamination meter.

- Yellow – Personnel who routinely perform radioactive work and possess the advanced knowledge to protect oneself and provide direct protection for up to two lesser-qualified people from radiation and radioactive contamination.

- Green – Personnel who routinely provide indirect Radiation Protection for others or direct Radiation Protection for more than two others.

Qualified trainers, using approved training packages designed to meet approved training objectives, deliver RP training. RP training is delivered in accordance with N-PROG-TR-0005, "Training" [98], which provides the structure, processes and tools for defining, developing, implementing, documenting, assessing and improving the training required to ensure staff have the appropriate knowledge, skill and attitudes for safe and efficient operation.

Key positions in the RP program organizations are given additional Radiation Protection related training to become qualified to perform in their specialized positions within the program. As described in Section 1.4.2.5 of N-PROG-RA-0013, the specialized Radiation Protection training and qualifications are described in the following documents [7]:

- N-QG-406-00002, Nuclear Qualification Guide for Responsible Health Physicist;
- N-TQD-402-00001, Nuclear Radiation Protection Technician Training and Qualification Description;
- N-TQD-405-00001, Nuclear Health Physics Technologist Training and Qualification Description;
- N-TQD-406-00001, Nuclear Health Physicist Training and Qualification Description
- N-TQD-440-00001, Nuclear Safety Devices and Equipment Technologist/Radiation Instrument Technician Training and Qualification Description.

Conclusion:

The conclusion of this Review Task assessment is that ALARA principles have been incorporated into both modifications of the reactor design and into operational programs and arrangements. The intent of Review Task #5 is met and therefore Pickering NGS is compliant.

4.2 L/R/C/S Reviews

As per Section 2.2 of this report, detailed reviews for five L/R/C/Ss with content applicable to Safety Factor 15 are provided in Reference [6]. Incremental Reviews against the L/R/C/Ss were conducted using the methodology explained in Section 3.2 of this report. Through the reviews conducted, alignment of programs, procedures, and practices in place at Pickering NGS relative to the L/R/C/Ss related to Radiation Protection identified in the assessment basis in [1] was evaluated. Associated findings applicable to Safety Factor 15 are summarized in Table 3 below.

Table 3: PSR2 L/R/C/S Review Results for Safety Factor 15

L/C/R/S Reviewed	PSR2 L/R/C/S Review Results for Safety Factor 15
CNSC G-129 (2004), "Keeping Radiation Exposures and Doses As Low As Reasonably Achievable (ALARA)"	There are no PSR2 gaps for CNSC G-129 Revision 1 (2004). Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CNSC G-129 Revision 1 (2004).
CNSC G-228 (2001), "Developing and Using Action Levels"	There are no PSR2 gaps for CNSC G-228 (2001). Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CNSC G-228 (2001).
SOR/2000-202 (Amended June 2015), "General Nuclear Safety and Control Regulations"	There are no PSR2 gaps for the General Nuclear Safety and Controls Regulations (Amended June 2015). Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with the General Nuclear Safety and Controls Regulations (Amended June 2015).
SOR/2000-203 (Amended June 2015), "Radiation Protection Regulations"	There are no PSR2 gaps for the Radiation Protection Regulations (Amended June 2015). Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with the Radiation Protection Regulations (Amended June 2015).
CNSC REGDOC-2.2.3 (2014), "Personnel Certification: Radiation Safety Officers"	There is no Class II nuclear facility or prescribed equipment within the bounds encompassed by the Pickering NGS PROL and there is no requirement to have a Radiation Safety Officer. Hence, a review of REGDOC-2.2.3 is not required because REGDOC-2.2.3 is not applicable to Pickering NGS.

It is also noted that there are no PSR2 gaps for CNSC RD-204 (2008), "Certification of Persons Working at Nuclear Power Plants", which addresses training and qualification requirements for Health Physicists. RD-204 was reviewed under Safety Factor 10 (Organization, the Management System and Safety Culture) as part of OPG Report P-REP-03680-00021 R000, "Pickering NGS PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 8, 10, 12, 13, and 14" [99].

4.3 OPG Program Effectiveness Reviews

The OPG Nuclear Program reviewed for the Radiation Protection Safety Factor is identified in Table 2, and details of the associated effectiveness reviews for this N-PROG are provided in Appendix B.

Additional supporting program reviews are documented in the associated Safety Factor Reports as cross-referenced in the Review Task sections of this report.

The audits, self-assessments, SCR reviews, and work management lessons learned processes as well as the safety performance and results achieved, demonstrate that the programs and processes supporting Radiation Protection are effective, and that continual learning is employed to address issues as they arise.

4.4 Additional Review Findings

As discussed in Section 3.4, the PSR2 Safety Factor 15 assessment also included a review of commitments previously made to the CNSC, open CNSC action items, and exemptions granted by the CNSC, as identified in the R04 Pickering LCH [4], to determine if there are any impacts associated with the operation of the Pickering Units past 2020. The review also included identification and review of previously identified programmatic Darlington PSR1 gaps related to Radiation Protection to determine impacts associated with operation of the Pickering Units past 2020. This assessment did not identify any gaps for Safety Factor 15.

Findings from the review of previously identified PSR1 gaps in the Pickering Units 5-8 Continued Operations Plan [8] are provided in Reference [9]. Findings from the review of Fukushima Action Items are provided in Reference [10]. Results from the Continued Operations Plan and Fukushima Action Items reviews will be considered in the Global Assessment process.

There were no PSR2 gaps identified in this Safety Factor 15 report that require additional consideration under other Safety Factors.

5.0 RESULTS AND CONCLUSIONS

OPG Governance, Programs, Policies, Procedures, Instructions and Guidelines related to Safety Factor 15 were reviewed for the five PSR2 Review Tasks in Section 4.1 of this report. These Review Tasks were derived from CNSC REGDOC-2.3.3 and IAEA SSG-25, and address all aspects of Radiation Protection, including reactor design, Radiation Protection equipment and instrumentation, Radiation Protection during nuclear emergencies, improvements of Radiation Protection based on Operating Experience, and ALARA. The Review Task assessments resulted in no PSR2 gaps. L/R/C/S and OPG Nuclear Program effectiveness reviews for Safety Factor 15 were prepared per Sections 4.2 and 4.3, respectively, and resulted in no PSR2 Gaps. Per Section 4.4, this report also included identification and review of previously identified programmatic Darlington PSR1 gaps related to Radiation Protection (to ascertain the implications of extending Pickering NGS operation beyond 2020), as well as a review of the R04 Pickering LCH [4] for any impacts of Pickering NGS operation beyond 2020 on: a) OPG commitments previously made to the CNSC, b) open CNSC action items, and c) exemptions granted by the CNSC (all related to Safety Factor 15), which resulted in no PSR2 gaps.

The review of Safety Factor 15 has confirmed that Radiation Protection has been adequately accounted for in the design and operation of Pickering NGS, that Radiation Protection provisions (including design and equipment) provide adequate protection of persons from the harmful effects of radiation, and that contamination and radiation exposures and doses to persons are monitored and controlled and maintained ALARA.

6.0 REFERENCES

[1] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[2] CNSC REGDOC-2.3.3, *Periodic Safety Reviews*, April 2015.

[3] IAEA Safety Guide No. SSG-25, *Periodic Safety Review for Nuclear Power Plants*, 2013.

[4] CNSC Report, LCH-PNGS-R004, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[5] OPG Report, P-REP-03680-00003 R000, *Pickering NGS PSR2: Definition of Safety Factor Review Tasks*, May 2016.

[6] OPG Report, P-REP-03680-0586480 R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 9, 11, and 15*, September 2016.

[7] OPG Program, N-PROG-RA-0013 R009, *Radiation Protection*, January 2015.

[8] OPG Plan, NK30-PLAN-00531-00001 R005, *Pickering 5-8 Continued Operations Plan*, December 2015.

[9] OPG Report, P-REP-03680-00024 R000, *Pickering 5-8 Continued Operations Plan Review in Support of PNGS Periodic Safety Review 2 (PSR2),* January 2017.

[10] OPG Report, P-REP-03680-00022 R000, *Fukushima Action Item Review in Support of PNGS Periodic Safety Review 2 (PSR2)*, January 2017.

[11] OPG Procedure, N-PROC-RA-0023 R018, Fleetview Program Health and Performance Reporting, August 2013.

[12] OPG Report, NK38-REP-03680-10185 R002, *Darlington NGS Integrated Implementation Plan (IIP)*, April 2015.

[13] OPG Report, NA44-SR-01320-00001 R015, *Pickering A Safety Report – Part II,* July, 2012.

[14] OPG Report, NK30-SR-01320-00002 R004, *Pickering B Safety Report – Part II,* October, 2012.

[15] OPG Design Manual, NK30-72200 R004, *Reactor Building D_2O Vapour Recovery*, November 1986.

[16] OPG Design Manual, NK30-34230.1 R000, *Filtered Air Discharge System, Main Process – Units 1,2,3,4,5,6,7, and 8*, August 1988.

[17] OPG Design Manual, P-DM-63420-00001 R000, *Radioactivity Monitoring and Containment Isolation*, August 2015.

[18] IAEA Report, INSAG-10, *Defence in Depth in Nuclear Safety,* June 1996.

[19] CNSC REGDOC-2.5.2, *Design of Reactor Facilities: Nuclear Power Plants*, May 2014.

[20] OPG Report, P-REP-03680-00008 R00, *Pickering NGS PSR2 Safety Factor 1 Report: Plant Design,* March, 2017.

[21] OPG Instruction, P-INS-09071-00002 R002, *Access Control,* February, 2016.

[22] OPG Program, N-PROG-MP-0008 R006B, *Integrated Aging Management,* May, 2016.

[23] OPG Program, N-PROG-MA-0025 R002, *Major Components,* March 2015.

[24] OPG Program, N-PROG-MA-0026 R002, *Equipment Reliability,* May 2015.

[25] OPG Program, N-PROG-MA-0017 R008, *Component and Equipment Surveillance*, June 2015.

[26] OPG Procedure, N-PROC-MA-0024 R016, *System Performance Monitoring,* December 2016.

[27] OPG Report, P-REP-03680-00005 R001, *Pickering NGS PSR2 Safety Factor 2 Report – Actual Condition of Structures, Systems, and Components Important to Safety*, March 2017.

[28] OPG Standard, N-STD-MA-0024 R000, *Obsolescence Management*, August 2015.

[29] OPG Procedure, N-PROC-MA-0077 R007, *Critical Equipment Identification and Categorization*, December 2016.

[30] OPG Report, P-REP-03680-00007 R00, *Pickering NGS PSR2 Safety Factor 4 Report: Aging,* July, 2016.

[31] OPG Report, NK38-SR-03500-10001 R004, *Darlington Safety Report, Part 1 and 2*, December 2013.

[32] OPG Design Manual, NA44-DM-78100 R00, *Pickering Generating Station - Decontamination Systems,* February 1972.

[33] OPG Design Manual, NK30-DM-78100 R01, *Pickering Generating Station B – Decontamination Systems,* February 1990.

[34] OPG Procedure, N-PROC-RA-0015 R015, *Contamination Control While Performing Work*, January, 2015.

[35] OPG Procedure, N-PROC-RA-0014 R023, *Radiological Zoning, Personnel/Material Monitoring and Transfer Permits*, June, 2015.

[36] CNSC RD/GD-369, *Licence Application Guide, Licence to Construct a Nuclear Power Plant,* August 2011.

[37] OPG Report, NA44-REP-P0693024, *Design Review to Assess the Effect of a Loss of Coolant Accident with Fuel Failures, Report No 82185*, April 1982.

[38] OPG Report, NK30-REP-P0526016, *Review of Plant Design and Operation for a Loss of Coolant Accident with Fuel Failures*, Report No 82278, September 1982.

[39] OPG Technical Basis Document, N-BDB-03600-00002 R00, *OPG Emergency Mitigating Equipment For Beyond Design Basis Accidents: Technical Basis Document,* October 2015.

[40] OPG Report, N-REP-09013-10011 R001, *Fukushima Action Item 1.9.1 – Pickering NGS Habitability*, October 2014.

[41] OPG Procedure, N-PROC-RA-0096 R003, *Lifecycle Management of Radiation Personal Protective Equipment,* September, 2015.

[42] OPG List, N-EL-03425.01-10000 R013, *Radiation Personal Protective Equipment Approved for Purchase,* December, 2015.

[43] OPG Procedure, N-PROC-RA-0025 R019*, Selection of Radiation Personal Protective Equipment*, October 2015.

[44] OPG Procedure, OPG-PROC-0132 R000, *Respiratory Protection*, September 2013.

[45] OPG Procedure, N-PROC-MA-0060 R005, *Control of Temporary Shielding,* July, 2011.

[46] OPG Procedure, N-PROC-RA-0066 R007, *Lifecycle Management of Radiation Protection Instruments*, 2015.

[47] OPG Instruction, N-INS-03425.41-10002 R003, *Performance Indicators For Radiation Instruments*, May 2014.

[48] OPG List, N-EL-03425.42-10000 R006, *List of Radiation Protection Instrumenation Approved for Purchase in Ontario Power Generation, Nuclear,* November, 2014.

[49] OPG List, N-EL-03425.42-10001 R003, *List of RP Instrumentation for Specialized Use in OPG Nuclear*, November, 2014.

[50] OPG Procedure, N-PROC-RA-0012 R021, *Dosimetry and Dose Reporting*, June 2011.

[51] OPG Instruction, N-INS-03428-10000 R001, *Audio-Visual Teledosimetry System (AVTS) Instructions*, January 2016.

[52] OPG Procedure, N-PROC-RA-0024 R021, *Hazard Surveys, Posting, Labeling, and Radiological Log*, February 2016.

[53] OPG Instruction, N-INS-09071-10004 R003, *Used of Fixed Radiation Protection Instruments*, December 2015.

[54] OPG Manual, N-MAN-03490-00001 R004, User Manual - Automated Near Boundary Gamma Monitoring System, January 2017.

[55] OPG Manual, N-MAN-03490-00002 R002, User Manual- Automated Source Term Gamma Monitoring System, September 2015.

[56] OPG Program, N-PROG-RA-0001 R014, *Consolidated Nuclear Emergency Plan*, May 2015.

[57] OPG Procedure, N-PROC-RA-0133 R000, *Management of Equipment Important to Emergency Response*, December 2014.

[58] OPG Instruction, P-INS-03491-00050 R002, *Unavailability of Equipment Important to Emergency Response – Pickering*, November 2015.

[59] OPG File, N-LEGL-03490-0413370, *Mutual Aid Agreement for Nuclear Emergency Support*, November 2012.

[60] OPG Design Manual, NK30-67876.1 R003, *Fixed Gaseous Process Radioactive Effluent Monitoring, Reactor Building, Service Wing and IFB Exhaust Duct*, April 1985.

[61] OPG Design Manual, NA44-DM-71750-10002 R000, *Unit 014 Liquid Effluent Sampling and Monitoring System*, April 2005.

[62] Government of Canada, SOR/2000-207, *Nuclear Substances and Radiation Devices Regulations*, Last Amended March 2015.

[63] OPG Instruction, N-INS-09071-10009 R013, *Requirements for the Calibration and Maintenance of Radiation Protection Instruments*, February 2015.

[64] OPG Program, N-PROG-MA-0004 R011, *Conduct of Maintenance*, April 2015.

[65] OPG Procedure, N-PROC-RA-0022 R032, *Processing Station Condition Records,* November 2014.

[66] Ontario Minstry of Community Safety & Correctional Services, *Provincial Nuclear Emergency Response Plan*, http://www.emergencymanagementontario.ca/english/emcommunity/response_resources/plans/provincial_nuclear_emergency_response_plan.html#P1564_98592, 2009.

[67] OPG Training and Qualification Description, N-TQD-503-00001 R017, *Nuclear Emergency Response Organization Training and Qualification Description*, December 2015.

[68] OPG Report, P-REP-01966-0630362 R000, *Radiation Protection Department Performance Improvement Report*, January 2017.

[69] OPG Report, P-REP-03680-00017 R00, *Pickering NGS PSR2 Safety Factor 13 Report: Emergency Planning*, December 2016.

[70] OPG Correspondence, N-CORR-00531-07025, *Supplementary Documents For The October 1st Commission Meeting On KI Tablets Distribution*, September 2015.

[71] OPG (Pickering) Instruction, P-INS-03491-00032 R006, *Relocation, Dismissal, Evacuation*, October 2015.

[72] OPG Correspondence, P-CORR-00531-04784, *Pickering NGS – CNSC Action Item 2016-48-7470 Status Update on Emergency Mitigating Equipment and Telecommunications Projects,* July 2016.

[73] OPG Procedure, N-PROC-RA-0019 R006, *Dose Limits and Exposure Control*, March 2015.

[74] OPG Procedure, N-PROC-RA-0027 R017, *Radioactive Work Planning, Execution and Close Out*, August 2012.

[75] OPG Form, N-FORM-11073 R004, *Emergency Task Briefing*, April 2016.

[76] OPG Guideline, N-GUID-09013-10003 R000, *Radiation Protection – Nuclear Emergency Responders Guide for Beyond Design Basis Events*, November 2015.

[77] OPG Instruction, N-INS-04162-10003 R003, *Emergency Response*, May 2016.

[78] OPG Standard, N-STD-RA-0036 R003, *Radioactive Materials Transportation Emergency Response Plan*, February 2016.

[79] OPG Procedure, N-PROC-RA-0035 R018, *Operating Experience Process*, 2014.

[80] OPG Correspondence, P-CORR-00531-03719, *Application for Renewal of Pickering Nuclear Generating Station Power Reactor Operating Licence,* July, 2012.

[81] OPG Procedure, N-PROC-MP-0083 R009, *Constructability, Operability, Maintainability and Safety (COMS)*, April, 2016.

[82] OPG Procedure, N-PROC-MP-0090 R012, *Modification Process*, April 2014.

[83] OPG Form, N-FORM-10959 R016, *Design Scoping Checklist*, 2016.

[84] OPG Report, P-REP-03680-00013 R00, *Pickering NGS PSR2 Safety Factor 9 Report: Use of Experience from Other Nuclear Power Plants and Research Findings*, October 2016.

[85] OPG Standard, N-STD-RA-0018 R007, *Controlling Exposure as Low as Reasonably Achievable*, November 2014.

[86] OPG Form, N-FORM-10958 R015, *Modification Outline*, 2016.

[87] OPG Standard, N-STD-OP-0042 R003, *Controlling Radiation Exposure to the Public and the Environment to As Low As Reasonably Achievable*, August 2014.

[88] OPG Form, N-FORM-10422 R007, *Environmental Impact Worksheet*, December, 2014.

[89] OPG Procedure, N-PROC-RA-0054 R015, *Control of Space Allocation for Transient Material and Extended Storage of Material within the Site,* April, 2015.

[90] OPG Procedure, N-PROC-MA-0013 R017, *Planned Outage Management,* December, 2016.

[91] OPG Report, P-REP-03680-00018 R00, *Pickering NGS PSR2 Safety Factor 14 Report: Radiological Impact on the Environment*, December 2016.

[92] OPG Report, N-REP-08100-10014 R002, *Dosimetry Service Annual Compliance Report 2013*, March 2014.

[93] OPG Report, N-REP-08100-10015 R000, *Dosimetry Service Annual Compliance Report 2014*, March 2015.

[94] OPG Report, N-REP-08100-10016 R000, *Dosimetry Service Annual Compliance Report 2015*, March 2016.

[95] OPG Correspondence, P-CORR-03680-0612567, *Recent ALARA Initiatives At Pickering NGS,* September 2016.

[96] OPG Standard, N-STD-RA-0015 R013, *Radiation Protection Qualifications*, October 2014.

[97] OPG Training and Qualification Description, N-TQD-502-00001 R018, *Nuclear Radiation Protection Training and Qualification Description*, October 2014.

[98] OPG Program, N-PROG-TR-0005 R016, *Training*, May 2016.

[99] OPG Report, P-REP-03680-00021 R000, *Pickering NGS PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 8, 10, 12, 13, and 14*, December 2016.

Appendix A: Nomenclature

ADL	Administrative Dose Limit
ALARA	As Low As Reasonably Achievable
AVTS	Audio-Visual Teledosimetry System
BDBA	Beyond Design Basis Accident
CANDU	CANada Deuterium Uranium
CNSC	Canadian Nuclear Safety Commission
COG	CANDU Owners Group
COMS	Constructability, Operability, Maintainability, and Safety
ECL	Exposure Control Level
EITER	Equipment Important to Emergency Response
EME	Emergency Mitigating Equipment
EPRI	Electric Power Research Institute
FADS	Filtered Air Discharge System
FAGM	Fixed Area Gamma Monitor
IAEA	International Atomic Energy Agency
IAM	Integrated Aging Management
IER	INPO Event Report
INPO	Institute of Nuclear Power Operations
ISR	Integrated Safety Review
LCH	Licence Conditions Handbook
L/R/C/S	Laws, Regulations, Codes and Standards
NEW	Nuclear Energy Worker
NGS	Nuclear Generating Station
N-PROG	Nuclear Program
NPP	Nuclear Power Plant
OPEX	Operating Experience
OPG	Ontario Power Generation
OPGN	Ontario Power Generation Nuclear
PARTS	Pickering A Return to Service
PNERP	Provincial Nuclear Emergency Response Plan
PE-ALARA	Public and Environment ALARA

PROL	Power Reactor Operating Licence
PSR	Periodic Safety Review
PSR1	Periodic Safety Review 1 (earlier OPG PSR work and other associated assessments)
PSR2	Periodic Safety Review 2 (subsequent PSR per CNSC REGDOC-2.3.3)
REP	Radiation Exposure Permit
RPPE	Radiological Personal Protective Equipment
RP	Radiation Protection
SCR	Station Condition Record
SER	Significant Event Report
SM	Shift Manager
SOER	Significant Operating Experience Report
SSC	Structures, Systems, and Components
WANO	World Association of Nuclear Operators

Appendix B: OPG Program Effectiveness Review Results

B.1 N-PROG-RA-0013, "Radiation Protection"

The Radiation Protection Program implements a series of standards and procedures for the conduct of activities within nuclear sites and with radioactive materials. The objectives of the Radiation Protection Program include the following:

a) Controlling occupational and public exposure:

   - Keeping individual doses below regulatory limits;
   - Avoiding unplanned exposures; and
   - Keeping collective doses ALARA, social and economic factors taken into account.

b) Preventing the uncontrolled release of contamination or radioactive materials from the nuclear sites through the movement of people and materials.

c) Demonstrating the achievement of a) and b) through monitoring.

The Health Physics department completed a self-assessment in June 2016, NO16-000655-SA [B.1.1], in order to assess the effectiveness of contamination control methods at Pickering NGS. The self-assessment concluded that there were opportunities for improvement in the areas of contamination surveys, zonal boundary definition, and standards for radioactive material storage areas. AR# 28188262 was initiated, identifying actions that have since been completed to address the identified improvements.

The Health Physics department completed a self-assessment in May 2015, NO15-000885-SA [B.1.2], in order to assess Radiation Protection Program compliance and effectiveness for Pickering and Darlington NGS as well as the Western Waste Management Facility. It was concluded that improvements have been made to the Radiation Protection Program and identified some opportunities for improvement in the areas of radiation hazard signage, postings and worker practices. AR# 28177826 was initiated, identifying actions that have since been completed to address the identified improvements.

Nuclear Oversight conducted a performance based audit of the Radiation Protection Program in July 2015, NO-2015-010 [B.1.3], for both Pickering and Darlington NGS, to determine whether requirements defined in governance are being effectively implemented. The audit identified performance improvement opportunities applicable to Pickering NGS in the areas of staff practices, calibration and labelling of Radiation Protection instruments, rubber area management, and hazard postings.

Five SCRs were initiated (P-2015-15739, P-2015-15785, N-2015-15791, N-2015-15729 and P-2015-15775). All corrective actions were completed to address the findings. An effectiveness review will be completed in Q2 2017 (AR# 28180495).

A CNSC Type II inspection on radiological hazard control was conducted in October 2015 for Pickering NGS [B.1.4]. CNSC staff concluded that there were no non-compliances with the Nuclear Safety and Control Act, applicable regulations, licence conditions, codes or standards. One Action Notice was generated due to a non-compliance with N-STD-AS-0002, "Procedure Use and Adherence", by not consistently adhering to approved procedures and practices as they pertain to the maintenance of emergency showers. OPG provided CNSC staff with a corresponding action plan and all actions associated with the corrective action plan have been completed to the satisfaction of the CNSC [B.1.5], [B.1.6].

There were instances identified where FAGMs were not re-calibrated if the FAGM passed a function test, resulting in the 12 month re-calibration period not being met. SCRs were initiated and this issue is being tracked and addressed under OPG's Corrective Action Program. Corrective actions have been completed to address the underlying issues (including updated predefined maintenance requirements to ensure calibration checks are performed at the required frequency, and the provision of additional spare instruments and parts). An effectiveness review of the FAGM calibration compliance plan will be undertaken per the requirements of the Corrective Action Program. OPG will also continue to address CNSC requirements in this area [B.1.7] under its regulatory program.

Regulatory Oversight Report for Canadian Nuclear Power Plants

As documented in the 2015 Regulatory Oversight Report for Canadian Nuclear Power Plants [B.1.8], the CNSC concluded that Radiation Protection and control measures at Pickering NGS met or exceeded all applicable performance objectives and regulatory requirements, kept doses to workers within both the one and five year regulatory limits, and maintained the estimated annual dose to the public to less than 1% of the regulatory limit of 1 mSv, well below the stated annual natural background level (of 1.8 mSv). The report assigned a Fully Satisfactory rating to Pickering NGS, comparing it favourably to industry-wide performance.

The Regulatory Oversight Report also stated that CNSC staff have verified that Pickering's five-year ALARA plan includes dose-reduction initiatives based on a review of operational experience, including an initiative to reduce overall collective radiation exposure. Compliance activities conducted by CNSC staff verified that ALARA is implemented into work planning and dose monitoring and control processes.

CNSC Performance Indicators

Pickering NGS Radiation Protection performance is tracked by the CNSC through the issuance of a quarterly Safety Performance Indicators Report. The Safety Performance Indicators include collective radiation exposure, personnel contamination events, unplanned dose, loose contamination events, as well as summaries of Emergency Response Organization drills completed. Over the first 3 quarters of 2016 [B.1.9], [B.1.10], [B.1.11], OPG doses to workers have been maintained within both the one and five year regulatory limits.

2016 Trending Analysis

Pickering NGS complies with the regulatory requirements to measure and record doses received by workers at Pickering NGS. Routine compliance verification activities indicate that performance in the area of worker dose control at Pickering is highly effective. No worker or member of the public received a radiation dose in excess of the regulatory dose limits or action levels established in the Pickering Radiation Protection program. Per P-CORR-03680-0634906 [B.1.12], collective radiation exposure for Pickering NGS over the period 2013-2016 reflects the extensive maintenance programs and modifications executed, particularly during planned maintenance outages to improve operations. The average collective radiation exposure and collective internal radiation exposure over the years 2014-2016 were reduced by 113.4 rem/year and 6.7 rem/year respectively compared to the period 2009-2012 (note that 2013 was excluded from this comparison since only 2 major planned outages were executed in 2013 and all other years had 3 major planned outages). Collective radiation exposure targets are established based on the anticipated outage and maintenance schedule for the year, hence the targets may vary from year to year (e.g., a larger amount of reactor / fuel channel maintenance and inspection work may result in higher anticipated personnel collective radiation exposure for that year and thus a higher target.). The figure below demonstrates that Pickering NGS collective radiation exposure was within target in each of the past 4 years.

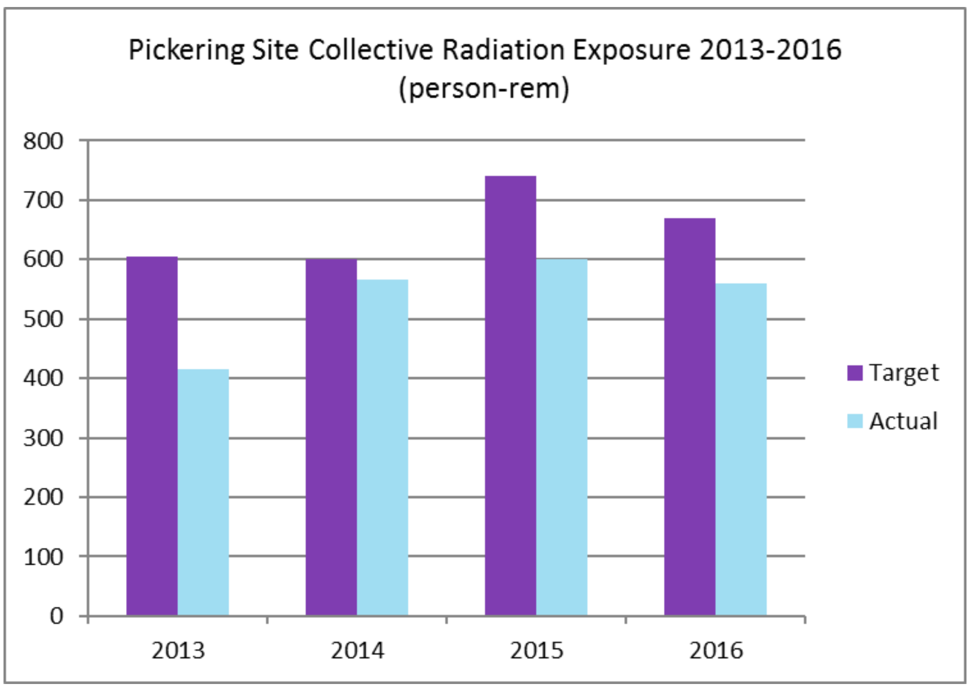


Figure 8 - Pickering Site Collective Radiation Exposure 2013-2016

In terms of SCR trending, per the Q4 2016 Radiation Protection Department Performance Improvement Report [B.1.13], from Q4 2015 to Q4 2016, SCR event based code analysis identified the following top three SCR contributors:

- Radiation Instrumentation - Major contributor identified as missing or lost gamma meters. Availability of the meters remains adequate.

- Dosimetry - The largest contributor was TLDs missing from the badge rack or lost TLD badges during work completion.

- Radiation Dose Control (Internal) - Major contributor identified as elevated tritium causing worker back-outs or delays.

In all cases, actions have been taken to address the trends.

Environmental Management Program

Per N-REP-03443-10015, "2015 Results of Environmental Monitoring Programs" [B.1.14], to ensure Pickering NGS activities are conducted in a manner that minimizes any adverse impact on the public and the environment, an Environmental Management Program has been established (consistent with CNSC requirements). For 2015, Pickering NGS radiological emissions for all radionuclides remained under 1% of their licensed Derived Release Limits.

References

[B.1.1] Self-Assessment, NO16-000655-SA, *Pickering NGS Contamination Control Report,* June 2016.

[B.1.2] Self-Assessment, NO15-000885-SA, *2015 Radiation Protection Program Self-Assessment,* May, 2015.

[B.1.3] Nuclear Oversight Audit, N-REP-01070-05501401 T06, *NO-2015-010 Radiation Protection Audit,* July, 2015.

[B.1.4] CNSC Letter, P-CORR-00531-04635, *Pickering NGS: CNSC Type II Compliance Inspection Report: PRPD-2015-014, Radiological Hazard Control,* January, 2016.

[B.1.5] OPG Letter, P-CORR-00531-04860, *Pickering NGS: Update on CNSC Action Item 2016-48-7398 and Request for Closure*, November, 2016.

[B.1.6] CNSC Letter, P-CORR-00531-04924, *Pickering NGS: CNSC Type II Inspection Report: Radiological Hazard Control, Action item 2016-48-7398,* December, 2016.

[B.1.7] CNSC Letter, P-CORR-00531-04979, *Pickering NGS: CNSC Type II System Inspection Report: PRPD-2016-023, Fixed Area Gamma Monitoring and Type I Semi-Portable Alarming Gamma Monitoring Systems, New Action Item 2017-48-9550*, March, 2017.

[B.1.8] CNSC Report, *Regulatory Oversight Report for Canadian Nuclear Power Plants: 2015*, October 2016.

[B.1.9] OPG Correspondence, P-CORR-00531-04917, *Pickering Quarterly Report on Safety Performance Indicators – Third Quarter of 2016*, December 2016.

[B.1.10] OPG Correspondence, P-CORR-00531-04830, *Pickering Quarterly Report on Safety Performance Indicators – Second Quarter of 2016*, September 2016.

[B.1.11] OPG Correspondence, P-CORR-00531-04769, *Pickering Quarterly Report on Safety Performance Indicators – First Quarter of 2016*, June 2016.

[B.1.12] OPG Correspondence, P-CORR-03680-0634906, *Radiation Protection Performance and Trends*, February 2017.

[B.1.13] OPG Report, P-REP-01966-0630362 R000, *Radiation Protection Department Performance Improvement Report*, January 2017.

[B.1.14] OPG Report, N-REP-03443-10015 R001, *2015 Results of Environmental Monitoring Program*, July 2016.

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ONTARIO POWER GENERATION	
ACCEPTED	✓
ACCEPTED AS NOTED	
REVISE AND RESUBMIT	
Signature	Date: 14 July 2016
Name: Mike Ruffolo, Manager	
Dept: Pickering Engineering - Aging Management & Strategic Initiatives	
OPG Proprietary	
Doc No.: P-REP-03680-00004	Rev: 000
This acceptance does not relieve the contractor from responsibility for errors or omissions or from any obligations or liability under this contract.	

Pickering NGS Periodic Safety Review 2: Code and Standard Reviews for Safety Factors 2 (Actual Condition of SSCs), 3 (Equipment Qualification) and 4 (Aging)

PS112/RP/010 R02

July 13, 2016

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Form 114 R26

Revision Summary - For Amec Foster Wheeler Report PS112/RP/010

Rev	Date	Author	Comments
R00	April 29, 2016	D. Vecchiarelli, A. Johnstone	Initial issue for OPG review and comment.
R01	May 26, 2016	A. Johnstone, S. Harvey	Updated report addressing OPG comments on R00 Report.
R02	July 13, 2016	A. Johnstone, S. Harvey	Updated report addressing OPG comments on R01 Report.

EXECUTIVE SUMMARY

OPG is performing a Periodic Safety Review (PSR) to support the possibility of continued operation of Pickering beyond 2020. The current PSR (referred to as "PSR2") is a subsequent PSR building on the review basis of earlier OPG Integrated Safety Reviews (ISRs) and other associated assessments. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

PSR2 will support and complement the licence renewal application for Pickering NGS. Fifteen Safety Factors will be assessed as part of the PSR. The objective of Safety Factor review reports is to document assessments against applicable modern Laws, Regulations, Codes and Standards (L/R/C/Ss), as well as PSR Review Tasks (as identified in the PSR2 Basis Document [1], and derived in Reference [2] based on CNSC REGDOC-2.3.3, *Periodic Safety Reviews* [3] and IAEA SSG-25, *Periodic Safety Review for Nuclear Power Plants* [4]), to confirm that the design, condition and operation of Pickering Unit 1,4, Unit 5-8, and common systems will support continued safe operation for the period of PSR2.

The process to identify the modern L/R/C/Ss that are applicable to the PSR2 Assessment Basis involved first creating a broad list from multiple sources (potential candidate L/R/C/Ss) and then filtering it to identify those that are most significant, and that are applicable to the PSR2 scope. The identification and selection criteria are defined in the PSR2 Basis Document [1]. The result of the identification and selection process was a set of modern L/R/C/Ss that became part of the PSR2 Assessment Basis. This Report provides the compliance reviews of L/R/C/Ss that are required to address PSR Safety Factors 2 (Actual Condition of SSCs), 3 (Equipment Qualification) and 4 (Aging). There is also some overlap with Safety Factor 1 (Plant Design) for a number of L/R/C/Ss considered, as per Table 1 in Section 2.0 of this Report.

The summary of findings is as follows:

- CSA N290.13-05, "Environmental Qualification of Equipment for CANDU Nuclear Power Plants": No gaps.
- CSA N285.4-14, "Periodic Inspection of CANDU Nuclear Power Plant Components": There are six gaps associated with Safety Factor 4.
- CSA N285.5-13, "Periodic Inspection of CANDU Nuclear Power Plant Containment Components": There are two gaps associated with Safety Factor 4.
- CSA N287.7-08, "In-Service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants": There are three gaps associated with Safety Factor 4.
- CNSC RD/GD-210, "Maintenance Programs for Nuclear Power Plants": No gaps.
- CNSC RD/GD-98, "Reliability Programs for Nuclear Power Plants": No gaps.

- CNSC REGDOC-2.6.3, "Aging Management": There are two gaps associated with Safety Factor 4.
- CSA N287.2-08, "Material Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants": No gaps.
- CSA N289.1-08, "General Requirements for Seismic Design and Qualification of CANDU Nuclear Power Plants": No gaps.
- CSA N289.2-10, "Ground Motion Determination for Seismic Qualification of Nuclear Power Plants": No gaps.
- CSA N289.3-10, "Design Procedures for Seismic Qualification of Nuclear Power Plants": There is one gap associated with Safety Factor 3.
- CSA N289.4-12, "Testing Procedures for Seismic Qualification of Nuclear Power Plants Structures, Systems, and Components": There is one gap associated with Safety Factor 3.
- CSA N289.5-12, "Seismic Instrumentation Requirements for Nuclear Power Plants and Nuclear Facilities": There is one gap associated with Safety Factor 3.
- CSA N285.8-15, "Technical Requirements for In-Service Evaluation of Zirconium Alloy Pressure Tubes in CANDU Reactors": There is one gap associated with Safety Factor 4.

Details of the assessment can be found in Table 2 and Appendix B of this report.

TABLE OF CONTENTS

1.0 INTRODUCTION

OPG is performing a Periodic Safety Review (PSR) in support of the evaluation of extended operation of the Pickering NGS units, beyond the year 2020, which is in accordance with the recent announcement by the government of the Province of Ontario.

The current PSR (referred to as "PSR2") is a subsequent PSR building on the review basis of earlier OPG Integrated Safety Reviews (ISRs) and other associated assessments (termed here "PSR1"). Specifically, PSR1 consists of:

- The Pickering B ISR, performed in support of refurbishment and continued operation of Pickering Units 5-8;
- Pickering 1, 4 integrated safety assessments performed during the Pickering A Return to Service (PARTS) work, in support of approval to restart Units 1 and 4; and
- The Darlington ISR, performed in support of refurbishment and continued operation of the Darlington units (programmatic parts applicable to Pickering).

PSR2 must satisfy the requirements in CNSC REGDOC-2.3.3, *Periodic Safety Reviews* [3], which in turn refers to International Atomic Energy Agency's (IAEA) Safety Guide No. SSG-25, *Periodic Safety Review for Nuclear Power Plants* [4]. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

The process to identify the modern Laws, Regulations, Codes and Standards (L/R/C/Ss) that are applicable to the PSR2 Assessment Basis involved first creating a broad list from multiple sources (potential candidate L/R/C/Ss) and then filtering it to identify those that are most significant, and that are applicable to the PSR2 scope. The identification and selection criteria are defined in the PSR2 Basis Document [1]. The result of the identification and selection process was a set of modern L/R/C/Ss that became part of the PSR2 Assessment Basis. The PSR2 Basis Document also identifies the modern version and date of the L/R/C/S and the type of review that will be completed in PSR2. The types of review are explained in Section 2.0 below.

This Report provides the compliance review of L/R/C/Ss that are required to address PSR Safety Factors 2 (Actual Condition of SSCs), 3 (Equipment Qualification) and 4 (Aging). The objectives of the reviews of these Safety Factors is as follows [1]:

- The objective of the review of Safety Factor 2 is to determine the actual condition of SSCs important to safety and so to consider whether they are capable and adequate to meet design requirements, throughout the period of

PSR2[1]. In addition, the review should verify that the condition of SSCs important to safety is properly documented, as well as reviewing the ongoing maintenance, surveillance and in-service inspection programmes, as applicable;

- The objective of the review of Safety Factor 3 is to determine whether plant equipment important to safety has been properly qualified (including for environmental conditions) and whether this qualification is being maintained through an adequate programme of maintenance, inspection and testing that provides confidence in the availability of safety functions throughout the period of PSR2; and

- The objective of the review of Safety Factor 4 is to determine whether aging aspects affecting SSCs important to safety are being effectively managed and whether an effective aging management program is in place so that all required safety functions will be available throughout the period of PSR2.

There is also overlap with Safety Factor 1 (Plant Design) for a number of L/R/C/Ss considered, as outlined in Table 1 in Section 2.0 below.

[1] Currently, Pickering Units 5-8 are approved to operate to 247,000 Effective Full Power Hours. This operation limit is expected to be reached on some units in 2020. For the purposes of PSR2, OPG assumes operation of Pickering NGS for up to eight additional years, from 2020 until 2028. OPG will make a decision regarding the permanent shut down dates for the six reactors following the performance of a technical evaluation that will include PSR2, and will communicate it to the CNSC by June 30, 2017 as required by the current Power Reactor Operating Licence.

2.0 REVIEW SCOPE AND METHODOLOGY

PSR2 is focused on the extension of Pickering NGS operations beyond 2020. Thus, it is important that the methodology for PSR2 be focused on addressing aspects of the review that are likely to have material impact in terms of identifying enhancements that will be reasonable and practicable to implement during the remaining commercial life of the plant. PSR2 will conduct reviews against a baseline of the PSR1 work. It is important to note that OPG conducts regular reviews of new and revised Codes and Standards, so a large amount of information is already available to assist in the Safety Factor compliance reviews. In OPG letter W.M. Elliott to P.A Webster and M. Santini, "Design Codes and Standards Effective Dates for OPG Nuclear Fleet" [5], OPG stated:

> *"...OPG commits to completing a code-over-code review (i.e., review of changes) of subsequent editions, addendum and/or updates of the Codes and Standards listed in [Attachment 1 of the referenced document]. Key emerging issues due to major changes in the codes will be addressed immediately, or as agreed with the CNSC on a case-by-case basis. Otherwise, OPG will confirm in a letter to the CNSC that these reviews have been completed and there are no significant technical issues..."*

As a result, many of the updated codes and standards issued since PSR1 have already had gap assessments performed, to varying degrees of detail, which will be utilized and cited in Pickering PSR2.

As a subsequent PSR, PSR2 focuses on changes in requirements, plant conditions, operating experience and new information, rather than repeating the activities of previous reviews. Since PSR2 is an update of previous ISRs, it incorporates reviews of L/R/C/Ss that have occurred as new versions have been issued. Therefore, clause-by-clause reviews of the majority of applicable L/R/C/Ss have already been completed and there is little value in repeating that process. If clause-by-clause reviews were to be undertaken in PSR2, a major portion of the review effort would be consumed by repackaging existing information that remains largely applicable and, therefore, is not contributing to the identification of new insights and enhancements. A more constructive approach has therefore been applied that maximizes the value and usefulness of the work by focusing attention where it is most beneficial, i.e., on identifying new issues. Since this assessment is a subsequent PSR, the focus is on identifying safety significant differences between what was previously assessed and what is now different within the current Pickering PSR2 Assessment Basis. In general, these differences relate to:

- More recent (new or revised) L/R/C/S versions than what was previously assessed;[2]
- Safety significant differences between Pickering and Darlington, if the Darlington ISR is the basis for the earlier L/R/C/S assessment;
- Implications of extending Pickering NGS operation beyond 2020; and
- Safety significant differences between Pickering Units 1,4 and Units 5-8.

In most cases PSR2 L/R/C/S reviews will be incremental in nature and performed by topic or subject matter for revised requirements. The rationale for this is that new or updated requirements that need to be included in PSR2 are predominantly replacements for other L/R/C/S that were previously assessed.

To align with the goals of a subsequent PSR, the following three tiers of reviews are applied for PSR2:

- <u>Clause-by-Clause review</u>: New L/R/C/Ss referenced in Pickering PROL 48.02/2018 (listed in Appendix C of the Licence Conditions Handbook) will be subjected to a clause-by-clause type review. In a clause-by-clause review, conformance with individual clauses is demonstrated by supporting evidence stating whether the requirements stipulated in the requirement document are met;
- <u>High Level review</u>: New L/R/C/Ss not referenced in Pickering PROL 48.02/2018 but which are in the PSR2 Assessment Basis will be subject to a high level review. In a high level review, the degree of conformance with clauses or groups of clauses in the Law, Regulation, Code or Standard is demonstrated by supporting evidence stating whether the intent of the requirements stipulated in the requirement document is met; and
- <u>Incremental review</u>: For L/R/C/Ss that have been reviewed in PSR1 but have had revisions since the last review, a topical review will be performed of the changes.

[2] "New" refers to a code or standard that was not previously considered in the context of earlier assessments. "Revised" refers to an updated version of a code or standard that was previously considered in the context of earlier assessments. Where a document has a new number/type, but addresses the same topic from the same organization, it is a "revised", not "new", document (e.g., if a REGDOC replaces a CNSC G or RD document).

Table 1 identifies the review type to be applied to each of the L/R/C/Ss in the PSR2 Assessment Basis that are applicable to Safety Factors 2, 3 and 4. Compliance assessments for each L/R/C/S are provided in Appendix B, and Results and Conclusions are summarized in Section 3.0.

Table 1: Applicable L/R/C/Ss for Pickering PSR2 Safety Factors 2, 3 and 4

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
L/R/C/Ss Referenced in Pickering NGS PROL 48.02/2018						
1	CSA N290.13	Environmental Qualification of Equipment for CANDU Nuclear Power Plants	N290.13-05	3, 4	Incremental	N290.13 addressed as part of Pickering B and Darlington ISRs
2	CSA N285.4	Periodic Inspection of CANDU Nuclear Power Plant Components	N285.4-14	1, 2, 4	Incremental	N285.4 addressed as part of Pickering B and Darlington ISRs
3	CSA N285.5	Periodic Inspection of CANDU Nuclear Power Plant Containment Components	N285.5-13	1, 2, 3, 4	Incremental	N285.5 addressed as part of Pickering B and Darlington ISRs
4	CSA N287.7	In-Service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants	N287.7-08	2, 3, 4	Incremental	N287.7 addressed as part of Pickering B and Darlington ISRs
5	CNSC RD/GD-210*	Maintenance Programs for Nuclear Power Plants	2012	3, 4	Incremental	S-210 and RD/GD-210 addressed as part of Darlington ISR
6	CNSC RD/GD-98	Reliability Programs for Nuclear Power Plants	2012	3, 4	Incremental	RD/GD-98 addressed as part of Darlington ISR and S-98 as part of Pickering B ISR
7	CNSC REGDOC-2.6.3*	Aging Management	2014	3, 4	Incremental	Transition plan in place and gap assessment between RD-334 and OPG Nuclear Integrated Aging Management governance has been performed
Additional L/R/C/Ss						
8	CSA N287.2	Material Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants	N287.2-08	1, 2, 3, 4	Incremental	N287.2 addressed as part of Pickering B and Darlington ISRs and PARTS

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
9	CSA N289.1	General Requirements for Seismic Design and Qualification of CANDU Nuclear Power Plants	N289.1-08	1, 3	Incremental	N289.1 addressed as part of Pickering B and Darlington ISRs
10	CSA N289.2	Ground Motion Determination for Seismic Qualification of Nuclear Power Plants	N289.2-10	1, 3	Incremental	N289.2 addressed as part of Pickering B and Darlington ISRs
11	CSA N289.3	Design Procedures for Seismic Qualification of Nuclear Power Plants	N289.3-10	1, 3	Incremental	N289.3 addressed as part of Pickering B and Darlington ISRs
12	CSA N289.4	Testing Procedures for Seismic Qualification of Nuclear Power Plants Structures, Systems, and Components	N289.4-12	1, 3	Incremental	N289.4 addressed as part of Pickering B and Darlington ISRs
13	CSA N289.5	Seismic Instrumentation Requirements for Nuclear Power Plants and Nuclear Facilities	N289.5-12	1, 3	Incremental	N289.5 addressed as part of Pickering B and Darlington ISRs
14	CSA N285.8	Technical Requirements for In-Service Evaluation of Zirconium Alloy Pressure Tubes in CANDU Reactors	N285.8-15	2, 4	Incremental	N285.8 addressed as part of Pickering B and Darlington ISRs

* Superseding documents to those currently in Pickering NGS PROL 48.02/2018.

L/R/C/Ss in the PSR2 Assessment Basis generally receive incremental reviews since PSR2 is an update of previous ISR assessments and clause-by-clause or high level reviews for the majority of the L/R/C/S in the PSR2 Assessment Basis have already been completed. Implementation plans (including gap analyses or code-over-code reviews) also exist for the latest editions of many L/R/C/Ss. As a result, incremental review is also used in circumstances where a L/R/C/S in the PSR2 Assessment Basis was not assessed in previous ISRs but an implementation plan currently exists for compliance.

The PSR2 incremental reviews in this Report include an assessment of the intent of recent changes to the L/R/C/Ss identified in Table 1 on a topic or subject-matter basis where there is potential to impact nuclear safety. The Incremental reviews provide:

- A summary of the purpose of the L/R/C/S;
- Pertinent background information about the current revision of the L/R/C/S that is being considered;

- Identification of which Safety Factor(s) are applicable to the current revision of the L/R/C/S;

- A description of which version(s) of the L/R/C/S were assessed for PSR1 (i.e., Darlington ISR (for programmatic content), Pickering B ISR and PARTS code reviews);

- Identification of whether the current version of the L/R/C/S is an update of a previous version of the L/R/C/S that was assessed in PSR1 (and if so, a description of the major changes in the latest revision is provided as discussed below);

- An assessment of the applicability of PSR1 assessment findings (gaps and conclusions), including the implications of extending Pickering NGS operation beyond 2020 if any;

- An assessment of the applicability of assessment findings that address more recent (post-PSR1) editions of the L/R/C/S, including any implementation or transition plans that are already committed to by OPG; and

- Where PSR1 and post-PSR1 assessments are not sufficient to address changes in the latest edition of the L/R/C/S, an assessment of the changes from the previously assessed edition of the L/R/C/S (including identification of any safety significant PSR2 gaps which result).

The Safety Factor 2, 3 and 4 L/R/C/S reviews will identify Compliances and Gaps as defined below:

- <u>Compliance</u>: Where an Incremental review has been performed, Compliance indicates that the change in the safety requirement, per the topical review, is met.

- <u>Gap</u>: Where an Incremental review has been performed, a Gap indicates that the change in the safety requirement, per the topical review, is not met.

The assessments documented in this Report assume that use of the word "shall" is used in an L/R/C/S to express a requirement, i.e., a provision that the licensee is obliged to satisfy in order to comply with the standard, "should" is used to express a recommendation or that which is advised but not required, "may" is used to express an option or that which is permissible within the limits of the standard, and "can" is used to express possibility or capability.

3.0 RESULTS AND CONCLUSIONS

The results of the PSR2 compliance assessments of the fourteen L/R/C/Ss listed in Table 1 are summarized in Table 2 below. Additional background information and details regarding the gaps listed in Table 2 are provided in Appendix B of this report.

Table 2: PSR2 L/R/C/S Compliance Assessment Results for Safety Factors 2, 3 and 4

L/R/C/S Reviewed	PSR2 Compliance Assessment
N290.13-05, "Environmental Qualification of Equipment for CANDU Nuclear Power Plants"	There are no PSR2 gaps for N290.13-05. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with N290.13-05.
N285.4-14, "Periodic Inspection of CANDU Nuclear Power Plant Components"	There are six PSR2 CSA N285.4 gaps which all relate to Safety Factor 4. The first five of these N285.4 gaps are applicable to compliance with N285.4-09 including Updates 1 and 2. The sixth N285.4 gap is applicable to compliance with N285.4-14. 1. N285.4 PIP Governance references N285.4-05, not N285.4-09 including Updates 1 and 2. This (programmatic) Darlington gap is a PSR2 gap against N285.4-09 including Updates 1 and 2. 2. There has been a significant change in the wording of clause 4.2.7 in CSA N285.4-09 including Updates 1 and 2. I-PROC-AS-0009, "Inspection Qualification of Non-Destructive Examination Processes" does not identify the authorized inspector as a qualifying authority as directed by clause 4.2.7. Instead it establishes the CANDU Inspection Qualification Bureau (CIQB) as the organization that would approve procedures and personnel. This (programmatic) Darlington gap is a PSR2 gap against N285.4-09 including Updates 1 and 2. 3. New erosion and corrosion inspection requirements in N285.4-09 including Updates 1 and 2 are not reflected in current PIP governance. NK38-REP-03680-10137 R000 states that: "It should be noted specifically that [this ISR Issue] is likely to have a major impact on piping PIPs because sub-clauses 7.4.7.X in CSA N285.4-09 including UPD1 and UPD2 include substantive changes. Under the new standard erosion and corrosion inspection exemptions can no longer be justified on the basis of [sic] that conditions are determined to be non-erosive and non-corrosive." This Darlington PIP gap will also need to be addressed in the Pickering PIPs. Therefore, this is a PSR2 gap against N285.4-09 including Updates 1 and 2. 4. Extended life inspection schedules in N285.4-09 including Updates 1 and 2 are not reflected in PIP governance. This (programmatic) Darlington gap is a PSR2 gap against N285.4-09 including Updates 1 and 2.

L/R/C/S Reviewed	PSR2 Compliance Assessment
	5. An assessment of the prior operating non-conforming state, as required by N285.4-09 including Updates 1 and 2, is required when dispositioning inspection results. This requirement has not been included in the feeder PIP plan. This Darlington PIP gap will also need to be addressed in the Pickering PIPs. Therefore, this is a PSR2 gap against N285.4-09 including Updates 1 and 2. 6. There is a PSR2 gap for Pickering NGS against N285.4-14 to address: o Revised requirements for pressure tube volumetric and dimensional inspection (Clause 12.2), pressure tube hydrogen equivalent determination (Clause 12.3) and pressure tube material property testing (Clause 12.4); o Clause 12.5 which specifies minimum annulus spacer surveillance examination and testing requirements; o Selection criteria for identifying candidate tube for pressure tube surveillance examination and testing (Annex E) to include selection criteria for annulus spacer surveillance examination and testing; and o Clause 7.4.8 which specifies requirements for inspection of Environmentally Assisted Cracking, and Clauses 7.5.1/7.5.2 which specify requirements for inspection of identical components.
N285.5-13, "Periodic Inspection of CANDU Nuclear Power Plant Containment Components"	There are two PSR2 CSA N285.5 gaps which both relate to Safety Factor 4: 1. There were a number of concessions granted from the CNSC for compliance with N285.5-M90 that will need to be reconciled for Pickering for the period of PSR2: (Since these gaps are all concession-related and associated with N285.5-M90, they are tracked under a single PSR2 gap.) o The Pickering B ISR gap associated with N285.5-M90 clause 4.5.1 is closed. However, the disposition of the gap refers to OPG receiving a concession from the CNSC on the inspection of components deemed to be inaccessible. A similar (updated) concession may be required for Pickering operation past 2020. Therefore, this is a gap for PSR2. o The Darlington ISR disposition of the gaps for N285.5-M90 clauses 8.4.2.1 and 8.4.2.2 refer to OPG receiving a concession from the CNSC that insulation will not be removed in the absence of visible damage to a component, and only "light weight" access covers will be removed. The Darlington ISR states: "This is a concession from the regulator which is not assured in the case of a refurbished plant. As such, this represents a gap". By the same logic it will need to be reconciled for Pickering for the period of PSR2 (life extension past 2020). o The Darlington ISR disposition of the gap for N285.5-M90 for clause 8.5.2.2 refers to an exception of the numerical rules of this clause for reasons of practicality, and that a concession was received from the

L/R/C/S Reviewed	PSR2 Compliance Assessment
	CNSC. The Darlington ISR stated "… it is categorized as a Gap, because a concession from the CNSC is not assured for a refurbished plant.". By the same logic it will need to be reconciled for Pickering for the period of PSR2. o Per the Darlington ISR disposition of the gap for N285.5-M90 clause 8.6.3, although CNSC acceptance was obtained, there is still a non-compliance with a portion of the clause related to the timing of inspections which is noted as needing to be reconciled for a refurbished station. The Darlington ISR stated "This represents a gap that will need to be reconciled with the regulator for a refurbished station." By the same logic it will need to be reconciled for Pickering for the period of PSR2. 2. The changes in N285.5-13 relative to N285.5-08 that are applicable to Fiberglass Reinforced Plastic material that is used at Pickering NGS have only been assessed for fitness for service to 2024 in the Pickering Continued Operations Plan. These changes related to aging management (monitoring and test programs) for FRP materials. As a result, additional assessment is required for Pickering to address FRP aging management at Pickering for operation to 2028, and to confirm the current program aligns with N285.5-13 clauses 8.2, 8.3.3, 8.3.4 and A.6.1.2. (Note: This gap only exists if Pickering NGS intends to operate past 2024.)
N287.7-08, "In-Service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants"	There are three PSR2 CSA N287.7 gaps which all relate to Safety Factor 4: 1. N287.7-08 clause 7.11.2 Table 1 involving non-compliance with accuracy and repeatability requirements for dewpoint temperature was a gap for Darlington. No evidence can be found that this has been addressed for Pickering NGS. This is therefore a gap for Pickering PSR2. 2. OPG initiated a Regulatory Management action to provide the CNSC with the latest Dow Corning 995 material test report in response to an Action Notice raised in the CNSC Type II Inspection. The work is currently in progress. Therefore, this is a gap for Pickering PSR2. 3. Actions #31, #32, and #33 from the Pickering Units 5-8 Continued Operations Plan are related to N287.7 and although complete, need to be re-assessed for Pickering operation past 2020. (IIP Action #31 involved submission of Periodic Inspection Plans and Life Cycle Management Plans for a number of safety-significant civil structures. IIP Action #32 involved submission of Aging Management Plans for concrete containment structures to the CNSC for acceptance. IIP Action #33 involved revising the Reactor Building Periodic Inspection Plan and submitting to the CNSC for acceptance.)
CNSC RD/GD-210, "Maintenance Programs for Nuclear Power Plants"	There are no PSR2 gaps for CNSC RD/GD-210. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with RD/GD-210.

L/R/C/S Reviewed	PSR2 Compliance Assessment
CNSC RD/GD-98, "Reliability Programs for Nuclear Power Plants"	There are no PSR2 gaps for CNSC RD/GD-98. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with RD/GD-98.
CNSC REGDOC-2.6.3, "Aging Management"	There are two PSR2 REGDOC-2.6.3 gaps which both relate to Safety Factor 4: 1. OPG is not compliant with N-PROC-MP-0060 Aging Management Process, Section 1.7 for "not reviewing and updating the Component Condition Assessments[3] within the review cycle of the component, and when new information or feedback from the program was received." OPG has since revised these CAs, which are now valid until 2020. OPG has stated they will develop an implementation plan to prevent reoccurrence of: a) not reviewing and revising the CAs within the review cycle, and b) not updating the CAs when pertinent new information becomes available. OPG stated they will provide an update and a target implementation date on this action to the CNSC by October 30, 2016. This is a gap for Pickering PSR2. 2. OPG is not compliant with N-PROC-MA-0077, "Critical Equipment Identification and Categorization", Section 1.2 because "the Reactor Safety (RS) category code and rationale for critical components was not always accurate or consistently applied in the CCAs[3]." OPG has stated they have since completed a review and update of the RS category code and rationale for a portion of the components to become fully compliant with N-PROC-MA-0077. However, OPG has stated that a review of the CAs will be conducted to ensure consistency with the revised Reactor Safety codes and that an update will be provided to the CNSC by October 30, 2016. This is a gap for Pickering PSR2.
N287.2-08, "Material Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants"	There are no PSR2 gaps for CSA N287.2-08. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with N287.2-08.
N289.1-08, "General Requirements for Seismic Design and Qualification of CANDU Nuclear Power Plants"	There are no PSR2 gaps for CSA N289.1-08. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with N289.1-08.
N289.2-10, "Ground Motion Determination for	There are no PSR2 gaps for CSA N289.2-10. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with N289.2-10.

[3] The terminology currently used is Condition Assessment (CA) instead of Component Condition Assessment (CCA).

L/R/C/S Reviewed	PSR2 Compliance Assessment
Seismic Qualification of Nuclear Power Plants”	
N289.3-10, “Design Procedures for Seismic Qualification of Nuclear Power Plants”	There is one PSR2 gap for Pickering NGS compliance with N289.3-10 which is applicable to Safety Factor 3: 1. Clause 4.4.4.5 of CSA N289.3-10 states: “The power spectral density (PSD) function of each time-history shall be calculated and shown to not have any significant gaps in energy over the frequency intervals outlined in Table 2….” The calculation of PSD is not addressed in the Pickering A or B PRA Based SMAs. The Pickering NGS A PRA Seismic Guide and the OPG PRA Guide do not identify any requirements for PSD. Also, evidence in the form of a calculation for time histories which represent the design ground motion was not found (which is a precursor for the PSD calculation). The lack of evidence of calculated time histories was also identified as a gap in the Darlington ISR (ISR Issues #D352 and #D617 - Documented evidence in the form of a calculation to show that the generated time history correctly represents the design ground response spectrum within the prescribed requirements has not been provided). The closure reference for #D352 and #D617 makes use of the detailed assessment performed in NK38-REP-03680-10224 R000 which is specific to Darlington. A similar assessment for Pickering NGS could not be found. As a result, there is a gap for PSR2 to provide similar evidence to show that: a) the generated time history used within seismic analyses of safety-related systems correctly represents the design ground response spectrum for the Pickering site in compliance with N289.3-10, and b) the PSD function of each time-history has been calculated and shown to not have any significant gaps in energy over the frequency intervals.
N289.4-12, “Testing Procedures for Seismic Qualification of Nuclear Power Plants Structures, Systems, and Components”	There is one PSR2 gap for Pickering NGS compliance with N289.4-12 which is applicable to Safety Factor 3: 1. Station-specific documents (including the Darlington seismic design guide, Darlington Reports and Darlington-specific technical specifications for seismic qualification) were used as the basis for compliance in the clause-by-clause Darlington code refresh review for clauses 4.2.1, 4.2.2.2, 4.2.3.1, 4.2.5, 4.3.2, 5.2.2.2.5, 5.7, 5.8.1, 5.8.1.2, 7.2.1, 7.7.1, 7.7.4 and 8.2. Pickering-specific seismic design guides, reports and technical specifications that are equivalent to those used to demonstrate Darlington compliance with the changes made in CSA N289.4-12 were identified. However, a detailed review to confirm that the Pickering-specific documents fully comply with the requirements of the clauses listed above is needed. As a result, this is a PSR2 gap.
N289.5-12, “Seismic Instrumentation Requirements for Nuclear Power Plants and Nuclear Facilities”	There is one PSR2 gap for Pickering NGS compliance with N289.5-12 which is applicable to Safety Factor 3: 1. Darlington ISR Issues #D622, D623 and D624 require no further action for Darlington as they were either classified as Acceptable Deviations or were closed. However, the issues are identified as a PSR2 gap for the following

L/R/C/S Reviewed	PSR2 Compliance Assessment
	reasons: (Note: These gaps are closely related and are therefore identified as a single PSR2 gap.) ○ Darlington ISR Issue #624 refers to specific Darlington instrumentation in order to classify the gaps as Acceptable Deviations. It must be demonstrated that Pickering seismic instruments have the same capabilities as the Darlington instruments (fleet-wide or Pickering-specific standards that would ensure that the Pickering seismic instruments have the same capabilities as the Darlington instruments could not be found). Therefore, this is identified as a gap for PSR2. ○ Darlington ISR Issue #D622 was deemed to be of low safety significance. The same rationale may apply at Pickering, but first it must be demonstrated that Pickering has the same set up of seismic instruments as Unit 0 at Darlington. Therefore, this is identified as a gap for PSR2. ○ Darlington ISR Issue #D623 was deemed to be of low safety significance. The same rationale may apply at Pickering, but first it must be demonstrated that similar accelerometers are used at Pickering, and that their locations are not affected by strong ambient vibration. Therefore, this is identified as a gap for PSR2.
N285.8-15, "Technical Requirements for In-Service Evaluation of Zirconium Alloy Pressure Tubes in CANDU Reactors"	There is one PSR2 gap for Pickering NGS compliance with N285.8-15 which is applicable to Safety Factor 4: 1. For the Pickering B ISR, no clause-by-clause review of the Standard was conducted on the basis that the pressure tubes will be replaced during the refurbishment outage for Pickering Units 5-8, and the condition of these components is well understood and managed through their own specific, detailed life cycle plans and fitness-for-service criteria. However, in November 2015, OPG issued Plan N-REP-31100-10061 R002 for Pickering NGS compliance with pressure tube in-service evaluation requirements in CSA N285.8-15. OPG had submitted a previous compliance plan for the long term use of the 2010 edition of CSA N285.8 and this compliance plan was accepted by the CNSC. The compliance plan was revised to document OPG's compliance to the 2015 edition of CSA N285.8. Since OPG has committed to fulfillment of the commitments in N-REP-31100-10061 R002, successful fulfillment by OPG of the commitments in the compliance plan is required for Pickering operation past 2020. This is therefore a gap for Pickering PSR2. In particular, the significant changes to CSA N285.8-15 per the CSA Impact Statement will need to be reflected in Pickering procedures, including: ○ Implementation of statistically based fatigue crack initiation evaluation curves for axial flaws (Clauses D.4.2, D.4.3, and D.3.6);

L/R/C/S Reviewed	PSR2 Compliance Assessment
	○ Implementation of closed-form engineering relation for threshold peak stress for Delayed Hydride Cracking (DHC) initiation (Clauses D.5 and 5.4.3.4); ○ Implementation of statistically based threshold relation for peak stress for crack initiation due to hydrided region overloads (Clause D.5); ○ Implementation of new fracture toughness models for axial through-wall flaws (Clause D.13.2); and ○ Implementation of Methods 1 and 2 Probabilistic Leak-Before-Break (Clauses 3.1, 7.3 and 7.4).

4.0 REFERENCES

1. OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

2. OPG Report, P-REP-03680-00003 R000, *Pickering NGS PSR2: Definition of Safety Factor Review Tasks*, May 30, 2016.

3. CNSC REGDOC-2.3.3, "Periodic Safety Reviews", April 2015.

4. IAEA Safety Guide No. SSG-25, "Periodic Safety Review for Nuclear Power Plants", 2013.

5. OPG Letter N-CORR-00531-05661, W. M. Elliott to P. A. Webster and M. Santini, *Design Codes and Standards Effective Dates for OPG Nuclear Fleet*, April 30, 2012.

APPENDIX A: NOMENCLATURE

AD	Acceptable Deviation
AECB	Atomic Energy Control Board
AECL	Atomic Energy of Canada Limited
AM	Aging Management
AMP	Aging Management Plan
AN	Action Notice
ASME	American Society of Mechanical Engineers
CANDU	CANada Deuterium Uranium
CAP	Corrective Action Plan
CA	Condition Assessment
CCSs	Concrete Containment Structure
CHR	Component Health Report
CNSC	Canadian Nuclear Safety Commission
CSA	Canadian Standards Association
CT	Calandria Tube
DBE	Design Basis Earthquake
DCR	Document Change Request
DEC	Design Extension Conditions
DNGS	Darlington Nuclear Generating Station
EQ	Environmental Qualification
IAEA	International Atomic Energy Agency
IAM	Integrated Aging Management
IIP	Integrated Implementation Plan
IRF	Issue Resolution Form
ISR	Integrated Safety Review
LCH	Licence Conditions Handbook
LCMP	Life Cycle Management Plan
L/R/C/S	Laws, Regulations, Codes and Standards
MCR	Main Control Room
NBC	National Building Code

NGS	Nuclear Generating Station
OEAD	Operational Engineering Assessment Division
OPG	Ontario Power Generation
PARTS	Pickering A Return to Service
PIP	Periodic Inspection Program
PMID	Predefined Maintenance Identification
PNGS	Pickering Nuclear Generating Station
PRA	Probabilistic Risk Assessment
PROL	Power Reactor Operating Licence
PSHA	Probabilistic Seismic Hazard Assessment
PSR	Periodic Safety Review
PSR1	Periodic Safety Review 1 (earlier OPG PSR work and other associated assessments)
PSR2	Periodic Safety Review 2 (subsequent PSR per CNSC REGDOC-2.3.3)
PT	Pressure Tube
REGM	Regulatory Management Action Request
ROV	Remotely Operated Vehicles
RS	Reactor Safety
SATM	Space Allocation and Transient Material
SCR	Station Condition Record
SF	Safety Factor
SHR	System Health Reports
SOE	Safe Operating Envelope
SPRA	Seismic Probabilistic Risk Assessment
SSCs	Structures, Systems and Components
VBO	Vacuum Building Outage

APPENDIX B: L/R/C/S REVIEWS FOR SAFETY FACTORS 2, 3 AND 4

B.1 CSA N290.13-05, "Environmental Qualification of Equipment for CANDU Nuclear Power Plants"

B.1.1 Background

The following paraphrased from the Preface of CSA N290.13-05 (R2015) [B.1-1] provides a brief overview of the purpose of this Standard and the requirements expressed therein:

> *The requirements for environmental qualification (EQ) of safety-related equipment in CANDU nuclear power plants are specified in the CSA N290 series of Standards. General quality assurance requirements for the life cycle of these plants are specified in the CSA N286 series of Standards. Standard N290.13-05 is supplemental to these two series of Standards.*
>
> *CSA N290.13 has been written as a general Standard for the establishment and maintenance of an environmental qualification program for safety-related equipment in the CANDU nuclear power plants that are within its scope. It provides generic requirements and methods for such qualification...*
>
> *Safety-related equipment that must be environmentally qualified has to meet functional safety requirements throughout its installed life. This requires a program of quality assurance, design qualification, production, transportation, storage, installation, maintenance, periodic testing, and surveillance. This Standard provides guidelines for the establishment of an ongoing program. Although it focuses on the qualification process, it also provides requirements for maintaining qualification once it has been established.*

All of CSA N290.13-05 (R2015) is directly relevant to Safety Factor 3 (Equipment Qualification). CSA N290.13-05 (R2015) also addresses the effects of aging on equipment required to mitigate the effects of a Design Basis Accident. The following clauses apply to Safety Factor 4 (Aging):

- Clause 6 - Preserving environmental qualification.
- Annex A - Arrhenius methodology in predicting material aging.

Compliance with CSA N290.13-05 including Update 1 (R2010) [B.1-2] is currently a licence requirement for Pickering NGS (per PROL 48.02/2018) as indicated in Appendix C.1 of the R04 Pickering Licence Conditions Handbook [B.1-3]. The current version of the Standard, N290.13-05 (R2015) [B.1-1] is a reaffirmation of the update introduced in 2009 [B.1-2] without any content changes.

The Impact Statement [B.1-4] for CSA N290.13-05 including Update 1 [B.1-2] provides a "Summary of significant changes from the previous edition" which identifies four changes to the Standard which are discussed in Section B.1.2 below and are not safety significant. In addition to findings resulting from review of the CSA N290.13-05 including Update 1 Impact Statement, the results of PSR1 N290.13 reviews (Pickering B and Darlington ISRs), as well as reviews performed since PSR1, have also been assessed for applicability to PSR2 in Section B.1.2.

As identified in [B.1-5], the Pickering PSR2 review of CSA N290.13-05 (R2015) is an Incremental Review. As discussed in Section 2.0 of this Report, PSR2 Incremental Review includes an assessment of the intent of recent changes to the L/R/C/S on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.
- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.1.2 Compliance Assessment for Pickering PSR2

B.1.2.1 Application of PSR1 Reviews

The versions of N290.13 subject to previous reviews conducted for Darlington and Pickering, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

A clause-by-clause review of the Pickering B Environmental Qualification program against the CSA N290.13-05 [B.1-6] version of the Standard was documented in NK30-REP-03680-00003 R000 [B.1-7]. Two Acceptable Deviations (ADs) were identified which resulted from differences in the wording used in applicable EQ Program documentation. The intent of the wording was concluded to be the same in both circumstances. These ADs were considered acceptable as they have no adverse impact on the Environmental Qualification program or on plant safety. As a result, NK30-REP-03680-00003 R000 concluded that the Pickering NGS B Environmental Qualification program complies with each clause of CSA N290.13-05 [B.1-6]. None of the ADs are impacted by Pickering operation past 2020, so there are no PSR2 gaps which result from the Pickering B ISR.

Pickering Units 1,4

No code review was performed for Pickering Units 1,4 against CSA N290.13. Also, no subsequent code reviews or code refresh reviews against the updated versions of N290.13 (i.e., [B.1-1] or [B.1-2]) were performed for either Pickering NGS A or B. However, Section 6.3 of the R04 Pickering Licence Conditions Handbook [B.1-3] states:

> *As per the agreement reached in CNSC letter dated June 22, 2012 (e-Doc 3947068) a number of design-related codes and standards, associated effective dates and conditions were established, including application of CSA N290.13-05. OPG is to provide the CNSC with the code-over-code reviews conducted for any subsequent editions, addendums and/or updates of the CSA N290.13-05 and Update No.1, with OPG's assessment of the changes and their significance upon completion of the review and assessment of significance. OPG is to submit such assessments on an annual basis. Refer to Appendix G, for additional details.*

Further, Section 5.3 of the Pickering PROL Application P-CORR-00531-03719 R000 [B.1-8] states:

> *OPG has in place an Environmental Qualification (EQ) Program, including the governing documents that systematically identify the equipment to be qualified, the environmental conditions to be used for qualification, the method(s) of qualification, and the documentation to ensure that the qualification is complete and can be maintained for the remaining life of the plant. The OPG EQ Program is compliant with the Canadian Standard Association CSA N290.13-05 as per the governing program document N-PROG-RA-0006, Environmental Qualification. All required systems, equipment, components, protective barriers, and structures were qualified to perform their safety functions under the environmental conditions defined by the Pickering design-basis accidents.*
>
> *The overall condition of the current EQ program at Pickering is acceptable. This assessment is based on weighted indicators that are consistent with industry-best practices.*

Since compliance with CSA N290.13-05 including Update 1 (R2010) is currently a licence requirement for Pickering NGS (per PROL 48.02/2018) as indicated in Appendix C.1 of the R04 Pickering Licence Conditions Handbook [B.1-3], and this is confirmed by review of the Pickering PROL Application, Pickering Units 1,4 (and Units 5-8) are also in compliance with N290.13-05 including Update 1 [B.1-2]. Further, assessments have been performed for Darlington which are programmatically applicable to Pickering, and which are used below to demonstrate Pickering NGS compliance with CSA N290.13-05 (R2015) [B.1-1].

Darlington NGS

The Darlington ISR performed assessments against two versions of CSA N290.13:

1) The original code review for the Darlington ISR, as documented in OPG Report NK38-REP-03680-10093 R000 [B.1-9], was performed against CSA N290.13-05 [B.1-6]. NK38-REP-03680-10093 R000 identified 8 gaps with the requirements in CSA N290.13-05 (February 2005) [B.1-6]. However, Appendix B of NK38-REP-03680-10093 R000 identified that OPG has completed a project to ensure program governance compliance with N290.13-05, including Update 1 [B.1-2], and that consequently those 8 gaps had been resolved as of the date that NK38-REP-03680-10093 R000 was issued in September 2011.

2) Following the issue of CSA N290.13-05 including Update 1 [B.1-2], a code refresh review was conducted and documented in OPG Report NK38-REP-03680-10154 R000 [B.1-10]. NK38-REP-03680-10154 R000 [B.1-10] identified that the changes to CSA N290.13-05 including Update 1 (R2010) [B.1-2] had no impact on the program governance compliance assessment and that as a result, OPG Nuclear governance continues to be compliant with the revised requirements of CSA N290.13-05 including Update 1 [B.1-2].

Therefore, there are no outstanding gaps for Darlington compliance with CSA N290.13-05 including Update 1 [B.1-2]. Since the current version of the Standard, N290.13-05 (R2015) [B.1-1] is a reaffirmation of the update introduced in 2009 [B.1-2] without any content changes, Darlington also complies with [B.1-1]. Further, as discussed in Section B.1.2.2 below, since all N290.13-05 requirements are programmatic (i.e., apply across OPG's Nuclear fleet operations), Pickering NGS (Units 1,4 and Units 5-8) also fully complies with N290.13-05 (R2015) [B.1-1].

B.1.2.2 Application of Post-PSR1 Reviews

As discussed above, CSA N290.13 was updated in 2009. The changes relative to CSA-N290.13-05 [B.1-6] included minor changes, mostly for clarification purposes. Information about the changes to the Standard can be found in the CSA N290.13-05 Impact Statement [B.1-4] for Update 1 [B.1-2]. The changes in Update 1 are primarily editorial in nature and do not trigger any new requirements. The only content-related change, which does not trigger new requirements and is not safety significant, is a revision to Clause 4.6, "Documentation of EQ Requirements". The revision adds the option of providing a reference source for the required equipment attributes instead of providing the necessary information as part of the mandatory list of equipment (revised text is underlined):

Based upon the identification of EQ requirements, a list of equipment shall be prepared. The list shall contain the following attributes for each piece of equipment or shall reference the source of such information:

(a) equipment identification (e.g., make, model number, manufacturer, stock identification, plant system, installed location, and relevant interfaces);

(b) safety function;

(c) applicable design basis accident;

(d) mission times; and

(e) normal and accident service conditions.

Since the current version of the Standard, N290.13-05 (R2015) [B.1-1] is a reaffirmation of the update introduced in 2009 [B.1-2], Darlington is compliant with the latest version of the Standard (N290.13-05 (R2015)).

With respect to Pickering, in 2015 OPG authorized an independent external assessment of the Environmental Qualification program at both Pickering and Darlington NGSs to ensure they were functioning acceptably as indicated in the current health reports. This assessment was documented in N-REP-03651-0564226 [B.1-11]. With specific reference to OPG's EQ governance, N-REP-03651-0564226 states the following in Section 4.1, "Program Governance":

> *Given that OPG's EQ program governance is written at the "N" level rather than at station specific level, the EQ program governance findings for the DNGS EQ program are equally valid for the PNGS EQ program.*

Therefore, since Darlington is compliant with the latest version of the Standard, and since all N290.13-05 requirements are programmatic (i.e., apply across OPG's Nuclear fleet operations), the EQ program at Pickering NGS (Units 1,4 and Units 5-8) also fully complies with N290.13-05 (R2015).[4]

Further, given that there are no technical differences between N290.13-05 including Update 1 and N290.13-05 (R2015), evidence that the requirements of the current version of N290.13

[4] N-REP-03651-0564226 [B.1-11] concluded that there has been a significant improvement in the Environmental Qualification Program at OPG. The Conclusions section of the report (Section 6) identified that findings from internal and external assessments have been resolved, but that some process improvements, if implemented, would further improve management oversight capability, while enhancing the sustainability of the EQ program. The improvement opportunities do not relate to compliance with the requirements stipulated in CSA N290.13 and are addressed separately in the Safety Factor Report for Equipment Qualification.

(N290.13-05 (R2015)) are being applied at OPG is provided by demonstrating that N290.13-05 including Update 1 are being applied in the following OPG governance:

1) OPG Engineering Standard N-STQ-03651-10013 R003, "Qualification Methods" [B.1-12], which provides:

   … guidelines on the qualification methods to be used in establishing environmental qualification (EQ). This standard is applicable to all electrical and mechanical equipment requiring environmental qualification.

   According to CSA Standard N290.13-05 and Update No. 1, fully sequential type testing is the preferred method of qualification, however, operating experience, analysis, ongoing qualification or a combination of these methods may be used. This standard provides guidance on the use of all these methods at Ontario Power Generation (OPG).

2) N-PROG-RA-0006 R008, "Environmental Qualification" [B.1-13], the Nuclear Program OPG uses, which states in Section 1.1.1 that the technical basis for the program is CSA Standard N290.13-05 including Update 1.

B.1.3 Compliance Summary for Pickering PSR2

There are no PSR2 gaps for N290.13-05 (R2015) [B.1-1]. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with N290.13-05 (R2015).

B.1.4 References

[B.1-1] CSA Standard N290.13-05 (R2015), *Environmental Qualification of Equipment for CANDU Nuclear Power Plants*, 2005; Update No. 1: October 2009.

[B.1-2] CSA Standard N290.13-05 (R2010), *Environmental Qualification of Equipment for CANDU Nuclear Power Plants,* 2005; Update No. 1: October 2009.

[B.1-3] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.1-4] CSA Impact Statement and Publication Notice, *Product: Amendment - Product Designation: N290.13-05 – Product title: Environmental Qualification of Equipment for CANDU Nuclear Power Plants – Date of Release: October 2009*, Date not provided.

[B.1-5] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.1-6] CSA Standard N290.13-05 (February 2005), *Environmental Qualification of Equipment for CANDU Nuclear Power Plants*, February 2005.

[B.1-7] OPG Report, NK30-REP-03680-00003 R000, *Pickering NGS-B Integrated Safety Review - Safety Factor for Equipment Qualification*, April 2007.

[B.1-8] OPG Letter, P-CORR-00531-03719 R000, G. Jager to M. A. Leblanc, *Application for Renewal of Pickering Nuclear Generating Station Power Reactor Operating Licence*, July 4, 2012.

[B.1-9] OPG Report, NK38-REP-03680-10093 R000, *Review of CAN/CSA-N290.13-05 (February 2005), Environmental Qualification of Equipment for CANDU Nuclear Power Plants for Darlington Integrated Safety Review*, September 2011.

[B.1-10] OPG Report, NK38-REP-03680-10154 R000, *Code Refresh Review of CSA-N290.13-05 (R2010) For DNGS ISR*, July 2013.

[B.1-11] OPG Report, N-REP-03651-0564226, *Assessment of OPG EQ Program*, September 2015.

[B.1-12] OPG Engineering Standard, N-STQ-03651-10013 R003, *Qualification Methods*, Implementation Date: October 20, 2010; Review Date: October 20, 2015.

[B.1-13] OPG Nuclear Program, N-PROG-RA-0006 R008, *Environmental Qualification*, May 2015.

B.2 CSA N285.4-14, "Periodic Inspection of CANDU Nuclear Power Plant Components"

B.2.1 Background

The following paraphrased from the Preface of CSA N285.4-14 [B.2-1] provides an overview of the purpose of this Standard and the requirements expressed therein:

> *This Standard defines requirements for the periodic inspection of pressure retaining systems, components, and supports that form part of a CANDU nuclear power plant. Periodic inspection is considered to include the fluid boundary portions of components and piping, including their supports that comprise:*
>
> *a) systems containing fluid that directly transports heat from nuclear fuel;*
> *b) systems essential for the safe shutdown of the reactor or the safe cooling of the fuel, or both, in the event of a process system failure; and*
> *c) other systems or components whose failure could jeopardize the integrity of the systems described in Item a) or b), or both.*
>
> *In addition, for components exposed to conditions beyond the known experience base, and where such components constitute part of a vital system, the components may be considered suitable for inclusion in the periodic inspection program, as supplementary inspections.*
>
> *This Standard addresses:*
>
> *a) failure aspects;*
> *b) classification of areas subject to inspection;*
> *c) provision for access;*
> *d) inspection techniques and procedures;*
> *e) personnel qualifications;*
> *f) frequency of inspection;*
> *g) responsibilities;*
> *h) documentation;*
> *i) records;*
> *j) evaluation of inspection results;*
> *k) dispositioning; and*
> *l) repair, replacement, and modification requirements.*

N285.4-14 is relevant to Safety Factor 2 (Actual Condition of SSCs). As stated in OPG Report NK30-REP-03680-00001 R000 [B.2-2], even though N285.4 applies to periodic inspection rather than to the design of the plant, there are elements of this Standard that have implications on plant design, and therefore N285.4 is also considered to be relevant to Safety Factor 1 (Plant Design). A number of clauses in N285.4-14 are also relevant to Safety Factor 4 (Aging), namely:

- Clause 7.4.7.3: Determination of wall-thinning susceptibility (specifically clauses 7.4.7.3.1, 7.4.7.3.3, and 7.4.7.3.4);
- Clause 7.4.7.4: Inspection requirements;
- Clause 7.4.7.5: Inspection area (specifically clauses 7.4.7.5.1, 7.4.7.5.2, and 7.4.7.5.3);
- Clause 8.2.1: Acceptance criteria - General;
- Clause 8.2.5: Dimensional inspection;
- Clause 12.4: Material property testing;
- Clause 12.5: Material surveillance of fuel channel annulus spacers;
- Clause 14.4.1.1: General (this clause documents the general requirements for the metallurgical examination of Steam Generator tubes); and
- Annex E: Guidance on the selection of pressure tubes for material property testing and spacer surveillance.

Compliance with CSA N285.4-05 [B.2-3] is currently a licence requirement for Pickering NGS (per PROL 48.02/2018) as indicated in Appendix C.1 of the R04 Pickering Licence Conditions Handbook [B.2-4]. The current revision of the Standard, N285.4-14 [B.2-1], is the sixth edition of N285.4 which supersedes the previous editions, published in 2009, 2005, 1994, 1983, and 1978.

In the more recent versions of the Standard, revisions have ranged from editorial to the addition of supplementary information as well as the addition of new requirements. The CSA N285.4-14 Impact Statement [B.2-5] provides a "Summary of significant changes from the previous edition" which identifies six primary changes to the Standard which are discussed in Section B.2.2 below. In addition to findings resulting from review of the CSA N285.4-14 Impact Statement, the results of PSR1 N285.4 reviews (Pickering B and Darlington ISRs), as well as reviews performed since PSR1, have also been assessed for applicability to PSR2 in Section B.2.2.

As identified in Reference [B.2-6], the Pickering PSR2 review of CSA N285.4-14 is an Incremental Review. As discussed in Section 2.0 of this Report, PSR2 Incremental Review includes an assessment of the intent of recent changes to the L/R/C/S on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.

- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.2.2 Compliance Assessment for Pickering PSR2

B.2.2.1 Application of PSR1 Reviews

The versions of N285.4 subject to previous ISR reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

The Pickering B ISR was performed using the N285.4-94 [B.2-7] and N285.4-05 [B.2-3] versions of the Standard. The Pickering B ISR review based on N285.4-05 [B.2-3], being the most recent (as well as the version referenced in current Pickering NGS PROL 48.02/2018) is discussed further with respect to the Pickering code reviews conducted for Safety Factors 1 and 2. OPG Report NK30-REP-03680-00001 R000 [B.2-2] documented a clause-by-clause review of N285.4-05 for Safety Factor 1 and concluded the following:

> *The existing CSA N285.4-05 inspection program documents were reviewed to determine the degree to which the program at PNGS B complies with the requirements of CSA N285.4-05. These program documents include a "Compliance Matrix" which specifically indicates the areas where the current program does not fully comply with the requirements of the standard. …the only area of non-compliance with the standard that is a design issue is for clause 3.7.1 [which specifies that the design and arrangement of components and piping shall provide for clearances adequate to permit all inspections]. OPG's planned resolution to this problem, stated in the previous section, has been accepted by the CNSC. Accordingly, CPUS recommend that OPG refrain from making design changes to the plant for the sole purpose of providing access to piping and components to facilitate future periodic inspections. The necessary design changes and work to rearrange equipment and*

piping would incur unnecessary costs and expose plant personnel to unnecessary radiation hazards.

OPG Report NK30-REP-03680-00002 R000 [B.2-8] documented a clause-by-clause review of N285.4-05 [B.2-3] Clauses 3 through 11 for Safety Factor 2. With respect to the non-compliance for Clause 3.7.1 of N285.4-05 identified in the code review for Safety Factor 1, Appendix D of NK30-REP-03680-00002 R000 [B.2-8] included the following update:

> *OPG provided clarification as per NK30-CORR-00531-04566 [Ref 2]. OPG agreed to include more details about evaluation and substitution criteria for inaccessible / partially accessible locations in the next revision of N-ED-03641.2-10000.*
>
> *NK30-PIP-03641.2-00001/00002/00003/00004 (for Units 5/6/7/8) Appendix E, "CAN/CSA-N285.4-05 Compliance Matrix" item 1 states: "Where there is limited access for inspection of a selected piping weld or component, the inspection will be completed to the extent possible and credit will be taken for full inspection of that component."*

Per Reference [B.2-9], OPG's planned resolution to this problem was accepted by the CNSC and this gap is now closed. This gap is not impacted by operation past 2020 and therefore there is no related gap for Pickering PSR2.

Section 3.1.9 of NK30-REP-03680-00002 R000 documented the findings from the code review for Clauses 3 through 11 as follows:

> *(a) Compliance Review of CSA N285.4*
>
> *… only Clauses 3 through 11 of this standard were included in this review. A tabularized detailed clause-by-clause summary of the results of the review is documented in Table 1 in Appendix D.*
>
> *Discrepancies: There were no "Discrepancies" identified.*
>
> *Acceptable Deviations: Compliance against 9 clauses were identified as Acceptable deviations.*

The nine Acceptable Deviations (ADs) identified in NK30-REP-03680-00002 R000 are not safety significant. Further, the ADs are not impacted by Pickering operation past 2020 and are therefore not addressed further in the context of Pickering PSR2. Therefore, there are no PSR2 applicable gaps resulting from the Pickering B ISR review.

Pickering Units 1,4

For Pickering A Return to Service, the active version of the Standard at the time was CSA-N285.4-94 [B.2-7]. As outlined in NA44-CORR-00531-00381 R000 [B.2-10], OPG did not perform a code review of the Standard on the basis that it "Pertains mostly to Operations aspects, or other aspects not having a direct or immediate effect on installed design features". However, NA44-CORR-00531-00381 R000 states that "Condition 5.2 of the Power Reactor Operating Licence requires OPG to inspect and perform material surveillance according [to] the technical requirements in CSA Standard N285.4-94…. Completion of the associated PIP is included as Regulatory Commitments in the scope of the Pickering A Return to Service project." Further, Section 6.1.3, "Periodic Inspection Program - Nuclear Plant and Containment Components Inspections" of the Pickering PROL Renewal Application [B.2-11] states:

> *The main objective of the periodic inspection programs is to ensure [OPG] satisfy the following Canadian Standards Association CAN/CSA Standards: CSA-N285.4-05, Periodic Inspection of CANDU Nuclear Power Plant Components.*
>
> *… The CSA N285.4 program consists of approximately 300-600 inspection items for each of the six operating units. Each scheduled item is normally inspected once within each 10-year cycle. Inspected components include: piping and vessel welds, pumps, valves, pipe and component supports, heat exchangers, and mechanical couplings.*

Section 6.2 of [B.2-11], "Major Components Life Cycle Management", states:

> *The Major Components program has implemented Periodic Inspection Program (PIP) plans for SG, FC and feeders, according to CSA Standard N285.4. The Program will continue to be executed until the end of commercial life with change expected primarily in the asset preservation category of work.*

Since N285.4-05 [B.2-3] is a licence requirement for Pickering NGS per Appendix C.1 of the R04 Pickering Licence Conditions Handbook [B.2-4], and compliance is confirmed in the Pickering PROL Renewal Application [B.2-11], Pickering Units 1,4 (and Units 5-8) are in compliance with N285.4-05. Further, there are no gaps that result from the Pickering B ISR as discussed above. The results of Darlington ISR reviews for the latest editions of the Standard, and their applicability to Pickering, are discussed below.

Darlington NGS

The Darlington ISR was performed using N285.4-05 including Update No. 1 (June 2007) [B.2-12] and N285.4-09 including Updates No. 1 (January 2010) and No. 2 (June 2011) [B.2-13]. These reviews are documented in Reports NK38-REP-03680-10057 R000 [B.2-14] and NK38-REP-03680-10137 R000 [B.2-15] respectively. Various gaps for Darlington were identified and documented in NK38-REP-03680-10057 R000 and NK38-REP-03680-10137 R000. Of these Darlington Reports, only NK38-REP-03680-10137 R000 (N285.4-09 assessment) was reviewed in detail for PSR2 since N285.4-05 was already assessed as part of the Pickering B ISR and Pickering B is in compliance with N285.4-05 as discussed above.

The clauses in N285.4 are primarily programmatic from the perspective of the specified requirements, while recognizing that required records and reports, including PIPs, are specific to each station. Therefore, the Darlington ISR N285.4 conclusions above are largely applicable to Pickering PSR2.

As described in NK38-REP-03680-10137 R000, the changes made in CSA N285.4-09 including Updates 1 and 2 [B.2-13] relative to N285.4-05 [B.2-12] included changes to 102 clauses. NK38-REP-03680-10137 R000 identified that changes to 35 of those clauses were editorial leaving the changes to the remaining 67 clauses which were reviewed in more detail within the report. Those 67 changes resulted in the identification of 43 Darlington ISR Gaps relative to the requirements of N285.4-09 which were then grouped under 8 areas (ISR Issues) where "OPG's governance for Darlington compliance with the requirements of CSA N285.4 will need to be updated in order to comply with the requirements of CSA N285.4-09 Update 2". The 8 Darlington ISR Issues identified in NK38-REP-03680-10137 R000 are applicable to Pickering (since they are either programmatic or relate to PIP updates (which also apply to Pickering)) and are therefore addressed below:

- **PSR2 N285.4 Gap #1**: OPG PIP Governance references N285.4-05, not N285.4-09 including Updates 1 and 2. This (programmatic) Darlington ISR gap is a PSR2 gap against N285.4-09 including Updates 1 and 2.

- **PSR2 N285.4 Gap #2**: There has been a significant change in the wording of clause 4.2.7 in CSA N285.4-09 including Updates 1 and 2. I-PROC-AS-0009, "Inspection Qualification of Non-Destructive Examination Processes" does not identify the authorized inspector as a qualifying authority as directed by clause 4.2.7. Instead it establishes the CANDU Inspection Qualification Bureau (CIQB) as the organization that would approve procedures and personnel. This (programmatic) Darlington gap is a PSR2 gap against N285.4-09 including Updates 1 and 2.

- **PSR2 N285.4 Gap #3**: New erosion and corrosion inspection requirements in N285.4-09 including Updates 1 and 2 are not reflected in current PIP governance. NK38-REP-03680-10137 R000 states that: "It should be noted specifically that [this ISR Issue] is likely to have a major impact on piping PIPs because sub-clauses 7.4.7.X in CSA N285.4-09 including UPD1 and UPD2 include substantive changes. Under the new standard erosion and corrosion inspection exemptions can no longer be justified on the basis of [sic] that conditions are determined to be non-erosive and non-corrosive." This Darlington PIP gap will also need to be addressed in the

Pickering PIPs. Therefore, this is a PSR2 gap against N285.4-09 including Updates 1 and 2.

- **PSR2 N285.4 Gap #4**: Extended life inspection schedules in N285.4-09 including Updates 1 and 2 are not reflected in PIP governance. This (programmatic) Darlington gap is a PSR2 gap against N285.4-09 including Updates 1 and 2.

- **PSR2 N285.4 Gap #5**: An assessment of the prior operating non-conforming state, as required by N285.4-09 including Updates 1 and 2, is required when dispositioning inspection results. This requirement has not been included in the feeder PIP plan. This Darlington PIP gap will also need to be addressed in the Pickering PIPs. Therefore, this is a PSR2 gap against N285.4-09 including Updates 1 and 2.

- The Darlington ISR identified that OPG Governance does not ensure that qualifications of examination personnel are included within inspection reports as required by clause 12.4.4.6 of N285.4-09 including Updates 1 and 2, and was therefore identified as a gap. However, clause 11.3.2.1 of N285.4-09 including Updates 1 and 2 already requires that qualifications of examination personnel be included in inspection reports, and clause 11.3.2.1 had not changed relative to N285.4-05 [B.2-12]. Further, Section 6.5 of the Pickering Units 1,4 and 5-8 Fuel Channel PIP plans ([B.2-16], [B.2-17], [B.2-18], [B.2-19], [B.2-20] and [B.2-21]) state:

  > *…Beyond this specific reporting requirement, test records and the issuance of reports shall comply with [N285.4] Clause 11.*

  As a result, this is not a PSR2 gap.

- The Darlington ISR identified that although the Darlington PIP recognizes feeder cracking as an OPEX issue, it does not establish a default classification of "high" for feeders as would be required under clause 7.1.3.3 (a) of CSA N285.4-09 including Updates 1 and 2. This gap was deemed as an Acceptable Deviation in the Darlington ISR with the following resolution [B.2-22]:

  > *The fact that Piping and Component PIPs do not classify feeders as having high fatigue factor is an "acceptable deviation" that has no safety significance. Feeders are now being inspected under Feeder PIPs… in accordance with the requirements of N285.4. The feeder inspections in these PIP plans recognize and account for the feeder cracking OPEX at other CANDU sites and supersede those in the Piping and Component PIP plans...No further action is required.*

  A similar resolution can be applied to Pickering as separate Feeder Pipe PIP plans have been issued and accepted by CNSC in P-CORR-00531-04492 R000 [B.2-23] for Pickering Units 1,4 and 5-8. Feeder inspections will be removed from the Piping and Component PIP plans in their next revision as communicated to the CNSC in N-CORR-

00531-06833 R000 [B.2-24]. The Unit 5-8 Feeder Pipe PIP plans recognize feeder cracking OPEX at other CANDU sites (i.e., weld cracking at Gentilly 2). While OPEX at Gentilly 2 is not mentioned in the Unit 1,4 Feeder Pipe PIP plans, weld cracking inspections are included in the inspection program. Therefore, although the feeders are not classified as having a high fatigue factor in Piping and Component PIP plans, this is not a gap for Pickering PSR2.

- CSA N285.4-09 including Updates 1 and 2 requires that stresses, including thermal expansion stresses, are included in evaluations. Although secondary stresses may have been included in the PIPs this is not explicit. The PIPs will need to ensure that secondary stresses, including thermal expansion stresses, are evaluated for all components. This was therefore identified as a Darlington PIP gap in NK38-REP-03680-10137 R000. However, N-REP-03641-10003 R000 [B.2-28] discussed later, which addresses the most recent version of N285.4-14, states that there is "No Impact" given that "This sub-clause was added to the 2009 edition of CSA N285.4 Standard in order to provide further clarification only and it does not introduce new requirement." Therefore, this is not a gap for Pickering PSR2.

NK38-REP-03680-10137 R000 goes on to state: "This change from N285.4-05 UPD1 will introduce a significant change to the PIPs for piping when the operating license is amended to require compliance with CSA N285.4-09 including UPD1 and UPD2."

The 8 Darlington ISR Issues were subsequently rolled-up into three Darlington IIP [B.2-25] gaps, namely IIP-OI 044, IIP-OI 049 and IIP-OI 050, which call for the update of Darlington PIPs and the qualification of inspection procedures for specific components to address the requirements of the 2014 edition of N285.4 [B.2-1]. These three Darlington IIP gaps, which have a completion date of 2019, are [B.2-25]: 1) IIP-OI 044: "Perform compliance activities to meet CSA N285.4 including appropriate assessments and PIP updates." , 2) IIP-OI 049: "The jurisdictional boundary between ASME III and the building structure defined for Darlington NGS does not meet the current requirements of ASME Section III.", and 3) IIP-OI 050: "Periodic inspection procedures for volumetric inspections of pressure tubes are to be documented and proven capable of yielding results to a sensitivity that is appropriate for the system or components being inspected. All inspection procedures used in periodic inspections need to be qualified. Inspection procedures applied to pressure tube inspections are to be qualified by the CANDU Inspection Qualification Bureau (CIQB)." These Darlington IIP gaps are not identified as gaps for PSR2 (the 8 ISR Issues above are instead) since the Darlington GAR/IIP may have used Darlington-specific rationale for these "roll-ups" which are not applicable to Pickering NGS.

Nevertheless, there is a gap for Pickering NGS compliance with N285.4-14. As discussed under Section B.2.2.2 below, OPG Letter N-CORR-00531-06613 R000 [B.2-26], "OPG Transition Plan to 2014 Edition of CSA Standard N285.4 - Periodic Inspection of CANDU

Nuclear Power Plant Components" states that "OPG intends to transition only Darlington NGS to the 2014 edition of CSA Standard N285.4" and that "Pickering NGS will remain compliant with the 2005 Edition of CSA N285.4 and Update No. 1 until the planned end of commercial operation in 2020" (at the time of the communication to the CNSC, OPG was planning to end commercial operation of Pickering NGS in 2019).

Regulatory Management Action Request #28168380 was submitted in September 2014 to address OPG's N285.4-14 Transition Plan for Darlington which is tasked with addressing the IIP gaps listed above (for Darlington only). Since OPG intends to transition Darlington NGS to comply with N285.4-14 by 2019, a similar transition plan may also be required for Pickering NGS for operation past 2020. There are PSR2 gaps for Pickering NGS compliance with the 2014 edition of N285.4. This is discussed further under Section B.2.2.2 below.

B.2.2.2 Application of Post-PSR1 Reviews

Following the Darlington ISR reviews, additional compliance assessments were completed for the N285.4-14 [B.2-1], first in July 2015 for Clauses 12 through 14 (in N-REP-03641-10002 R000 [B.2-27]) and then in November 2015 for Clauses 1 through 11 (in N-REP-03641-10003 R000 [B.2-28]). Although these N285.4-14 reviews were completed specifically for Darlington, the dispositions are programmatic in nature (affecting OPG Nuclear Program documentation and Governance) or impacting identical station-specific documents for both Darlington and Pickering (i.e., the PIPs). Therefore, these reviews are applicable to Pickering NGS.

The compliance assessment of Clauses 1 through 11 [B.2-28] included a gap analysis between N285.4-14 and N285.4-05 including Update No. 1 [B.2-12] to identify, for Clauses 1 through 11, all changes between these two editions of the standard and the impact of the changes on Darlington's N285.4 Periodic Inspection Program. With respect to the Darlington ISR gaps identified earlier in OPG Reports NK38-REP-03680-10057 R000 [B.2-14] and NK38-REP-03680-10137 R000 [B.2-15], N-REP-03641-10003 R000 [B.2-28] states:

> *Although some of the compliance gaps have since been resolved and are no longer considered compliance gaps relative to N285.4-14, other compliance gaps identified relative to the earlier editions of N285.4 remain relative to N285.4-14.*

The compliance assessment of Clauses 1 through 11 [B.2-28] concluded:

> *The existing Darlington N285.4 PIP documentation complies with the 2005 edition of the Standard, but as the 2014 edition not only includes changes to some of the existing requirements (e.g. revised sample sizes for the inspection of identical components, adjusted inspection intervals) it also introduces new requirements (e.g. the need to identify locations for potential pipe wall thinning and to incorporate the inspection of those locations in the PIP Plans, and PIP documents). All the PIP Plans, program*

documentation and governance and PIP-related documents and procedures (e.g. inspection procedures) need to be reviewed and revised to reflect the new or revised requirements of N285.4-14, which range from straightforward reference updates to extensive revision.

The compliance assessment of Clauses 12 through 14 [B.2-27] documents the findings of the gap analysis between N285.4-14 [B.2-1] and N285.4-05 including Update No. 1 [B.2-12], and also addresses CNSC comments raised in the Code Refresh Review of N285.4-09 Update No. 2 for the Darlington ISR [B.2-15]. In general, the review found that various Darlington PIP documents will need to be updated to reflect updated requirements in N285.4-14. Similar to OPG Letter N-CORR-00531-06613 R000 [B.2-26], the compliance assessment of Clauses 12 through 14 [B.2-27] stated that OPG intends to transition Darlington NGS to N285.4-14 by 2019, and that Pickering NGS will remain compliant with the CSA N285.4-05 and Update No. 1 until the planned end of commercial operation (assumed at the time to be 2019).

As discussed earlier, the CSA N285.4-14 Impact Statement [B.2-5] identifies six primary changes from the previous edition of the Standard which provides additional insight about the potential impact of changes in N285.4-14. A summary of each change, as well as a brief assessment of safety significance, is addressed below:

- N285.4-14 Impact Statement Change #1 [B.2-5]: "Clarified inspection ultrasonic scan requirements to remove ambiguous language. Provided guidance for use of phased array ultrasonic inspection methodology. Provides direction for inspection staff. Changes made to Clauses 2.2, 4.2, 4.3, 8.2.1, and 8.2.4; update of Figure 3. Impact: - Clarified inspection ultrasonic scan requirements to remove ambiguous language. - Provided guidance for use of phased array ultrasonic inspection methodology. - Changes made to Clauses 2.2, 4.2, 4.3, 8.2.1, and 8.2.4; update of Figure 3."

  These changes are for clarification and guidance only (to remove ambiguity) and are not safety significant. Therefore, there is no gap for Pickering PSR2.

- N285.4-14 Impact Statement Change #2 [B.2-5]: "Updated Figure A-1 to reflect current practices in defining inspection requirements. Provides direction to staff setting periodic inspection programs. Impact: Provides increased clarity in non-mandatory annex that provides an overview of the philosophy."

  These changes are for increased clarity of a non-mandatory Annex only and are not safety significant. Therefore, there is no gap for Pickering PSR2.

- N285.4-14 Impact Statement Change #3 [B.2-5]: "Updated minimum requirements for SG tube surveillance examination and testing (Clause 14.4.1 and new Annex H). Possible update to industry best practices. Impact: - Modified Clause 14.4.1, added new Annex H. - Permits use of Integrated Material Surveillance Program (IMSP) to minimize number of SG tube removals. - Allows licensee to request exemption from SG tube removals if can demonstrate, via a technical justification, that active SG tubing degradation is known, stable, predictable and managed by SG LCMP actions over next surveillance interval (Regulatory acceptance is required for an IMSP or exemption; Rules for IMSP preparation or exemption request are clearly defined)."

N-PROG-MA-0025 R002, "Major Components" [B.2-29] points to N-PLAN-33110-10009 R006, "Steam Generators Life Cycle Management Plan" [B.2-30], which states the following in Section 1.10:

> *CSA N285.4 Clause 14 [N-PLAN-33110-10009] requires periodic removal of SG tubes for material surveillance (every six years without an integrated plan). In 2011, OPG performed a review of the potential for changes to the Clause 14 material surveillance requirements and initiated dialogue with the CSA N285B Technical Committee (TC) regarding the next version of CSA N285.4 to be issued in 2014. OPG authored a position paper on this subject [N-PLAN-33110-10009]. Options were developed in the paper for possible modification to Clause 14 N285.4 tube removal requirements. The OPG position paper was distributed to external stakeholders in Q1 2012 [N-PLAN-33110-10009]. In 2012, OPG led a CSA N285B TC task team that produced a modification to Clause 14 and a new informative annex to allow a licensee to prepare and request regulatory acceptance of an exemption from the requirement for a surveillance tube removal provided certain conditions are satisfied. These modifications were included in the draft 2014 CSA N285.4 standard that underwent public review in 2013 and have been approved by CSA N285B TC for inclusion in the standard.*
>
> *Periodic removal of SG tubes remains as the default requirement in the 2014 CSA N285.4 standard until such time as a licensee has prepared, submitted and obtained CNSC approval of an exemption from the tube removal requirements. OPG prepared a technical justification and requested CNSC acceptance of an exemption from CSA N285.4-05 requirements for steam generator surveillance tube removal, for the Periodic Inspection Program (PIP) interval of 2009-2015, during 2015 Unit 5 and Unit 6 planned maintenance outages [N-PLAN-33110-10009]. CNSC accepted OPG's request for exemption of tube removal if the following conditions were met [N-PLAN-33110-10009]: 1. Inspection results of planned 2015 outages of Unit 5 and Unit 6 will unquestionably validate current technical justification and fully satisfy the requirements from the Annex H of CSA N285.4-14 and OPG proposed criteria. 2. In case that the 2015 inspection results deviate from the criteria defined in Annex H of CSA N285.4-14 , OPG will follow guidance of current CSA N285.4-05 and perform tube removal. Assessment of the P5, 2015 steam generator inspection results confirmed that OPG's technical justification remains valid and satisfies CNSC's condition #1 described above. PIP primary side tube removal in one SG was deleted*

from the approved P1551 outage scope of work per the scope change #2 [N-PLAN-33110-10009]. P6 steam generator inspection results will be assessed following P1561.

Further, N-REP-03641-10002 R000 [B.2-27] assessed this clause for Darlington and concluded: "Compliant. Impact - None unless exemption is pursued. If exemption is desired follow Clause 14.4.1.3 and Annex H". Based on the above, OPG is aware of, and has been interfacing with the CNSC to satisfactorily account for this new requirement. As a result, there is no gap for Pickering PSR2.

- N285.4-14 Impact Statement Change #4 [B.2-5]: "Revised requirements for pressure tube volumetric and dimensional inspection (Clause 12.2), pressure tube hydrogen equivalent determination (Clause 12.3) and pressure tube material property testing (Clause 12.4). Updates also made to associated Annexes. Updated to include industry best practices and incorporate lessons learned from reactor which have operated one full life cycle. Impact: Revised requirements for pressure tube inspections, [Heq] measurements and material property testing. - Based on good Operating Experience, and lessons learned for FCLM project. - 20% reduction in total inspections and number of inspection outages in first 30 years of operation. - Clarified volumetric inspection requirements, eliminating potential compliance gaps."

  N-PROC-MA-025 R002, "Major Components" [B.2-29], points to N-PLAN-01060-10002 R016, "Fuel Channel Life-Cycle Management Plan" [B.2-31], which states: "In this procedure wherever reference is made to Clause 12 and sub-clauses, the reference is to the edition of CSA N285.4 in the station specific Power Reactor Operating License" which is N285.4-05, not N285.4-14. As discussed earlier, N-REP-03641-10002 R000 [B.2-27] reviewed these N285.4-14 clauses and determined that various Darlington PIP documents will need to be updated to reflect updated requirements in N285.4-14. There is a PSR2 gap (i.e., PIP updates required) for Pickering NGS compliance with N285.4-14 Clauses 12.2, 12.3 and 12.4. Compliance requirements for alignment with N285.4-14 are identified under **PSR2 N285.4 Gap #6.**

- N285.4-14 Impact Statement Change #5 [B.2-5]: "Added new Clause 12.5 specifying minimum annulus spacer surveillance examination and testing requirements. Addresses OPEX item on material property degradation of annulus spacers (garter springs). Impact: - Spacer surveillance requirements added (new Clause 12.5). - Requirements modeled on PT material property testing (Clause 12.4)."

  N-REP-03641-10002 R000 [B.2-27] reviewed this clause and determined that "requirements will need to be incorporated into DNGS PIP plans and associated documentation in transition to 2014 Edition of standard." There is a PSR2 gap (i.e., PIP updates required) for Pickering NGS compliance with N285.4-14 Clause 12.5. As discussed above, compliance requirements for alignment with N285.4-14 are identified under **PSR2 N285.4 Gap #6**.

- N285.4-14 Impact Statement Change #6 [B.2-5]: "Updated selection criteria for identifying candidate tube for pressure tube surveillance examination and testing (Annex E) to include address [sic] selection criteria for annulus spacer surveillance examination and testing. Possible update to industry best practices. Impact: Modified Annex E (channel selection requirements for material surveillance) to include requirements for spacer surveillance. - Can be integrated with tube removals for pressure tube material property testing to minimize impacts."

  This is an update with respect to aligning with industry best practices, and similar to the previous two Impact Statement changes, PIP documents will need to be updated to reflect updated requirements in N285.4-14. As discussed above, compliance requirements for alignment with N285.4-14 are identified under **PSR2 N285.4 Gap #6**.

It is noted that the CSA N285.4-14 Impact Statement [B.2-5] does not identify clauses 7.4.8 (which relates to new requirements for inspection for Environmentally Assisted Cracking), or clauses 7.5.1/7.5.2 (which relate to new requirements for Inspection of identical components), as being a significant change from the previous edition of the Standard. However, N-REP-03641-10003 R000 [B.2-28] states:

> *Inspection for Environmentally Assisted Cracking (EAC): As the aforementioned, Clause 7.4.8 introduces new requirement to assess the potential for EAC to be present in all N285.4 PIP systems or components, and then determine its extent, the sample size, inspection intervals, inspection methods and procedures , reporting and acceptance criteria. As above, if a potential for EAC is identified as credible degradation. This information then needs to be incorporated in the N285.4 program.*
>
> *Inspection of identical components: According to Clauses 7.5.1 and 7.5.2, the minimum number of identical components/areas/welds to be inspected for each identical unit has been changed. Not only has the Figure 1 graph been replaced by Table 2 for specifying the sample size, but the sampling population in Table 2 is different and typically calls for a smaller number of identical components to be inspected. Consequently the PIP Plans and PIP documents are to be adjusted accordingly.*

Therefore, PIP documents will need to be updated to reflect updated requirements in N285.4-14. As discussed above, compliance requirements for alignment with N285.4-14 are identified under **PSR2 N285.4 Gap #6**.

In addition to the above N285.4 PSR2 gaps, in May 2015 the CNSC communicated a concern in CNSC Letter P-CORR-00531-04474 R000 [B.2-32] about OPG's use of N285.4 with respect to credit taken for inaugural inspections. Reference [B.2-32] states:

… inaugural inspections, performed to establish a baseline prior to SSCs going into service, cannot replace or be credited for the periodic inspections required to verify degradation of SSCs after going into service.

… CNSC staff request that OPG stops using inaugural inspections to fulfill the periodic inspections program requirements. Future periodic inspection reports will be monitored to verify that OPG is complying with CNSC's request.

There is no commitment or open action from OPG related to this CNSC request and compliance will be monitored by the CNSC as indicated above. Therefore, this is not a gap for PSR2.

B.2.3 Compliance Summary for Pickering PSR2

There are six PSR2 CSA N285.4 gaps which all relate to Safety Factor 4. The first five of these N285.4 gaps are applicable to compliance with N285.4-09 including Updates 1 and 2 [B.2-13]. The sixth N285.4 gap is applicable to compliance with N285.4-14 [B.2-1].

1. N285.4 PIP Governance references N285.4-05, not N285.4-09 including Updates 1 and 2. This (programmatic) Darlington gap is a PSR2 gap against N285.4-09 including Updates 1 and 2.

2. There has been a significant change in the wording of clause 4.2.7 in CSA N285.4-09 including Updates 1 and 2. I-PROC-AS-0009, "Inspection Qualification of Non-Destructive Examination Processes" does not identify the authorized inspector as a qualifying authority as directed by clause 4.2.7. Instead it establishes the CANDU Inspection Qualification Bureau (CIQB) as the organization that would approve procedures and personnel. This (programmatic) Darlington gap is a PSR2 gap against N285.4-09 including Updates 1 and 2.

3. New erosion and corrosion inspection requirements in N285.4-09 including Updates 1 and 2 are not reflected in current PIP governance. NK38-REP-03680-10137 R000 states that: "It should be noted specifically that [this ISR Issue] is likely to have a major impact on piping PIPs because sub-clauses 7.4.7.X in CSA N285.4-09 including UPD1 and UPD2 include substantive changes. Under the new standard erosion and corrosion inspection exemptions can no longer be justified on the basis of [sic] that conditions are determined to be non-erosive and non-corrosive." This Darlington PIP gap will also need to be addressed in the Pickering PIPs. Therefore, this is a PSR2 gap against N285.4-09 including Updates 1 and 2.

4. Extended life inspection schedules in N285.4-09 including Updates 1 and 2 are not reflected in PIP governance. This (programmatic) Darlington gap is a PSR2 gap against N285.4-09 including Updates 1 and 2.

5. An assessment of the prior operating non-conforming state, as required by N285.4-09 including Updates 1 and 2, is required when dispositioning inspection results. This requirement has not been included in the feeder PIP plan. This Darlington PIP gap will also need to be addressed in the Pickering PIPs. Therefore, this is a PSR2 gap against N285.4-09 including Updates 1 and 2.

6. There is a PSR2 gap for Pickering NGS against N285.4-14 to address:

    - Revised requirements for pressure tube volumetric and dimensional inspection (Clause 12.2), pressure tube hydrogen equivalent determination (Clause 12.3) and pressure tube material property testing (Clause 12.4);

    - Clause 12.5 which specifies minimum annulus spacer surveillance examination and testing requirements;

    - Selection criteria for identifying candidate tube for pressure tube surveillance examination and testing (Annex E) to include selection criteria for annulus spacer surveillance examination and testing; and

    - Clause 7.4.8 which specifies requirements for inspection of Environmentally Assisted Cracking, and Clauses 7.5.1/7.5.2 which specify requirements for inspection of identical components.

B.2.4 References

[B.2-1] CSA Standard N285.4-14, *Periodic Inspection of CANDU Nuclear Power Plant Components*, May 2014.

[B.2-2] OPG Report, NK30-REP-03680-00001 R000, *Pickering NGS-B Integrated Safety Review – Plant Design Safety Factor*, August 2007.

[B.2-3] CAN/CSA Standard N285.4-05, *Periodic Inspection of CANDU Nuclear Power Plant Components*, June 2005.

[B.2-4] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.2-5] CSA Impact Statement for Publication, *Product: New Edition – Product Designation: CSA* N285.4 *– Product Title: Periodic Inspection of CANDU Nuclear Power Plant Components – Date of release: May 2014*, Date not provided.

[B.2-6] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.2-7] CAN/CSA Standard N285.4-94*, Periodic Inspection of CANDU Nuclear Power Plant Components*, December 1994.

[B.2-8] OPG Report, NK30-REP-03680-00002 R000, *Pickering NGS-B Integrated Safety Review - Actual Condition of Systems, Structures and Components Safety Factor Report*, May 2008.

[B.2-9] CNSC Letter, NK30-CORR-00531-01864 R000, CNSC eDocs #3322802, T. E. Schaubel to P. Pasquet, Pickering B Periodic Inspection Program Submissions Action Item 2007-8-08, January 6, 2009.

[B.2-10] OPG Letter, NA44-CORR-00531-00381 R000, R. J. Strickert to J.S.C. Tong, *Pickering A - Updated Basis for Return to Service Document*, April 20, 2001.

[B.2-11] OPG Letter, P-CORR-00531-03719 R000, G. Jager to M. A. Leblanc, *Application for Renewal of Pickering Nuclear Generating Station Power Reactor Operating Licence*, July 4, 2012.

[B.2-12] CAN/CSA Standard N285.4-05 including Update 1, *Periodic Inspection of CANDU Nuclear Power Plant Components*, June 2005; Update 1: June 2007.

[B.2-13] CSA Standard N285.4-09, *Periodic Inspection of CANDU Nuclear Power Plant Components*, June 2009; Update 1: January 2010; Update 2: June 2011.

[B.2-14] OPG Report, NK38-REP-03680-10057 R000, *Review of CAN/CSA-N285.4-05 incl. UPD1 (June 2007), Periodic Inspection of CANDU Nuclear Power Plant Components for Darlington Integrated Safety Review*, August 2011.

[B.2-15] OPG Report, NK38-REP-03680-10137 R000, *Code Refresh Review of CSA N285.4-09 UPD2, Periodic Inspection of CANDU Nuclear Power Plant Components, for DNGS ISR*, July 2013.

[B.2-16] OPG Plan, NA44-PIP-31100-00001 R003, *Pickering Nuclear 1-4, Unit 1 Fuel Channel Pressure Tubes Periodic Inspection Program Plan*, May 30, 2015.

[B.2-17] OPG Plan, NA44-PIP-31100-00004 R003, *Pickering Nuclear 1-4, Unit 4 Fuel Channel Pressure Tubes Periodic Inspection Program Plan*, October 14, 2014.

[B.2-18] OPG Plan, NK30-PIP-31100-00001 R002, *Pickering Nuclear 5-8, Unit 5 Fuel Channel Pressure Tubes Periodic Inspection Program Plan*, April 25, 2012.

[B.2-19] OPG Plan, NK30-PIP-31100-00002 R002, *Pickering Nuclear 5-8, Unit 6 Fuel Channel Pressure Tubes Periodic Inspection Program Plan*, April 25, 2012.

[B.2-20] OPG Plan, NK30-PIP-31100-00003 R002, *Pickering Nuclear 5-8, Unit 7 Fuel Channel Pressure Tubes Periodic Inspection Program Plan*, April 25, 2012.

[B.2-21] OPG Plan, NK30-PIP-31100-00004 R002, *Pickering Nuclear 5-8, Unit 8 Fuel Channel Pressure Tubes Periodic Inspection Program Plan*, April 25, 2012.

[B.2-22] OPG Report, NK38-REP-00770-0471441 R000, *Nuclear Refurbishment Issue Resolution Form – Darlington Issue # D418 - Feeders are Not Classified as Having High Fatigue Factor Due to Feeder Cracking OPEX at Other Plants,* August 2013.

[B.2-23] CNSC Letter, P-CORR-00531-04492 R000, e-Doc 4780461, File No. 2.01, M. Santini to B. McGee, *Pickering NGS: Acceptance of CSA N285.4-05 Compliant Periodic Inspection Programs for Fuel Channel Feeder Pipes*, June 16, 2015.

[B.2-24] OPG Letter, N-CORR-00531-06833 R000, W.S. Woods to M. Santini and F. Rinfret, *Darlington and Pickering NGS – Updates to CSA N285.4 and CSA N285.5 Periodic Inspection Programs*, June 17, 2015.

[B.2-25] OPG Report, NK38-REP-03680-10185 R002, *Darlington NGS Integrated Implementation Plan (IIP)*, April 2015

[B.2-26] OPG Letter, N-CORR-00531-06613 R000, W.M. Elliott to M. Santini and F. Rinfret, *OPG Transition Plan to 2014 Edition of CSA Standard N285.4 – Periodic Inspection of CANDU Nuclear Power Plant Components*, September 17, 2014.

[B.2-27] OPG Report, N-REP-03641-10002 R000, *CSA N285.4-14 Compliance Assessment for Clauses 12-14*, July 2015.

[B.2-28] OPG Report, N-REP-03641-10003 R000, *CSA N285.4-14 Compliance Assessment*, November 2015.

[B.2-29] N-PROG-MA-0025 R002, "Major Components".

[B.2-30] N-PLAN-33110-10009 R006, "Steam Generators Life Cycle Management Plan", December 12, 2015.

[B.2-31] N-PLAN-01060-10002 R016, "Fuel Channels Life Cycle Management Plan", October 30, 2015.

[B.2-32] CNSC Letter P-CORR-00531-04474 R000, e-Doc 4759048 File No. 4.01.02, *Pickering Planned Outages P1351, P1441 and P1481, Final Periodic Inspection Reports for CSA N285.4, N285.5 & N287.7 (Action Items 2014-48-5288, 2014-48-5576 and 2014-48-5588)*, May 20, 2015.

B.3 CSA N285.5-13, "Periodic Inspection of CANDU Nuclear Power Plant Containment Components"

B.3.1 Background

The following paraphrased from the Preface of CSA N285.5-13 [B.3-1] provides a brief overview of the purpose of this Standard and the requirements expressed therein:

> *This Standard defines requirements for the periodic inspection of containment system components, including containment pressure suppression systems, in CANDU nuclear power plants.*
>
> *This Standard specifies requirements for*
>
> a) *inspection;*
> b) *accessibility;*
> c) *inspection methods and procedures;*
> d) *personnel qualifications;*
> e) *inspection criteria;*
> f) *inspection program development;*
> g) *inspection frequency;*
> h) *evaluation of inspection results;*
> i) *disposition of defects;*
> j) *repairs;*
> k) *documentation; and*
> l) *records.*

N285.5-13 is relevant to Safety Factor 2 (Actual Condition of SSCs). As stated in OPG Report NK30-REP-03680-00001 R000 [B.3-2], even though N285.5 applies to periodic inspection rather than to the design of the plant, there are elements of the Standard that have implications on plant design, and therefore N285.5 is also considered relevant to Safety Factor 1 (Plant Design). A number of clauses in N285.5-13 are also relevant to Safety Factor 3 (Equipment Qualification) and Safety Factor 4 (Aging), including:

- Clause 5.2: Inspection methods (specifically Clauses 5.2.3, 5.2.4, and 5.2.5);
- Clause 8.2: Plastic materials;
- Clause 8.3: Assessment and determination of inspection methods (specifically Clauses 8.3.1.3, 8.3.2.1, 8.3.3.1, and 8.3.4); and

- Annex A: Periodic inspection, material property monitoring, and test programs for Fibreglass Reinforced Plastic (FRP) containment components.

Compliance with CSA N285.5-08 including Update 1 [B.3-3] is currently a licence requirement for Pickering NGS (per PROL 48.02/2018) as indicated in Section 7.1 and Appendix C.1 of the R04 Pickering Licence Conditions Handbook [B.3-4]. The current revision of the Standard is N285.5-13 [B.3-1]. This is the fourth edition of CSA N285.5, and it supersedes the previous editions, published in 2008, 1990, and 1988.

The CSA N285.5-13 Impact Statement [B.3-5] provides a "Summary of Significant Changes from the Previous Edition" which identifies two primary changes to the Standard which are discussed in Section B.3.2 below. In addition to findings resulting from review of the CSA N285.4-14 Impact Statement, the results of PSR1 N285.5 reviews (Pickering B and Darlington ISRs), as well as reviews performed since PSR1, have also been assessed for applicability to PSR2 in Section B.3.2.

As identified in Reference [B.3-6], the Pickering PSR2 review of CSA N285.5-13 is an Incremental Review. As discussed in Section 2.0 of this Report, PSR2 Incremental Review includes an assessment of the intent of recent changes to the L/R/C/S on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.

- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.3.2 Compliance Assessment for Pickering PSR2

B.3.2.1 Application of PSR1 Reviews

The versions of N285.5 subject to previous reviews conducted for Darlington and Pickering, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

For Pickering B ISR Safety Factors 1 and 2, a clause-by-clause review against N285.5-M90 (R2000) [B.3-7] was performed. The reviews were documented in reports NK30-REP-03680-00001 R000 [B.3-2] and NK30-REP-03680-00002 R000 [B.3-8], respectively.

Report NK30-REP-03680-00001 R000 [B.3-2] concluded the following for Safety Factor 1:

The existing CSA N285.5-M90 inspection program documents were reviewed to determine the degree to which the program at PNGS B complies with the requirements of CSA N285.5-M90… only one area of non-compliance with the program is a design issue and that is for clause 4.5.1. OPG's planned resolution to this problem, stated in the previous section has been accepted by the CNSC.

The gap associated with clause 4.5.1 is closed. However, as discussed under the Darlington review below, the disposition of the gap for clause 4.5.1 refers to OPG receiving a concession from the CNSC on the inspection of components deemed to be inaccessible. A similar (updated) concession may be required for Pickering operation past 2020. This is therefore identified as **PSR2 CSA N285.5 Gap #1.**

NK30-REP-03680-00002 R000 [B.3-8] identified a number of Acceptable Deviations (ADs) for Clauses 4.5.1, 8.4.2.1, 8.4.2.2, 8.5.2.2, 8.6.3, 8.6.5, and 8.6.6. (Note: These ADs were also identified for Darlington and are discussed further in the "Darlington NGS" review subsection below. As discussed there, although compliance against these clauses were deemed to be ADs during the Pickering B ISR, further consideration in the context of PSR2 (Pickering operation past 2020) has led to these being reclassified as a PSR2 gap relating to ongoing CSA N285.5 concessions). In addition, ADs were identified in NK30-REP-03680-00002 R000 [B.3-8] for:

- Clause 4.1.2 (a) which describes requirements prefaced by "should" rather than "shall" but notes that all requirements must be performed unless an exception is granted by the CNSC. This is not a safety significant gap in the context of Pickering PSR2.

- Clause 4.5.2 which defines requirements for clearances to permit access for inspections. "PIP requirements identified in NK30-PIP-03642.2-00001 and P-PIP-03642.2-00001 have been accepted by CNSC" [B.3-8].

- Clause 5.3 (b) which requires that procedures that deviate from requirements must be submitted to the regulatory authority for approval. "Deviation from the standard is identified in NK30-PIP-03642.2-00001 and P-PIP-03642.2-00001 and has been approved by CNSC" [B.3-8].

- Clauses 6.1.2.1 through 6.1.2.5 which address authorized inspectors who are not part of OPG.

- Clause 7.1.1 which notes that requirements for inaugural inspections were not performed in some cases since components have been in service since 1970 and inaugural inspections were not required. This deviation has been accepted by the CNSC [B.3-8].

- Clause 8.6.1 which requires that a complete periodic inspection be made within a 5 year period commencing one year after the reactor unit achieves first criticality. This deviation has been accepted by the CNSC [B.3-8].

Given the above, these six ADs do not result in gaps for Pickering PSR2.

<u>Pickering Units 1,4</u>

For Pickering A Return to Service, the active version of the Standard at the time was CSA-N285.5-M90 (R2000) [B.3-7]. Per NA44-CORR-00531-00381 R000 [B.3-9] OPG did not perform a code review of the Standard on the basis that it "Pertains mostly to Operations aspects, or other aspects not having a direct or immediate effect on installed design features". However, NA44-CORR-00531-00381 states that "Condition 5.2 of the Power Reactor Operating Licence requires OPG to inspect and perform material surveillance according [to] the technical requirements in CSA Standard N285.4-94 and N285.5-M90. Completion of the associated Periodic Inspection Program (PIP) is included as Regulatory Commitments in the scope of the Pickering A Return to Service project." Further, Section 6.1.3, "Periodic Inspection Program - Nuclear Plant and Containment Components Inspections", of the Pickering PROL Renewal Application [B.3-10] states:

> *The main objective of the periodic inspection programs is to ensure [OPG] satisfy the following Canadian Standards Association CAN/CSA Standards: … CSA-N285.5-M90, Periodic Inspection of CANDU Nuclear Power Plant Containment Components.*
>
> *The CSA N285.5 program consists of approximately 1000 inspection items for Unit 0 and 200-600 inspection items for each of the six operating units. Each item is normally inspected once within each 10-year cycle. Inspected components include: containment penetration seal welds, pipe supports, piping/ducting, valves, containment dampers and other components.*
>
> *The CSA N285.5-08 compliance project is on target and will be completed in December 2012 in accordance with the transition plan accepted by the CNSC.*

Since CSA N285.5-08 [B.3-3] is a licence requirement for Pickering NGS per Appendix C.1 of the R04 Pickering Licence Conditions Handbook [B.3-4], and compliance is confirmed in the Pickering PROL Renewal Application [B.3-10], Pickering Units 1,4 (and Units 5-8) are currently in compliance with N285.5-08 [B.3-3]. The results of Darlington ISR reviews for the latest versions of the Standard, and their applicability to Pickering, are discussed below.

Darlington NGS

The Darlington ISR review of N285.5 was initially performed using version N285.5-M90 (R2005) [B.3-11] which was documented in OPG Report NK38-REP-03680-10058 R000 [B.3-12]:

> *The review found that Darlington NGS Governance is compliant with most of the clauses in CSA N285.5, with the exception of ten that have been categorized as gaps. The gaps are to clauses in which Darlington NGS has obtained regulatory acceptance…*

The clauses in N285.5 are primarily programmatic from the perspective of the specified requirements, while recognizing that required records and reports, including PIPs, are specific to each station. Therefore, the Darlington ISR N285.5 conclusions above are largely applicable to Pickering PSR2.

The clauses of N285.5-M90 (R2005) identified as having gaps in the Darlington ISR were 4.5.1, 8.4.2.1, 8.4.2.2, 8.4.5.2, 8.4.5.4, 8.5.2.2, 8.6.2, 8.6.3, 8.6.5 and 8.6.6. These clauses are assessed below in the context of Pickering operation past 2020:

- The disposition of the gap for clause 4.5.1 refers to OPG receiving a concession from the CNSC on the inspection of components deemed to be inaccessible (Item 1 for Clause 4.5.1). However, the Darlington ISR states: "… it remains a gap in Darlington governance as there is no assurance that a similar concession would be forthcoming in a refurbished plant". By the same logic it will need to be reconciled for Pickering for the period of PSR2 (life extension past 2020). As discussed under the Pickering B ISR review above, this was already identified as **PSR2 CSA N285.5 Gap #1**.

- The disposition of the gaps for clauses 8.4.2.1 and 8.4.2.2 refer to OPG receiving a concession from the CNSC that insulation will not be removed in the absence of visible damage to a component, and only "light weight" access covers will be removed. The Darlington ISR states: "This is a concession from the regulator which is not assured in the case of a refurbished plant. As such, this represents a gap". By the same logic it will need to be reconciled for Pickering for the period of PSR2. Since this gap is concession-related and associated with N285.5-M90 (similar to the gap above), it will also be tracked under **PSR2 CSA N285.5 Gap #1**.

- The disposition of the gap for clause 8.5.2.2 refers to Item 4 in Appendix F of Reference [B.3-13], which declares an exception of the numerical rules of this clause for reasons of practicality, and that a concession was received from the CNSC. The Darlington ISR states: "… it is categorized as a Gap, because a concession from the CNSC is not assured for a refurbished plant." By the same logic it will need to be reconciled for Pickering for the period of PSR2. Since this gap is concession-related and associated with N285.5-M90, it will also be tracked under **PSR2 CSA N285.5 Gap #1**.

- For clause 8.6.3, although CNSC acceptance was obtained, there is still a non-compliance with a portion of the clause related to the timing of inspections which is noted as needing to be reconciled for a refurbished station. The Darlington ISR states: "This represents a gap that will need to be reconciled with the regulator for a refurbished station." By the same logic it will need to be reconciled for Pickering for the period of PSR2. Since this gap is concession-related and associated with N285.5-M90, it will also be tracked under **PSR2 CSA N285.5 Gap #1**.

For clause 8.6.6, since the Darlington governance defines commencement (for the inspection program) as one year after first net power and or criticality, it does not comply with the requirement in the Standard, which defines commencement as when first criticality is achieved. However, N-PROC-MA-0064 [B.3-15], "Administrative Requirements For The Periodic Inspection Of Nuclear Power Plant Containment Components", was revised subseqent to the Darlington ISR and does not define the commencement of the inspection program as one year after first net power or criticality. Therefore, there is no PSR2 gap.

The disposition of the gap for Clause 8.6.2 refers to the Darlington VBO frequency being on a 12-year interval with the approval of the CNSC, which is a deviation from the 10-year requirement in N285.5-M90. The compliance discussion noted that this represents a gap that will need to be reconciled for a refurbished station. However, N285.5-13 has since updated clause 8.6.2 such that a 12-year interval is permitted. This is not a gap for Pickering PSR2 because the Pickering VBO is on a 10-year interval (i.e., on more frequent intervals than required).

The Gap related to clause 8.4.5.2 is closed and is not impacted by Pickering operation past 2020. Finally, the remaining two gaps in N285.5-M90 (R2005) for clauses 8.4.5.4, and 8.6.5 are classified as ADs per the Darlington Final ISR Report NK38-REP-03680-10104 R000 [B.3-14] and are not identified as being at risk by refurbishment (life extension), so they are not relevant in the context of PSR2.

Following the original code review for the Darlington ISR, a code refresh review was conducted based on the current revision of the Standard N285.5-13 [B.3-1] which was documented in OPG Report NK38-REP-03680-10138 R000 [B.3-16]. A clause-by-clause comparison of all clauses in N285.5-13 [B.3-1] relative to CSA N285.5-M90 (R2005) [B.3-11] was performed for the code refresh review. Three of the four changes in N285.5-13 relative to N285.5-08 are not applicable to Darlington since they were related to aging management (monitoring and test programs) for Fibreglass Reinforced Plastic (FRP) materials, which has been assessed at Pickering as part of Reference [B.3-17] for operation to 2024. As a result, additional assessment is required to address FRP aging management at Pickering for operation to 2028, and to confirm the current program aligns with N285.5-13 clauses 8.2, 8.3.3, 8.3.4 and A.6.1.2. This is therefore identified as **PSR2 CSA N285.5 Gap #2** (Note: This gap only exists if Pickering NGS intends to operate past 2024).

There were significant changes (approximately 40) in N285.5-08 relative to N285.5-M90 (R2005). However, the current Pickering NGS program is compliant to N285.5-08 as discussed earlier.

The code refresh review report concluded the following (paraphrased) [B.3-16]:

1. *Darlington is in compliance with the changed requirements in CSA N285.5-13. However, the OPG Nuclear governance and the PIP plan must be updated to reference the 2013 version of the standard. This can be done upon licence renewal once the 2013 N285.5 is included in the PROL.*

2. *There are no changes between CSA N285.5-08, Update 1 and N285.5-13 which are applicable to Darlington. Compliance with the N285.5-05, update 1 standard is based on:*

   (i) specific sections in NK38-PIP-03642.2-10001 R002 [B.3-18] and N-PROC-MA-0064 R005 [B.3-15], and

   (ii) Section 1.1.3 of OPG governance document, N-PROC-MA-0064 R005 [B.3-15] which states 'all inspections, examinations or testing required to ensure acceptability of containment components, shall be performed in accordance with the edition of CAN/CSA-N285.5 stated in the operating license.'

The above conclusions from NK38-REP-03680-10138 R000 were recorded for tracking purposes as Darlington ISR Issue #D564, and assigned Gap #02192 per the Nuclear Refurbishment Issue Resolution Form - Darlington [B.3-19]:

> *The review of the changed clauses in the code refresh review report [R1] concludes that Darlington is in compliance with the changed requirements in CSA N285.5-13. However, applicable OPG Nuclear governance and PIP plans do not make reference to CSA N285.5-13; hence Darlington is considered to be administratively non-compliant i.e. Darlington has been determined to be compliant with all of applicable clauses of N285.5-13, however, this edition of the N285.5 standard is not referenced in OPG governance documents and PIP plans.*

There are no gaps against N285.5-13 per the Darlington IIP Report NK38-REP-03680-10185 R002 [B.3-20]. The Darlington CSA N285.5-13 ISR conclusions above are largely applicable to Pickering PSR2 with the exception of clauses related to Fiberglass Reinforced Plastic which were not applicable to Darlington but are applicable to Pickering as discussed above. This is discussed further below.

B.3.2.2 Application of Post-PSR1 Reviews

Per the CSA N285.5-13 Impact Statement [B.3-5], the following is a summary of the significant changes from the previous edition of the Standard:

- N285.5-13 Impact Statement Change #1 [B.3-5]: New Annex A to identify the inspection and material testing requirements for Fibreglass Reinforced Plastic components added. The outlined requirements in Annex A "provide consistent framework for industry to perform periodic inspection, measurements material testing, and acceptance criteria for FRP."

- N285.5-13 Impact Statement Change #2 [B.3-5]: Update Standard for structure, content, clarity and alignment with other CSA N285 Standards. In particular, "New clause 4.6.3 added to adopt similar requirements in CSA N285.4-09."

As discussed under Section B.3.2.1 above, the changes in N285.5-13 relative to N285.5-08 that are applicable to FRP material that is used at Pickering NGS (but was not relevant to Darlington) have only been assessed for fitness for service to 2024 [B.3-17], so additional assessment is required in the context of Pickering PSR2 for operation to 2028. This was identified earlier as **PSR2 CSA N285.5 Gap #2**.

With respect to new clause 4.6.3, it states: "In cases when this Standard is being applied to an existing plant or to an existing periodic inspection program written to an earlier edition of CSA N285.5, the updated program documents shall identify: a) the requirements in this Standard that cannot be practically implemented; and b) measures taken to compensate for the requirements that cannot be practically implemented." This is not safety significant in the context of PSR2.

B.3.3 Compliance Summary for Pickering PSR2

There are two PSR2 CSA N285.5 gaps which both relate to Safety Factor 4:

1. There were a number of concessions granted from the CNSC for compliance with N285.5-M90 that will need to be reconciled for Pickering for the period of PSR2. Since these gaps are all concession-related and associated with N285.5-M90, they are tracked under a single PSR2 gap:

    - The Pickering B ISR gap associated with N285.5-M90 clause 4.5.1 is closed. However, the disposition of the gap refers to OPG receiving a concession from the CNSC on the inspection of components deemed to be inaccessible. A similar (updated) concession may be required for Pickering operation past 2020. Therefore, this is a gap for PSR2.

- The Darlington ISR disposition of the gaps for N285.5-M90 clauses 8.4.2.1 and 8.4.2.2 refer to OPG receiving a concession from the CNSC that insulation will not be removed in the absence of visible damage to a component, and only "light weight" access covers will be removed. The Darlington ISR states: "This is a concession from the regulator which is not assured in the case of a refurbished plant. As such, this represents a gap". By the same logic it will need to be reconciled for Pickering for the period of PSR2 (life extension past 2020).

- The Darlington ISR disposition of the gap for N285.5-M90 for clause 8.5.2.2 refers to an exception of the numerical rules of this clause for reasons of practicality, and that a concession was received from the CNSC. The Darlington ISR stated "… it is categorized as a Gap, because a concession from the CNSC is not assured for a refurbished plant.". By the same logic it will need to be reconciled for Pickering for the period of PSR2.

- Per the Darlington ISR disposition of the gap for N285.5-M90 clause 8.6.3, although CNSC acceptance was obtained, there is still a non-compliance with a portion of the clause related to the timing of inspections which is noted as needing to be reconciled for a refurbished station. The Darlington ISR stated "This represents a gap that will need to be reconciled with the regulator for a refurbished station." By the same logic it will need to be reconciled for Pickering for the period of PSR2.

2. The changes in N285.5-13 relative to N285.5-08 that are applicable to Fiberglass Reinforced Plastic material that is used at Pickering NGS have only been assessed for fitness for service to 2024 in Reference [B.3-17]. These changes related to aging management (monitoring and test programs) for FRP materials. As a result, additional assessment is required for Pickering to address FRP aging management at Pickering for operation to 2028, and to confirm the current program aligns with N285.5-13 clauses 8.2, 8.3.3, 8.3.4 and A.6.1.2. (Note: This gap only exists if Pickering NGS intends to operate past 2024.)

B.3.4 References

[B.3-1] CSA Standard N285.5-13, *Periodic Inspection of CANDU Nuclear Power Plant Containment Components*, July 2013.

[B.3-2] OPG Report, NK30-REP-03680-00001 R000, *Pickering NGS-B Integrated Safety Review – Plant Design Safety Factor*, August 2007.

[B.3-3] CSA Standard N285.5-08 , *Periodic Inspection of CANDU Nuclear Power Plant Containment Components*, November 2008; Update No. 1, January 2011.

[B.3-4] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.3-5] CSA Impact Statement, *Notification of N285.5-13*, Date not provided.

[B.3-6] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.3-7] CAN/CSA Standard N285.5-M90 (R2000), *Periodic Inspection of CANDU Nuclear Power Plant Containment Components*, October 1990.

[B.3-8] OPG Report, NK30-REP-03680-00002 R000, *Pickering NGS-B Integrated Safety Review - Actual Condition of Systems, Structures and Components Safety Factor Report*, May 2008.

[B.3-9] OPG Letter, NA44-CORR-00531-00381 R000, R. J. Strickert to J.S.C. Tong, *Pickering A - Updated Basis for Return to Service Document*, April 20, 2001.

[B.3-10] OPG Letter, P-CORR-00531-03719 R000, G. Jager to M. A. Leblanc, *Application for Renewal of Pickering Nuclear Generating Station Power Reactor Operating Licence*, July 4, 2012.

[B.3-11] CAN/CSA Standard N285.5-M90 (R2005), *Periodic Inspection of CANDU Nuclear Power Plant Containment Components*, October 1990.

[B.3-12] OPG Report, NK38-REP-03680-10058 R000, *Review of CAN/CSA-N285.5-M90 – (R2005) October 1990, Periodic Inspection of CANDU Nuclear Power Plants Containment Components for Darlington Integrated Safety Review*, August 2011.

[B.3-13] OPG Report, NK38-PIP-03642.2-10001 R001, *Darlington Nuclear Generating Station, Periodic Inspection Program for Unit 0 and Units 1 to 4 Containment Components*, October 2006.

[B.3-14] OPG Report, NK38-REP-03680-10104 R000, *Darlington NGS Integrated Safety Review (ISR) - Final ISR Report*, October 2011.

[B.3-15] OPG Report, N-PROC-MA-0064 R005, *Administrative Requirements For The Periodic Inspection Of Nuclear Power Plant Containment Components, October 2013*.

[B.3-16] OPG Report, NK38-REP-03680-10138 R000, *Code Refresh Review of CSA N285.5-13 (July 2013) Periodic Inspection of CANDU Nuclear Power Plant Containment Components*, January 2014.

[B.3-17] OPG Reference NA44-CORR-34320-0242520 R000, (NSS P0990/RP/002), *Pickering A Fibreglass Reinforced Plastic Aging Management Program: Stategy and Recommendations for FRP Components for the 2010 Vacuum Building Outage,* April 2008.

[B.3-18] OPG Report, NK38-PIP-03642.2-10001 R002, *Darlington Nuclear Generating Station - Periodic Inspection Program for Unit 0 and 1 to 4 Containment Components*, August 2012.

[B.3-19] OPG Report, NK38-REP-00770-0486906 R000, *Nuclear Refurbishment Issue Resolution Form – Darlington Issue # D564 - Darlington's Governance Documents and PIP Plans Reference Previous Edition of Code*.

[B.3-20] OPG Report, NK38-REP-03680-10185 R002, *Darlington NGS Integrated Implementation Plan (IIP)*, April 2015.

B.4 CSA N287.7-08 including Update No. 1, "In-Service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants"

B.4.1 Background

The following paraphrased from the Preface of CSA N287.7-08 (including Update No. 1) (R2013) [B.4-1] provides a brief overview of the purpose of this Standard and the requirements expressed therein:

> *This Standard provides requirements for in-service examinations and positive pressure leakage-rate testing of concrete containment structures of a containment system that are designated as class containment components.*

N287.7-08 is relevant to Safety Factor 2 (Actual Condition of SSCs). N287.7-08 is also applicable to Safety Factor 3 (Equipment Qualification) and Safety Factor 4 (Aging) since CNSC web site [B.4-2] identifies the Standard as being relevant to the Safety Control Area for "Fitness for Service".

Compliance with CSA N287.7-08 [B.4-3] is currently a licence requirement for Pickering NGS (per PROL 48.02/2018) as indicated in Appendix C.1 of the R04 Pickering Licence Conditions Handbook [B.4-4]. The current revision of the Standard is N287.7-08 (including Update No. 1) (R2013) [B.4-1]. This is the fourth edition of CSA N287.7 which supersedes the previous editions, published in 1996, 1980, and 1976. In the more recent versions of the Standard, the changes have ranged from editorial, to the addition of supplementary information as well as the addition of new requirements. An Impact Statement was not prepared by the CSA for N287.7-08. However, the results of PSR1 N287.7 reviews (Pickering B and Darlington ISRs), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.4.2.

As identified in Reference [B.4-5], the Pickering PSR2 review of CSA N287.7-08 is an Incremental Review. As discussed in Section 2.0 of this Report, PSR2 Incremental Review includes an assessment of the intent of recent changes to the L/R/C/S on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.
- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.4.2 Compliance Assessment for Pickering PSR2

B.4.2.1 Application of PSR1 Reviews

The versions of N287.7 subject to previous reviews conducted for Darlington and Pickering, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

The Pickering B ISR review of N287.7 was performed for Safety Factor 2, comprising a clause-by-clause review against N287.7-96 (R2005) [B.4-6]. The review was documented in OPG Report NK30-REP-03680-00002 R000 [B.4-7]. This report also included a review of environmentally qualified components, seismic qualification status of SSCs and aging management of critical components. Appendix D, Table 3 of NK30-REP-03680-00002 R000 documents the following findings: 103 clauses in direct compliance, Clause 4.5.1 is an Acceptable Deviation (AD) and 10 clauses with gaps (i.e., Clauses 3.1(c), 4.2, 5.1.3(c), 5.1.4, 5.2.4, 5.3.1, 5.3.3, 6.13.2, 8(j), and 8(k)).

SCR N-2008-01932 [B.4-8] was generated to track the documentation changes which were required to ensure compliance with N287.7-96 (R2005) [B.4-6]. The SCR assignment (Action Request assignment 28086700-01) to update N-PROC-MA-0066, "Administrative Requirements for In-Service Examination and Testing for Concrete Containment Structures" [B.4-9] to be fully compliant with CSA N287.7-M96 was completed in February 9, 2009 and also included a review of N287.7-M96 compliance for Pickering A. SCR N-2008-01932 [B.4-8] noted the following actions initiated to resolve the gaps associated with N287.7-M96 (R2005) [B.4-6]:

> *Nuclear Generation Division (NGD) focus has been on whether or not Pickering B is compliant with CSA Standard N287.7-M96. Revising NK30-PIP-03643.2-0001 will ensure that this document will cover the CSA Standard N287.7-M96 requirements. However, a preliminary review of other documentation associated with N287.7-M96 found that minor revisions were required in some of the other related documentation. These are listed below:*
>
> 1. *NK30-CTP-21100-00001 RB Pressure Test Prerequisites (IPTE)*
> 2. *NK30-CTP-21100-00002 Reactor Building Pressure Test (IPTE)*
> 3. *NK30-CTP-21100-00003 RB Pressure Test Post-requisites (IPTE)*
> 4. *P-REP-34200-10003 Vacuum Building and Pressure Relief Duct Inspection Requirements*

5. *N-PROC-MA-0066, Administrative Requirements for In-Service Examination and Testing for Concrete Containment Structures*

6. *N-PROC-MA-0052, Flaw Dispositioning.*

Some of the CSA N287.7-M96 clauses to be considered during documentation revision are clauses 3.1 (c), 4.2, 5.2.4 and 5.3.3.

The SCR associated with these gaps is closed, and is not impacted by Pickering operation past 2020. Based on the above, there are no gaps for Pickering PSR2 that relate to the Pickering B ISR clause-by-clause review of N287.7-96 (R2005).

<u>Pickering Units 1,4</u>

For Pickering A Return to Service, the active version of the Standard at the time was CSA-N287.7-96 (R2000) [B.4-10]. Per NA44-CORR-00531-00381 R000 [B.4-11], OPG did not perform a code review of the Standard on the basis that it "Pertains mostly to Operations aspects, or other aspects not having a direct or immediate effect on installed design features". However, Section 6.1.3, "Periodic Inspection Program - Nuclear Plant and Containment Components Inspections", of the Pickering PROL Renewal Application [B.4-12] states:

> *The main objective of the periodic inspection programs is to ensure [OPG] satisfy the following Canadian Standards Association CAN/CSA Standards: … CSA-N287.7-96 or -08, In-service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants.*
>
> *The CSA N287.7 program addresses inspection and testing of concrete containment structures. Separate PIP Plans have been created and submitted to the CNSC for the vacuum building and Pressure Relief Duct (PRD), the reactor buildings, and the vacuum building post-tensioning system. The PIP Plans identify the Civil Containment Structures and components to be inspected, describe relevant mechanisms potentially affecting these components, identify inspection methods and acceptance criteria, and define the reporting requirements. Access and other supporting functions required to perform the inspections are also provided.*
>
> *Inspections/testing of Vacuum Building (VB) and PRD containment structures were performed during the 2010 Vacuum Building Outage. Inspection activities involved concrete components, vacuum building joint sealant, vacuum building roof seal and pressure relief duct joint seals. There were only minor findings from the 2010 VBO inspections, with repairs completed or findings assessed as acceptable by the Plant Design Department. In addition, vacuum building in-leakage testing was performed during the 2010 VBO and results were acceptable. The Pickering VB and PRD PIP Plan, NA44-PIP-03643.2-00002 R000 (written to N287.7-96), is currently being adhered to.*

Inspections/testing of Units 1 and Unit 4 Reactor Buildings (RBs) were performed during 2010 (P1011) and 2009 (P941) planned outages, respectively. Inspections/testing of the Units 5-8 Reactor Buildings (RBs) were performed during the period of 2008-2011. Minor follow-up activities for repair and monitoring were undertaken as detailed in the submitted inspection reports.

Inspections/testing of the Vacuum Building post-tensioned rods (PTRs) were performed in 2010. CNSC AI 2010-4-18 tracks open Action Notices related to the post-tensioning system PIP plan, NA44-PIP-03643.2-00003 R000 (written to N287.7-08).

OPG revised its Aging Management Plan (AMP) N-PLAN-01060-10004 R002 [B.4-13], which establishes, implements, and improves OPG's aging management strategy for containment structures built under CSA N287 series of standards. The AMP notes that OPG's aging management governance, along with nuclear procedure N-PROC-MA-0066 [B.4-9], ensure that the Pickering PIPs are compliant with CSA N287.7. Based on the above, Pickering Units 1,4 and Units 5-8 PIPs are currently in compliance with either N287.7-96 [B.4-6] or N287.7-08 [B.4-3]. There are no gaps for Pickering that result from the past Pickering B ISR review of N287.7-96, as discussed above. Further, as discussed earlier, CSA N287.7-08 [B.4-3] is currently a licence requirement for Pickering NGS (per PROL 48.02/2018) as indicated in Appendix C.1 of the R04 Pickering Licence Conditions Handbook [B.4-3]. The CNSC has accepted the Pickering 1,4 and Pickering Units 5-8 PIPs with respect to the Compliance Verification Criteria related to N287.7-08 per the Pickering NGS Licence Conditions Handbook [B.4-4], which states:

CNSC staff have accepted the Pickering NGS-A and B PIP documents (e-Doc 4452432).

The results of Darlington ISR reviews and their applicability to Pickering (including compliance against N287.7-08) are discussed below, since Darlington ISR programmatic conclusions are applicable to Pickering PSR2.

Darlington NGS

The Darlington ISR was performed against N287.7-08 [B.4-3] and is documented in OPG Report NK38-REP-03680-10061 R000 [B.4-14] which states that this version of the Standard was a non-PROL code at the time:

CAN/CSA N287.7-08, "In-Service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants" [R-6] is a Non-PROL code and the use of the term "Compliant" in the Compliance Categorization is therefore to be read as being the same as the term "Indirect Compliance".

The findings from the Darlington ISR code review were:

... that eight clauses in CSA N287.7-08 are not applicable to Darlington and also Darlington is non-compliant with three other clauses of CSA N287.7-08. It is noted that N287.7-08 is a fairly recent revision with some new clauses, and that OPG has informed the CNSC that it will update its governance so that it is in full compliance with the new standard. Other gaps have already received CNSC approval, but are noted as gaps since there is no assurance that a similar concession from the CNSC would be forthcoming for a refurbished plant.

Four of the eight clauses that are not applicable to Darlington are also not applicable to Pickering:

- Clause 6.1.8.1 is not applicable to Darlington because the Emergency Water Storage Tank (EWST), which forms part of the Dousing System, is drained for inspection. The same is true for Pickering, i.e., the Dousing Tank is drained for inspection, therefore this is not a gap for PSR2.

- Clause 6.2.4 is not applicable at Darlington because test beams are not used. The same is true for Pickering, therefore this is not a gap for PSR2.

- Clause 7.6.2 is not applicable to Darlington because Darlington uses the absolute method using the mass plot analysis technique for leakage-rate testing (per clause 7.6.1). The same is true for Pickering, therefore this is not a gap for PSR2.

- Clause 7.12.3 is not applicable to Darlington because Darlington does not use two independent systems to validate the leakage test results. The same is true for Pickering, therefore this is not a gap for PSR2.

The remaining four of the eight clauses that are not applicable to Darlington are Clauses 6.2.1, 6.2.2, 6.2.3 and Clause 7.4. However, clauses 6.2.1, 6.2.2 and 6.2.3 need to be assessed for Pickering due to design differences as discussed below:

- N287.7-08 Clause 6.2.1 requires that: "Pre-stressing systems used as principal reinforcement in concrete containment structures shall be subject to an integrity evaluation for conformance to the design specifications to determine the effects of certain time-related factors, such as: (a) shrinkage and creep of the concrete; (b) stress relaxation; and (c) deterioration." Clause 6.2.2 requires: "In addition to the requirements of Clause 6.1, the consequences of full or partial loss of pre-stress shall be considered." Clause 6.2.3 requires: "Where instrumented monitoring is used to verify the integrity of the pre-stressing system, a monitoring program that is specific to the application shall be developed (see Annex F)."

  Clauses 6.2.1, 6.2.2 and 6.2.3 are not applicable to Darlington because the concrete pre-stressing systems are not used as the principle reinforcement in concrete structures.

Although the Reactor Buildings at Pickering do not have a pre-stressing system, the Vacuum Building does. However, the pre-stressing system in the Vacuum Building at Pickering is not used as principal reinforcement, and therefore Clause 6.2 of N287.7-08 is not applicable. Section 7.0 of NA44-PIP-03643.2-00003 R002, "Pickering Nuclear GS – Vacuum Building Post Tensioning Rods Periodic Inspection Program" [B.4-15] states:

> *Both vertical and horizontal PTRs [post-tensioning rods] in the VB [Vacuum Building] structure ring girder are not principal reinforcement of the VB concrete containment structure and are therefore not subject to an integrity evaluation in accordance with the clause 6.2 of CSA/CAN N287.7.*

NA44-PIP-03643.2-00003 R002 was prepared in accordance with CSA N287.7-08 [B.4-16] and was accepted by the CNSC on July 16, 2014 [B.4-17]. Therefore, there is no gap for PSR2.

With respect to N287.7-08 Clause 7.4, this allows omission of positive Vacuum Building pressure testing if certain conditions are met. It is not applicable to Darlington because the Darlington Vacuum Structure leakage test is performed at positive pressure, as required in Regulatory Document R-7, "Requirements for Containment Systems for CANDU Nuclear Power Plants". Positive pressure testing is not possible at Pickering so the conditions listed (sub clauses a-g) must all be met. As discussed earlier, the 1996 version of the Standard (which did not include clause 7.4, which covers topics such as visual inspection, confirmation of lack of known defects, in-leakage measurements, and testability of access hatches) was assessed for the Pickering B ISR. However, compliance with N287.7-08-related VB pressure testing requirements is addressed by NA44-REP-25100-00009 R000, "Pickering NGS Vaccum Building In-Service Leaakge Rate Test Requirements in Accordance with CSA N287.7-08" [B.4-18] which was developed in accordance with N287.7-08 and has been accepted by the CNSC. Therefore, this is not a gap for PSR2.

The three clauses of N287.7-08 [B.4-3] identified as having gaps in the Darlington ISR were clauses 6.1.8.2, 6.1.4, and 7.13. Clause 6.1.8.2 requires that remotely operated underwater vehicle and/or divers with imaging equipment be used to provide images of the wet side of the containment structure. This was regarded as a Darlington gap since underwater inspections will be required in the reception bay. Darlington is in the process of updating governance to comply. This gap is not applicable to Pickering since the Pickering IFBs do not interface with containment structures unlike the reception bay at Darlington.

Although OPG received concession approval from the CNSC, clauses 6.1.4 and 7.13 were listed as Darlington gaps because there is no assurance that a similar concession from the CNSC would be forthcoming for a refurbished plant. Clause 6.1.4 addresses in-service examination by means of visual inspection of accessible areas inside and outside the containment structure. However, there is no provision in Darlington governance to set requirements for invasive testing or analytical methods, as this was not a requirement in N287.7-96 (R2005) [B.4-6]. As discussed below, the code refresh review based on the current revision of the Standard N287.7-08 (R2013) [B.4-1] documented that Darlington has become compliant with Clause 6.1.4. Pickering is also compliant with Clause 6.1.4 as "visual inspections of accessible areas inside and outside the containment structure" are performed, thereby meeting the requirements of the clause.

Clause 7.13 addresses the requirement for deformation of the containment structure to be monitored during pressurization and depressurization to ensure the elastic behaviour of the containment structure. Deformation is checked and measurements are taken if present (buckling, misalignment). However, the Pickering positive pressure test plan does not specify measurements for deformation to confirm elastic behaviour of the containment structure. Pickering's periodic inspection plans (NK30-PIP-03643.2-00001 R003 [B.4-19], NA44-PIP-03643.2-00001 R002 [B.4-20], and NA44-PIP-03643.2-00002 R002 [B.4-21]) state:

> *Where movement, deformation, misalignment, abrasion, etc, not meeting criteria described in Tables 1 to 4 [of the PIP], are found on the concrete or steel structures, pertinent dimensional measurements should be taken and recorded.*

The PIPs do not appear to specify requirements for on-going measurements which would facilitate monitoring of deformation during the pressurization cycle to ensure the elastic behaviour of the containment structure. However, Clause 7.13 of CSA N287.7-08 describing deformation monitoring is a non-mandatory requirement as indicated by the use of the word "should" in the requirement. Clause 7.13 refers to Annex F for more information and this is a non-mandatory annex. Furthermore, Annex F has not been imposed as a mandatory requirement, as per license requirements. As such, this clause is not applicable and this is not a gap for PSR2.

The disposition above is based on the DNGS Nuclear Refurbishment Issue Resolution Form for Issue D011 (NK38-REP-00770-0417579 R002 [B.4-22]) which states:

> *Clause 7.13 of CSA-N287.7-08 describing deformation is a non-mandatory requirement as indicated by the use of the word should in the requirement. Clause 7.13 refers to Annex F for more information and this is a non-mandatory annex. Annex F applies to a station with permanent instrumentation installed for evaluating containment deformation, which is not the case at DNGS. Further this Annex F has not been imposed as a mandatory requirement, as per licence requirements. Because clause 7.13 is non-mandatory and because the DNGS design does not include the means to measure containment deformation, this clause is not applicable. No further action is required.*

Following the original code review for the Darlington ISR, a clause-by-clause code refresh against the current revision of the Standard N287.7-08 including Update 1 (R2013) [B.4-1] was conducted in OPG Report NK38-REP-03680-10145 R000 [B.4-23] which stated the following:

There was only one change made in Update 1 2010 (R2013). The revision made was to the resolution of dry bulb temperature (Clause 7.11.2, Table 1). Darlington has demonstrated compliance with this clause in an assessment completed in May 2013. The same assessment identified non-compliance with accuracy and repeatability requirements for dewpoint temperature. Therefore, this is identified as a gap. Actions are being taken to procure new hygrometer probes that will meet the requirements of Clause 7.11.2 Table 1 in CSA N287.7-08 Update 1 2010 (R2013).

The gaps identified in the last code review report have been addressed and the Periodic Inspection Programs (PIPs) updated accordingly. The PIPs and the OPG Nuclear governance must be updated to reference Update 1 2010 (R2013) version of the standard. This can be done upon licence renewal once Update 1 2010 (R2013) is included in the PROL.

The assessment which Darlington used to demonstrate compliance with the dry bulb temperature issue [B.4-23] also addressed Pickering (the assessment states that the current temperature elements used for leakage rate testing meet the required resolution of 0.06°C in CSA N287.7-08 Update 1), and therefore this is not a PSR2 gap. However, the same assessment identified non-compliances for Pickering for the repeatability of the existing pressure transmitters and accuracy of the existing moisture transmitters, as discussed below.

On February 11, 2016 and March 14, 2016, OPG submitted concession requests [B.4-25] [B.4-26] to the CNSC for deviation from Clause 7.11.2, Table 1 requirements of CSA N287.7-08 for the use of existing pressure and moisture transmitters during the 2016 Pickering Unit 4 and Unit 8 containment leak rate tests. The repeatability of the existing station pressure transmitters and accuracy of the existing station moisture transmitters have specifications which differ from the CSA requirements identified in Clause 7.11.2, Table 1 of N287.7-08. The CNSC accepted the above concession requests on February 15, 2016 and April 4, 2016 respectively [B.4-27][B.4-28].

OPG requested a clarification from the CSA committee on the repeatability value in Table 1 of N287.7-08 for future exemptions for the use of the existing pressure transmitters. Based on the CSA response, the CNSC may determine a path forward for the CSA standard and future concession requests by licensees [B.4-29]. An update on the pending response from the CSA could not be found. Until CSA clarification is received and a path forward is directed by the CNSC, CNSC consent will continue to be required in the future to exempt the repeatability requirements in Clause 7.11.2, Table 1 of N287.7-08. Since OPG must also continue to request exemption from the accuracy requirements for the existing moisture transmitters for future containment leak rate tests, compliance with Clause 7.11.2 Table 1 in CSA N287.7-08 has not been addressed for Pickering NGS and is therefore **PSR2 CSA N287.7 Gap #1**. Note that the gap against N287.7-08 (R2013) identified in the code refresh review report [B.4-23] is listed in the Darlington IIP report NK38-REP-03680-10185 R002 [B.4-30] as completed per Darlington IIP gap IIP-OI-065 [B.4-31].

As discussed above, the code refresh review report states that the gap for clause 6.1.4 noted in the NK38-REP-03680-10061-R000 [B.4-14] has been addressed:

> *Periodic Inspection Programs documents for RB, Unit 0 and VB containment structures … have been revised to include provisions for non-destructive testing, invasive testing and analytical methods. These can be found in Sections 5.6.2, 5.6.3 and 5.6.4 in each of the PIPs. Therefore, Darlington is compliant with this clause.*

As discussed above, Pickering is compliant with Clause 6.1.4 as "visual inspections of accessible areas inside and outside the containment structure" are performed, thereby meeting the requirements of the clause.

B.4.2.2 Application of Post-PSR1 Reviews

In September 2009, OPG issued report N-REP-21000-10000 [B.4-32] documenting a clause-by-clause review identifying differences between the 1996 [B.4-6] and 2008 [B.4-3] versions of CSA N287.7, as well as differences between N287.7-08 and existing PIP documents for both Pickering and Darlington concrete containment structures. The following recommendations were made to incorporate changes identified in N287.7-08 into existing PIPs and governance on the administrative program:

> *Revisions to the existing Pickering and Darlington PIPs are required for compliance to CSA N287.7-08.*
>
> *Governance document N-PROC-MA-0066 and the technical specification for the inspection of post-tensioning for the Darlington Vacuum Building will have to be reviewed for compliance with CSA N287.7-08 and updated to comply with CSA N287.7-08.*

As discussed in Section B.4.2.1, the Pickering PROL Renewal Application [B.4-12] states that the Pickering PIPs are currently in compliance with either N287.7-96 [B.4-6] or N287.7-08 [B.4-3]. There are no gaps for Pickering that result from the past Pickering B ISR review of N287.7-96. Further, as discussed earlier, CSA N287.7-08 [B.4-3] is currently a licence requirement for Pickering NGS (per PROL 48.02/2018) as indicated in Appendix C.1 of the R04 Pickering Licence Conditions Handbook [B.4-3]. The CNSC has accepted the Pickering 1,4 and Pickering Units 5-8 PIPs with respect to the Compliance Verification Criteria related to N287.7-08 per the Pickering NGS Licence Conditions Handbook [B.4-4], which states:

> *CNSC staff have accepted the Pickering NGS-A and B PIP documents (e-Doc 4452432).*

Therefore, the N-REP-21000-10000 [B.4-32] recommendation to revise "the existing Pickering and Darlington PIPs are required for compliance to CSA N287.7-08" is not a gap for Pickering PSR2.

Compliance against the current version of the standard (N287.7-08 including Update 1 (R2013) [B.4-1]), and its applicability to Pickering, was discussed earlier.

In 2013, the CNSC conducted a Type II Inspection on the implementation of the Pickering Units 5-8 PIPs for N287. 7 [B.4-33]. Two relevant Action Notices (ANs) were raised by the CNSC and subsequently closed [B.4-34] (hence, AN2 and AN4 are not gaps):

1. AN2: ensure PIP documents make reference to the correct and up to date documents and meet the recording requirements of OPG procedures in order to become compliant with N-STD-MA-0021 sub-section 1.2.7 (b) 1.

2. AN4: ensure alignment of record reviewer qualification level with station document requirements in order to become compliant with N-STD-MA-0021 sub-section 1.4.

Also, per NK30-CORR-00531-07118 R000 [B.4-35], OPG initiated a Regulatory Management action to provide the CNSC with the latest Dow Corning 995 material test report in response to AN3 raised in the CNSC Type II Inspection [B.4-34] noted above. AN3 concerns the Dow Corning sealant repair material used in the Unit 6, Embedded Part 12679 repair. The work is currently in progress at the Kinectrics laboratories, and OPG plans to submit this to the CNSC by June 30, 2016 (i.e., the action is still in progress). Therefore, this is identified as **PSR2 CSA N287.7 Gap #2**.

In 2011, the CNSC conducted a Type II Inspection on CSA N287.7 PIPs for Concrete Containment Structure Inspections and Leakage Rate Tests and raised six Action Notices for OPG [B.4-36]. OPG responded to the CNSC that "these items have been reviewed and deemed minor in terms of risk to safety and represent opportunities for improvement" [B.4-37]. Therefore, this is not a gap for Pickering PSR2.

Finally, the Action List from the Pickering Units 5-8 Continued Operations Plan [B.4-38] identifies three actions related to N287.7: Appendix A Actions #31, #32, and #33. IIP Action #31 involved submission of Periodic Inspection Plans and Life Cycle Management Plans for a number of safety-significant civil structures. IIP Action #32 involved submission of Aging Management Plans for concrete containment structures to the CNSC for acceptance. IIP Action #33 involved revising the Reactor Building Periodic Inspection Plan and submitting to the CNSC for acceptance. All three of these actions have been closed; however, they only address fitness for service "to end of mission time" (which will need to be extended for Pickering operation past 2020) or older (non-current) versions of CSA N287.7. Therefore, they will need to be re-addressed for Pickering operation past 2020. This is identified as **PSR2 CSA N287.7 Gap #3**.

B.4.3 Compliance Summary for Pickering PSR2

There are three PSR2 CSA N287.7 gaps which all relate to Safety Factor 4:

1. N287.7-08 clause 7.11.2 Table 1 involving non-compliance with accuracy and repeatability requirements for dewpoint temperature was a gap for Darlington. No evidence can be found that this has been addressed for Pickering NGS. This is therefore a gap for Pickering PSR2.

2. OPG initiated a Regulatory Management action to provide the CNSC with the latest Dow Corning 995 material test report in response to an Action Notice raised in the CNSC Type II Inspection. The work is currently in progress. Therefore, this is a gap for Pickering PSR2.

3. Actions #31, #32, and #33 from the Pickering Units 5-8 Continued Operations Plan are related to N287.7. (IIP Action #31 involved submission of Periodic Inspection Plans and Life Cycle Management Plans for a number of safety-significant civil structures. IIP Action #32 involved submission of Aging Management Plans for concrete containment structures to the CNSC for acceptance. IIP Action #33 involved revising the Reactor Building Periodic Inspection Plan and submitting to the CNSC for acceptance.) All three of these actions have been closed; however, they only address fitness for service "to end of mission time" (which will need to be extended for Pickering operation past 2020) or older (non-current) versions of CSA N287.7. Therefore, the actions will need to be re-addressed for Pickering operation past 2020 and this is a PSR2 gap.

B.4.4 References

[B.4-1] CSA Standard N287.7-08 (R2013), *In-Service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants*, May 2008; Update No. 1: September 2010.

[B.4-2] CNSC Acts and Regulations web page: *Regulatory Documents*, *http://cnsc-ccsn.gc.ca/eng/acts-and-regulations/regulatory-documents/index.cfm#R14,* April 2016.

[B.4-3] CSA Standard N287.7-08, *In-Service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants*, May 2008.

[B.4-4] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.4-5] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.4-6] CSA Standard N287.7-96 (R2005), *In-Service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants*, July 1996.

[B.4-7] OPG Report, NK30-REP-03680-00002 R000, *Pickering NGS-B Integrated Safety Review - Actual Condition of Systems, Structures and Components Safety Factor Report*, May 2008.

[B.4-8] OPG Station Condition Record, SCR N-2008-01932, *Documentation Changes Required to Ensure Compliance with CSA Standard N287.7*, March 31, 2008.

[B.4-9] OPG Procedure, N-PROC-MA-0066 R005, *Administrative Requirements for In-Service Examination and Testing for Concrete Containment Structures,* June 13, 2014*.*

[B.4-10] CAN/CSA Standard N287.7-96 (R2000), In-Service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants, July 1996.

[B.4-11] OPG Letter, NA44-CORR-00531-00381 R000, R. J. Strickert to J.S.C. Tong, Pickering A - Updated Basis for Return to Service Document, April 20, 2001.

[B.4-12] OPG Letter, P-CORR-00531-03719 R000, G. Jager to M. A. Leblanc, *Application for Renewal of Pickering Nuclear Generating Station Power Reactor Operating Licence*, July 4, 2012.

[B.4-13] OPG Plan, N-PLAN-01060-10004 R002, *Aging Management Plan for Concrete Containment Structures*, November 2014.

[B.4-14] OPG Report, NK38-REP-03680-10061 R000, Review of CAN/CSA-N287.7-08 (May 2008), *In-Service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants for Darlington Integrated Safety Review*, September 2011.

[B.4-15] OPG Plan, NA44-PIP-03643.2-00003 R002, *Pickering Nuclear GS – Vacuum Building Post Tensioning Rods Periodic Inspection Program*, April 2014.

[B.4-16] OPG Letter, NA44-CORR-00531-07372 R000, *Pickering NGS – Vacuum Building Post Tensioned Rods Inspections – Submission of Periodic Inspection Program*, June 20, 2014.

[B.4-17] CNSC Letter, File 2.01, e-Doc 4385286, (OPG File No. P-CORR-00531-04287 R000), *Pickering NGS: Vacuum Building Post Tensioned Rods Periodic Inspection Program Document*, July 16, 2014.

[B.4-18] OPG Report, NA44-REP-25100-00009 R000, *Pickering NGS Vaccum Building In-Service Leaakge Rate Test Requirements in Accordance with CSA N287.7-08,* November 2, 2012.

[B.4-19] OPG Plan, NK30-PIP-03643.2-00001 R003, *Pickering Nuclear GSB – Reactor Building Periodic Inspection Program*, February 2014.

[B.4-20] OPG Plan, NA44-PIP-03643.2-00001 R002, *Pickering Nuclear GSA – Reactor Building Periodic Inspection Program*, February 2014.

[B.4-21] OPG Plan, NA44-PIP-03643.2-00002 R002, *Pickering Nuclear GS – PRD & VB Periodic Inspection Program*, February 2014.

[B.4-22] OPG Report, NK38-REP-00770-0417579 R002, *Nuclear Refurbishment Issue Resolution Form – Darlington – Issue #D011 – Changes to In-Service Examination and Testing Requirements for Concrete Containment Structures*, April 2015.

[B.4-23] OPG Memorandum, N-CORR-34200-0429300 R02, *Compliance Assessment of Massleak II vs 2.2.9 and leakage rate test instrumentation,* February 18, 2014.

[B.4-24] OPG Report, NK38-REP-03680-10145 R000, *Code Refresh Review of CSA N287.7-08 Update 1 (2010) In-Service Examination And Testing Requirements For Concrete Containment Structures For CANDU Nuclear Power Plants*, January 2014.

[B.4-25] OPG Letter, P-CORR-00531-04665 R000, B. McGee to M. Santini, *Pickering NGS – Concession Request for Instrumentation Related to Unit 4 Containment Leak Rate Testing*, February 11, 2016.

[B.4-26] OPG Letter, NK30-CORR-00531-07207 R000, B. McGee to M. Santini, *Pickering NGS – Concession Request for Instrumentation Related to Unit 8 Containment Leak Rate Testing*, March 14, 2016.

[B.4-27] CNSC Letter, File No. 2.01, e-Doc 4937932 (OPG File No. NA44-CORR-00531-07599 R000), M. Santini to B. McGee, *Pickering NGS: Concession Request for Instrumentation Related to Unit 4 Containment Leak Rate Testing*, February 15, 2016.

[B.4-28] CNSC Letter, File No. 2.01, e-Doc 4970098 (OPG File No. NK30-CORR-00531-07225 R000), M. Santini to B. McGee, *Pickering NGS: Concession Request for Instrumentation Related to Unit 8 Containment Leak Rate Testing*, April 4, 2016.

[B.4-29] OPG Correspondence, NK38-CORR-00531-17171 R000, *Clarification – Darlington NGS: CNSC Concession Request for Instrumentation Related to Containment/Vacuum Building Leak Rate Testing*, December 15, 2014.

[B.4-30] OPG Report, NK38-REP-03680-10185 R002*, Darlington NGS Integrated Implementation Plan (IIP)*, April 2015.

[B.4-31] OPG Report, NK38-REP-03680-10241, *IIP-OI 065 – Line Item Completion – Hygrometer Probe Requirements*, April 2015.

[B.4-32] OPG Report, N-REP-21000-10000 R000, *Gap Analysis Between CSA Standards CAN/CSA-N287.7-96 AND CAN/CSA-N287.7-08*, September 2009.

[B.4-33] CNSC Letter, File No. 4.01.02, e-Doc 4176788, (OPG File No. NK30-CORR-00531-06645 R000), *Pickering 5-8 - CNSC Type II Inspection on the Implementation of the CSA N285.4, N285.5 and N287.7 Periodic Inspection Programs (PIPs)*, July 25, 2013.

[B.4-34] CNSC Letter, File No. 4.01.03, e-Doc 4536057, (OPG File No. NK30-CORR-00531-06851 R000), *Pickering Units 5 to 8: CNSC Type II Compliance Inspection, Implementation of CSA N285.4, N285.5 and N287.7 Periodic Inspection Programs, Action Item 2013-8-4515*, October 17, 2014.

[B.4-35] OPG Letter, NK30-CORR-00531-07118 R000, *Response to CNSC Action Item 2013-8-4515, AN3 - Pickering Units 5 to 8: Implementation of CSA N285.4, N285.5 and N287.7 Periodic Inspection Program, Report PRPD-2013-182*, October 9, 2015.

[B.4-36] CNSC Letter, File No. 4.01.02, e-Doc 3904324, (OPG File No. NA44-CORR-00531-06929 R000), *Pickering NGS-A - CNSC Type II Compliance Inspection Report: Pickering A Generating Station, CSA N287.7 Periodic Inspection Programs (PIPs) for Concrete Containment Structure Inspections and Leakage Rate Tests", Report #PRPD-PICKA-2011-135, New Action Item 2012-4-3271*, May 3, 2012.

[B.4-37] OPG Letter, NA44-CORR-00531-06943 R000, *Pickering NGS - CNSC Type II Inspection Report on CSA N287.7 Implementation at PNGS-A - OPG Response to Action Item 2012-4-3271*, July 3, 2012.

[B.4-38] OPG Plan, NK30-PLAN-00531-00001 R005, *Pickering 5-8 Continued Operations Plan*, December 2015.

B.5 CNSC RD/GD-210, "Maintenance Programs for Nuclear Power Plants"

B.5.1 Background

The following paraphrased from the Preface of CNSC RD/GD-210 [B.5-1] provides a brief overview of the purpose of this Standard and the requirements expressed therein:

> *Effective maintenance is essential for the safe operation of a nuclear power plant. The range of maintenance activities includes monitoring, surveillance, inspection, testing, assessment, calibration, service, overhaul, repair and replacement of parts. The scope of the maintenance program covers all structures, systems or components (SSCs) within the bounds of the nuclear power plant.*
>
> *Regulatory document RD/GD-210 sets out the requirements of the CNSC with regard to maintenance programs for nuclear power plants (NPPs). A nuclear power plant maintenance program consists of policies, processes and procedures that provide direction for maintaining SSCs of the plant.*

RD/GD-210 is relevant to Safety Factor 3 (Equipment Qualification) and Safety Factor 4 (Aging).

Compliance with CNSC Regulatory Standard S-210, "Maintenance Programs for Nuclear Power Plants" [B.5-2] is currently a licence requirement for Pickering NGS (per PROL 48.02/2018) as indicated in Appendix C.2 of the R04 Pickering Licence Conditions Handbook [B.5-3]. The current revision of the document, which supersedes S-210, is RD/GD-210 (November 2012) [B.5-1]. RD/GD-210 reaffirms the existing requirements found in S-210, and adds information and guidance on how these requirements may be met. These guidelines provide additional information licensees may use in developing a Maintenance Program.

The results of PSR1 S-210 and RD/GD-210 reviews (Pickering B and Darlington ISRs), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.5.2. As identified in Reference [B.5-4], the Pickering PSR2 review of RD/GD-210 is an Incremental Review. As discussed in Section 2.0 of this Report, PSR2 Incremental Review includes an assessment of the intent of recent changes to the L/R/C/S on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.
- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.5.2 Compliance Assessment for Pickering PSR2

B.5.2.1 Application of PSR1 Reviews

The versions of RD/GD-210 or S-210 subject to previous reviews conducted for Darlington and Pickering, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

A review of S-210 was not included as part of the Pickering B ISR. However, as discussed below, the Darlington ISR prepared a clause-by-clause review against the current revision of RD/GD-210, the findings of which are programmatically applicable to Pickering NGS.

Darlington NGS

The Darlington ISR was originally performed against S-210 [B.5-2], comprising a clause-by-clause review documented in OPG Report NK38-REP-03680-10006 R000 [B.5-5]. The Section 4.0 conclusions of NK38-REP-03680-10006 R000 identified one gap:

> *The Clause by Clause Review of the CNSC S-210 document showed that the governing documents adequately addressed most sections of the code. Due to outstanding SCR actions on Time Based preventative maintenance, it was determined that one area is non-compliant and a gap exists for section 5.3.1.*

Appendix B of NK38-REP-03680-10006 R000 lists the ISR gaps that have been reclassified or added and addresses the impact of these changes. In the case of the gap against Section 5.3.1 noted above, it was reclassified as "Compliant" with SCRs related to the gap closed and all actions completed per the supporting rationale and references provided in Appendix B of the report. Thus, the Darlington ISR demonstrated OPG governance and practices are compliant with S-210 [B.5-2].

Following the original code review for the Darlington ISR, a clause-by-clause code refresh review against the current revision of RD/GD-210 (2012) [B.5-1] was conducted and documented in OPG Report NK38-REP-03680-10164 R000 [B.5-6]. NK38-REP-03680-10164 R000 concluded the following regarding compliance with RD/GD-210:

- *The Clause by Clause Review of the CNSC RD/GD-210 (November 2012) document showed that the governing documents adequately addressed all sections of the code.*
- *Documents were found to be current and appropriately posted.*

NK38-REP-03680-10164 R000 also confirmed that SCRs related to the gap from the review of Darlington against CNSC S-210 have been closed and all actions completed.

The clauses in RD/GD-210 are programmatic from the perspective of the specified requirements while recognizing that maintenance programs are specific to each station. Therefore, the Darlington ISR conclusions are applicable to Pickering PSR2.

B.5.2.2 Application of Post-PSR1 Reviews

In 2015, the CNSC conducted a Type II Compliance Inspection on maintenance work execution at Pickering NGS [B.5-7]. The CNSC concluded:

> *Based on the scope of the field inspections conducted February 23-27, 2015, CNSC staff concludes that the licensee met the regulatory requirements of RD/GD-210 that were within the scope of the inspection. It did not however meet requirements of one of its documents that address completing SATM [Space Allocation for Transient Material] permit applications during work package development or work planning process.*

The Action Notice (AN) raised by the CNSC under Action Item 2015-48-6450 does impact OPG's compliance with RD/GD-210 as the AN concerns an OPG compliance with its own governance related to space allocation for transient material:

> *In order for Ontario Power Generation to become compliant with N-INS-09070-10001 R002, CNSC staff request Ontario Power Generation to develop and implement a corrective action plan to ensure that when required, work is performed with a SATM permit.*

Reference [B.5-8] contains OPG's response to the AN, which states that the finding in the AN was previously identified by OPG during an internal Nuclear Audit (NO-2014-017). To address their audit finding, OPG initiated SCR P-2014-30948, "Space Allocation and Transient Material (SATM) non-compliances", and subsequent Corrective Action Plan (CAP) and Assignment Actions (per Action Request # 28174181). OPG's response in Reference [B.5-8] states that this existing CAP addresses the issue in the AN. In Reference [B.5-9] the CNSC staff agreed with OPG's CAP which covers the concern expressed in the AN, but stated that "CNSC staff will wait for the completion of the CAP before closing AN1". Action Request # 28174181 was subsequently completed on February 11, 2016. Therefore, there is no gap for Pickering PSR2. In addition, operation of Pickering NGS past 2020 does not impact compliance with this AN.

Since Pickering A Return to Service [B.5-10] preceded the advent of the Regulatory Standard S-210 [B.5-2], no code review of the Standard was performed in support of the Pickering A Restart. However, the past ISR conclusions are applicable for Pickering NGS as a whole.

B.5.3 Compliance Summary for Pickering PSR2

There are no PSR2 gaps for CNSC RD/GD-210 (2012) [B.5-1]. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with RD/GD-210 (2012).

B.5.4 References

[B.5-1] CNSC Regulatory Document RD/GD-210, *Maintenance Programs for Nuclear Power Plants*, November 2012.

[B.5-2] CNSC Regulatory Standard S-210, *Maintenance Programs for Nuclear Power Plants*, July 2007.

[B.5-3] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.5-4] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.5-5] OPG Report, NK38-REP-03680-10006 R000, *Review of CNSC S-210 (July 2007) Maintenance Programs for Nuclear Power Plant for Darlington Integrated Safety Review*, September 2011.

[B.5-6] OPG Report, N-REP-03680-10164 R000, *Code Refresh Review of CNSC-RD/GD210 210 (November 2012) Maintenance Programs For Nuclear Power Plant*, February 2014.

[B.5-7] CNSC Letter, File No. 4.01.03, e-Doc 4719160, (OPG File No. P-CORR-00531-04463 R000), M. Santini to B. McGee, *Pickering NGS: CNSC Type II Compliance Inspection Report: PRPD-2015-002, Maintenance Work Execution, New Action Item 2015-48-6450*, May 8, 2015.

[B.5-8] OPG Letter, P-CORR-00531-04486, B. McGee to M. Santini, *Pickering NGS: CNSC Type II Compliance Inspection, Maintenance Work Execution Report: PRPD-2015-002 - CNSC Action Item 2015-48-6450*, July 7, 2015.

[B.5-9] CNSC Letter, e-Doc 4802616 (OPG File No. P-CORR-00531-04522), M. Santini to B. McGee, *Pickering NGS: CNSC Type II Compliance Inspection, Maintenance Work Execution Report: PRPD-2015-002 - CNSC Action Item 2015-48-6450*, July 15, 2015.

[B.5-10] OPG Letter, NA44-CORR-00531-00381 R000, R. J. Strickert to J.S.C. Tong, *Pickering A - Updated Basis for Return to Service Document*, April 20, 2001.

B.6 CNSC RD/GD-98, "Reliability Programs for Nuclear Power Plants"

B.6.1 Background

The following paraphrased from Section 1, "Introduction", of CNSC Regulatory Document RD/GD-98 (2012) [B.6-1] provides a brief overview of the purpose of this Standard and the requirements expressed therein:

> *A reliability program assures that the systems important to safety shall meet their defined design, and performance criteria at acceptable levels of reliability throughout the lifetime of the facility. Regulatory document RD/GD-98 sets out the requirements and guidance of the CNSC for the development and implementation of a reliability program, including reliability assessment, modelling, evaluation, and monitoring.*

RD/GD-98 is applicable to Safety Factor 3 (Equipment Qualification) and Safety Factor 4 (Aging) since CNSC web site [B.6-2] identifies the Regulatory Document as being relevant to the Safety Control Area for "Fitness for Service".

Compliance with RD/GD-98 is currently a licence requirement for Pickering NGS (per PROL 48.02/2018) as indicated in Appendix C.2 of the R04 Pickering Licence Conditions Handbook [B.6-3]. The current revision of the Regulatory Document is RD/GD-98 (2012) [B.6-1]. RD/GD-98 captures the existing requirements previously found in CNSC Regulatory Standard S-98 (Revision 1), "Reliability Programs for Nuclear Power Plants" [B.6-4]. As per Darlington ISR code refresh review report NK38-REP-03680-10167 R000 [B.6-5], there are no changes to the mandatory requirements of RD/GD-98 (2012) relative to S-98 (Revision 1). The non-mandatory changes made in RD/GD-98 (2012) relative to S-98 (Revision 1) consist of a new section on guidelines. These guidelines provide additional information licensees may use in developing a Reliability Program.

The results of PSR1 S-98 and RD/GD-98 reviews (Pickering B and Darlington ISRs), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.6.2. As identified in Reference [B.6-6], the Pickering PSR2 review of RD/GD-98 is an Incremental Review. As discussed in Section 2.0 of this Report, PSR2 Incremental Review includes an assessment of the intent of recent changes to the L/R/C/S on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.
- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.6.2 Compliance Assessment for Pickering PSR2

B.6.2.1 Application of PSR1 Reviews

The versions of RD/GD-98 or S-98 subject to previous reviews conducted for Darlington and Pickering, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

<u>Pickering Units 5-8</u>

The Pickering B ISR performed a clause-by-clause review of S-98 (Revision 1) [B.6-4]. The review was documented in report NK30-REP-03680-00005 R000 [B.6-7] and concluded:

> *Recognizing that at a high level S-98 has been established and OPG is currently complying with the standard, there still remain details with respect to methods and guidance that are being further developed. This was noted in Reference [R-44], where further improvements were deemed desirable, but were not considered compliance issues. However, for this reason, both the CNSC and the industry are developing 'guide' type documents that will provide more guidance with respect to acceptable approaches for complying with S-98. It is noted in Reference [R-46], that the COG working group is recommending that further guidance be developed in the following areas…Hence, this review finds that Pickering B is in Direct Compliance with all clauses of S-98.*

Based on the above, there are no PSR2 gaps relating to Pickering B ISR compliance with S-98 (Revision 1) [B.6-4].

<u>Pickering Units 1-4</u>

For Pickering A Return to Service, the active version of the Regulatory Document at the time was as an Atomic Energy Control Board (AECB) Consultative Document, C-98 [B.6-8]. Per NA44-CORR-00531-00381 R000 [B.6-9], OPG did not perform a code review of the Standard on the basis that it "Pertains mostly to Design Support Analysis". However, given the programmatic applicability as discussed under the Darlington review below, a review specifically for Pickering A is not required. Furthermore, the Pickering B ISR conclusions are applicable to all of Pickering NGS.

Section 4.6, "Risk and Reliability Program" of the Pickering PROL Renewal Application [B.6-10] notes that implementation standard N-STD-RA-0033, "Reliability Monitoring and Reporting of Systems Important to Safety" [B.6-11] has been created to provide direction on carrying out reliability program activities at the station consistent with Regulatory Standard S-98, which was the predecessor of RD/GD-98. As stated below for the Darlington ISR, there are no changes to the mandatory requirements of CNSC RD/GD-98 (R2012) relative to S-98 Revision 1. Therefore, OPG Nuclear governance remains compliant with CNSC RD/GD-98 (R2012) and N-STD-RA-0033 is consistent with RD/GD-98.

Furthermore, the Pickering Licence Conditions Handbook [B.6-3] notes OPG's required compliance with RD/GD-98 and the CNSC's associated yearly review of the Pickering annual reliability reports [B.6-12] [B.6-13]:

> *CNSC Regulatory Document RD/GD-98, "Reliability Programs for Nuclear Power Plants", has recently been issued and replaces S-98 in the regulatory framework. Requirements set out in the newly issued document remain unchanged from those established in S-98.*
>
> *OPG's key governing documents for the reliability program are listed in the written notification table below. OPG has developed the lists of systems important to safety for both Pickering NGS-A and Pickering NGS-B as required by RD/GD-98 (using a methodology developed by COG). The systems important to safety, along with their unavailability target, are documented in NA44-REP-03611-00004 "Pickering A Systems Important to Safety" and NK30-REP-03611-00024 "Pickering B Systems Important to Safety", respectively.*
>
> *CNSC staff will review the annual report on risk and reliability required by REGDOC 3.1.1 to ensure the performance of systems important to safety meet their reliability requirements. See section 4.3 for version control of REGDOC 3.1.1.*

Darlington NGS

The Darlington ISR was originally performed using S-98 (Revision 1) [B.6-4], comprising a clause-by-clause review documented in OPG Report NK38-REP-03680-10005 R000 [B.6-14]. Section 4.0, "Conclusions of Darlington ISR code review report", identified three gaps:

> *… three gaps in Darlington's reliability program activities. It should be noted that these gaps are not directly related to plant design; they are related to reporting and performance characterization. Two of these gaps are the result of differences between the technical methodology specified in the OPG Governance and the requirements of the CNSC standard, while the third gap is a failure in a specific instance of practice to adhere to the requirements of the OPG procedures.*

Appendix B of NK38-REP-03680-10005 R000 lists the ISR gaps that were reclassified or added and addresses the impact of these changes on the code review report. In the case of the three gaps noted above, all three were reclassified as "Compliant" with supporting rationale and references. Thus, the Darlington ISR demonstrated OPG governance and practices were compliant with S-98 (Revision 1) [B.6-4].

Following the original code review for the Darlington ISR, a clause-by-clause code refresh review against the current revision of RD/GD-98 (2012) [B.6-1] was conducted and documented in OPG Report NK38-REP-03680-10167 R000 [B.6-5]. NK38-REP-03680-10167 R000 concluded that OPG is compliant with RD/GD-98 (2012):

> *There are no changes to the mandatory requirements of CNSC RD/GD-98 (R2012) relative to S-98 Revision 01-2005. Therefore, OPG Nuclear governance remains compliant with CNSC RD/GD-98 (R2012).*

The clauses in RD/GD-98 (2012) are programmatic from the perspective of the specified requirements while recognizing that reliability monitoring and reporting documented in Annual Reliability Reports (e.g., [B.6-12], [B.6-13], [B.6-15]) are specific to each station. Therefore, the Darlington ISR conclusions are applicable to Pickering NGS and no PSR2 gaps result from the Darlington ISR reviews.

B.6.2.2 Application of Post-PSR1 Reviews

In 2014, the CNSC conducted a Type II Compliance Inspection of the reliability program at Pickering NGS [B.6-16]. One relevant Action Notice (AN) was raised by the CNSC under Action Item 2015-48-5864 to ensure that all component failures were identified. OPG responded [B.6-17] to the CNSC that OPG procedures already exist at Pickering that address the AN, that is P-INS-03611-00001 R000, "Pickering Reliability Instruction" [B.6-18], and P-GUID-03611-00004 R000, "Fault Data Monitoring and Recording" [B.6-19] (which is a supporting guide to P-INS-03611-00001). OPG's response to the CNSC concluded [B.6-17]:

> *With respect to the monitoring of the performance of components, OPG procedures P-INS-03611-00001 and P-GUID-03611-00004 are in alignment with Section 3.6.3.2 of Regulatory Document RD/GD-98, "Reliability Programs for Nuclear Power Plants" which states:*
>
> *"The performance or condition of all components of SIS should be monitored. This monitoring of component reliability should include: ... Assessment and recording of every failure of a component that could affect the reliability of the whole system to which it belongs, as soon as practicable after the failure has been discovered."*
>
> *In summary, OPG complies with OPG procedures with respect to the identification and recording of component failures for the Systems Important to Safety. Therefore, OPG has determined that no corrective action plan is required.*

Based on the above, OPG requested closure of the Action Item 2015-48-5864, which the CNSC accepted per P-CORR-00531-04615 R000 [B.6-20]:

Canadian Nuclear Safety Commission (CNSC) staff have completed their review of Ontario Power Generation (OPG) Inc.'s response [1] to CNSC Action Item 2015-48-5864 and find it satisfactory. Action Item 2015-48-5864 is therefore closed."

Pickering operation past 2020 does not impact the above mentioned compliance with RD/GD-98.

B.6.3 Compliance Summary for Pickering PSR2

There are no PSR2 gaps for CNSC RD/GD-98 (2012) [B.6-1]. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with RD/GD-98 (2012).

B.6.4 References

[B.6-1] CNSC Regulatory Document RD/GD-98, *Reliability Programs for Nuclear Power Plants*, June 2012.

[B.6-2] CNSC Acts and Regulations web page: *Regulatory Documents*, http://cnsc-ccsn.gc.ca/eng/acts-and-regulations/regulatory-documents/index.cfm#R14, April 2016.

[B.6-3] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.6-4] CNSC Regulatory Standard S-98 (Revision 1), *Reliability Programs for Nuclear Power Plants*, July 2005.

[B.6-5] OPG Report, NK38-REP-03680-10167 R000, *Code Refresh Review of CNSC-RD/GD98 (R2012), Reliability Programs for Nuclear Power Plants, for DNGS ISR*, July 2013.

[B.6-6] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.6-7] OPG Report, NK30-REP-03680-00005 R000, *Pickering NGS B - Integrated Safety Review - Safety Analysis Review*, June 2007.

[B.6-8] AECB Consultative Document C-98 Revision 1, *Reliability of Systems Important to Safety for Nuclear Reactor Facilities*, 1998.

[B.6-9] OPG Letter, NA44-CORR-00531-00381 R000, R. J. Strickert to J.S.C. Tong, *Pickering A - Updated Basis for Return to Service Document*, April 20, 2001.

[B.6-10] OPG Letter, P-CORR-00531-03719 R000, G. Jager to M. A. Leblanc, *Application for Renewal of Pickering Nuclear Generating Station Power Reactor Operating Licence*, July 4, 2012.

[B.6-11] OPG Standard, N-STD-RA-0033 R002, *Reliability Monitoring and Reporting of Systems Important to Safety,* October 2013.

[B.6-12] OPG Report, NA44-REP-09051.1-00014 R000, *2014 Annual Reliability Report - Pickering Units 1 & 4*, March 2015.

[B.6-13] OPG Report, NK30-REP-09051.1-00014 R000, *2014 Annual Reliability Report - Pickering Units 5 to 8*, March 2015.

[B.6-14] OPG Report, NK38-REP-03680-10005 R000, *Review of CNSC S-98 Revision 1 (July 2005): Reliability Programs for Nuclear Power Plants for Darlington Integrated Safety Review*, September 2011.

[B.6-15] OPG Report, NK38-REP-01500-10015 R000, *2014 S-99 Annual Reliability Report*, March 2015.

[B.6-16] CNSC Letter, File No. 4.01.03, e-Doc 4590770, (OPG File No. P-CORR-00531-04354 R000), M. Santini to B. McGee, *Pickering NGS: CNSC Type II Compliance Inspection Report: PRPD-2014-215, Reliability, New Action Item 2015-48-5864*, January 21, 2015.

[B.6-17] OPG Letter, P-CORR-00531-04445 R000, B. McGee to M. Santini, *Pickering NGS: CNSC Type II Compliance Inspection Report, Reliability Program, Report PRPD-2014-215 - CNSC Action Item 2015-48-5864*, April 21, 2015.

[B.6-18] OPG Instruction, P-INS-03611-00001 R000, *Pickering Reliability Instruction*, November 2014.

[B.6-19] OPG Guide, P-GUID-03611-00004 R000, *Fault Data Monitoring and Recording*, June 2014.

[B.6-20] CNSC Letter, File No. 4.01.02, e-Doc 4883897, (OPG File No. P-CORR-00531-04615 R000), M. Santini to B. McGee, *Pickering NGS: CNSC Type II Compliance Inspection Report, Reliability Program, Closure of CNSC Action Item 2015-48-5864*, December 3, 2015.

B.7 CNSC REGDOC-2.6.3, "Aging Management"

B.7.1 Background

The following provides a brief overview of the purpose of CNSC REGDOC-2.6.3, "Aging Management" [B.7-1] and the requirements expressed therein:

> *REGDOC-2.6.3 sets out requirements and guidance for the appropriate and proactive management of aging throughout the different phases of a power reactor facility's lifecycle. This document provides a framework within which codes and standards can be applied so that physical aging and obsolescence of structures, systems and components important to safety are effectively managed.*

REGDOC-2.6.3 replaces RD-334, "Aging Management for Nuclear Power Plants" [B.7-2], which was published in June 2011. As per CNSC news release "CNSC publishes REGDOC-2.6.3, Aging Management" [B.7-3]:

> *REGDOC-2.6.3 does not contain new regulatory requirements. Some requirements in RD-334 have transitioned to guidance as appropriate. New guidance has been added to supplement the requirements, including some information related to lessons learned from Fukushima, providing licensees and applicants with information on how the requirements may be met.*

All of REGDOC-2.6.3 is directly relevant to Safety Factor 4 (Aging). A number of clauses in REGDOC-2.6.3 also require consideration of aging effects in the specifications for equipment qualification which are applicable to Safety Factor 3 (Equipment Qualification).

Although compliance with REGDOC-2.6.3 is not currently a licence requirement for Pickering NGS, compliance with RD-334 is as indicated in Appendix C.1 of the R04 Pickering Licence Conditions Handbook [B.7-4]. In the 2014 Regulatory Oversight Report for Canadian Nuclear Power Plants [B.7-5] the CNSC state:

> *All operating NPPs are reviewing and updating their processes and programs in accordance with the updated regulatory document [REGDOC-2.6.3]. All operating NPPs have component-specific aging management programs, also known as lifecycle management programs, for the major primary heat transport components of their CANDU reactors (feeders, pressure tubes and steam generators) and for concrete containment structures and BOP safety-related civil structures. CNSC staff conducted onsite inspections in accordance with the compliance verification program to confirm the licensees' implementation of their aging management programs.*

The results of PSR1 RD-334 and REGDOC-2.6.3 reviews performed since PSR1 have been assessed for applicability to PSR2 in Section B.7.2 below. As identified in Reference [B.7-6], the

Pickering PSR2 review of REGDOC-2.6.3 is an Incremental Review. As discussed in Section 2.0 of this Report, PSR2 Incremental Review includes an assessment of the intent of recent changes to the L/R/C/S on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.
- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.7.2 Compliance Assessment for Pickering PSR2

B.7.2.1 Application of PSR1 Reviews

Neither CNSC RD-334 nor REGDOC-2.6.3 were reviewed as part of the Darlington or Pickering B ISRs, or as part of Pickering A Return to Service. However, as discussed under Section B.7.2.2 below, a review of RD-334 has been performed since PSR1 which is applicable to both Darlington and Pickering (due to the programmatic nature of the requirements) as well as REGDOC-2.6.3 (since requirements have not changed from RD-334 to REGDOC-2.6.3).

B.7.2.2 Application of Post-PSR1 Reviews

In order to facilitate the implementation of RD-334 at OPG, the Engineering Programs Integration department performed a gap assessment between the RD-334 and OPG Nuclear Integrated Aging Management (IAM) governance in 2012. The methodology and the results of the gap assessment are documented in N-REP-01060-10012 R000 [B.7-7], with Table 1 of the document providing the RD-334 gap assessment results in the format of a clause-by-clause review. The gap assessment found that OPG Nuclear IAM governance is aligned with the aging management requirements of RD-334 given that both documents were written based on the IAEA Safety Guide NS-G-2.12 "Aging Management for Nuclear Power Plants" [B.7-8]. IAM governance and its interfacing programs were found to cover all requirements of RD-334 except the following:

- Extended Shutdown: The requirements for "Extended Shutdown" (i.e., reactor shutdown lasting for a period exceeding one year, excluding shutdowns for regular maintenance outages) are provided in the RD-334. At the time of the gap assessment, IAM governance did not specify requirements for extended shutdowns. Document Change Request (DCR) 116405 was placed against N-PROG-MP-0008, "Integrated Aging Management", and N-PROC-MP-0060, "Aging Management Process", to incorporate the required changes. Both documents have since been updated accordingly and this gap is therefore closed.

- Decommissioning: Although a corporate level decommissioning charter was in place and a fleet wide decommissioning governance has since been developed, OPG IAM governance did not address Decommissioning at the time of the gap assessment. Decommissioning is now planned and executed as a part of the OPG corporate level governance using W-PROG-WM-0003, "Decommissioning Program". Further, IAM governance now recognize decommissioning program level documents and list them in the N-PROG-MP-0008 interfacing programs. Therefore, this gap is now closed.

- Mutual link to Design Management: RD-334 prescribes design-related requirements: N-PROG-MP-0009, "Design Management" was in compliance with the RD-334 design related requirements by being in compliance with the current codes and standards. However, the terms aging management, aging, and degradation mechanisms were not explicitly mentioned in the N-PROG-MP-0009 or other implementing Design Management documents. DCR 116024 was put in place to address the Design Management program interface with the IAM program, and to readily demonstrate the mutual link between both programs. N-PROG-MP-0009 has since been updated accordingly and this gap is therefore closed.

Per N-REP-01060-10012 R000 [B.7-7], the Major Component Engineering department also reviewed RD-334 requirements versus the Major Components governance, i.e., N-PROG-MA-0025, "Major Components", N-PROC-MA-0100, "Major Components Life Cycle Management Plan", and the supporting Technical Basis documents. The review did not identify any additional gaps. Based on the above, there are no outstanding gaps related to compliance with RD-334 based on the findings of Reference [B.7-7].

Reference [B.7-9] summarizes the findings of a CNSC inspection carried out on the IAM program at Pickering NGS from July 20 to 24, 2015. OPG's response to each finding is provided in Reference [B.7-10] as summarized below:

- Finding #1: "OPG is not compliant with Power Reactor Operating Licence 48.01/2018 Licence Condition 7.1 because the governance N-PROG-MP-0008 Integrated Aging Management Program does not refer to the current regulatory requirement, RD-334 Aging Management, identified in the Licence and Licence Conditions Handbook." OPG has since revised N-PROG-MP-0008 Section 1.3.3 and N-PROC-MP-0060 Section 1.0 which now include the reference to RD-334. Therefore, this gap is now closed.

- Finding #2: "OPG is not compliant with RD-334 Section 4.7 (REGDOC-2.6.3 Section 4.7) because their governance N-PROG-MP-0008 Integrated Aging Management does not refer to a specific program for management of technological obsolescence." OPG has since revised N-PROG-MP-0008 Section 1.6.10 and N-PROC-MP-0060 Section 4.3.1 which now include the reference to N-STD-MA-0024," Obsolescence Management" for the management of technological obsolescence. Therefore, this gap is now closed.

- Finding #3: "OPG is not compliant with RD-334 Section 3.5 (REGDOC-2.6.3 Section 3.5) because the governance N-PROG-MP-0008 Integrated Aging Management does not address decommissioning." As discussed earlier, OPG has revised N-PROG-MP-0008, Section 1.3.4 which now covers decommissioning and extended shutdowns. Therefore, this gap is now closed.

- Finding #4: "OPG is not compliant with RD-334 Section 4.6 (REGDOC-2.6.3 Section 4.6) because the governance N-PROG-MP-0008 does not specifically address the nine attributes for effective aging management." OPG stated they believe that the nine attributes for effective aging are addressed through the combination of the OPG programs and processes, such as Condition Assessments, Station Condition Records and corrective action programs, Self-Assessments, System Health Reports / Component Health Reports and inspection, testing, monitoring and maintenance programs and strategies. However, Reference [B.7-10] states that OPG is in the process of initiating a formal review and documenting the relationship between the nine attributes for effective aging management and the existing OPG aging management programs and processes. Per Reference [B.7-11], the action is now complete and no further updates are planned. Therefore, there is no gap for PSR2. (Note: There is no Finding #5 specified in Reference [B.7-10].)

- Finding #6: "OPG is not compliant with N-PROC-MP-0060 Aging Management Process, Section 1.7 for not reviewing and updating the Component Condition Assessments (CCAs)[5]: within the review cycle of the component, and when new information or feedback from the program was received." OPG has reviewed the extent of condition of not reviewing and revising the CAs within the review cycle and found that 432 CAs were overdue for revision. OPG has since revised these CAs, which are now valid until 2020. Further, OPG is updating CAs to address potential operation past 2020 as part of Pickering PSR2. OPG has stated they will develop an implementation plan to prevent reoccurrence of: a) not reviewing and revising the CAs within the review cycle, and b) not updating the CAs when pertinent new information becomes available. OPG stated they will provide an update and a target implementation date on this action to the CNSC by October 30, 2016. Therefore, this is an open gap (**PSR2 CNSC REGDOC-2.6.3 Gap #1**).

- Finding #7: "OPG is not compliant with N-PROC-MA-0077, "Critical Equipment Identification and Categorization", Section 1.2 because the Reactor Safety (RS) category code and rationale for critical components was not always accurate or consistently applied in the CCAs." OPG has stated that they have since completed a review and update of the RS category code and rationale for a portion of the

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[5] The terminology currently used is Condition Assessment (CA) instead of Component Condition Assessment (CCA).

components to become fully compliant with N-PROC-MA0077, "Critical Equipment Identification and Categorization". All components previously coded as RS1, RS2 or RS3 have been updated with the correct RS code. Components coded as RS4 or N/A have been updated with the correct RS code. Any components that were ranked RS1 per the Fussell-Vesely or Risk Achievement Work criteria were also updated for Pickering Units 1,4 and 5-8. Further, to ensure that RS codes remain accurate the normal update process will be used after any future revisions of Probabilistic Safety Assessments, Operational Safety Requirements or N-PROC-MA-0077. This will ensure that RS codes remain accurate and consistent. OPG has stated that a review of the CAs will be conducted to ensure consistency with the revised RS codes and an update will be provided to the CNSC by October 30, 2016. Therefore, this is an open gap (**PSR2 CNSC REGDOC-2.6.3 Gap #2**).

- Finding #8: "OPG is not compliant with N-INS-01071-10000 System Health Reporting for not ensuring AM [Aging Management] actions identified in the HX [Heat Exchanger] Component Health Reports (CHRs) dashboard were captured in the PHTS [Primary Heat Transport System] System Health Reports (SHR)." OPG performed a self-assessment (P15-000853) and found that the majority of SHRs are in compliance with the requirements specified in Section 1.9.1 of N-INS-01 07110000, "System Health Reporting" and actions are addressed in the Aging Management section of SHRs. However, there were found to be instances when the SHR does not show AM related actions from CHRs. Station Condition Record SCR-P-2015-11922 was initiated to document the findings and the two recommendations identified to address the findings from this self-assessment. As identified in the recommendations, OPG has completed a roll out of a briefing card regarding the expectations that CHRs Aging Management related information is to be included in SHRs and DCR #0000131304 has been approved to enhance N-INS-01071-10000 to include relevant information from Condition Assessments on Aging Management related actions in the SHRs. This action is complete and no further updates are planned. Therefore, this gap is now closed.

With respect to demonstrating compliance specifically with REGDOC-2.6.3, this was first addressed in 2014 per NK38-REP-09701-0523075 [B.7-12] where "Appendix C Terms of Reference - Aging Management Program Integration with Reliability Program", which states:

Two phases are proposed to integrate aging management into the ongoing plant operation, these are:

Phase I Assessment

a. Conduct a gap analysis for the requirements of Draft REGDOC-2.6.3 against the maintenance, engineering components, and system health and reliability programs.

b. *Review the gaps against OPG benchmarking results performed in 2010 and the GALL report to ensure that there is a comprehensive list of gaps.*

c. *Perform a cost evaluation between enhancing the existing programs to close the gaps versus maintaining the Component Condition Assessments as a means of meeting REGDOC-2.6.3. Alternately, consider changes to CCAs based on the gap closure analysis.*

d. *Recommend a path forward based on the business case study*

Phase II Implementation

Implement recommendations from phase I

Stakeholders identified at the time were from Refurbishment Engineering, Darlington and Pickering NGS. The team was "lead by Engineering Program Integration with members from Refurbishment Engineering, Darlington Aging Management and Pickering Engineering". The outcome of the Stakeholder review team assessment was an "Approved Implementation Plan and schedule for implementation" for REGDOC-2.6.3 which is discussed in Reference [B.7-13]. Attachment 1 of [B.7-13] provides details of all the required steps of the Implementation Plan and associated milestones, namely:

- Conducting a detailed gap analysis between REGDOC-2.6.3 and OPG governance, documents, practices, and procedures.
- Preparing a White Paper summary for the affected governance, procedures, documents and practices.
- Conducting process mapping exercise with stakeholders to identify process gaps with future governance structure per white paper summary.
- Validating process maps with stakeholders and draft Change Management Plan.
- Ensuring review and acceptance of the Change Management Plan by stakeholders, i.e., Program and Process owners.
- Revising affected governance to begin implementation of the Change Management Plan.

OPG stated in [B.7-13] that the target completion date for the REGDOC-2.6.3 implementation plan is July 15, 2017. A Regulatory Management Action Request (REGM) has been entered to track this item. After implementation, [B.7-13] states that an annual effectiveness review will be performed for two consecutive years on the revised processes and procedures to ensure they execute as intended, and are sustaining. A REGM has also been entered to track these items with a final completion date of December 3, 2019.

Since [B.7-3] states that REGDOC-2.6.3 does not contain any new regulatory requirements in comparison to RD-334 (some requirements in RD-334 have transitioned to guidance as appropriate, and new guidance has been added to supplement the requirements), there are not expected to be any additional safety significant findings that will result from the above mentioned detailed gap assessment of REGDOC-2.6.3 that were not already identified above during past gap assessments and CNSC inspections of RD-334 compliance.

B.7.3 Compliance Summary for Pickering PSR2

Reference [B.7-9] summarizes the findings of a CNSC inspection carried out on the IAM program at Pickering NGS from July 20 to 24, 2015. OPG has two remaining Gaps to close as a result of the CNSC inspection that are applicable to Pickering PSR2 [B.7-10]. Both of these gaps relate to Safety Factor 4:

1. OPG is not compliant with N-PROC-MP-0060 Aging Management Process, Section 1.7 for "not reviewing and updating the Component Condition Assessments[6] within the review cycle of the component, and when new information or feedback from the program was received." OPG has since revised these CAs, which are now valid until 2020. OPG has stated they will develop an implementation plan to prevent reoccurrence of: a) not reviewing and revising the CAs within the review cycle, and b) not updating the CAs when pertinent new information becomes available. OPG stated they will provide an update and a target implementation date on this action to the CNSC by October 30, 2016. This is a gap for Pickering PSR2.

2. OPG is not compliant with N-PROC-MA-0077, "Critical Equipment Identification and Categorization", Section 1.2 because "the Reactor Safety (RS) category code and rationale for critical components was not always accurate or consistently applied in the CCAs." OPG has stated they have since completed a review and update of the RS category code and rationale for a portion of the components to become fully compliant with N-PROC-MA-0077. However, OPG has stated that a review of the CAs will be conducted to ensure consistency with the revised Reactor Safety codes and that an update will be provided to the CNSC by October 30, 2016. This is a gap for Pickering PSR2.

B.7.4 References

[B.7-1] CNSC REGDOC-2.6.3, *Aging Management*, March 2014.

[B.7-2] CNSC RD-334, *Aging Management for Nuclear Power Plants*, June 2011.

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[6] The terminology currently used is Condition Assessment (CA) instead of Component Condition Assessment (CCA).

[B.7-3] CNSC, *News Releases: CNSC publishes REGDOC-2.6.3, Aging Management*, http://nuclearsafety.gc.ca/eng/resources/news-room/news-releases/index.cfm?news_release_id=496, March 19, 2014.

[B.7-4] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.7-5] CNSC Letter, e-Doc word 4778715 / pdf 4778718 (OPG File No. P-CORR-00531-04495), L. Levert to B. McGee, *Presentation of the 2014 NPP Report*, June 16, 2015.

[B.7-6] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.7-7] OPG Report, N-REP-01060-10012 R000, *Gap Assessment – Integrated Aging Management Governance vs. RD-334*, November 5, 2012.

[B.7-8] IAEA Safety Guide NS-G-2.12, *Aging Management for Nuclear Power Plants*, 2009.

[B.7-9] CNSC Letter e-Doc 4852894, File 4.01.03 (OPG File No. P-CORR-00531-04562), M. Santini to B. McGee, *Pickering NGS: CNSC Type II Compliance Inspection Report: PRPD-2015-015, Integrated Aging Management Program, New Action Item 201548-7043*, October 6, 2015

[B.7-10] OPG Letter P-CORR-00531-04596, B. McGee to M. Santini, *CNSC Action Item 2015-48-7043 - Pickering NGS: CNSC Type II Compliance Inspection Report, Integrated Aging Management Program, # PRPD-2015-015*, December 4, 2015.

[B.7-11] OPG Letter P-CORR-00531-04710, B. McGee to M. Overton, *CNSC Action Item 2015-48-7043, Directive D2 Update - Type II Compliance Inspection Report, Integrated Aging Management Program, # PRPD-2015-015*, May 13, 2016.

[B.7-12] OPG Report, NK38-REP-09701-0523075, *Review of CNSC Draft REGDOC-2.3.3*, January 12 2015.

[B.7-13] OPG Letter NK38-CORR-00531-17440 P, B. Duncan to F. Rinfret, *Darlington NGS - Transition Implementation Plan for New Regulatory Document REGDOC-2.6.3, "Fitness for Service - Aging Management"*, June 11, 2015.

B.8 CSA N287.2-08, "Material Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants"

B.8.1 Background

The following paraphrased from the Preface of CSA N287.2-08 (R2013) [B.8-1] provides a brief overview of the purpose of this Standard and the requirements expressed therein:

> *The N287 standards were initiated in response to the recognition by the utilities and industries concerned with nuclear structures in Canada of a need for consistent standards for the design, construction, and testing of concrete containment structures for CANDU nuclear power plants.*
>
> *CSA N287.2 provides the requirements for materials used in concrete containment structures of containment systems designed as class containment components, parts, and appurtenances in CANDU nuclear power plants. Together with the other CSA N287 Standards, it provides the requirements for concrete containment structures for CANDU nuclear power plants.*

All of N287.2-08 is directly relevant to Safety Factor 2 (Actual Condition of SSCs), and potentially applicable to Safety Factor 3 (Equipment Qualification) and Safety Factor 4 (Aging) since CNSC web site [B.8-2] lists the Standard as being relevant to the Safety Control Areas on "Fitness for Service" and "Physical Design". On this basis, N287.2-08 is deemed applicable to Safety Factors 1, 2, 3 and 4.

CSA N287.2 is identified in Appendix E.1 of the R04 Pickering Licence Conditions Handbook [B.8-3] as "Guidance and Criteria". The current revision of the Standard is N287.2-08 (R2013) [B.8-1]. This is the fifth edition of CSA N287.2 which supersedes the previous editions, published in 1991, 1982, 1977, and 1976. Changes since the last version of the Standard in 1991 have ranged from editorial, to the addition of supplementary information as well as the addition of new requirements. An Impact Statement was not prepared by the CSA for N287.2-08.

The results of PSR1 N287.2 reviews (Pickering B and Darlington ISRs), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.8.2 below. As identified in Reference [B.8-4], the Pickering PSR2 review of CSA N287.2-08 is an Incremental Review. As discussed in Section 2.0 of this Report, PSR2 Incremental Review includes an assessment of the intent of recent changes to the L/R/C/S on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.

- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.8.2 Compliance Assessment for Pickering PSR2

B.8.2.1 Application of PSR1 Reviews

The versions of N287.2 subject to previous reviews conducted for Darlington and Pickering, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering

Pickering Units 5-8

OPG Report NK30-REP-03680-00001 R000 [B.8-5] documents a clause-by-clause review against CSA N287.2-M91 (R2003) [B.8-6]. The findings of the review were:

> *The compliance evaluation of the CSA N287.2-M91 Standard has determined that it does not introduce any new requirements that impact the design basis of the CCSs [Concrete Containment Structures].*
>
> *Reviewing the various material standards referenced in CSA N287.2 to identify and assess changes introduced since the PNGS B CCSs were design and built is not considered worthwhile.*

The context for these findings is that Pickering NGS B was designed to National Building Code 1970 requirements, and though N287.2-M91 (R2003) [B.8-6] introduced new requirements none impact the design basis of the Concrete Containment Structures. For the 18 N287.2-M91 clauses that make reference to additional material standards as listed below, NK30-REP-03680-00001 R000 recommended that the new material standards be categorized as Acceptable Deviations (ADs) (by crediting in-service inspections and the aging management program during life extension with the ability to detect and monitor any safety significant aging mechanism and to provide assurance of continued fitness for service of the Concrete Containment Structures):

> *a) 4.1 General Requirements (Concrete Materials).*
>
> *b) 4.2 Materials for Concrete.*
>
> *c) 4.2.3.1 General Aggregates.*
>
> *d) 4.2.3.2 High Density Aggregates.*
>
> *e) 4.2.4 Admixtures.*
>
> *f) 4.2.5.1 Supplementary Cementing Materials.*

g) 4.3.1 Cement.

h) 4.3.3 Aggregates.

i) 5.1 General, Non prestressed Reinforcements.

j) 5.6.1 Rebar Weld Splicing.

k) 7.1.1 Nonmetallic Liner Systems.

l) 7.2 Compatibility of Nonmetallic Liner Systems.

m) 8.1.1 General, Studs & Metallic Embedded Parts.

n) 8.1.2 Anchorage Systems.

o) 8.2 Material Selection (metallic).

p) 9.1 Welding Materials.

q) 10.1.1 Organic Materials.

r) 11.1 General, Anchorage Systems.

As a result, NK30-REP-03680-00001 R000 [B.8-5] made the following recommendations:

> *It is recommended that an assessment of the N287.7 Periodic Inspection Program for Concrete Components and the NPC system In-service Inspection and Test program be carried out to determine whether they provide adequate assurance of continued fitness for service of the CCSs.*
>
> *It is not recommended that OPG carry out a review of the material standards referenced in CSA N287.2 to identify and assess changes introduced since the PNGS B CCSs were designed and built. There appears to be little of value to be gained from performing such a review.*

The Pickering B ISR actions were incorporated into the Pickering B IIP (see Appendix A of the Pickering Units 5-8 Continued Operation Plan NK30-PLAN-00531-00001 R005 [B.8-7]). IIP Action #31 resulted in submission of PIPs and Life Cycle Management Plans (LCMPs) for the safety-significant civil structures that are under the scope of N291-08, but not covered by N287.7. IIP Action #32 resulted in submission of Aging Management plans for the CCSs to the CNSC for review. Although both actions are complete, they will need to be considered in the context of Pickering operation past 2020 and are therefore identified as gaps for Pickering PSR2. However, these gaps are already identified under the review for CSA N287.7-08 in Section B.4 of this Report and are therefore are not addressed further here.

Pickering Units 1,4

For Pickering A Return to Service, the active version of the Standard at the time was CSA-N287.2-M91 (R1998) [B.8-8]. Per NA44-CORR-00531-00381 R000 [B.8-9], OPG identified N287.2-M91 as a code to be reviewed on the basis that it has "Direct and Immediate Effect on Installed Design Features". The code review was documented in OPG Report 44RS-00531-ASD-001-Revision 4 [B.8-10] and the findings were similar to those for Pickering B discussed above:

> *Based on the present study and the evaluation of documents, it is concluded that the design of Pickering 'A' containment structure meets the intent of the current CSA standards. As the design was done years ago utilizing NBC 1965, it is not possible to meet every aspect of newly developed codes. However after a review of the requirements in old and new codes and project specific documents it can be concluded that the changes and additions that have occurred in new codes do not have impact on the performance of the containment structure, namely the pressure retaining capability and leak tightness.*

Based on above, there are no PSR2 gaps which result from past Pickering N287.2 reviews.

Darlington NGS

The Darlington ISR was performed using version N287.2-08 [B.8-11] (equivalent to the latest version N287.2-08 [B.8-1], which was simply reaffirmed in 2013) and documented in report NK38-REP-03680-10035 R000 [B.8-12]. NK38-REP-03680-10035 R000 notes that this version of the Standard underwent a clause-by-clause (non-PROL) intent review.

The conclusions of the Darlington ISR code review of N287.2-08 were:

> *Based on the results of the review of the Darlington NGS concrete containment structures against CSA Standard N287.2-08, it is concluded that the Darlington NGS concrete containment structures are compliant, except for the gaps identified in Appendix C.*
>
> *The identified gaps are related to new requirements for seismic qualification tests, static tensile tests and static shear tests for mass produced embedded anchors and repair materials for concrete and grout.*

The clauses in N287.2-08 (R2013) are programmatic from the perspective of the specified requirements, while recognizing that design programs are specific to each station. Therefore, the programmatic Darlington ISR conclusions are applicable to Pickering PSR2 as discussed below.

The gaps arising from the Darlington ISR are related to new code requirements for seismic qualification tests, static tensile tests and static shear tests for mass produced embedded anchors and repair materials for concrete and grout. A number of the gaps identified for clauses of N287.2-08 were closed in either Appendix B of NK38-REP-03680-10035 R000 [B.8-12] (i.e., clause 5.4) or OPG Report NK38-REP-03680-10035-ADD-001 R000 [B.8-13] (e.g., clauses 12.2.3 to 12.2.6, inclusive), which documents the CNSC staff comments and OPG responses related to NK38-REP-03680-10035 R000. The unclosed gaps (e.g., clauses 12.3.2.1 to 12.3.2.4, inclusive) are part of Darlington ISR Issue #D014, as recorded in the Darlington Final ISR Report NK38-REP-03680-10104 R000 [B.8-14]. Issue #D014 was assessed to be an Acceptable Deviation (AD) for Darlington [B.8-14] based on the following logic:

> *DNGS Concrete Containment Structures were designed and constructed in accordance with the CAN/CSA-N287 Standards of the day, and although the gaps indicate that they did not include the prescriptive specifications for, and testing of, the materials used in concrete anchors according to CAN/CSAN287.2-08 and CAN/CSA-N287.3-93, the anchor manufacturers did meet testing requirements at that time. The Periodic Inspection of the concrete structures and routine leakage tests of the containment envelope verify that there is no significant deterioration in the integrity of the Concrete Containment Structures. No further action required.*

The above AD rationale also applies to Pickering NGS. The Darlington AD compliance statement for Issue #D014 points to Darlington-specific documentation (NK38-CORR-00531-14185, "Darlington NGS - Qualified Concrete Expansion Anchors" [B.8-15]) for evidence that post-installed mechanical anchors qualified for use at OPG are compliant to the revised CSA N287.2-08 testing requirements. N-GUID-20000-10000 R001, "Guide for Design of Post Installed Anchors for Safety Related, Seismically Qualified Systems and Components" [B.8-16], which is applicable to Pickering NGS, states:

> *Post-installed anchors do not have predictable pullout capacities, and are therefore required to be tested for reliability and simulated seismic events. Anchors not meeting the material and test requirements of CSA N287.2 are not qualified and should not be used in safety related systems, structures or components (located inside or outside containment). New anchors should be qualified by analysis and design as per current standards.*

There is therefore no gap for Pickering PSR2.

B.8.2.2 Application of Post-PSR1 Reviews

No post-PSR1 reviews were performed as the latest version was assessed for the Darlington ISR (which is programmatically applicable to Pickering NGS).

B.8.3 Compliance Summary for Pickering PSR2

There are no PSR2 gaps for CSA N287.2-08 (R2013) [B.8-1]. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with N287.2-08 (R2013).

B.8.4 References

[B.8-1] CSA Standard N287.2-08 (R2013), *Material Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants*, March 2008.

[B.8-2] CNSC Acts and Regulations web page: *Regulatory Documents*, *http://cnsc-ccsn.gc.ca/eng/acts-and-regulations/regulatory-documents/index.cfm#R14*, April 2016.

[B.8-3] CNSC Report, LCH-PNGS R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.8-4] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.8-5] OPG Report, NK30-REP-03680-00001 R000, *Pickering NGS-B Integrated Safety Review – Plant Design Safety Factor*, August 2007.

[B.8-6] CAN/CSA Standard, CAN/CSA N287.2-M91 (R2003), *Material Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants*, December 1991.

[B.8-7] OPG Plan, NK30-PLAN-00531-00001 R005, *Pickering 5-8 Continued Operations Plan*, December 2015.

[B.8-8] CAN/CSA Standard, CAN/CSA N287.2-M91 (R1998), *Material Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants*, December 1991.

[B.8-9] OPG Letter, NA44-CORR-00531-00381 R000, R. J. Strickert to J.S.C. Tong, *Pickering A - Updated Basis for Return to Service Document*, April 20, 2001.

[B.8-10] OPG Report, 44RS-00531-ASD-001 R004, *Review of Picketing 'A' Design Against Current Codes and Standards (C1-007-03-01-007)*, November 2000

[B.8-11] CSA Standard N287.2-08, *Material Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants*, March 2008.

[B.8-12] OPG Report, NK38-REP-03680-10035 R000, *Review of CSA Standard N287.2-08 (March 2008), Material Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants for Darlington Integrated Safety Review*, September 2011.

[B.8-13] OPG Report, N-REP-03680-10035-ADD-001 R000, *Addendum to the CSA N-287.2-08 Code Review Report for Darlington ISR*, January 2014.

[B.8-14] OPG Report, NK38-REP-03680-10104 R000, Darlington NGS Integrated Safety Review (ISR) – Final ISR Report, October 2011.

[B.8-15] OPG Letter, NK38-CORR-00531-14185 R000, W. Robbins to P.A. Webster, *Darlington NGS – Qualified Concrete Expansion Anchors*, July 15, 2008.

[B.8-16] OPG Guideline, N-GUID-20000-10000 R001, *Guide for Design of Post Installed Anchors for Safety Related, Seismically Qualified Systems and Components*, September 21, 2015.

B.9 CSA N289.1-08, "General Requirements for Seismic Design and Qualification of CANDU Nuclear Power Plants"

B.9.1 Background

The following, paraphrased from the Preface of CSA N289.1-08 (R2013) including Update 1 [B.9-1], provides a brief overview of the purpose of this Standard and the requirements expressed therein:

> *The CSA N289 series specifies means for the seismic qualification of those nuclear power plant structures, systems, and components necessary for safe shutdown, fuel cooling, the containment of potential releases of radioactive material, and the monitoring and control of essential safety-related functions in the event of an earthquake.*
>
> *CSA N289.1 has been restructured to act as an introduction to the CSA N289 series and to supplement the Standards in this series with current seismic qualification concepts and methodologies. The standard sets forth the general requirements for seismic design and qualification of CANDU nuclear power plants.*

All of N289.1-08 is directly relevant to Safety Factor 1 (Plant Design) since CNSC web site [B.9-2] identifies the Standard as being relevant to the Safety Control Area on "Physical Design". This Standard also addresses seismic qualification of SSCs and is therefore related to Safety Factor 3 (Equipment Qualification).

CSA N289.1 is identified in Appendix E.1 of the R04 Pickering Licence Conditions Handbook [B.9-3] as "Guidance and Criteria". The current revision of the Standard is N289.1-08 (R2013) including Update 1 [B.9-1]. It supersedes the previous edition, published in 1980. Changes since the last version of the Standard in 1980 have ranged from editorial, to the addition of supplementary information as well as the addition of new requirements.

The CSA N289.1-08 Impact Statement for N289.1-08 (R2013) including Update 1 [B.9-4] provides a "Summary of significant changes from the previous edition" which identifies several changes to the Standard as described in Section B.9.2 below. The results of PSR1 N289.1 reviews (Pickering B and Darlington ISRs), as well as reviews performed since PSR1, have also been assessed for applicability to PSR2 in Section B.9.2.

As identified in [B.9-5], the Pickering PSR2 review of CSA N289.1-08 is an Incremental Review. As discussed in Section 2.0 of this Report, PSR2 Incremental Review includes an assessment of the intent of recent changes to the L/R/C/S on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.

- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.9.2 Compliance Assessment for Pickering PSR2

The versions of N289.1 subject to previous reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

B.9.2.1 Application of PSR1 Reviews

Pickering NGS

Pickering Units 5-8

OPG Reports NK30-REP-03680-00001 R000 [B.9-6] and NK30-REP-03680-00005 R000 [B.9-7] document code reviews of CSA version N289.1-80 (R2008) [B.9-8] for the Plant Design and Safety Analysis Safety Factors, respectively. The review documented in NK30-REP-03680-00001 R000 [B.9-6] is a clause-by-clause review that used the following approach:

> *Station specific documents that identify seismic requirements for Safe Operating Envelope (SOE) system designs were used to perform the assessment. It was decided to include the safety report and the design requirements documents listed, because these documents provide more detail and were prepared much more recently than the design guide.*

The findings and conclusions of the review were [B.9-6]:

> *The documents … that describe the seismic requirements for the design of the Pickering B SOE systems were evaluated against the Standard CAN3-N289.1-80 and were found to comply with all the design requirements specified therein. On the assumption that the documents referenced accurately describe the seismic designs of the relevant systems in the plant, it is concluded that these systems are compliant with the Standard.*

The review documented in NK30-REP-03680-00005 R000 [B.9-7] assessed N289.1-80 (R2008) against the three Safety Factors related to the Safety Analysis subject area (i.e., Deterministic Safety Analysis, Probabilistic Safety Analysis, and Hazard Analysis):

- *The extent to which the plant conforms to modern high-level safety goals and requirements,*

- *The extent to which the licensing basis remains valid in terms of meeting the current regulatory requirements,*

- *The effectiveness of the arrangements that are in place to maintain plant safety for long-term operation.*

The findings and conclusions of the review were [B.9-7]:

> *This standard requires that the plant be designed and constructed to ensure that the effects of an earthquake do not result in unacceptable radiation exposure to the public. In terms of Safety Analysis, this standard has limited applicability. However, per Clause 4.1, it is ultimately the seismic success path that specifies the credited Systems, Structures and Components (SSCs) for a seismic event and the level of qualification required.*
>
> *Once the success path is established, the level of qualification for SSCs credited with mitigating consequence of a Design Basis Earthquake (DBE) or the Site Design Earthquake (SDE) is identified (the SDE being specific to the post-LOCA [Loss of Coolant Accident] mission).*
>
> *These credits are further subdivided into functional requirements for a seismic event to perform:*
>
> - *During an earthquake,*
> - *During or after an earthquake, and*
> - *After an earthquake.*
>
> *The above requirements have all been adopted in the Pickering B Seismic Design Guide for seismic qualification as DBE(a) and DBE(b).*
>
> *Clause 1.2 of the standard identifies that any SSCs that are not seismically qualified must be built to at least the requirements of the National Building Code (NBC).*

Based on the above, no PSR2 gaps are identified based on the Pickering B ISR assessments.

<u>Pickering Units 1,4</u>

For Pickering A Return to Service, the active version of the Standard at the time was CSA-N289.1-80 (R1998) [B.9-9]. Per NA44-CORR-00531-00381 R000 [B.9-10], "Pickering A was not reviewed against CSA CAN3-N289.1-80, because completion of the Seismic Margin Assessment makes such a review unnecessary." As noted in the Pickering PROL Renewal Application [B.9-11], for Pickering A the common containment structures were designed to exceed the National Building Code 1965 seismic design provisions and were subsequently confirmed analytically to meet seismic design requirements. No PSR2 gaps are identified based on the Pickering A Return to Service assessments.

Darlington NGS

The clauses in N289.1-08 are primarily programmatic from the perspective of the specified requirements, while recognizing that Seismic Qualified Equipment Lists are specific to each station. Therefore, the Darlington ISR conclusions are primarily applicable to Pickering PSR2 as discussed below.

The Darlington ISR was performed using two different versions of N289.1:

1) The first review used version N289.1-80 (R2008) [B.9-8] and is documented in OPG Report NK38-REP-03680-10039 R000 [B.9-12] which notes that this version of the Standard underwent a clause-by-clause (non-PROL) intent review. The review found there were no material differences between the version re-affirmed in 2008 and the previous version from 1980.

   The conclusions of the Darlington ISR code review of N289.1-80 (R2008) were [B.9-12]:

   > *In this safety review, six clauses are considered as blank as they do not have any requirements for compliance. The rest of the clauses, except one, are found to be either directly compliant or at least meet the intent of the respective clauses of the given Standard [6]. Details of the review are presented in Appendix C of this report.*
   >
   > *This review identified one gap with regard to Clause 5.1. This clause is concerned with the responsibilities of the Owner of the nuclear power plant with regard to compliance with the CAN CSA N289 series of seismic codes.*

   The gap identified for Clause 5.1 of N289.1-80 (R2008) was closed in Appendix B of the code review report [B.9-12] for N289.1-80 (R2008) (December 1980). However, OPG Report NK38-REP-03680-10039-ADD-01 R000 [B.9-13], which documents the CNSC staff comments and OPG responses related to the N289.1-80 (R2008) code review report [B.9-12], introduced new gaps on some clauses that were previously identified as compliant. The affected clauses were 4.1, 4.1(b), 5.2.53, 5.3.10, and 6.1, to which gaps were assigned due to a lack of a consolidated Seismic Qualified Equipment List:

   > *ISR Gaps will be raised against relevant clauses to assess the adequacy of existing list(s) and to assess the need for a consolidated list.*

This Darlington ISR gap and its applicability in the context of PSR2 are addressed in the discussion of the second code review below.

2) The second code review for the Darlington ISR used version N289.1-08 [B.9-14] and is documented in OPG Report NK38-REP-03680-10110 R000 [B.9-15]. This version of the Standard underwent a clause-by-clause (non-PROL) intent review, and considered the Plant Design and Deterministic Safety Analysis Safety Factors. The code review identified a few changes and new methods/terminologies that had been incorporated in N289.1-08. The conclusions of the Darlington ISR code review of N289.1-08 were [B.9-15]:

   This review identified gaps against the intent of eight clauses viz. 5.2.5.1, 5.3.2, 6.1, 6.5.6.1, 6.5.6.2.2, 6.5.6.4, 6.5.7.3 and 6.6. Either Darlington NGS does not meet requirement/s for compliance or no documentary evidence for compliance could be found for these clauses. The definition of design basis earthquake (DBE) has been changed in the latest Standard in the more conservative direction [1]. DBE is now defined at a minimum of 1 x 10^{-4} probability per year thus requiring a higher earthquake level.

   The Darlington ISR gaps identified above were assigned to the following Darlington ISR Issues in the Darlington Final ISR Report NK38-REP-03680-10104 R000 [B.9-16]:

ISR Issue #	Gap	Clause
D063	01525	5.2.5.1
	01526	5.3.2
	01527	6.1
	01532	6.6
D281	01528	6.5.6.1
	01529	6.5.6.2.2
	01530	6.5.6.4
	01531	6.5.7.3

   Based on a review of the existing OPG governance, no further action was required to address the gaps in Darlington ISR Issue #D281, and so they were reclassified as Acceptable Deviations (ADs) per Appendix M of the Darlington Final ISR Report [B.9-16].

   The gaps in Darlington ISR Issue #D063 were reclassified as closed, with the justification provided in Appendix E of the Darlington Final ISR Report Addendum 002 Report NK38-REP-03680-10104-ADD-002 R000 [B.9-17]. In the case of Gap #01525, which was declared against Clause 5.2.5.1 of CSA N289.1 which requires "A seismic classification list shall be created and shall define the seismic category and earthquake level for each SSC", the gap was addressed as follows [B.9-17]:

OPG is currently developing a seismic classification list to meet the criteria in CSA N289.1-8. This Gap #01525 is reassigned to a new Issue D345, which is created as a result of the CNSC comment during the review of the ISR. The Issue D345 has been resolved separately in accordance with the ISR Gap resolution process. The creation of the List will be tracked to completion in the IIP under Issue D345.

As stated above, Gap #01525 was reassigned to a new Issue #D345, which will track the creation of a Seismic Qualified Equipment List for Darlington. OPG is currently developing a Seismic Qualified Equipment List to meet the criteria in CSA N289.1-08 for Darlington.

The impact on Pickering of the Darlington ISR gaps related to CSA N289.1-08 relate to the need for a consolidated Seismic Qualified Equipment List. Seismic Probabilistic Risk Assessments (SPRAs) have been completed for Pickering A and B ([B.9-18], [B.9-19]) in which the Seismic Equipment List for each Station was prepared. The Seismic Equipment Lists identify the SSCs required for the seismic success path and meet the intent of the seismic classification list requirement. Therefore, this is not a gap for Pickering PSR2.

In November 2014, a study documented in OPG Report NK38-REP-03680-10197 R001 [B.9-20] examined the aggregate effects of ADs that are related to the seismic, pressure boundary, or accident management programs. The purpose of this study was to determine whether there are aggregate effects of these ADs. This report concluded there were no aggregate effects from ADs associated with the seismic, pressure boundary or accident management barriers and, therefore, no impacts on public safety. The ADs related to N289.1-08 (September 2008) [B.9-14] listed above for the Darlington ISR Issue #D281 were part of the study. There are no implications for Pickering PSR2.

No PSR2 gaps are identified based on the Darlington ISR assessments.

B.9.2.2 Application of Post-PSR1 Reviews

Per the CSA N289.1-08 Impact Statement [B.9-4] for N289.1-08 (R2013) including Update 1 [B.9-1], the following is a summary of the significant changes from the previous edition of the Standard:

- Incorporation of lessons from Fukushima in the body of the standard, further detailed in a new annex on Readiness for Beyond Design Basis seismic events. The impact is updated guidance for Beyond Design Basis seismic event evaluation and requirement for readiness for such events (for example periodic hazard evaluation minimum 10 year cycle, suggested margins for BDB over design basis). The standard requires consideration of seismic-induced multiple hazard interactions such as fire, flooding, blasts, tsunami, seiche and power blackout.

- Incorporation of Design Extension Conditions (DECs) as a subset of Beyond Design Basis events. This change clarifies checking/review level earthquake as a Design Extension Condition.

- Annex C on Evaluation for Beyond Design Basis events updated to incorporate current technology. This change aligns with the current technology, regulatory requirements and international practice (i.e. peer review, irradiated fuel storage bay evaluation, scaling of in-structure response spectra, etc.).

- Seismic Instrumentation and automatic shutdown addressed via reference to the updated CSA N289.5 standard. This change improves the usefulness of seismic monitoring records for post-earthquake evaluation, with respect to a decision to shut down the plant and future engineering designs.

- Incorporated current references to the extent possible. This change reflects the state-of-the-art technology via updated and international references.

These five changes were all assessed during the clause-by-clause review of N289.1-08 (R2013) [B.9-21] discussed below. As discussed below two issues were identified as a result of the clause-by-clause assessment. The other CSA code changes were not safety significant for Pickering.

OPG has committed to submitting to the CNSC annual "Code-over-Code" reviews to identify significant technical changes to the requirements of the set of engineering design-related code and standard effective dates, as agreed with the CNSC. The Code-over-Code reviews for 2014 included a review [B.9-21] of the latest version of CSA N289.1-08 (R2013) including Update 1 [B.9-1]. For this version of the Standard there were two significant issues that were identified related to Clauses 5.3.11 and 5.4.3. OPG raised internal Action Request (AR) # 28168507-09 to address the impact of these issues on OPG facilities by December 2015. This AR is now complete. Clauses 5.3.11 and 5.4.3 have an impact on Safety Factor 3 (Equipment Qualification) as demonstrated by the requirements set forth in these clauses to assess the performance of SSCs [B.9-21]:

- *Clause 5.3.11 Readiness for beyond design basis seismic event: Each facility shall have a periodic evaluation to demonstrate readiness to cope with the potential consequences of a beyond design basis seismic event.*

  The Standard specifies, as a minimum, the evaluation will be carried out once every 10 years. This is considered to be potentially at risk for a refurbished plant and also at risk for Pickering life extension to 2028. However, both Pickering Units 1,4 and Pickering Units 5-8 have completed Seismic Probabilistic Risk Assessments (SPRAs) (reports NA44-REP-03611-00022 R000 [B.9-18] and NK30-REP-03611-00013 R001 [B.9-19], respectively) that meet the intent of Clause 5.3.11. Hence, this is not a gap for Pickering PSR2.

- *Clause 5.4.3 Seismic capacity evaluation: Seismic capacity evaluation of existing nuclear power plant SSCs shall be performed in accordance with the methods specified in Clause 5.3, as applicable.*

  There is no PSR2 gap for Clause 5.4.3 since Darlington and Pickering meet the requirements by recently having issued the SPRAs; the documented Pickering assessments are identified below.

The seismic qualification methodologies specified in this Standard are probabilistic in nature "in order to more rigorously quantify the engineering capacity of the safety-related SSCs with respect to the seismic demand at the plant site." Clause 5.4.1.1 indicates that Seismic Margin Assessment and SPRA methodologies have been developed to establish the seismic margin of nuclear power plant SSCs relative to the seismic hazard at a plant site. Both Pickering Units 1,4 and Pickering Units 5-8 have completed SPRAs (reports NA44-REP-03611-00022 R000 [B.9-18] and NK30-REP-03611-00013 R001 [B.9-19], respectively) that meet the requirements of the Standard.

No PSR2 gaps are identified based on the post-PSR1 assessments.

B.9.3 Compliance Summary for Pickering PSR2

There are no PSR2 gaps for CSA N289.1-08 (R2013) [B.9-1]. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with N289.1-08 (R2013).

B.9.4 References

[B.9-1] CSA Standard N289.1-08 (R2013), *General Requirements for Seismic Design and Qualification of CANDU Nuclear Power Plants*, September 2008; *Update No. 1*: September 2014.

[B.9-2] CNSC Acts and Regulations web page: *Regulatory Documents, http://cnsc-ccsn.gc.ca/eng/acts-and-regulations/regulatory-documents/index.cfm#R14*, April 2016.

[B.9-3] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.9-4] CSA Impact Statement for Publication, *Product: Amendment – Product Designtation: CSA N289.1-08 – Product Title: General Requirements for Seismic*

Design and Qualification of CANDU Nuclear Power Plants - Date of Release: September 2014, Date not provided.

[B.9-5] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.9-6] OPG Report, NK30-REP-03680-00001 R000, *Pickering NGS-B Integrated Safety Review – Plant Design Safety Factor*, August 2007.

[B.9-7] OPG Report, NK30-REP-03680-00005 R000, *Pickering NGS B - Integrated Safety Review - Safety Analysis Review*, June 2007.

[B.9-8] CAN3/CSA Standard N289.1-80 (R 2008), *General Requirements for Seismic Qualification of CANDU Nuclear Power Plants*, December 1980.

[B.9-9] CAN3/CSA Standard N289.1-80 (R 1998), *General Requirements for Seismic Qualification of CANDU Nuclear Power Plants*, Decmber 1980.

[B.9-10] OPG Letter, NA44-CORR-00531-00381 R000, R. J. Strickert to J.S.C. Tong, *Pickering A - Updated Basis for Return to Service Document*, April 20, 2001.

[B.9-11] OPG Letter, CD#P-CORR-00531-03719, *Application for Renewal of Pickering Nuclear Generating Station Power Reactor Operating Licence*, July 4, 2012.

[B.9-12] OPG Report, NK38-REP-03680-10039 R000, *Review of CAN/CSA-N289.1-80 (R2008) (December 1980), General Requirements for Seismic Qualification for CANDU Nuclear Power Plants for Darlington Integrated Safety Review*, September 2011.

[B.9-13] OPG Report, NK38-REP-03680-10039-ADD-01 R000, *Addendum to the CAN/CSA N289.1 (1980) Code Review Report for Darlington ISR*, February 2014.

[B.9-14] CSA Standard N289.1-08, *General Requirements for Seismic Qualification of CANDU Nuclear Power Plants*, September 2008.

[B.9-15] OPG Report, NK38-REP-03680-10110 R000, *Review of CAN/CSA-N289.1-2008, General Requirements for Seismic Design and Qualification of CANDU Nuclear Power Plants for Darlington Integrated Safety Review*, August 2011.

[B.9-16] OPG Report, NK38-REP-03680-10104 R000, *Darlington NGS Integrated Safety Review (ISR) – Final ISR Report*, October 2011.

[B.9-17] OPG Report, NK38-REP-03680-10104-ADD-002 R000, *Darlington NGS Integrated Safety Review (ISR) – Final ISR Report Addendum 002*, November 2013.

[B.9-18] OPG Report, NA44-REP-03611-00022 R000, *PRA-Based Seismic Margin Assessment of PNGS-A*, January 2014.

[B.9-19] OPG Report, NK30-REP-03611-00013 R001, *PRA-Based Seismic Margin Assessment of PNGS-B*, April 2015.

[B.9-20] OPG Report, NK38-REP-03680-10197 R001, *Evaluation of Seismic, Pressure Boundary and Accident Management ISR Acceptable Deviation Issues for Aggregate Effects between Issues*, November 2014.

[B.9-21] OPG Report, N-REP-00590-00003 R001, *Code-Over-Code Review Report: CSA N289.1 Amendment No 1 -2014 for the Year 2014*, March 2015.

B.10 CSA N289.2-10, "Ground Motion Determination for Seismic Qualification of CANDU Nuclear Power Plants"

B.10.1 Background

The following paraphrased from the Preface of CSA N289.2-10 (R2015) [B.10-1] provides a brief overview of the purpose of this Standard and the requirements expressed therein:

> *For nuclear power plants, investigation is required to obtain the seismological and geological information necessary to determine the seismic ground motion that will be used in seismic qualification of safety-related plant structures and systems, and the potential for seismically induced phenomena that can have a direct or indirect effect on plant safety or operation.*
>
> *CSA N289.2 sets requirements on how to determine the appropriate seismic ground motion parameters for a particular site.*

N289.2-10 is directly relevant to Safety Factor 1 (Plant Design) since CNSC web site [B.10-2] identifies the Standard as being relevant to the Safety Control Area on "Physical Design". However, as stated above, the primary purpose of this Standard is to describe the "investigations required to obtain the seismological and geological information necessary to determine the seismic ground motion that will be used in seismic qualification of safety-related plant structures and systems." The Standard is also applicable to Safety Factor 3 (Equipment Qualification).

CSA N289.2 is identified in Appendix E.1 of the R04 Pickering Licence Conditions Handbook [B.10-3] as "Guidance and Criteria". The current revision of the Standard is N289.2-10 (R2015) [B.10-1]. It supersedes the previous edition, published in 1981. Changes since the last version of the Standard in 1981 have ranged from editorial, to the addition of supplementary information as well as the addition of new requirements.

The Impact Statement for CSA N289.2-10 [B.10-4] provides a "Summary of significant changes from the previous edition" which identifies several primary changes to the Standard described in Section B.10.2 below. In addition to findings resulting from review of the CSA N289.2-10 Impact Statement, the results of PSR1 CSA N289.2 reviews (Pickering B and Darlington ISRs), as well as reviews performed since PSR1, have also been assessed for applicability to PSR2 in Section B.10.2.

As identified in Reference [B.10-5], the Pickering PSR2 review of CSA N289.2-10 is an Incremental Review. As discussed in Section 2.0 of this Report, PSR2 Incremental Review includes an assessment of the intent of recent changes to the L/R/C/S on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.

- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.10.2 Compliance Assessment for Pickering PSR2

The versions of N289.2 subject to previous reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

B.10.2.1 Application of PSR1 Reviews

Pickering NGS

Pickering Units 5-8

OPG Report NK30-REP-03680-00005 R000 [B.10-6] documents the review of CSA N289.2-M81 (R2003) [B.10-7] for the Safety Analysis Safety Factor. The review documented in NK30-REP-03680-00005 R000 [B.10-6] assessed the Standard against the three Safety Factors related to the Safety Analysis subject area (i.e., Deterministic Safety Analysis, Probabilistic Safety Analysis, and Hazard Analysis) for:

- *The extent to which the plant conforms to modern high-level safety goals and requirements,*

- *The extent to which the licensing basis remains valid in terms of meeting the current regulatory requirements,*

- *The effectiveness of the arrangements that are in place to maintain plant safety for long-term operation.*

With respect to the applicability of N289.2-M81 (R2003) [B.10-7] to the Safety Analysis Safety Factor, the code review report [B.10-6] states:

> *In terms of Safety Analysis, N289.2 does not specify any specific requirements relating to the seismic success path. It is focused on a systematic approach to identify the seismic hazard, and as such, there are no specific clauses relating to Safety Analysis. The remainder of this standard is applicable to Siting.*

However, as stated in Section B.10.1, the Standard has limited applicability to Safety Factor 1, as only Clause 4.4.1.4 refers to this Safety Factor:

The effects of applicable seismically induced phenomena shall be mitigated by siting, layout, and design of the nuclear power plant structures, systems, and components.

The compliance review in the code review report [B.10-6] states the level of compliance with N289.2-M81 (R2003) [B.10-7] is an Acceptable Deviation (AD):

In terms of Safety Analysis, N289.2 does not specify any specific requirements relating to the seismic success path. It is focused on a systematic approach to identify the seismic hazard. And as such, there are no specific clauses relating to Safety Analysis. However, it is recognized that the magnitude and frequency of the spectra impacts on the qualification of the Systems, Structures, and Components and that if their required qualification were to change, the success path may need to change. There is no evidence of any additional information or insight in terms of quantification of the seismic hazard that would invalidate Pickering B's safety case for seismic events. Notwithstanding, the fact that there remains some ongoing activities in terms of establishing seismic hazard uncertainties results in the level of compliance being an Acceptable Deviation.

Follow-up on the status of the aforementioned ongoing activities was required to confirm the level of compliance being an AD. NK30-REP-03680-00005 R000 [B.10-6] includes the following explanation:

OPG has investigated, as part of the Seismic Hazard Resolution Project (SHRP), three sources of seismic hazard uncertainty; a) Rouge River faults, b) accuracy of historical seismicity data, and c) adequacy of regional seismic monitoring. The first two hazard uncertainties were eliminated by the SHRP. Recent regional seismic monitoring enhancements (e.g., POLARIS Project, which will add a further 23 seismometers to the Southern Ontario monitoring program) support resolution of the third.

The Pickering B Integrated Implementation Plan NK30-PLAN-03680-00002 R000 [B.10-8] also has Action I03 to complete the PRA-based seismic margin assessment for Pickering B. Seismic Probabilistic Risk Assessments (SPRAs) have been issued for both Pickering Units 1,4 (OPG Report NA44-REP-03611-00022 R000 [B.10-9]) and Pickering Units 5-8 (OPG Report NK30-REP-03611-00013 R001 [B.10-10]). While Action I03 is one of the actions to close the gaps identified in codes and standards with respect to the PRA, it is applicable here to support the compliance being an AD. Thus, there is no gap related to Gap #407 for Pickering PSR2. No PSR2 gaps are identified based on the Pickering B ISR.

<u>Pickering Units 1,4</u>

For Pickering A Return to Service, the active version of the Standard at the time was CSA-N289.2-M81 (R1998) [B.10-11]. Per NA44-CORR-00531-00381 R000 [B.10-12], this version of the Standard "Pertains mostly to design support analysis" and states that "Recent completion of the Seismic Margin Assessment makes review of this CSA Standard not necessary". As noted in the Pickering PROL Renewal Application [B.10-13], for Pickering A the common containment structures were designed to exceed the National Building Code 1965 seismic design provisions and were subsequently confirmed analytically to meet seismic design requirements. No PSR2 gaps are identified based on the Pickering A Return to Service assessments.

Darlington NGS

The Darlington ISR was originally performed against N289.2-M81 (R2008) [B.10-14] and is documented in OPG Report NK38-REP-03680-10040 R000 [B.10-15] which notes that this version of the Standard underwent a clause-by-clause (non-PROL) high-level intent review.

The clauses in N289.2 are primarily programmatic from the perspective of the specified requirements, while recognizing, for example, that Soil-Structure Interaction is specific to each station. Therefore, significant portions of the Darlington ISR assessment are applicable to Pickering as discussed below.

NK38-REP-03680-10040 R000 identified the following gaps [B.10-15]:

1. *Clause 3.2.1.1: Historical earthquake data*

   This clause requires that all the relevant sources are used to obtain as much historical and current seismic data as can be available for determining the site seismicity. Data to the present day has not been included. Therefore, this is a gap in compliance.

2. *Clause 3.2.1.2: Re-evaluation of earthquake history*

   This clause requires that the state-of -the art information is utilized in re-evaluation of the seismic parameters. In keeping with the recent development in seismic methodology… recently (2008) a Probabilistic Seismic Hazard Analysis (PSHA) has been performed at Darlington site. However, the earthquake parameters have not been re-evaluated on the basis of current methods of earthquake analysis.

3. *Clause 3.2.1.3: Use of instrumental seismology in assessing the earthquake parameters*

   This clause specifies the requirement for assessing the completeness and reliability of the historical recorded and qualitative data. The more recent seismological data from seismograph stations are not included in the assessment of the earthquake parameters; therefore this constitutes a gap in compliance.

4. *Clause 3.2.1.4: Detailed earthquake history of the DNGS site*

   This clause specifies detailed requirement for documenting the earthquake history of the region and required conservatism in making judgement based on qualitative *information. Since the time of construction of Darlington, 30 years of seismic data has not been included in the seismic risk analyses; therefore, this constitutes a gap.*

All of these clauses pertain to the incorporation of recent earthquake data into re-evaluating the earthquake parameters for DNGS.

The Darlington ISR gaps identified above were assigned to the following Darlington ISR Issue in the Darlington Final ISR Report NK38-REP-03680-10104 R000 [B.10-16]:

ISR Issue #	Gap	Clause
D074	00800	3.2.1.1
	00801	3.2.1.2
	00802	3.2.1.3
	00803	3.2.1.4

For Gaps #00800, #00801, #00802, and #00803, the Darlington Final ISR Report NK38-REP-03680-10104 R000 [B.10-16] also provided a proposed resolution for compliance:

The combination of the original seismic design and the current seismic monitoring practice provides high confidence that DNGS can withstand a seismic event and that the seismic activity in the region is within the design basis assumptions. The Seismic Probabilistic Risk Assessment will use the latest analysis methods and seismic data to assess the risk to the station from a seismic event and confirm the adequacy of DNGS to withstand an earthquake, or suggest possible design modifications in order to ensure the proper level of safety is maintained. Periodic updates to the seismic hazard have also been included in governance to ensure continuity.

Management action AR# 28127916 has been raised to track the Darlington Risk Assessment (DARA) project to completion in order to confirm that it satisfactorily resolved Issue D074.

Gaps #00800, #00801, #00802, and #00803 were reclassified as closed in Appendix E of the Darlington Final ISR Report Addendum 002 report NK38-REP-03680-10104-ADD-002 R000 [B.10-17] based on the existence of the Darlington Probabilistic Seismic Hazard Assessment (PSHA) and Darlington Seismic Probabilistic Risk Assessment (SPRA) which address the identified gaps. There is no gap for Pickering PSR2 since SPRAs have recently been issued for Pickering Units 1,4 and Pickering Units 5-8.

OPG Report NK38-REP-03680-10040-ADD-001 R000 [B.10-18] documents the CNSC staff comments and OPG responses related to the N289.2-M81 (R2008) code review report [B.10-15]. The report notes that OPG agreed to an action "to assign compliance with clause 4.2.1.1 as an ISR Gap". The action addresses CNSC comments on clause 4.2.1.1 (i.e., that Soil

Structure Interactions (SSIs) effects in terms of analysis approaches for the evaluation of site response analysis for the free-field case were not explicitly considered). This is related to ISR Issue #D415, Gap #01674 on clause 4.2.1.1 of N289.2-M81 (R2008), which was reclassified as an Acceptable Deviation in Appendix F of NK38-REP-03680-10104-ADD-001 R000 [B.10-19] since "despite the gap, the SPRA confirms that the DNGS buildings are sound enough to meet the intent of the clause." As discussed earlier, SPRAs have also been issued for both Pickering Units 1,4 (OPG Report NA44-REP-03611-00022 R000 [B.10-9]) and Pickering Units 5-8 (OPG Report NK30-REP-03611-00013 R001 [B.10-10]). Further, NA44-REP-02004-0119 R000, "Seismic Soil-Structure Interaction Analysis of Pickering NGS A Reactor Building" [B.10-20], documents the Soil Structure Interactions analysis for the Seismic Margin Assessment of Pickering A Reactor Building components. NK30-REP-21001-00002 R000, "Reactor Building Seismic Analysis" [B.10-21], documents the Pickering B Reactor Building seismic analysis, which includes Soil Structure Interactions effects. Therefore, there is no gap for Pickering PSR2 relating to Soil Structure Interactions effects.

Following the original code review for the Darlington ISR, a code refresh review based on version N289.2-10 [B.10-22] was conducted and documented in OPG Report NK38-REP-03680-10150 R000 [B.10-23]. The code refresh review comprised a clause-by-clause review of all changes in N289.2-10 (May 2010) [B.10-22] relative to N289.2-M81 (R2008) [B.10-14]. The review of the changed clauses did not identify any gaps relative to the requirements of N289.2-10 (May 2010) [B.10-22]. Furthermore, the gaps listed above on Clauses 3.2.1.3 and 3.2.1.4 of N289.2-M81 (R2008) [B.10-14] were reclassified as "Indirect Compliance". (Note that Clauses 3.2.1.3 and 3.2.1.4 of N289.2-M81 (R2008) [B.10-14] are similar to Clauses 4.2.1.3 and 4.2.1.4, respectively, of N289.2-10 [B.10-22] that was assessed in the code refresh review.) The code refresh review of the changed clauses in this code concluded the following:

> *The review confirms OPG Nuclear governance is in compliance with the requirements of CSA N289.2-10 relative to CSA N289.2 M81 (R2008).*

In November 2014, a study documented in OPG Report NK38-REP-03680-10197 R001 [B.10-24] examined the aggregate effects of ADs that are related to the seismic, pressure boundary, or accident management programs. The purpose of this study was to determine whether there are aggregate effects of these ADs. This report concluded there were no aggregate effects from ADs associated with the seismic, pressure boundary or accident management barriers and, therefore, no impacts on public safety. In the case of the ADs related to N289.2-M81 (R2008) [B.10-14] under Issue Resolution Form (IRF) #D415, they were excluded from the aggregation assessment for the following reason:

> *The gaps were raised in response to CNSC comments about whether Soil Structure Interaction effects had been adequately considered the DNGS seismic analysis. The rationale for considering these gaps as an AD is that the DNGS Site Probabilistic Seismic Hazard Assessment [R-62] confirms that DNGS conforms to CNSC S-294 [R-63] and to*

various international guides on seismic PRA, and that the results are considered acceptable. Consequently these gaps are considered to have no impact on the seismic resistance of qualified structures.

The combined safety significance category of the AD is 4.

This AD is therefore excluded from further review in this aggregation assessment.

As discussed earlier, the clauses in N289.2-10 are primarily programmatic from the perspective of the specified requirements, while recognizing, for example, that Soil-Structure Interaction is specific to each station.

B.10.2.2 Application of Post-PSR1 Reviews

The Impact Statement for CSA N289.2-10 [B.10-4] provides the following summary of the significant changes from the previous edition of the Standard:

- The term CANDU was removed from the title to be more technology neutral;

- Seismic hazard assessment procedures are updated and detailed to reflect international practices;

- The requirement to specify the Design Basis Seismic Ground Motion has been removed. The Standard instead provides seismic hazard estimates for a range of ground motion parameters, a range of probabilities, and a range of confidence intervals. The impact is: improves usage of the results, since various design requirements in N289.3 can use different outputs from a single application of N289.2. Removal of explicit reference to nuclear design parameters means N289.2 could be used for the seismic hazard assessment of non-nuclear facilities;

- Volcanism and surface faulting are added considerations. This change reflects current international practice and extends applicability to the standard within Canada and worldwide;

- The required site, site vicinity, and regional investigations are updated. This change reflects current international practice;

- Scenario earthquakes are added. This change provides a set of "earthquakes" representative of the probabilistic seismic hazard results that may be useful as check/review level earthquakes;

- Requires a full treatment of uncertainty in the evaluation of seismic hazard. This change reflects current seismic hazard assessment practice;

- Explicit clause on documentation is added to reflect the current regulatory requirements; and
- An Annex keyed to the Standard's clauses together with a flowchart has been added. This provides additional guidance and imposes understanding of the standard and its procedures.

The latest changes incorporated into CSA N289.2-10 were part of the clause-by-clause code refresh review [B.10-23] using N289.2-10 (May 2010) [B.10-22] since the R2015 version [B.10-1] introduced no additional changes (i.e., the Darlington clause-by-clause code refresh review using the CSA N289.2-10 (May 2010) version is the latest version of the standard to be assessed). No gaps were found as discussed above.

OPG has committed to submitting to the CNSC annual Code-over-Code reviews to identify significant technical changes to the requirements of the set of engineering design-related code and standard effective dates, as agreed with the CNSC. The code-over-code review methodology documents [B.10-25] and [B.10-26] list Standard N289.2-10 (R2015) [B.10-1], signifying that future versions of the Standard will be included in subsequent Code-over-Code reviews. A Code-over-Code review for CSA N289.2-10 (R2015) relative to CSA N289.2-10 (May 2010) has not been prepared as it is not required (i.e., there are no changes to address).

Further, OPG Nuclear Standard N-STD-MP-0025 R001 [B.10-27] sets forth the general integrated approach to seismic requirements for the applicable SSCs within OPG's nuclear facilities. The approach and framework for OPG activities related to the general requirements for seismic design and qualification of OPG nuclear facilities are in accordance with the CSA N289 series of standards.

B.10.3 Compliance Summary for Pickering PSR2

There are no PSR2 gaps for CSA N289.2-10 (R2015) [B.10-1]. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CSA N289.2-10 (R2015).

B.10.4 References

[B.10-1] CSA Standard N289.2-10 (R2015), *Ground Motion Determination for Seismic Qualification of Nuclear Power Plants*, January 2010.

[B.10-2] CNSC Acts and Regulations web page: *Regulatory Documents*, *http://cnsc-ccsn.gc.ca/eng/acts-and-regulations/regulatory-documents/index.cfm#R14,* April 2016.

[B.10-3] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.10-4] CSA Impact Statement, *Notification of CSA N289.2-10 Publication*, Date not provided.

[B.10-5] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.10-6] OPG Report, NK30-REP-03680-00005 R000, *Pickering NGS B - Integrated Safety Review - Safety Analysis Review*, June 2007.

[B.10-7] CAN3/CSA Standard N289.2-M81 (R2003), *Ground Motion Determination for Seismic Qualification of CANDU Nuclear Power Plants*, April 1981.

[B.10-8] OPG Plan, NK30-PLAN-03680-00002 R000, *Pickering B- Integrated Implementation Pla*n, December 2011.

[B.10-9] OPG Report, NA44-REP-03611-00022 R000, *PRA-Based Seismic Margin Assessment of PNGS-A*, January 2014.

[B.10-10] OPG Report, NK30-REP-03611-00013 R001, *PRA-Based Seismic Margin Assessment of PNGS-B*, April 2015.

[B.10-11] CAN3/CSA Standard N289.2-M81 (R1998), *Ground Motion Determination for Seismic Qualification of CANDU Nuclear Power Plants*, April 1981.

[B.10-12] OPG Letter, NA44-CORR-00531-00381 R000, R. J. Strickert to J.S.C. Tong, *Pickering A - Updated Basis for Return to Service Document*, April 20, 2001.

[B.10-13] OPG Letter, CD#P-CORR-00531-03719, Application for Renewal of Pickering Nuclear Generating Station Power Reactor Operating Licence, July 4, 2012.

[B.10-14] CAN3/CSA Standard N289.2-M81 (R2008), *Ground Motion Determination for Seismic Qualification of CANDU Nuclear Power Plants*, April 1981.

[B.10-15] OPG Report, NK38-REP-03680-10040 R000, *Review of CAN/CSA-N289.2-M81 (R2008) (April 1981), Ground Motion Determination for Seismic Qualification of CANDU Nuclear Power Plants for Darlington Integrated Safety Review*, August 2011.

[B.10-16] OPG Report, NK38-REP-03680-10104 R000, *Darlington NGS Integrated Safety Review (ISR) - Final ISR Report*, October 2011.

[B.10-17] OPG Report, NK38-REP-03680-10104-ADD-002 R000, *Darlington NGS Integrated Safety Review (ISR) - Final ISR Report Addendum 002*, November 2013.

[B.10-18] OPG Report, NK38-REP-03680-10040-ADD-001 R000, *Addendum to the CAN/CSA N289.2 M81 Code Review Report for Darlington ISR*, February 2014.

[B.10-19] OPG Report, NK38-REP-03680-10104-ADD-001 R000, *Darlington NGS Integrated Safety Review (ISR) – Final ISR Report Addendum*, June 2013.

[B.10-20] OPG Report, NA44-REP-02004-0119 R000, *Seismic Soil-Structure Interaction Analysis of Pickering NGS A Reactor Building*, March 1996.

[B.10-21] OPG Report, NK30-REP-21001-00002 R000, *Reactor Building Seismic Analysis*, February 2012.

[B.10-22] CSA Standard N289.2-10, *Ground Motion Determination for Seismic Qualification of Nuclear Power Plants*, May 2010.

[B.10-23] OPG Report, NK38-REP-03680-10150 R000, *Code Refresh Review of CSA N289.2-10, Ground Motion Determination for Seismic Qualification of Nuclear Power Plants, for DNGS ISR*, July 2013.

[B.10-24] OPG Report, NK38-REP-03680-10197 R001, *Evaluation of Seismic, Pressure Boundary and Accident Management ISR Acceptable Deviation Issues for Aggregate Effects between Issues*, November 2014.

[B.10-25] OPG Letter, N-CORR-00531-06858 R000, "W. M. Elliott to M. Santini, K. Glenn and F. Rinfret", *2014 Code-Over-Code Review for OPG Nuclear Fleet*, April 2015.

[B.10-26] OPG Report, N-REP-00590-058059 R000, *2015 COC - OPG Compilation*, March 2016.

[B.10-27] OPG Nuclear Standard, N-STD-MP-0025 R001, *General Requirements for Seismic Qualification of OPG Nuclear Facilities*, October 2015.

B.11 CSA N289.3-10, "Design Procedures for Seismic Qualification of CANDU Nuclear Power Plants"

B.11.1 Background

The following paraphrased from the Preface of CSA N289.3-10 (Update No. 2; November 2015) [B.11-1] provides a brief overview of the purpose of this Standard and the requirements expressed therein:

> *Nuclear power plants are required to be designed and seismically qualified in a manner using analytical techniques that meet a quality and standard commensurate with the safety principles necessary to comply with the Canadian nuclear safety philosophy.*
>
> *CSA N289.3 provides design requirements and methods, specifies the requirements, criteria and methods of analysis for the following:*
>
> - *Determining the design response spectra and ground motion time-histories to be used in the analysis;*
> - *Establishing design criteria for SSCs, and supports that require seismic qualification; and*
> - *Performing seismic analyses, including the effects of the soil-structure-interaction.*

N289.3-10 is directly relevant to Safety Factor 1 (Plant Design) since the CNSC web site [B.11-2] identifies the Standard as being relevant to the Safety Control Area on "Physical Design". Also, since this Standard addresses seismic qualification of SSCs it is related to Safety Factor 3 (Equipment Qualification).

CSA N289.3 is identified in Appendix E.1 of the R04 Pickering Licence Conditions Handbook [B.11-3] as "Guidance and Criteria". The current revision of the Standard is N289.3-10 (Update No. 2; November 2015) [B.11-1]. It supersedes the previous edition, published in 1981. Changes since the last version of the Standard in 1981 have ranged from editorial, to the addition of supplementary information as well as the addition of new requirements.

The CSA N289.3-10 (Update No. 2; November 2015) Impact Statement [B.11-4] provides a "Summary of significant changes from the previous edition" which identifies several primary changes to the Standard described in Section B.11.2.2 below. In addition to findings resulting from review of the CSA N289.3-10 Impact Statement, the results of PSR1 N289.3 reviews (Pickering B and Darlington ISRs), as well as reviews performed since PSR1, have also been assessed for applicability to PSR2 in Section B.11.2.

As identified in Reference [B.11-5], the Pickering PSR2 review of CSA N289.3-10 is an Incremental Review. As discussed in Section 2.0 of this Report, PSR2 Incremental Review includes an assessment of the intent of recent changes to the L/R/C/S on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.
- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.11.2 Compliance Assessment for Pickering PSR2

The versions of CSA N289.3 subject to previous reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

B.11.2.1 Application of PSR1 Reviews

Pickering NGS

Pickering Units 5-8

OPG Report NK30-REP-03680-00001 R000 [B.11-6] documents the code review of CSA N289.3-M81 (R2003) [B.11-7] for the Plant Design Safety Factor. The review of the Plant Design Safety Factor identified two types of gaps:

- *Gaps stemming from new Design Code requirements.*
- *Gaps related Documentation issues.*

  The ISR Plant Design safety factor review concludes that the design of the Pickering NGS-B was initially robust and continues to be that way. In addition, programs for maintaining the design integrity of the Station have been adequately implemented.

The gaps identified were against Clauses 4.7.1, 4.7.2, 4.7.3 (Gap #64, "no evidence that factor of safety applied consistently") and 5.13.2 (Gap #65, "number of cycles for fatigue analysis not in compliance"). Action Plans for these gaps are identified in the Pickering B Integrated Implementation Plan NK30-PLAN-03680-00002 R000 [B.11-8]:

- Gap #64 - Overturning and Sliding Stability Factor of Safety-not less than 1.25 used in calculations:

  Action: I01 Perform a review of the relevant seismic analysis reports regarding seismic overturning and sliding stability and document the level of compliance of *structures with CSA N289.3-M81, Clause 4.7.1 to 4.7.3. If seismic reports are not available, perform required calculations to demonstrate compliance.*

- Gap #65 - Required Minimum Number of Seismic Cycles-15/25:

  Action: I02 Review the relevant seismic analysis reports regarding number of cycles for a fatigue analysis and document compliance with CSA N289.3-M81, Clause 5.13.2, and revise OPG Design Guide, DG-30-68000-2, "Pickering G.S. 'B' Seismic Qualification of Safety Related Systems", R02, December 1979, Section 6.4 to document the requirements of CSA N289.3-M81 clause 5.13.2.

The Pickering Continued Operations Plan (NK30-PLAN-00531-00001 R005 [B.11-9]) states: "I01 and I02 - 112013: Refer to Appendix B of R001 of the Pickering Consolidated End of life Action Log (CAL), P-LIST-09314-00001, for completion notes and additional action details. This action is complete." Therefore, Gaps #64 and 65 are now closed. Further, these gaps are not impacted by Pickering operation past 2020.

Thus, for the Pickering B ISR, gaps were identified, but these were deemed not safety significant as they stem from new design code requirements or were related to documentation issues. Furthermore, the design of Pickering B was initially robust and continues to be that way. In addition, programs for maintaining the design integrity of the Station have been adequately implemented.

<u>Pickering Units 1,4</u>

For Pickering A Return to Service, the active version of the Standard at the time was CSA-N289.3-M81 (R1998) [B.11-10]. Per NA44-CORR-00531-00381 R000 [B.11-11] this version of the Standard "Pertains mostly to design support analysis" and states "Recent completion of the Seismic Margin Assessment makes review of this CSA Standard not necessary". As noted in the Pickering PROL Renewal Application [B.11-12], for Pickering A the common containment structures were designed to exceed the National Building Code 1965 seismic design provisions and were subsequently confirmed analytically to meet seismic design requirements.

Darlington NGS

The Darlington ISR was performed using two different versions of the Standard as discussed below. The clauses in N289.3 are primarily programmatic from the perspective of the specified requirements, while recognizing, for example, that seismic hazard models are specific to each station. Therefore, significant portions of the Darlington ISR assessments are applicable to Pickering.

1) The first review used version N289.3-M81 (R2008) [B.11-13] and is documented in report NK38-REP-03680-10008 R000 [B.11-14], which notes that this version of the Standard underwent a clause-by-clause (non-PROL) intent review. The review identified the following gap:

   Clause 5.10: This clause pertains to account for torsional effects on unsymmetric components. No documentary evidence of compliance with this clause could be found.

   During the initial ISR Gap resolution process, it was determined that the gap with clause 5.10 could not be reclassified as compliant. Therefore, the ISR Gap remained as per the code review report [B.11-14] for N289.3-M81 (R2008). Gap 00174 was declared against Clause 5.10 of CAN/CSA-N289.3-M81 which requires that for unsymmetrical systems torsional effects be included.

   Gap 00174 was assigned to the following Darlington ISR Issue in the Darlington Final ISR Report NK38-REP-03680-10104 R000 [B.11-15]:

ISR Issue #	Gap	Clause
D063	00174	5.10

   The Darlington Final ISR Report NK38-REP-03680-10104 R000 [B.11-15] provided a proposed resolution for compliance:

   The completion of this Seismic Probabilistic Risk Assessment will provide compliance with CNSC S-294 and is consistent with CAN/CSA-N289.1-08 methods. The Seismic Probabilistic Risk Assessment will confirm the adequacy of the DNGS design to withstand an earthquake, or suggest possible design modifications in order to ensure the proper level of safety is maintained.

   Management action AR# 28127914 has been raised to track the Darlington Risk Assessment project to completion in order to confirm that it satisfactorily resolved Issue D063.

   AR# 28127914 is complete as of December 15, 2012. In the case of Pickering, SPRAs have been issued for both Pickering Units 1,4 (OPG Report NA44-REP-03611-00022 R000 [B.11-16]) and Pickering Units 5-8 (OPG Report NK30-REP-03611-00013 R001 [B.11-17]), so this Darlington ISR gap is not applicable for PSR2.

2) The second code review for the Darlington ISR used version N289.3-10 (Update No. 1; August 2012) [B.11-18] and is documented in report NK38-REP-03680-10151 R000 [B.11-19]. This version of the Standard underwent a clause-by-clause (non-PROL) intent review, and considered two Safety Factors: Plant Design and Equipment Qualification. The

conclusions of the Darlington ISR code review of N289.3-10 (Update No. 1; August 2012) were:

> *Darlington NGS is in compliance with all of the clauses except for the following:*
>
> *Clauses 4.4.1, 4.4.2, 4.4.3, 4.4.4.1 to 4.4.4.6, 5.3.3, 5.3.4, 5.3.5, 5.6.1, 5.6.3, 5.6.4, 6.5.2.2, 6.7.1, 6.10.1, 6.10.2, and 6.10.3.*

Appendix D of NK38-REP-03680-10151 R000 [B.11-19] indicates that the Darlington ISR Issues covering the above gaps are #D352, #D617, #D415, #D618, #D414, and #D063.

Report NK38-REP-03680-10008-ADD-001 R000 [B.11-20] documents the CNSC staff comments and OPG responses related to the N289.3-M81 (R2008) [B.11-13] code review report [B.11-14]. The report identified a number of news gaps for Clauses 3.43, 4.2, 5.2.2.1, 5.7.1, and 5.11, and a "General" gap *against the CSA N289.3 code review report to address the CNSC concerns with the systems and structures that were considered in the report*.

Report NK38-REP-03680-10104-ADD-001 R000 [B.11-21], which documents the CNSC staff comments and OPG responses related to the Darlington Final ISR Report [B.11-15], addressed CNSC comments on N289.3-M81 (R2008) and provided the following updates to existing Darlington ISR Issues:

- #D352 - Resolution: OPG to check the time histories used in seismic analyses of safety-related System, Structure and Components against original and recent versions of the code. An assessment is required to determine whether time histories meet code requirements.
- #D414 – This gap related to seismic time histories, was reclassified as an Acceptable Deviation per Appendix F.
- #D415 - This gap to Soil-Structure Interactions, was reclassified as an Acceptable Deviation per Appendix F.

Both #D414 and #D415 were reclassified as ADs by pointing to the Darlington SPRA as evidence of indirect compliance. As discussed earlier, SPRAs have been issued for both Pickering Units 1,4 (OPG Report NA44-REP-03611-00022 R000 [B.11-16]) and Pickering Units 5-8 (OPG Report NK30-REP-03611-00013 R001 [B.11-17]). Therefore, there is no gap for Pickering PSR2 relating to Soil Structure Interactions effects.

The resolution of Darlington ISR Issue #D352 was completed and documented in report NK38-REP-03680-10224 R000 [B.11-22] which states the following:

A series of calculations were performed on the set of seismic time-histories provided by OPG that was used in the design of DNGS to demonstrate compliance with Clause 3.4.3 of CSA N289.3-M81 and with the requirements of Clauses 4.4.4.3 and 4.4.4.4 of CSA N289.3-10. Calculations were performed to: (a) digitize the tri-partite Darlington Ground Response Spectrum, (b) convert the time history text file into a format that can be used as input to ANSYS, using the ANSYS APDL command language, and, (c) perform ANSYS transient analyses to generate the required 5 % response. The results indicate that the seismic time history files are compliant with Clause 3.4.3 of CSA N289.3-M81 and Clauses 4.4.4.3 and 4.4.4.4 of CSA N289.3-10.

The Darlington NGS Integrated Implementation Plan report NK38-REP-03680-10185 R002 [B.11-23] also documents closure of Darlington ISR Issue #D352 under IIP-OI-052, which also includes related ISR Issue #D617 (Seismic Time History Requirements). The closure reference for #D352 and #D617 makes use of the detailed assessment performed in NK38-REP-03680-10224 R000 which is specific to Darlington. A similar assessment for Pickering NGS could not be found. As a result, there is an associated gap for PSR2 to provide similar evidence to show that the generated time history used within seismic analyses of safety-related systems correctly represents the design ground response spectrum for the Pickering site in compliance with N289.3-10 (**PSR2 CSA N289.3 Gap #1**).

The Darlington IIP report [B.11-23] also notes that ISR Issue #D618 (Soil Liquefaction Potential) under IIP-OI-071 remains as a code gap, with the following Action Plan tracked by Action Request 28175301-01 for completion by 2019:

Review the available information to verify that the liquefaction potential for fill materials in the Protected Area related to safety related systems and structures is low. Otherwise, complete a liquefaction assessment study.

It needs to be demonstrated that a similar gap does not exist for Pickering due to the plant-specific nature of the gap. This investigation is discussed in section B.11.2.2 and the conclusion is that there is no PSR2 gap.

Gap #00174 in Darlington ISR Issue #D063 was reclassified as closed, with the justification provided in Appendix E of the Darlington Final ISR Report Addendum 002 report NK38-REP-03680-10104-ADD-002 R000 [B.11-24]. The rationale included evidence that the original design considered torsional effects based on application of AECL seismic design guidance, DG-91-01040-1-39113. Whether this argument applies to Pickering is not pertinent because the issue is resolved by the completed Pickering Seismic Probabilistic Risk Assessments (SPRAs) listed later in this paragraph. Also, in accordance with Clause 5.4 of CSA N289.1-08, an acceptable method of re-evaluating existing Nuclear Plants for seismic considerations is SPRA, for which the issued SPRA for Darlington (report NK38-REP-03611-10051 [B.11-25] confirms that the original design and seismic qualification of Darlington provide an adequate level of Safety from the earthquake hazard at the DNGS site. In the case of Pickering, SPRAs have been issued for both Pickering Units 1,4 (OPG Report NA44-REP-03611-00022 R000 [B.11-16]) and Pickering Units 5-8 (OPG Report NK30-REP-03611-00013 R001 [B.11-17]), so this Darlington ISR gap is not applicable for Pickering.

Appendix E of the Darlington Final ISR Report Addendum 002 report NK38-REP-03680-10104-ADD-002 R000 [B.11-24] also documented a new Darlington ISR Issue #D500 (Gap #02128):

Gap 02128: The gap addresses the CNSC concerns with the adequacy of the N289.3-M81 code review report [R-1], specifically the lack of a listing of systems and structures that were considered in the report.

Gap #02128 was addressed and closed by the provision of a list of systems and structures proposed for inclusion in the code refresh review report. This list was documented in NK38-CORR-03680-0486309 R000 [B.11-26], which states:

This letter provides a list of systems and structures proposed for inclusion within the scope of the CSA N289.3-10, "Design Procedures for Seismic Qualification of Nuclear Power Plants", including Update No. 1 (2012) Code Refresh Report.

As discussed above, SPRAs have been completed for Pickering U1,4 and 5-8 ([B.11-16], [B.11-17]) in which the Seismic Equipment List for each Station was prepared. The Seismic Equipment Lists identify the SSCs required for the seismic success path and meet the intent of the seismic classification list requirement. Therefore, this is not a gap for PSR2.

In November 2014, a study documented in OPG Report NK38-REP-03680-10197 R001 [B.11-27] examined the aggregate effects of ADs that are related to the seismic, pressure boundary, or accident management programs. The purpose of this study was to determine whether there are aggregate effects of these ADs. This report concluded there were no aggregate effects from ADs associated with the seismic, pressure boundary or accident management barriers and, therefore, no impacts on public safety. The ADs related to N289.3-M81 (R2008) [B.11-13] listed above for Darlington ISR Issue #D281 were part of the study.

B.11.2.2 Application of Post-PSR1 Reviews

The CSA N289.3-10 (Update No. 2; November 2015) Impact Statement [B.11-4] provides a summary of the significant changes from the previous edition of the Standard which are noted in the following table. Since these changes to the code were not assessed in the latest code update review described in the previous section, each change is assessed below.

#	CSA "Summary of Significant Changes from Previous Edition"	Assessment of Safety Significance of Changes
1	Incorporation of requirements for addressing differences between Standard-shape ground spectra and Uniform Hazard Spectra (esp. at high frequency) to reflect current seismological information and technology. This change provides an enhanced level of safety by addressing potential impact of high frequency seismic input.	Clause 4.3.2 of CSA N289.3-10 specifies requirements for addressing the differences between Standard-shape ground spectra and Uniform Hazard Spectra (especially at high frequency). This is addressed below. <u>Pickering Units 1,4</u> Per "Pickering NGS-A PRA-Based Seismic Margin Assessment" [B.11-16], the Uniform Hazard response spectrum (UHRS) defines the review level earthquake (the review level earthquake is a very low probability (less than 10^{-4} per year) seismic ground motion derived from assessment of seismic hazard at the Pickering site) and forms the basis of the in-structure response needed in estimating the seismic demand on equipment. The UHRS represents the acceleration at the specified probability of occurrence at all response frequencies. Per the definition above, the UHRS reflects all the expected response frequencies (which includes high frequency). The high frequency content for UHRS is further supported by the PNGS-A PRA Guide-Seismic [B.11-28], which identifies the Uniform Hazard Spectra as possessing high frequency responses (i.e., section 2.3.3 "…..due to the high frequency content of the Uniform Hazard Spectra".). Therefore, since the UHRS (which shows high frequency content) forms the basis of the in-structure response needed in estimating the seismic demand on equipment, the ground response spectra in the Pickering A PRA based SMA, reflect the expected high frequency content of the seismic ground motions. No PSR2 gap exists against the requirements of the Clause. <u>Pickering Units 5-8</u> Applying similar logic used for Pickering A above, per "Pickering NGS B PRA based SMA" [B.11-17]: *"The RLE selected for this PRA based SMA is the 10, 000 year return 84th percentile UHRS updated in 2011."* The OPG PRA Seismic Guide (N-GUID-03611-10001 [B.11-29]) which provides guidance for the Pickering B PRA based SME, identifies the UHRS as containing high frequency" (i.e. Section 2.3.3.1, "….since the UHS shows significant high frequency contents at frequencies over 10Hz….". Therefore, since the UHRS (which shows high frequency content) forms the basis of the in-structure response needed in estimating the seismic demand on equipment,

#	CSA "Summary of Significant Changes from Previous Edition"	Assessment of Safety Significance of Changes
		the ground response spectra in the Pickering B PRA based SMA reflects the expected high frequency content of the seismic ground motions. No PSR2 gap exists against the requirements of the Clause.
2	Incorporation of more detailed requirements for Power Spectral Density and its use. This change clarified requirements that are consistent with signal processing terminology.	Clause 4.4.4.5 of CSA N289.3-10 states: "*The power spectral density (PSD) function of each time-history shall be calculated and shown to not have any significant gaps in energy over the frequency intervals outlined in Table 2…."* The calculation of PSD is not addressed in the Pickering A PRA Based SMA [B.11-16] or the Pickering B PRA Based SMA [B.11-17]. Also, the Pickering NGS A PRA Seismic Guide [B.11-28] and the OPG PRA Guide [B.11-29] do not identify any requirements for PSD. Also, evidence in the form of a calculation for time histories which represent the design ground motion was not found (which is a precursor for the PSD calculation). The lack of evidence of calculated time histories was also identified as a gap in the Darlington ISR (*#D352 and #D617 – Documented evidence in the form of a calculation to show that the generated time history correctly represents the design ground response spectrum within the prescribed requirements has not been provided).* This is therefore a gap for PSR2 (identified earlier as **PSR2 CSA N289.3 Gap #1**).
3	Improvement of requirements for soil liquefaction to articulate the requirements for the Design Basis level. This change articulated requirements for liquefaction at Design Basis earthquake levels.	Clause 5.6.1 of CSA N289.3-10 states: "*Adverse effects of soil liquefaction at a nuclear power plant site shall be precluded at the DBE level".* The adverse effects of soil liquefaction are not addressed in the Pickering A PRA Based SMA [B.11-16] or the Pickering B PRA Based SMA [B.11-17]. Also, the Pickering NGS A PRA Seismic Guide [B.11-28] and the OPG PRA Guide [B.11-29] do not identify any requirements for soil liquefaction. However, as identified in NK30-CORR-00531-04876, "Pickering NGS-B – Integrated Safety Review (ISR) – CNSC Review of Acceptable Deviations and Discrepancies for the Plant Design Safety Factor Report" [B.11-30], it indicates that the potential for liquefaction of the subsurface materials at Pickering B site does not have to

#	CSA "Summary of Significant Changes from Previous Edition"	Assessment of Safety Significance of Changes
		be evaluated since the plant is located in an area of flat geologically stable ground and a seismic ground motion assessment was performed (Section 2.3 of Part 1 of the Pickering A Safety Report was referenced). This gap assessment was subsequently accepted by the CNSC. In addition, Section 1.0 of Geotechnical Report 74148, "Pickering Generating Station B Interim Report Preliminary Engineering Phase Geotechnical Evaluation of Site Features" [B.11-31] states the following regarding the Pickering B site: "the potential of liquefaction occurring on this site is considered to be remote and of minor importance since the reactors and turbines will be founded on piles to bedrock". Since Pickering A and B share common site geology as reflected in Section 2.3, Part 1 of the Pickering A and B Safety Report, the above arguments are also applicable for Pickering A (i.e., soil liquefaction is not applicable for Pickering A as well). Additionally, Section 2.4.5 of the Pickering NGS B Design Manual, "Seismic Route" (NK30-DM-21002-10001 [B.11-32]) states, "Pickering DBE seismic ground acceleration levels are insufficient to induce soil liquefaction or other vibration-related disturbances…..". Hence, soil liquefaction at the DBE level can be precluded for both Pickering A and B. No PSR2 gap exists against the requirements of the Clause.
4	Incorporation of improved requirements reflective of currently applicable codes and their use in SSCs' design and specifically in the design of supports and anchorage. This change adds requirements that allow for use of Canadian steel code (CSA S16) to design supports.	The inclusion of CSA S16 appears in CSA N289.3-10 Clauses 7.3.4(d), 7.4.3.1, 7.4.3.3 and 7.4.3.4. The text in the "impact of changes" column as well as the Clauses themselves, indicates that CSA S16 requirements are now allowed for use as required (i.e., does not impose more restrictive requirements), due to the improvement in the code itself. Examples exist in which compliance with these requirements have been met: • Seismic analysis of 7132-HX501, -HX502 support frame (NA44-CALC-71320-10001 [B.11-33]), in which a code check against CSA S16 was performed (section 5.0) • Unit 7 Boiler Blow Down Pipe Civil Pipe Supports Miscellaneous Steel (NK30-CALC-21459-00005 [B.11-

#	CSA "Summary of Significant Changes from Previous Edition"	Assessment of Safety Significance of Changes
		34]), states in the Conclusion section that all analysis, modifications and new designs meet the current code S16-09. Hence, the incorporation of improved applicable codes (i.e. CSA S16) in the design of supports and anchorage has been applied for both Pickering A and B. No PSR2 gap exists against the requirements of the Clause.
5	Incorporation of provisions that allow for extension to the highest frequency of interest from the current 33Hz in the standard to values corresponding to input from modern seismic hazard studies. This change extends the frequency range to be consistent with current input from seismic hazard assessments.	The text in the "impact of changes" column, indicates that the highest frequency of interest is no longer limited to 33Hz, but rather should reflect the current input from the seismic hazard assessment which may be greater than 33Hz. Although this value is typically 33Hz for standard shaped spectra (as indicated in Table 2 of CSA N289.3-10), higher values can be used depending on the values used in the seismic hazard study. The 33Hz value has not been used as a limit in the Pickering seismic assessments. No PSR2 gap exists against the requirements of the Clause.
6	Clarification of the requirements for consideration of seismic fatigue. This change increases clarity of provisions for seismic fatigue.	CSA N289.3-10 Clause 7.3.3.1 states: "*Seismic fatigue analysis of ASME Class 1 components (vessels, pumps, valves and piping) shall not be required when the range of primary plus secondary stresses due to the seismic load alone (inertia plus anchor movement) is limited to 3 Sm (the design stress intensity) or equivalent.* Seismic fatigue analysis has been carried out as per the requirements of the Clause. For example: • Per 44RS-60350-AR-004, "Pickering A Return to Service - Seismic Analysis of Instrument Panel" [B.11-35], section 7.8 states "*seismic fatigue evaluation is not required since the range of stress due to seismic loads alone is less than 3 Sm*" • Per NK30-REP-33319-00001, "Pickering B PHT 3331 Feed Circuit Nuclear Class 1 Pipe Stress Analysis Report" [B.11-36], Section 6.2 states "*in the event some components exceed the 3 Sm limit (for seismic primary plus secondary stress range), then a less conservative detailed fatigue analysis for these components may be conducted*".

#	CSA "Summary of Significant Changes from Previous Edition"	Assessment of Safety Significance of Changes
		• Per NK30-REP-71389-0094106, "Stress Report – Pickering NGS B Emergency Water System" [B.11-37], section 6.2 states, *"in the event some components exceed the 3 Sm limit (for seismic primary plus secondary stress range), then a less conservative detailed fatigue analysis for these components may be conducted".* Hence, there is evidence that seismic fatigue analysis was carried out as per the requirements of this Clause for Pickering A and B. No PSR2 gap exists against the requirements of the Clause.

B.11.3 Compliance Summary for Pickering PSR2

There is one PSR2 gap for Pickering NGS compliance with N289.3-10 which is applicable to Safety Factor 3:

1. Clause 4.4.4.5 of CSA N289.3-10 states: "The power spectral density (PSD) function of each time-history shall be calculated and shown to not have any significant gaps in energy over the frequency intervals outlined in Table 2…." The calculation of PSD is not addressed in the Pickering A or B PRA Based SMAs. The Pickering NGS A PRA Seismic Guide and the OPG PRA Guide do not identify any requirements for PSD. Also, evidence in the form of a calculation for time histories which represent the design ground motion was not found (which is a precursor for the PSD calculation). The lack of evidence of calculated time histories was also identified as a gap in the Darlington ISR (ISR Issues #D352 and #D617 – Documented evidence in the form of a calculation to show that the generated time history correctly represents the design ground response spectrum within the prescribed requirements has not been provided). The closure reference for #D352 and #D617 makes use of the detailed assessment performed in NK38-REP-03680-10224 R000 which is specific to Darlington. A similar assessment for Pickering NGS could not be found. As a result, there is a gap for PSR2 to provide similar evidence to show that: a) the generated time history used within seismic analyses of safety-related systems correctly represents the design ground response spectrum for the Pickering site in compliance with N289.3-10, and b) the PSD function of each time-history has been calculated and shown to not have any significant gaps in energy over the frequency intervals.

B.11.4 References

[B.11-1] CSA Standard N289.3-10, *Design Procedures for Seismic Qualification of Nuclear Power Plants*, January 2010; Update No. 2, November 2015.

[B.11-2] CNSC Acts and Regulations web page: *Regulatory Documents*, *http://cnsc-ccsn.gc.ca/eng/acts-and-regulations/regulatory-documents/index.cfm#R14*, April 2016.

[B.11-3] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.11-4] CSA Impact Statement, *Notification of CSA N289.3 Public Review; Product: Amendment – Product Designation N289.3-10 – Product Title: Design Procedures for Seismic Qualification of Nuclear Power Plants*, Date not provided.

[B.11-5] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.11-6] OPG Report, NK30-REP-03680-00001 R000, *Pickering NGS-B - Integrated Safety Review – Plant Design Safety Factor*, August 2007.

[B.11-7] CAN3/CSA Standard N289.3-M81 (R2003), *Design Procedures for Seismic Qualification of CANDU Nuclear Power Plants*, January 1981.

[B.11-8] OPG Plan, NK30-PLAN-03680-00002 R000, *Pickering B- Integrated implementation Plan*, December 2011.

[B.11-9] OPG Plan, Plan, NK30-PLAN-00531-00001 R005, *Pickering 5-8 Continued Operations Plan*, December 2015.

[B.11-10] CAN3/CSA Standard N289.3-M81 (R1998), *Design Procedures for Seismic Qualification of CANDU Nuclear Power Plants*, January 1981.

[B.11-11] OPG Letter, NA44-CORR-00531-00381 R000, R. J. Strickert to J.S.C. Tong, *Pickering A - Updated Basis for Return to Service Document*, April 20, 2001.

[B.11-12] OPG Letter, CD#P-CORR-00531-03719, *Application for Renewal of Pickering Nuclear Generating Station Power Reactor Operating Licence*, July 4, 2012.

[B.11-13] CAN3/CSA Standard N289.3-M81 (R2008), *Design Procedures for Seismic Qualification of CANDU Nuclear Power Plants*, January 1981.

[B.11-14] OPG Report, NK38-REP-03680-10008 R000, *Review of CAN/CSA-N289.3-M81 (R2008) (January 1981), Design Procedures for Seismic Qualification of CANOU Nuclear Power Plants for Darlington Integrated Safety Review*, August 2011.

[B.11-15] OPG Report, NK38-REP-03680-10104 R000, *Darlington NGS Integrated Safety Review (ISR) – Final ISR Report*, October 2011.

[B.11-16] OPG Report, NA44-REP-03611-00022 R000, *PRA-Based Seismic Margin Assessment of PNGS-A*, January 2014.

[B.11-17] OPG Report, NK30-REP-03611-00013 R001, *PRA-Based Seismic Margin Assessment of PNGS-B*, April 2015.

[B.11-18] CSA Standard N289.3-10, *Ground Motion Determination for Seismic Qualification of Nuclear Power Plants*, May 2010; Update No. 1, August 2012.

[B.11-19] OPG Report, NK38-REP-03680-10151 R000, *Code Review Of CSA N289.3-10 Design Procedures For Seismic Qualification Of Nuclear Power Plants*, April 2014.

[B.11-20] OPG Report, NK38-REP-03680-10008-ADD-001 R000, *Addendum to the CAN/CSA N289.3 M81 Code Review Report for Darlington ISR*, February 2014.

[B.11-21] OPG Report, NK38-REP-03680-10104-ADD-001 R000, *Darlington NGS Integrated Safety Review (ISR) – Final ISR Report Addendum*, June 2013.

[B.11-22] OPG Report, NK38-REP-03680-10224 R000, *Spectrum-enveloping N289.3 Code Compliance of Darlington NGS Seismic Time Histories*, July 2014.

[B.11-23] OPG Report, NK38-REP-03680-10185 R002, *Darlington NGS Integrated Implementation Plan (IIP)*, April 2015.

[B.11-24] OPG Report, NK38-REP-03680-10104-ADD-002 R000, *Darlington NGS Integrated Safety Review (ISR) - Final ISR Report Addendum 002*, November 2013.

[B.11-25] OPG Report, NK38-REP-03611-10051 R002, *Darlington NGS A Seismic Probabilistic Risk Assessment (DARA-Seismic)*, June 2015.

[B.11-26] OPG Letter, NK38-CORR-03680-0486309 R000, *List of Systems and Structures for Inclusion within the CSA N289.3 2010 Code Refresh*, December 2013.

[B.11-27] OPG Report, NK38-REP-03680-10197 R001, *Evaluation of Seismic, Pressure Boundary and Accident Management ISR Acceptable Deviation Issues for Aggregate Effects between Issues*, November 2014.

[B.11-28] OPG Guideline, NA44-GUID-03611-00015 R001, *Pickering NGS A Probabilistic Risk Assessment Guide – Seismic*, November 2013.

[B.11-29] OPG Guideline, N-GUID-03611-10001 Volume 7 R001,*OPG Probabilistic Risk Assessment (PRA) Guide – Seismic*, March 2011.

[B.11-30] CNSC Letter, NK30-CORR-00531-04876 R00, *Pickering NGS-B – Integrated Safety Review (ISR) – CNSC Review of Acceptable Deviations and Discrepancies for the Plant Design Safety Factor Report,* June 27, 2008.

[B.11-31] OPG Report, NK30-REP-P0732002, *Pickering Generating Station B Interim Report Preliminary Engineering Phase Geotechnical Evaluation of Site Features (Report No. 74148)*, June 1, 1974.

[B.11-32] OPG Design Manual, NK30-DM-21002-10001 R00, *Seismic Route,* April 1999.

[B.11-33] OPG Engineering Calculation, NA44-CALC-71320-10001 R000, *Seismic Analysis of 7132-HX501, -HX502 Support Frame,* April 2005.

[B.11-34] OPG Engineering Calculation, NK30-CALC-21459-00005 R002, *Unit 7 Boiler Blow Down Pipe Civil Pipe Supports Miscellaneous Steel,* November 2012.

[B.11-35] AECL Analysis Report, 44RS-60350-AR-004 R000, *Seismic Analysis of Instrument Panel (1-60350-PL-2115/2215),* February 2002.

[B.11-36] OPG Report, NK30-REP-33319-00001 R001, *Pickering B PHT 3331 Feed Circuit Nuclear Class 1 Pipe Stress Analysis Report,* November 1982.

[B.11-37] OPG Report, NK30-REP-71389-0094106 R001, *Emergency Water Supply System Nuclear Class 1 Pipe Stress Analysis Report Per ASME Section III NB 3600 and CSA N289.3,* November 1984.

B.12 CSA N289.4-12, "Testing Procedures for Seismic Qualification of CANDU Nuclear Power Plants"

B.12.1 Background

The following paraphrased from the Preface of CSA N289.4-12 [B.12-1] provides a brief overview of the purpose of this Standard and the requirements expressed therein:

> *The purpose of CSA N289.4 is to provide a basis for the development of specifications for seismic qualification by testing, and to aid purchasers, suppliers, and testing laboratories in selecting the appropriate test method(s) for performing acceptable seismic qualification tests that meet a quality and standard commensurate with the safety principles necessary to comply with the Canadian nuclear safety philosophy.*
>
> *CSA N289.4 provides the design requirements and methods for seismic qualification of specific components and systems by testing methods.*

N289.4-12 is directly relevant to Safety Factor 1 (Plant Design) since the CNSC web site [B.12-2] identifies the Standard as being relevant to the Safety Control Area on "Physical Design". Also, as this Standard addresses seismic qualification of SSCs, it applicable to Safety Factor 3 (Equipment Qualification).

CSA N289.4 is identified in Appendix E.1 of the R04 Pickering Licence Conditions Handbook [B.12-3] as "Guidance and Criteria". The current revision of the Standard is N289.4-12. It supersedes the previous edition, published in 1986. Changes since the last version of the Standard in 1986 have ranged from editorial, to the addition of supplementary information as well as the addition of new requirements.

The CSA N289.4-12 Impact Statement [B.12-4] provides a "Summary of significant changes from the previous edition" which identifies several primary changes to the Standard described in Section B.12.2 below. In addition to findings resulting from review of the CSA N289.4-12 Impact Statement, the results of PSR1 N289.4 reviews (Pickering B and Darlington ISRs), as well as reviews performed since PSR1, have also been assessed for applicability to PSR2 in Section B.12.2.

As identified in Reference [B.12-5], the Pickering PSR2 review of CSA N289.4-12 is an Incremental Review. As discussed in Section 2.0 of this Report, PSR2 Incremental Review includes an assessment of the intent of recent changes to the L/R/C/S on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- <u>Compliance:</u> Compliance indicates that the change in the safety requirement, per the topical review, is met.

- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.12.2 Compliance Assessment for Pickering PSR2

The versions of N289.4 subject to previous reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

B.12.2.1 Application of PSR1 Reviews

Pickering NGS

Pickering Units 5-8

OPG Report NK30-REP-03680-00003 R000 [B.12-6] documents the code review of CSA N289.4-M86 (R2003) [B.12-7] for the Equipment Qualification Safety Factor. The code review report findings were:

> *The review report concluded that Equipment qualification measures to assure safe operation under accident and earthquake environmental conditions and normal operational conditions have been adequately implemented. The various programs, procedures and controls that establish and maintain these measures conform to the requirements of CSA standards and are consistent with the recommendations of international guides.*

Therefore, there are no PSR2 gaps identified based on the Pickering B ISR.

Pickering Units 1,4

For Pickering A Return to Service, the active version of the Standard at the time was CSA-N289.4-M86 (R1998) [B.12-8]. Per NA44-CORR-00531-00381 R000 [B.12-9], this version of the Standard "Pertains mostly to design support analysis" and it is stated that "Recent completion of the Seismic Margin Assessment makes review of this CSA Standard not necessary". As noted in the Pickering PROL Renewal Application [B.12-10], for Pickering A the common containment structures were designed to exceed the National Building Code 1965 seismic design provisions and were subsequently confirmed analytically to meet seismic design requirements. No PSR2 gaps are identified based on the Pickering A Return to Service assessments.

Darlington NGS

The Darlington ISR was performed using N289.4-M86 (R2008) [B.12-11] and is documented in OPG Report NK38-REP-03680-10009 R000 [B.12-12], which notes that this version of the Standard underwent a clause-by-clause (non-PROL) intent review.

The clauses in N289.4 are largely programmatic from the perspective of the specified requirements. Therefore, the Darlington ISR conclusions are applicable to Pickering PSR2 as discussed below.

NK38-REP-03680-10009 R000 found Darlington NGS is in compliance with all of the clauses of N289.4-M86 (R2008) except the following clause identified as a gap (paraphrased) [B.12-12]:

> *Clause 4.1.4: Multi-axis and Multi-frequency Coupling.*
>
> *The multiplication factor for uni-axial testing in the sine sweep test is 1.2, which is lower than the 1.4 specified in CAN/CSA-N289.4-M86. No documentation was found justifying the lower multiplication factor.*

The gap identified for Clause 4.1.4 was assigned as Gap #00175 under Darlington ISR Issue #D063 in the Darlington Final ISR Report NK38-REP-03680-10104 R000 [B.12-13]:

ISR Issue #	Gap	Clause
D063	00175	4.1.4

For Gap #00175, the Darlington Final ISR Report NK38-REP-03680-10104 R000 [B.12-13] also provided a proposed resolution for compliance:

> *The completion of this Seismic Probabilistic Risk Assessment will provide compliance with CNSC S-294 and is consistent with CAN/CSA-N289.1-08 methods. The Seismic Probabilistic Risk Assessment will confirm the adequacy of the DNGS design to withstand an earthquake, or suggest possible design modifications in order to ensure the proper level of safety is maintained.*
>
> *Management action AR# 28127914 has been raised to track the Darlington Risk Assessment project to completion in order to confirm that it satisfactorily resolved Issue D063.*

Gap #00175 was reclassified as closed with the completion of the Darlington Seismic Probabilistic Risk Assessment (SPRA), which also closed Management action AR# 28127914. Closure of Gap #00175 is documented with the following justification provided in Appendix E of the Darlington Final ISR Report Addendum 002 report NK38-REP-03680-10104-ADD-002 R000 [B.12-14]:

> *In accordance with Clause 5.4 of CSA N289.1-08, an acceptable method of re-evaluating existing Nuclear Plants for seismic considerations is SPRA. SPRA is also an acceptable method of seismic qualification per Clause 5.3.9. The recently issued SPRA, NK38-REP-03611-10051, confirms that the original design and seismic qualification of DNGS provides an adequate level of Safety from the earthquake hazard at the DNGS site.*

Regardless of this change in multiplication factor, the original testing has produced an adequate design and continuing qualification has been confirmed via SPRA.

With closure of Gap #00175 for Darlington supported by its SPRA, there is also no gap for Pickering PSR2 since both Pickering Units 1,4 and Pickering Units 5-8 have recently issued SPRAs.

Following the original code review for the Darlington ISR, a code refresh review based on the latest version of the Standard N289.4-12 [B.12-1] was conducted and documented in OPG Report NK38-REP-03680-10152 R000 [B.12-15]. The code refresh review comprised a clause-by-clause review of all changes in N289.4-12 relative to N289.4-M86 (R2008) [B.12-11]. NK38-REP-03680-10152 R000 concluded the following [B.12-15]:

The review confirms OPG Nuclear governance is in compliance with the changes in the requirements of CSA-N289.4 -12 [R-1] relative to CAN3-N289.4-M86 [R-2] with the exception of one ISR Gap.

The gap identified in NK38-REP-03680-10152 R000 [B.12-15] was assigned as Gap #02041 under Darlington ISR Issue #D427, as indicated below. The scope of this ISR Issue relates to the requirement to account for aging degradation effects that may impair seismic functionality in seismic qualification by testing.

ISR Issue #	Gap	Clause
D427	02041	4.2.6

Darlington ISR Issue #D427 was reclassified as an Acceptable Deviation (AD) in report NK38-REP-03680-0477236 R000 [B.12-16] based on the following rationale:

A review of OPG governance on Integrated Aging Management [R-6] is in place and effective. OPG's commitment [R-8] to using the most current published standard for Testing procedures for seismic qualification of nuclear power plant structures, systems, and components [R-2] will ensure that clause 4.2.6 will be followed. This demonstrates the intent of Clause 4.2.6 of N289.4-12 will be met and concludes that this ISR Issue can be considered an acceptable deviation with no further action required.

The Darlington code refresh demonstrated compliance with Clause 4.2.6 of CSA N289.4-12 using fleet-wide, OPG Nuclear governance documents (including Integrated Aging Management) which are applicable to Pickering. Consequently there is no PSR2 gap related to Clause 4.2.6.

Station-specific documents (including the Darlington seismic design guide, Darlington Reports and Darlington-specific technical specifications for seismic qualification) were used as the basis for compliance in the clause-by-clause Darlington code refresh review for clauses 4.2.1, 4.2.2.2, 4.2.3.1, 4.2.5, 4.3.2, 5.2.2.2.5, 5.7, 5.8.1, 5.8.1.2, 7.2.1, 7.7.1, 7.7.4 and 8.2. Pickering-specific seismic design guides, reports and technical specifications that are equivalent to those used to demonstrate Darlington compliance with the changes made in CSA N289.4-12 were identified. However, a detailed review to confirm that the Pickering-specific documents fully comply with the requirements of the clauses listed above is needed. As a result, this is a PSR2 gap **(PSR2 CSA N289.4-12 Gap #1)**."

In November 2014, a study documented in OPG Report NK38-REP-03680-10197 R001 [B.12-17] examined the aggregate effects of ADs that are related to the seismic, pressure boundary, or accident management programs. The purpose of this study was to determine whether there are aggregate effects of these ADs. This report concluded there were no aggregate effects from ADs associated with the seismic, pressure boundary or accident management barriers and, therefore, no impacts on public safety. The AD related to N289.4-12 [B.12-1] listed above for Darlington ISR Issue #D427 was part of the study.

B.12.2.2 Application of Post-PSR1 Reviews

The CSA N289.4-12 Impact Statement [B.12-4] provides a summary of the significant changes from the previous edition of the Standard:

- The scope has been broadened to have the standard apply to nuclear power plant structure systems and component irrespective of plant design type (technology neutral). This is reflected in the changed title of the standard. This change accommodates non CANDU designs;

- By reference to CSA N289.1-08 and providing detailed guidance in annexes to this standard, qualification by similarity, experience and industry test database has been acknowledged in the standard as being acceptable when justified. This change incorporates the current Canadian and international practice to permit seismic qualification by similarity, experience and industry test database as being acceptable methods. This may result in minimal cost and better use of resources;

- For the SSCs requiring functionality demonstration, preference of qualification by shake testing is emphasized. This change is aligned with recent regulatory documents (e.g. RD-337) and with current international industry practice;

- While retaining the acceptability of seismic testing using single frequency and uni- and bi-axial testing as in the old edition, the new edition emphasises preference for multi-axis random motion that more closely simulates the earthquake motion. This change reflects current industry practice for using multi-axis random motion test methodology that more closely simulates the earthquake motion;

- In recognition of possible high frequency contents in a site-specific ground motion response spectrum (GMRS), the cut-off frequency in testing of equipment that is sensitive to high frequency effects is extended to above 33Hz. This change reflects the recent understanding of the earthquake characteristics in central/eastern North America. The testing requirements and test simulations are required to incorporate high frequency effects;

- The Standard provides more detailed guidance solicited from industry experience to better align and harmonize with the international practices in qualification by seismic testing. This change provides clarity in implementation and streamlined compliance with regulator expectations;

- More detailed requirements for preparation of a test specification, test plan and documentation of a test report have been included in the Standard. The impact of this change is that improved quality of documentation will facilitate engineering work; and

- Additional guidance has been added to the Standard in the annexes on qualification by similarity, qualification by experience database, qualification of valves, qualification of dampers, isolators and dynamic restraints and qualification of cabinet mounted equipment. This change makes the standard more user-friendly.

The Darlington ISR code refresh review based on N289.4-12 discussed earlier addresses these latest changes. The applicability of that work to Pickering was demonstrated in many cases. However, a PSR2 gap was identified to confirm Pickering-specific documents are compliant with 13 clauses that had used Darlington-specific documents as the basis of compliance. The PSR2 gap is identified in Section B.12.2.1.

OPG has committed to submitting to the CNSC annual Code-over-Code reviews to identify significant technical changes to the requirements of the set of engineering design-related code and standard effective dates, as agreed with the CNSC. The code-over-code review methodology documents [B.12-18] and [B.12-19] list Standard N289.4-12 [B.12-1], signifying that future versions of the Standard will be included in subsequent Code-over-Code reviews. A Code-over-Code review of CSA N289.4-12 relative to a more recent version has not been prepared because CSA N289.4-12 is the latest version.

Further, OPG Nuclear Standard N-STD-MP-0025 R001 [B.12-20] sets forth the general integrated approach to seismic requirements for the applicable SSCs within OPG's nuclear facilities. The approach and framework for OPG activities related to the general requirements for seismic design and qualification of OPG nuclear facilities are in accordance with the CSA N289 series of standards.

No additional PSR2 gaps are identified based on the post PSR1 assessments.

B.12.3 Compliance Summary for Pickering PSR2

There is one PSR2 gap for Pickering NGS compliance with CSA N289.4-12 which is applicable to Safety Factor 3:

1. Station-specific documents (including the Darlington seismic design guide, Darlington Reports and Darlington-specific technical specifications for seismic qualification) were used as the basis for compliance in the clause-by-clause Darlington code refresh review for clauses 4.2.1, 4.2.2.2, 4.2.3.1, 4.2.5, 4.3.2, 5.2.2.2.5, 5.7, 5.8.1, 5.8.1.2, 7.2.1, 7.7.1, 7.7.4 and 8.2. Pickering-specific seismic design guides, reports and technical specifications that are equivalent to those used to demonstrate Darlington compliance with the changes made in CSA N289.4-12 were identified. However, a detailed review to confirm that the Pickering-specific documents fully comply with the requirements of the clauses listed above is needed. As a result, this is a PSR2 gap.

B.12.4 References

[B.12-1] CSA Standard N289.4-12, *Testing Procedures for Seismic Qualification of Nuclear Power Plant Structures, Systems, and Components*, August 2012.

[B.12-2] CNSC Acts and Regulations web page: *Regulatory Documents*, *http://cnsc-ccsn.gc.ca/eng/acts-and-regulations/regulatory-documents/index.cfm#R14*, April 2016.

[B.12-3] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.12-4] CSA Impact Statement and Publication Notice, *Product: New Edition – Product Designation: N289.4-12 - Product Title: Testing Procedures for Seismic Qualification of Nuclear Power Plant Structures, Systems, and Components, - Date of Release: August 2012,* Date not provided.

[B.12-5] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.12-6] OPG Report, NK30-REP-03680-00003 R000, *Pickering NGS-B Integrated Safety Review – Safety Factor for Equipment Qualification*, April 2007.

[B.12-7] CAN3/CSA Standard N289.4-M86 (R2003), *Testing Procedures for Seismic Qualification of CANDU Nuclear Power Plants*, June 1986.

[B.12-8] CAN3/CSA Standard N289.4-M86 (R1998), *Testing Procedures for Seismic Qualification of CANDU Nuclear Power Plants*, June 1986.

[B.12-9] OPG Letter, NA44-CORR-00531-00381 R000, R. J. Strickert to J.S.C. Tong, *Pickering A - Updated Basis for Return to Service Document*, April 20, 2001.

[B.12-10] OPG Letter, CD#P-CORR-00531-03719, Application for Renewal of Pickering Nuclear Generating Station Power Reactor Operating Licence, July 4, 2012.

[B.12-11] CAN3/CSA Standard N289.4-M86 (R2008), *Testing Procedures for Seismic Qualification of CANDU Nuclear Power Plants*, June 1986.

[B.12-12] OPG Report, NK38-REP-03680-10009 R000, *Review of CAN/CSA-N289.4-M86 (R2008) (June 1986), Testing Procedures for Seismic Qualification of CANDU Nuclear Power Plants for Darlington Integrated Safety Review*, August 2011.

[B.12-13] OPG Report, NK38-REP-03680-10104 R000, *Darlington NGS Integrated Safety Review (ISR) – Final ISR Report*, October 2011.

[B.12-14] OPG Report, NK38-REP-03680-10104-ADD-002 R000, *Darlington NGS Integrated Safety Review (ISR) - Final ISR Report Addendum 002*, November 2013.

[B.12-15] OPG Report, NK38-REP-03680-10152 R000, *Code Refresh Review of CSA N289.4-12, Testing Procedures for Seismic Qualification of Nuclear Power Plant Structures, Systems, and Components, for DNGS ISR*, July 2013.

[B.12-16] OPG Report, NK38-REP-00770-0477236 R002, *Nuclear Refurbishment Issue Resolution Form – Darlington – Issue #D427 - Seismic Qualification - General*, August 2013.

[B.12-17] OPG Report, NK38-REP-03680-10197 R001, *Evaluation of Seismic, Pressure Boundary and Accident Management ISR Acceptable Deviation Issues for Aggregate Effects between Issues*, November 2014.

[B.12-18] OPG Letter, N-CORR-00531-06858 R000, W. M. Elliott to M. Santini, K. Glenn and F. Rinfret, *2014 Code-Over-Code Review for OPG Nuclear Fleet*, April 2015.

[B.12-19] OPG Report, N-REP-00590-058059 R000, *2015 COC - OPG Compilation*, March 2016.

[B.12-20] OPG Nuclear Standard, N-STD-MP-0025 R001, *General Requirements for Seismic Qualification of OPG Nuclear Facilities*, October 2015.

B.13 CSA N289.5-12, "Seismic Instrumentation Requirements for CANDU Nuclear Power Plants"

B.13.1 Background

The following paraphrased from the Preface of CSA N289.5-12 [B.13-1] provides a brief overview of the purpose of this Standard and the requirements expressed therein:

> *The purpose of CSA N289.5 is to aid nuclear facilities owners in the determination of the extent and nature of instrumentation to be installed. It is also intended to aid owners and equipment suppliers by specifying instrumentation commensurate with Canadian nuclear safety principles.*
>
> *CSA N289.5 provides a basis for specifying requirements for seismic instrumentation to monitor site-specific seismic responses.*

N289.5-12 is directly relevant to Safety Factor 1 (Plant Design) since CNSC web site [B.13-2] identifies the Standard as being relevant to the Safety Control Area on "Physical Design". Also, since this Standard addresses seismic qualification of SSCs, it is applicable to Safety Factor 3 (Equipment Qualification).

CSA N289.5 is identified in Appendix E.1 of the R04 Pickering Licence Conditions Handbook [B.13-3] as "Guidance and Criteria". The current revision of the Standard is N289.5-12 [B.13-1]. It supersedes the previous edition, published in 1991. Changes since the last version of the Standard in 1991 have ranged from editorial, to the addition of supplementary information as well as the addition of new requirements.

The CSA Impact Statement for N289.5-12 [B.13-4] provides a "Summary of significant changes from the previous edition" which identifies several primary changes to the Standard described in Section B.13.2.2 below. In addition to findings resulting from review of the CSA N289.5-12 Impact Statement, the results of PSR1 N289.5 reviews (Pickering B and Darlington ISRs), as well as reviews performed since PSR1, have also been assessed for applicability to PSR2 in Section B.13.2.1.

As identified in Reference [B.13-5], the Pickering PSR2 review of CSA N289.5-12 is an Incremental Review. As discussed in Section 2.0 of this Report, PSR2 Incremental Review includes an assessment of the intent of recent changes to the L/R/C/S on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- <u>Compliance:</u> Compliance indicates that the change in the safety requirement, per the topical review, is met.

- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.13.2 Compliance Assessment for Pickering PSR2

The versions of N289.5 subject to previous reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

B.13.2.1 Application of PSR1 Reviews

Pickering NGS

Pickering Units 5-8

OPG Report NK30-REP-03680-00001 R000 [B.13-6] documents the code review of CSA version N289.5-M91 (R2003) [B.13-7] for the Plant Design Safety Factor. The code review report findings were:

> *The design of the Seismic Monitoring System was evaluated against the Standard CSA N289.5-91 on a clause by clause basis. The Seismic Monitoring System was found to comply with the requirements of the Standard. The one exception to the Seismic Monitoring System is that the system Components are becoming obsolete and the system is experiencing maintainability issues (References 9, 10, 11 and 12). Pickering Engineering is currently reviewing this matter and will be identifying a path forward strategy.*

Appendix B of the Pickering B Integrated Implementation Plan NK30-PLAN-03680-00002 R000 [B.13-8] describes the improvement action G09-01 established to resolve the seismic monitoring equipment upgrade gap. The improvement plan is documented under ISR Issue G9 - Seismic Monitoring System:

> *A Seismic Monitoring System Replacement project was initiated to improve the reliability of the current seismic monitoring system and to address the findings from the CNSC Type 2 inspection of this system. The project scope is to complete the design, procurement, installation, staff training, and testing to upgrade the Seismic Monitoring System to meet the intent of CSA Standard N289.5-M91. The detailed design of the Seismic Monitoring System upgrade is currently in progress and is expected to be completed by March 2012.*

The Pickering Continued Operations Plan (NK30-PLAN-00531-00001 R005 [B.13-9]) states:

> *2014: This action is complete. The new seismic monitoring system was installed, commissioned, and placed in service as documented in P-REP-61150-00002, 'Available For Service Report PN Seismic Monitoring System Replacement Project*

(Syscom Instruments) Project 13·49129 Unit 018', MASTER EC: 111865, DESIGN ECS: 113888, 116978, 113887, 117638. This work was completed by December 6, 2013.

Therefore, there are no PSR2 gaps which result from the Pickering B ISR.

<u>Pickering Units 1,4</u>

For Pickering A Return to Service, the active version of the Standard at the time was CSA N289.5-M91 (R1998) (January 1991) [B.13-10]. Per NA44-CORR-00531-00381 R000 [B.13-11], for this version of the Standard "Pickering A was not reviewed against CSA CAN3-N289.5-M91, because seismic instrumentation has been installed at Pickering site and, therefore, such a review was unnecessary". As noted in the Pickering PROL Renewal Application [B.13-12], for Pickering A the common containment structures were designed to exceed the National Building Code 1965 seismic design provisions and were subsequently confirmed analytically to meet seismic design requirements. No PSR2 gaps are identified based on Pickering A Return to Service assessments.

Darlington NGS

The Darlington ISR was performed using N289.5-M91 (R2008) [B.13-13] and is documented in report NK38-REP-03680-10041 R000 [B.13-14], which notes that this version of the Standard underwent a clause-by-clause (non-PROL) intent review.

The clauses in N289.5 are primarily programmatic from the perspective of the specified requirements. Therefore, significant portions of the Darlington ISR assessment are applicable to Pickering as discussed below.

NK38-REP-03680-10041 R000 found that Darlington NGS is in compliance with all of the clauses of N289.5-M91 (R2008) except the following clauses identified as a gap [B.13-14]:

The review finds that Darlington, for the most part, complies with the standard. However, one gap has been identified. This relates to surveillance Clauses 6.2.2 and 6.2.5 where no evidence of on-power testing or regular scheduled surveillance of the system could be identified in OPG's electronic systems. Commissioning tests were performed and results are available and a Control Maintenance Procedure for testing exists. However, no Preventative Maintenance IDs were identified and no evidence of regular testing could be located.

During the initial ISR Gap resolution process, it was determined that the gap listed above could not be reclassified as compliant. Therefore, the ISR Gap remained for N289.5-M91 as identified in NK38-REP-03680-10041 R000 [B.13-14].

The Darlington ISR gap identified above was assigned to the following Darlington ISR Issue in the Darlington Final ISR Report NK38-REP-03680-10104 R000 [B.13-15]:

ISR Issue #	Gap	Clause
D064	00701	6.2.2
	00702	6.2.5

Darlington ISR Issue #D064 is defined as:

> *The gaps in this Issue indicated that on-power testing and routine inspection and recalibration of the entire Seismic Monitoring System was not being done. No scheduled testing or Predefined Maintenance Identifications were identified for the system to demonstrate conformance with the testing requirements.*

Appendix M of the Darlington Final ISR Report NK38-REP-03680-10104 R000 [B.13-15] documents that this Issue has been reclassified as an Acceptable Deviation (AD):

> *Station Condition Record D-2010-11677 was raised following the in-depth scrutiny of the Seismic Monitoring System and addresses this issue. It indicates that immediate action was taken to institute regular weekly walkdowns by the System Responsible Engineer to confirm the status of the Seismic Monitoring System.*
>
> *In addition, Change Request # 2010-00593 has been entered into the Preventive Maintenance Living Program and submitted to set up routine maintenance activities, in accordance with the vendor manual recommendations. By establishing weekly walkdowns, scheduled testing and routine maintenance activities, the Seismic Monitoring System is expected to perform as per design.*
>
> *No further action required. ISR Issue reclassified as an Acceptable Deviation.*

This Darlington ISR gap is potentially applicable to Pickering and was therefore investigated to confirm that seismic instrumentation tests are listed in the Pickering PMIDs. These PMIDs are complete for Pickering NGS per P-REP-61150-00002, "Available for Service Report PN Seismic Monitoring System Replacement Project (Syscom Instruments) Project 13·49129 Unit 018" [B.13-16]. Therefore, this is not a gap for PSR2.

Following the original code review for the Darlington ISR, a code refresh review based on the latest version N289.5-12 [B.13-1] was conducted and documented in OPG Report NK38-REP-03680-10153 R000 [B.13-17]. The code refresh review comprised a clause-by-clause review of all changes in N289.5-12 relative to N289.5-M91 (R2008) [B.13-13]. NK38-REP-03680-10153 R000 found the following:

The changes in the requirements are minor and reflect improvements for clarity or for formatting purposes. The review of the changed clauses in this code review report also confirms that OPG Nuclear governance is in compliance with the requirements of CSA-N289.5 -12 [R-1].

In summary, the review did not identify any additional gaps relative to the requirements of CSA-N289.5-12 [R-1].

OPG Report NK38-REP-03680-10153-ADD-001 R000 [B.13-18] documents the results of a review of Darlington NGS governance regarding seismic qualification against only Clause 5 of N289.5-12 [B.13-1] in response to comments from the CNSC related to the code refresh review report [B.13-17]. The CNSC comment is summarized as follows from OPG Report NK38-REP-00770-0517256 R001 [B.13-19]:

The code refresh report for CSA N289.5-12 did not include a review of Section 5 of the code. Section 5 is titled 'New nuclear power plants and on-site nuclear facilities' and although it would not normally be applicable to an existing station, it is a requirement of the review of modern codes and standards for the Darlington Integrated Safety Review. The code refresh report for CSA N289.5-12 should have included a review of Section 5 of that standard.

NK38-REP-03680-10153-ADD-001 R000 [B.13-18] identified the following gaps against Clause 5 of N289.5-12:

- *Clause 5.1.5: Recording devices installed at Darlington are triggered and not continuous; therefore the intent of this clause is not met.*
- *Clauses 5.2.3.1.2, 5.2.3.2.3 (a) and (c): The intent of these code requirements regarding the number of seismic monitoring system sensors is not met.*
- *Clause 5.2.3.1.9: It could not be confirmed if all accelerometers are free from any strong ambient vibration and/or what measures were taken to ensure that strong ambient vibrations do not preclude recording of earthquake data, therefore the intent of this clause is not met.*
- *Clauses 5.2.3.2.3 (b): There is no remote display and annunciation functions for each reactor unit in the MCR [Main Control Room]. Only the U0 [Unit 0] panel in the MCR has seismic system annunciations available, therefore the intent of this clause is not met.*

The gaps against Clause 5 of N289.5-12 [B.13-18] were closed or reclassified as Acceptable Deviations (ADs) per the following reports:

Issue	Gap	Clause	Status	Reference
D621	02327	5.1.5	Acceptable Deviation	NK38-REP-00770-0528825 R000 [B.13-20]
D622	02328	5.2.3.2.3	Acceptable Deviation	NK38-REP-00770-0528826 R000 [B.13-21]
D623	02329	5.2.3.1.9	Closed	NK38-REP-00770-0528827 R000 [B.13-22]
D624	02330	5.2.3.1.2	Acceptable Deviation	NK38-REP-00770-0528828 R000 [B.13-23]
	02331	5.2.3.2.3 (a)		
	02332	5.2.3.2.3 (b)		

The above gaps against Clause 5 of N289.5-12 [B.13-1] require no further action for Darlington. However, three of these are identified as a PSR2 gap (**PSR2 CSA N289.5 Gap #1**) for the following reasons: (Note: These gaps are closely related and are therefore identified as a single PSR2 gap).

- Darlington ISR Issue D621 refers to specific seismic instrumentation recording device trigger settings for Darlington instrumentation and then classifies the gap as an Acceptable Deviation. Reference [B.13-24] states that the initial setting parameters for the Pickering seismic monitoring system for recording seismic motion (per Master EC 111865 discussed under the Pickering Units 5-8 review section above) were determined to meet the requirements of N289.5-12. Therefore, this is not a gap for PSR2.

- Darlington ISR Issue D622 was deemed to be of low safety significance. The same rationale may apply at Pickering. However, it must first be demonstrated that Pickering has the same set-up of seismic instruments as Unit 0 at Darlington. Therefore, this is a gap for PSR2.

- Darlington ISR Issue D623 was deemed to be of low safety significance. The same rationale may apply at Pickering. However, it must first be demonstrated that similar accelerometers are used at Pickering, and that their locations are not affected by strong ambient vibration. Therefore, this is a gap for Pickering PSR2.

- Darlington ISR Issue D624 refers to specific Darlington instrumentation. It must be demonstrated that Pickering seismic instruments have the same capabilities as the Darlington instruments (fleet-wide or Pickering-specific standards that would ensure that the Pickering seismic instruments have the same capabilities as the Darlington instruments could not be found). Therefore, this is identified as a gap for PSR2.

In November 2014, a study documented in report NK38-REP-03680-10197 R001 [B.13-25] examined the aggregate effects of ADs that are related to the seismic, pressure boundary, or accident management programs. The purpose of this study was to determine whether there are aggregate effects of these ADs. This report concluded there were no aggregate effects from ADs associated with the seismic, pressure boundary or accident management barriers and, therefore, no impacts on public safety. The ADs related to N289.5-M91 (R2008) [B.13-13] (under ISR Issue #D064) and N289.5-12 [B.13-1] (under Issues #D621 to #D624) listed above were not part of the study. Reference [B.13-25] states the following for the exclusion of the ADs under ISR Issue #D064 per report NK38-REP-00770-0421410 R001 [B.13-26]:

> *… there is no potential for this AD to aggregate into a significant impact with another AD related to seismic design because the SMS plays no role in dynamically influencing any seismic safety barriers. It serves primarily to indicate the occurrence of low-level seismic motions, and to record response levels in 9 locations of the station. The recorded dynamic response measurements aid in evaluating the condition of SSCs after seismic events that are less intense than the Design Basis Earthquake level, to establish viability of continued operation.*
>
> *With the implementation of surveillance and maintenance measures, the gaps are no longer present so there is no potential for aggregation.*
>
> *This AD is therefore excluded from further review in this aggregation assessment.*

B.13.2.2 Application of Post-PSR1 Reviews

The CSA Impact Statement for N289.5-12 [B.13-4] provides a summary of the significant changes from the previous edition of the Standard:

- Scope changes from addressing only CANDU nuclear power plants to addressing a range of nuclear facilities including nuclear power plants; small reactors; enriched fuel processing, fuel fabrication, and storage facilities; and high- and intermediate-level radioactive waste storage facilities;
- Seismic instrumentation is now mandatory for nuclear power plants and some other nuclear facilities;
- Seismic instrumentation is not considered a safety related system;
- The minimum requirement for instrumentation is now one instrument at the free field, even in low seismic areas;
- Requires that the instrumentation system operate for the life of the facility including maintenance periods;
- Requires continuous recording devices, as opposed to triggered devices;
- Updates the requirements for the sensor characteristics based on current technology;

- Updates the requirements for the seismic instrumentation system characteristics based on current technology;
- Provides detailed guidance on the number and location of sensors, including multi-unit cases; and
- Additional guidance for design, installation, maintenance, testing, and record keeping.

The Darlington ISR code refresh review based on N289.5-12 discussed earlier addresses these latest changes. Furthermore, the applicability of that work to Pickering was demonstrated. **PSR2 CSA N289.5 Gap #1** resulted from that assessment.

OPG has committed to submitting to the CNSC annual Code-over-Code reviews to identify significant technical changes to the requirements of the set of engineering design-related code and standard effective dates, as agreed with the CNSC. The code-over-code methodology documents [B.13-27] and [B.13-28] list Standard N289.5-12 [B.13-1], signifying that future versions of the Standard will be included in subsequent code-over-code reviews. A Code-over-Code review of CSA N289.5-12 relative to a more recent version has not been prepared because CSA N289.5-12 is the latest version.

Further, OPG Nuclear Standard N-STD-MP-0025 R001 [B.13-29] sets forth the general integrated approach to seismic requirements for the applicable SSCs within OPG's nuclear facilities. The approach and framework for OPG activities related to the general requirements for seismic design and qualification of OPG nuclear facilities are in accordance with the CSA N289 series of standards.

B.13.3 Compliance Summary for Pickering PSR2

There is one PSR2 gap for Pickering NGS compliance with N289.5-12 which is applicable to Safety Factor 3:

1. Darlington ISR Issues #D622, D623 and D624 require no further action for Darlington as they were either classified as Acceptable Deviations or were closed. However, they are identified as a PSR2 gap for the following reasons: (Note: These gaps are closely related and are therefore identified as a single PSR2 gap).

   - Darlington ISR Issue #624 refers to specific Darlington instrumentation in order to classify the gaps as Acceptable Deviations. It must be demonstrated that Pickering seismic instruments have the same capabilities as the Darlington instruments (fleet-wide or Pickering-specific standards that would ensure that the Pickering seismic instruments have the same capabilities as the Darlington instruments could not be found). Therefore, this is identified as a gap for PSR2.

- o Darlington ISR Issue #D622 was deemed to be of low safety significance. The same rationale may apply at Pickering. However, it must first be demonstrated that Pickering has the same set up of seismic instruments as Unit 0 at Darlington. Therefore, this is identified as a gap for PSR2.
- o Darlington ISR Issue #D623 was deemed to be of low safety significance. The same rationale may apply at Pickering. However, it must first be demonstrated that similar accelerometers are used at Pickering, and that their locations are not affected by strong ambient vibration. Therefore, this is identified as a gap for PSR2.

B.13.4 References

[B.13-1] CSA Standard N289.5-12, Seismic Instrumentation Requirements for Nuclear Power Plants and Nuclear Facilities, August 2012.

[B.13-2] CNSC Acts and Regulations web page: Regulatory Documents, http://cnsc-ccsn.gc.ca/eng/acts-and-regulations/regulatory-documents/index.cfm#R14, April 2016.

[B.13-3] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.13-4] CSA Impact Statement and Publication Notice, *Product: New Edition – Product Designation: N289.5 – Product Title: Seismic Instrumentation Requirements for Nuclear Power Plants and Nuclear Facilities – Date of Release: August 2012*, Date not provided.

[B.13-5] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.13-6] OPG Report, NK30-REP-03680-00001 R000, *Pickering NGS-B Integrated Safety Review – Plant Design Safety Factor*, August 2007.

[B.13-7] CAN/CSA Standard N289.5-M91 (R2003), *Seismic Instrumentation Requirements for CANDU Nuclear Power Plants*, January 1991.

[B.13-8] OPG Plan, NK30-PLAN-03680-00002 R000, *Pickering B- Integrated Implementation Pla*n, December 2011.

[B.13-9] OPG Plan, NK30-PLAN-00531-00001 R005, *Pickering 5-8 Continued Operations Plan*, December 2015.

[B.13-10] CAN/CSA Standard N289.5-M91 (R1998), *Seismic Instrumentation Requirements for CANDU Nuclear Power Plants*, January 1991.

[B.13-11] OPG Letter, NA44-CORR-00531-00381 R000, R. J. Strickert to J.S.C. Tong, *Pickering A - Updated Basis for Return to Service Document*, April 20, 2001.

[B.13-12] OPG Letter, CD#P-CORR-00531-03719, Application for Renewal of Pickering Nuclear Generating Station Power Reactor Operating Licence, July 4, 2012.

[B.13-13] CAN/CSA Standard N289.5-M91 (R2008), *Seismic Instrumentation Requirements for CANDU Nuclear Power Plants*, January 1991.

[B.13-14] OPG Report, NK38-REP-03680-10041 R000, *Review of CAN/CSA-N289.5-M91 (R2008) (January 1991), Seismic Instrumentation Requirements for CANDU Nuclear Power Plants for Darlington Integrated Safety Review*, August 2011.

[B.13-15] OPG Report, NK38-REP-03680-10104 R000, *Darlington NGS Integrated Safety Review (ISR) – Final ISR Report*, October 2011.

[B.13-16] OPG Report, P-REP-61150-00002 R001, *Available for Service Report PN Seismic Monitoring System Replacement Project (Syscom Instruments) Project 13·49129 Unit 018*, December 2013.

[B.13-17] OPG Report, NK38-REP-03680-10153 R000, *Code Refresh Review of CSA-N289.5-12 for Seismic Instrumentation Requirements for CANDU Nuclear Power Plants*, February 2014.

[B.13-18] OPG Report, NK38-REP-03680-10153-ADD-001 R000, *Review of Clause 5 of CSA N289.5-12 Seismic Instrumentation Requirements for Nuclear Power Plants and Nuclear Facilities for Darlington Integrated Safety Review*, March 2015.

[B.13-19] OPG Report, NK38-REP-00770-0517256 R001, *Section 5 of the CSA 289.5-12 Code Refresh*, February 2015.

[B.13-20] OPG Report, NK38-REP-00770-0528825 R000, *Nuclear Refurbishment Issue Resolution Form – Darlington – Issue # D621 - Continuous Recording Seismic Devices*, February 2015.

[B.13-21] OPG Report, NK38-REP-00770-0528826 R000, *Nuclear Refurbishment Issue Resolution Form – Darlington – Issue # D622 - Lack of Remote Display and Seismic Annunciation Functions for Each Reactor Unit*, February 2015.

[B.13-22] OPG Report, NK38-REP-00770-0528827 R000, *Nuclear Refurbishment Issue Resolution Form – Darlington – Issue # D623 - Location of Accelerometers in Relation to Ambient Vibrations*, February 2015.

[B.13-23] OPG Report, NK38-REP-00770-0528828 R000, *Nuclear Refurbishment Issue Resolution Form – Darlington – Issue # D624 - Number of Seismic Monitoring System Sensors*, February 2015.

[B.13-24] OPG Letter P-CORR-61150-0443299 R000, *RE: PNGS Initial Setup Parameters of Seismic Monitoring System (SMS)*, November 23, 2012.

[B.13-25] OPG Report, NK38-REP-03680-10197 R001, *Evaluation of Seismic, Pressure Boundary and Accident Management ISR Acceptable Deviation Issues for Aggregate Effects between Issues*, November 2014.

[B.13-26] OPG Report, NK38-REP-00770-0421410 R000, *Nuclear Refurbishment Issue Resolution Form – Darlington – Issue # D064 - Seismic Monitoring System Testing*, April 2011.

[B.13-27] OPG Letter, N-CORR-00531-06858 R000, W. M. Elliott to M. Santini, K. Glenn and F. Rinfret, *2014 Code-Over-Code Review for OPG Nuclear Fleet*, April 2015.

[B.13-28] OPG Report, N-REP-00590-058059 R000, *2015 COC - OPG Compilation*, March 2016.

[B.13-29] OPG Nuclear Standard, N-STD-MP-0025 R001, *General Requirements for Seismic Qualification of OPG Nuclear Facilities*, October 2015.

B.14 CSA N285.8-15, "Technical Requirements for In-Service Evaluation of Zirconium Alloy Pressure Tubes in CANDU Reactors"

B.14.1 Background

The following paraphrased from the Preface of CSA N285.8-15 [B.14-1] provides a brief overview of the purpose of this Standard and the requirements expressed therein:

> *CSA N285.8 is one of a series of CSA N285 Standards that provide consistent rules for the design, fabrication, installation, inspection, and assessment of pressure-retaining systems and components in CANDU nuclear power plants.*
>
> *The purpose of CSA N285.8 is to ensure structural integrity of cold-worked Zr-2.5 wt% Nb alloy pressure tubes in operating CANDU reactor. CSA N285.8 specifies mandatory technical requirements and non-mandatory evaluation procedures for fitness-for-service assessments. The main body contains the mandatory rules and acceptance criteria for in-service evaluation of zirconium alloy pressure tubes in CANDU reactors. The annexes contain the non-mandatory evaluation procedures, material properties and derived quantities, and a form for providing notification of the evaluation to the authority having jurisdiction.*

N285.8-15 is applicable to zirconium alloy pressure tubes in CANDU reactors, and is relevant to Safety Factor 2 (Actual Condition of SSCs) and Safety Factor 4 (Aging).

CSA N285.8-15 is identified in Appendix E.1 of the R04 Pickering Licence Conditions Handbook [B.14-2] as "Guidance and Criteria". However, Section 7.1 "Programs to ensure Fitness for Service and In-Service Inspection" of the LCH [B.14-2] states: "Where N285.4 refers to N285.8, OPG shall comply with N285.8-10". As discussed in Section B.2, compliance with CSA N285.4 is a licensing requirement and therefore any N285.8-10 call-ups are in practice license requirements.

The current revision of the Standard is N285.8-15 [B.14-1]. This is the third edition of CSA N285.8. It supersedes the previous editions, published in 2010 and 2005. Changes since the 2005 version of the Standard have ranged from editorial, to the addition of supplementary information as well as the addition of new requirements.

The CSA N285.8-15 Impact Statement [B.14-3] provides a "Summary of significant changes from the previous edition" which identifies five primary changes to the Standard which are discussed in Section B.14.2 below. In addition to findings resulting from review of the CSA N285.4-14 Impact Statement, the results of PSR1 N285.8 reviews (Pickering B and Darlington ISRs), as well as reviews performed since PSR1, have also been assessed for applicability to PSR2 in Section B.14.2.

As identified in [B.14-4], the Pickering PSR2 review of CSA N285.8-15 is an Incremental Review. As discussed in Section 2.0 of this Report, PSR2 Incremental Review includes an assessment of the intent of recent changes to the L/R/C/S on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.
- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.14.2 Compliance Assessment for Pickering PSR2

B.14.2.1 Application of PSR1 Reviews

The versions of N285.8 subject to previous reviews conducted for Darlington and Pickering, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

OPG Reports NK30-REP-03680-00001 R000 [B.14-5] and NK30-REP-03680-00002 R000 [B.14-6] document the Pickering B ISR Safety Factor 1 and 2 reviews, respectively, of CSA N285.8-05 [B.14-7]. With respect to the Plant Design Safety Factor, NK30-REP-03680-00001 R000 [B.14-5] concluded that:

> *CSA Standard N285.8-05 does not directly address HT pressure tube design requirements.*
>
> *PNGS B reactors' HT pressure tubes will be replaced during the life extension outage if OPG proceeds with the life extension.*
>
> *Accordingly, a clause by clause review of the CSA Standard N285.8-05 is not required.*

NK30-REP-03680-00001 R000 [B.14-5] recommended the following:

> *OPG should take the opportunity to use current codes and standard when preparing specifications for HT pressure tubes replacement.*

With respect to the Actual Condition of SSCs Safety Factor, NK30-REP-03680-00002 R000 [B.14-6] stated in the Scope section that a review against N285.8-05 for the pressure tubes was unnecessary based on the following rationale:

The fuel channels, pressure tubes, calandria tubes, feeders and steam generators will be replaced during the refurbishment outage for Pickering NGS-B. The condition of these components is well understood and managed through their own specific, detailed life cycle plans and fitness-for-service criteria. As such, an assessment of the current condition of these components is unnecessary and is not included in this report. Since CSA N285.8-05 is only applicable to fuel channels and Clauses 12, 13 and 14 of CSA N285.4-05 only apply to fuel channels, feeders and steam generators, respectively, these are not included in the scope of the review documented in this report. Life Cycle Plans for these components are routinely submitted to the CNSC.

The design basis of the components to be replaced will be reviewed in accordance with N-PROG-MP-0001, "Engineering Change Control" [R-10], as part of the process to prepare detailed procurement specifications for these components. Current expectations are that, at minimum, the materials and/or material properties of these components will differ from components currently installed at Pickering NGS-B. As per N-PROG-MP-0001, the requirements of the versions of codes and standards applicable to Pickering NGS-B at the time will be applied during these design reviews. This includes the codes and standards applicable to maintaining the condition of these components.

Based on the above, no PSR2 gaps are identified based on the Pickering B ISR.

<u>Pickering Units 1,4</u>

Since Pickering A Return to Service [B.14-8] preceded the first edition of N285.8 that was issued in June 2005, no code review of the Standard was performed in support of the Pickering A Restart. However, the results of Darlington ISR N285.8 reviews are assessed for applicability to Pickering NGS below. Further, Section 7.1 "Programs to ensure Fitness for Service and In-Service Inspection" of the Pickering LCH [B.14-2] states:

CNSC staff have accepted the Pickering NGS-A and the Pickering-B NGS PIP documents… With respect to N285.4-05 clause 12.2.5.1.3, CNSC staff have accepted (e-Doc 4280370 and e-Doc 4523259) OPG's revised compliance plan N-REP-31100-10061 R001 (N-CORR00531-06244, e-Doc 4180866 & N-CORR-00531-06522, e-Doc 4447955) for the use of CSA standard N285.8-10 Update 2 "In-Service Evaluation of Zirconium Alloy Pressure Tubes", as the evaluation method used for the fitness-for-service assessment of the Fuel Channels in Pickering A and B units. See Recommendations and Guidance (section "CSA N285.4-05, Clause 12") for more information on how CNSC will be reviewing the probabilistic assessments.

CNSC has accepted (e-Doc 4272552 & 4369355) the OPG proposed "Approach to Fitness for Service Assessment for Pressure Tubes" described in OPG submission N-CORR-00531-06304 (e-Doc 4250561), subject to the conditions described in CNSC letter e-Doc 4369355. With respect to N285.4-05 clause 12.4.4.2, CNSC staff have accepted (e-Doc

3895468) OPG's procedural updates and technical justifications for pressure tube material testing submitted in e-Doc 3848127, N-CORR-00531-05488.

The compliance plan and submissions discussed in the Pickering LCH [B.14-2] are addressed in Section B.14.2.2 below. No PSR2 gaps are identified based on the Pickering A Return to Service.

Darlington NGS

The Darlington ISR was originally performed using version N285.8-05 including Update No. 1 (May 2007) [B.14-9] and documented in OPG Report NK38-REP-03680-10059 R000 [B.14-10], which notes that this version of the Standard underwent a clause-by-clause (non-PROL) review.

The conclusions of the Darlington ISR code review were:

> *A Clause-by-Clause review of CSA N285.8-05 "Technical requirements for in-service evaluation of zirconium alloy pressure tubes in CANDU reactors" (updated May 2007) has been performed in this report. It was found that Darlington Nuclear Generating Station is compliant with all clauses of CSA N285.8-05 except for Clause 6, which has been identified as a gap.*

The gap identified for Clause 6 of N285.8-05 including Update No. 1 (May 2007) was closed in Appendix B of NK38-REP-03680-10059 R000 [B.14-10] as follows:

> *As required by the Darlington NGS Nuclear Power Reactor Operating licence, PROL-13.00/2013 [R-1], Darlington NGS performs pressure tube Periodic Inspections in accordance with the technical requirements provided in CAN/CSA-N285.4 "Periodic inspection of CANDU nuclear power plant components" [R-2].*
>
> *Clause 12 of CAN/CSA-N285.4-09 [R-2] requires that when the results of pressure tube inspection do not satisfy the acceptance criteria, the licensee shall evaluate the inspection results to determine the acceptability for continued operation as per CAN/CSA-N285.8 [R-3]. Currently at Darlington NGS, the pressure tube to calandria tube gap margins are maintained and have not required invoking Clause 6 of the CAN/CSA-N285.8-05 [R-3] standard.*
>
> *However, if gap margins are challenged, then the requirements of Clause 6 of CAN/CSA-N285.8-05 [R-3] would be used for evaluation of pressure tube to calandria tube contact as required by CAN/CSA-N285.4 "Periodic inspection of CANDU nuclear power plant components" [R-2].*
>
> *This issue has therefore been re-categorized as "Closed" and no further action is required.*

However, OPG Report NK38-REP-03680-10059-ADD-001 R000 [B.14-11], which documents the CNSC staff comments and OPG responses related to the N285.8-05 including Update No. 1 code review report [B.14-10], introduced new gaps on some clauses that were previously identified as compliant. The affected clauses were 7.3.3.1, 7.3.3.2, and 7.3.3.3, which were treated collectively as a Gap for inclusion in the gap resolution and issue prioritization process. This remaining gap relates to predicting time to Pressure Tube (PT) - Calandria Tube (CT) contact, using the as-installed locations of tight-fitting annulus spacers in Darlington Fuel Channels (FCs). OPG agreed to treat it as a gap for Clauses 7.3.3.1 through 7.3.3.3 inclusive. This gap is applicable to Pickering NGS. However, OPG has issued a plan for compliance with pressure tube in-service evaluation requirements in N285.8-15 [B.14-1] that addresses this gap as outlined in Section B.14.2.2 below.

A clause-by-clause (non-PROL) code refresh review based on version N285.8-10 [B.14-12] was performed for the Darlington ISR and is documented in report NK38-REP-03680-10140 R000 [B.14-13]. The assessment considered N-PROC-MA-0044 "Fuel Channel Life Cycle Management" [B.14-14], with lower-level disposition processes as directed by N-PROC-MA-0052 "Flaw Dispositioning" [B.14-15]. Future activities to ensure continued compliance are planned and guided by N-PLAN-01060-10002 "Fuel Channel Aging and Life Cycle Management Strategy and Plan" [B.14-16].

The code refresh review determined that only 10 mandatory clauses had revisions compared to version N285.8-05 including Update No. 1 (May 2007). Three are minor non-intent changes, such as the renumbering and reordering of clauses and four were found to be "intent" changes. In addition, Clauses 8.3, 8.4, and 8.5 were found to have technical revisions and also are "intent" changes. For the "intent" changes, Darlington was found to be in compliance both with the four clauses and with Clauses 8.3, 8.4, and 8.5. The conclusions of the code refresh review report [B.14-13] were:

> *Although four clauses in CSA N285.8-10 [R-1] were found to contain "intent" changes in the code review, DNGS was found to be in compliance both with the four clauses and also with three other clauses (8.3, 8.4.and 8.5), which had technical changes.*
>
> *Therefore, no ISR Gaps relative to CSA N285.8 – 10 [R-1] were identified in OPG's governance of the DNGS Pressure Tubes.*

To the extent that the Darlington ISR identified that OPG Nuclear fleet documents are compliant, this conclusion is applicable to Pickering NGS as well. However, since the assessment of Darlington compliance relied in part on Darlington specific analysis reports and component dispositions, a similar assessment is required for Pickering. As discussed in Section B.14.2.2 below, OPG has issued a detailed plan for compliance [B.14-17] with pressure tube in-service evaluation requirements in N285.8-15 [B.14-1] that will address this gap.

B.14.2.2 Application of Post-PSR1 Reviews

The CSA N285.8-15 Impact Statement [B.14-3] provides a summary of the significant changes from the previous edition of the Standard as follows:

- Implementation of statistically based fatigue crack initiation evaluation curves for axial flaws (Clauses D.4.2, D.4.3, and D.3.6): "This change will allow evaluation of fatigue crack initiation in probabilistic core assessments of flaws."
- Implementation of closed-form engineering relation for threshold peak stress for Delayed Hydride Cracking (DHC) initiation (Clauses D.5 and 5.4.3.4): "This change will provide closed-form engineering equations for evaluation of delayed hydride cracking (DHC) for use in probabilistic core assessments of flaws."
- Implementation of statistically based threshold relation for peak stress for crack initiation due to hydrided region overloads (Clause D.5): "This change will allow evaluation of crack initiation due to hydrided region overloads in probabilistic core assessments of flaws."
- Implementation of new fracture toughness models for axial through-wall flaws (Clause D.13.2): "This change will allow predictions of fracture toughness of pressure tubes with low to high levels of hydrogen equivalent concentration."
- Implementation of Methods 1 and 2 Probabilistic Leak-Before-Break (Clauses 3.1, 7.3 and 7.4): "This change will provide specific requirements on performing probabilistic leak-before-break analysis."

Compliance with the above changes has not yet been demonstrated for either Darlington or Pickering. As discussed earlier, the latest clause-by-clause review performed for N285.8 was based on version N285.8-10 [B.14-12] as documented in NK38-REP-03680-10140 R000 [B.14-13], and the above changes to the latest version of the Standard (N285.8-15 [B.14-1]) were therefore not addressed. However, in November 2015 OPG issued Plan N-REP-31100-10061 R002 [B.14-17] for compliance with pressure tube in-service evaluation requirements in N285.8-15 [B.14-1] and submitted the compliance plan to the CNSC for acceptance [B.14-18]. OPG has submitted a previous revision compliance plan for the long term use of the 2010 edition of N285.8 [B.14-19] and this compliance plan was accepted by the CNSC [B.14-20]. The compliance plan was revised to document OPG's compliance to the latest 2015 edition of CSA N285.8 in Reference [B.14-17]:

> *OPG had been using in-service evaluation procedures in the 2010 Edition of CSA Standard N285.8. The 2015 Edition of CSA N285.8 contains updates and improvements for in-service evaluation procedures based on the experience obtained from the use of*

the 2010 Edition as well as recent inspection and R&D results. OPG plans to continue using CSA N285.8 on a long-term basis.

This document outlines OPG's plan for compliance with pressure tube in-service evaluation requirements in CSA N285.8-15, and summarizes OPG's commitments for supporting activities for long-term use of CSA N285.8-15. Submission of this revised CSA N285.8 Compliance Plan is being tracked under CNSC Action Item 2014-OPG-4862.

As identified in N-REP-31100-10061 R002, the purpose of the N285.8-15 compliance plan is to:

- Confirm OPG's continued intention to use CSA N285.8 on a long-term basis;
- Define the evaluation methodology for disposition of pressure tube flaws;
- Define the evaluation methodology for disposition of pressure tube to calandria tube contact;
- Establish a protocol for submission of the future probabilistic core assessment updates; and
- Summarize OPG's commitments for supporting activities for the long-term use of CSA N285.8-15.

Table 2 of N-REP-31100-10061 R002 lists the supporting activities and OPG commitments for long-term use of CSA N285.8. The timeline for submission of the compliance plan was agreed to by the CNSC in Reference [B.14-21]. Table 1 of N-REP-31100-10061 R002 lists the schedule for updating the core assessments for Pickering 1,4 (May 31, 2016) and Pickering Units 5-8 (December 15, 2016). The core assessments for Pickering Units 1,4 were recently submitted to the CNSC in References [B.14-22] and [B.14-23]. Since OPG has committed to fulfilment of the commitments in N-REP-31100-10061 R002 [B.14-17], completion of the above-mentioned Pickering Units 5-8 action is required for Pickering NGS operation past 2020. This is therefore a gap for Pickering PSR2 (**PSR2 CSA N285.8-15 Gap #1**).

B.14.3 Compliance Summary for Pickering PSR2

There is one PSR2 gap for Pickering NGS compliance with N285.8-15 which is applicable to Safety Factor 4:

1. For the Pickering B ISR, no clause-by-clause review of the Standard was conducted on the basis that the pressure tubes will be replaced during the refurbishment outage for Pickering Units 5-8, and the condition of these components is well understood and managed through their own specific, detailed life cycle plans and fitness-for-service criteria. However, in November 2015, OPG issued Plan N-REP-31100-10061 R002 for Pickering NGS compliance with pressure tube in-service evaluation requirements in

CSA N285.8-15 [B.14-1]. OPG had submitted a previous compliance plan for the long term use of the 2010 edition of CSA N285.8 and this compliance plan was accepted by the CNSC. The compliance plan was revised to document OPG's compliance to the 2015 edition of CSA N285.8. Since OPG has committed to fulfillment of the commitments in N-REP-31100-10061 R002, successful fulfillment by OPG of the commitments in the compliance plan is required for Pickering operation past 2020. This is therefore a gap for PSR2. In particular, the significant changes to CSA N285.8-15 per the CSA Impact Statement will need to be reflected in Pickering procedures, including:

- Implementation of statistically based fatigue crack initiation evaluation curves for axial flaws (Clauses D.4.2, D.4.3, and D.3.6);
- Implementation of closed-form engineering relation for threshold peak stress for Delayed Hydride Cracking (DHC) initiation (Clauses D.5 and 5.4.3.4);
- Implementation of statistically based threshold relation for peak stress for crack initiation due to hydrided region overloads (Clause D.5);
- Implementation of new fracture toughness models for axial through-wall flaws (Clause D.13.2); and
- Implementation of Methods 1 and 2 Probabilistic Leak-Before-Break (Clauses 3.1, 7.3 and 7.4).

B.14.4 References

[B.14-1] CSA Standard N285.8-15, *Technical Requirements for In-Service Evaluation of Zirconium Alloy Pressure Tubes in CANDU Reactors*, January 2015; Errata: January 2016.

[B.14-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.14-3] CSA Draft Impact Statement and Public Review Notice, *Product: New Edition – Product Designation: CSA N285.8-15 – Product Title: Technical Requirements for In-Service Evaluation of Zirconium Alloy Pressure Tubes in CANDU Reactors*, Date not provided.

[B.14-4] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.14-5] OPG Report, NK30-REP-03680-00001 R000, *Pickering NGS-B Integrated Safety Review – Plant Design Safety Factor*, August 2007.

[B.14-6] OPG Report, NK30-REP-03680-00002 R000, *Pickering NGS-B Integrated Safety Review - Actual Condition of Systems, Structures and Components Safety Factor Report*, May 2008.

[B.14-7] CSA Standard N285.8-05, *Technical Requirements for In-Service Evaluation of Zirconium Alloy Pressure Tubes in CANDU Reactors*, June 2005.

[B.14-8] OPG Letter, NA44-CORR-00531-00381 R000, R. J. Strickert to J.S.C. Tong, *Pickering A - Updated Basis for Return to Service Document*, April 20, 2001.

[B.14-9] CSA Standard N285.8-05, *Technical Requirements for In-Service Evaluation of Zirconium Alloy Pressure Tubes in CANDU Reactors*, June 2005; Update No. 1, May 2007.

[B.14-10] OPG Report, NK38-REP-03680-10059 R000, *Review of CAN/CSA-N285.8-05 incl. UPD1 (May 2007), Technical Requirements for In-Service Evaluation of Zirconium Alloy Pressure Tubes in CANDU Reactors for Darlington Integrated Safety Review*, September 2011.

[B.14-11] OPG Report, N-REP-03680-10059-ADD-001 R000, *Addendum to the CAN/CSA N285.8 05 Code Review Report for Darlington ISR*, January 2014.

[B.14-12] CSA Standard N285.8-10, *Technical Requirements for In-Service Evaluation of Zirconium Alloy Pressure Tubes in CANDU Reactors*, July 2010.

[B.14-13] OPG Report, NK38-REP-03680-10140 R000, *Code Refresh Review of CSA N285.8-10, Technical Requirements for In-Service Evaluation of Zirconium Alloy Pressure Tubes in CANDU Reactors, for DNGS ISR*, July 2013.

[B.14-14] OPG Nuclear Procedure, N-PROC-MA-0044 R003, *Fuel Channel Life Cycle Management*, March 2012.

[B.14-15] OPG Nuclear Procedure, N-PROC-MA-0052 R007, *Flaw Dispositioning*, December 2011.

[B.14-16] OPG Plan, N-PLAN-01060-10002-R016, *Fuel Channel Aging and Life Cycle Management Plan*, November 2015.

[B.14-17] OPG Report, N-REP-31100-10061 R002, *Compliance Plan For Long-Term Use Of CSA N285.8 for In-Service Evaluation Of Zirconium Alloy Pressure Tubes*, November 2015.

[B.14-18] OPG Letter, N-CORR-00531-17932 R000, W.S. Woods to M. Santini and F. Rinfret, *Revised OPG CSA N285.8 Compliance Plan Submission – Closure of CNSC Action Item 2014-OPG-4862*, November 30, 2015.

[B.14-19] OPG Letter, N-CORR-00531-06244 R000, W.M. Elliott to M. Santini and F. Rinfret, *Revised OPG CSA N285.8 Compliance Plan Submission*, July 30, 2013.

[B.14-20] CNSC Letter, e-Doc 4280370, File 4.01.03, (OPG File No. N-CORR-00531-06507 R000), M. Santini and F. Rinfret to W.M. Elliott, *Darlington and Pickering NGS: OPG Revised CSA N285.8 Compliance Plan, New Action Item 2014-OPG-4862*, March 20, 2014.

[B.14-21] CNSC Letter, e-Doc 4523259, File 2.01 (OPG File No. N-CORR-00531-06733 R000), M. Santini and F. Rinfret to W.M. Elliott, *Darlington and Pickering NGS: OPG CSA N285.8 Compliance Plan, Action Item 2014-0PG-4862*, November 19, 2014.

[B.14-22] OPG Letter, NA44-CORR-00531-07632 R000, B. McGee to H. Overton, *Pickering 1&4 – Submission of Pickering Unit 4 Probabilistic Core Assessment*, May 20, 2016.

[B.14-23] OPG Letter, NA44-CORR-00531-07617 R000, B. McGee to M. Santini, *Pickering 1&4 – Submission of Pickering Unit 1 Probabilistic Core Assessment, Probabilistic Leak-Before-Break Assessment and Extension of Leak-Before-Break Limit Beyond July 31, 2016*, April 25, 2016.

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Amec Foster Wheeler reference: PS112/104/000002 R01
Security Class: Amec Foster Wheeler Confidential

amec foster wheeler

ONTARIO POWER GENERATION	
ACCEPTED	X
ACCEPTED AS NOTED	
REVISE AND RESUBMIT	
K Bramma for — Signature	26 Sep 16 — Date
Name: Mike Ruffolo, Manager	
Dept: Pickering Engineering - Aging Management & Strategic Initiatives	
OPG Proprietary	
Doc No.: P-REP-03680-0586480	Rev: 000
This acceptance does not relieve the contractor from responsibility for errors or omissions or from any obligations or liability under this contract.	

September 23, 2016

Mike Ruffolo
Pickering NGS
P42 - Engineering Services Building II
1675 Montgomery Park Rd
Pickering, ON L1V 2R5

Dear Mr. Ruffolo,

Re: Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 9, 11, and 15

Please find in Attachment A of this letter the reviews of 6 Laws, Regulations, Codes and Standards (L/R/C/Ss) which form part of the Pickering Periodic Safety Review 2 (PSR2) Assessment Basis [1]. Table 1 of this letter identifies the L/R/C/S that were assessed, as well as the modern version and date of each L/R/C/S and the type of review that was completed.

If there are any questions on the attached reviews, please do not hesitate to contact me to discuss.

Sincerely,

SEAN DONNELLY FOR:
Andrew Johnstone
Senior Analyst
Station Operations and Licensing

Prepared by:	Various Staff (Amec Foster Wheeler)	Verified by:	Damien Moule Associate Analyst Station Operations and Licensing
Reviewed by:	SEAN DONNELLY FOR: Stan B. Harvey P. Eng. Senior Advisor Engineering and Analysis	Reviewed by:	Rob Ross Senior Technical Expert Station Support Programs
		Approved by:	Ron Henry Director (Acting) Station Support Programs

References

[1] OPG Report P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

Enclosure:

Attachment A: PSR2 L/R/C/S Reviews

cc: S. Donnelly

Table 1: L/R/C/S Assessed for Safety Factors 9, 11 and 15

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis	Attachment (Page #)
		L/R/C/Ss Referenced in Pickering PROL 48.02/2018					
1	CSA N286	Management System Requirements for Nuclear Facilities	N286-12	5, 6, 9, 10, 11	Incremental	N286 addressed as part of Pickering B and Darlington ISRs.	A.1 (Pg. 6)
		Additional L/R/C/Ss					
2	CNSC G-129	Keeping Radiation Exposures and Doses "As Low As Reasonably Achievable (ALARA)"	2004	8, 15	Incremental	G-129 addressed as part of Pickering B and Darlington ISRs.	A.2 (Pg. 13)
3	CNSC G-228	Developing and Using Action Levels	2001	8, 14, 15	Incremental	G-228 addressed as part of Pickering B and Darlington ISRs.	A.3 (Pg. 16)
4	SOR/2000-202	The General Nuclear Safety and Control Regulations	Amended in June 2015	10, 15	Incremental	SOR/2000-202 addressed as part of Pickering B and Darlington ISRs.	A.4 (Pg. 20)
5	SOR/2000-203	The Radiation Protection Regulations	Amended in June 2015	8, 15	Incremental	SOR/2000-203 addressed as part of Pickering B and Darlington ISRs.	A.5 (Pg. 24)
6	CNSC REGDOC-2.2.3	Personnel Certification: Radiation Safety Officers	2014	15	High Level	REGDOC-2.2.3 not addressed as part of Pickering B or Darlington ISRs.	A.6 (Pg. 29)

Attachment A: PSR2 L/R/C/S Reviews

The applicable Laws, Regulations, Codes and Standards relevant to PSR2 are listed in the PSR2 Basis Document [A-1]. Table 1 identifies the L/R/C/S addressed in this letter, as well as the modern version and date of each L/R/C/S considered and the type of review performed.

All of the PSR2 reviews are high level or incremental in nature. The definitions of High Level Review and Incremental Review are as follows:

- High Level Review: New L/R/C/Ss not referenced in Pickering Power Reactor Operating License (PROL) 48.02/2018 but which are in the PSR2 Assessment Basis will be subject to a high level review. In a high level review, the degree of conformance with clauses or groups of clauses in the L/R/C/S is demonstrated by supporting evidence stating whether the intent of the requirements stipulated in the requirement document is met; and

- Incremental Review: For L/R/C/Ss that have been reviewed in PSR1 but have had revisions since the last review, a topical review will be performed of the changes.

The PSR2 incremental reviews in this letter include an assessment of the intent of recent changes to the L/R/C/Ss on a topic or subject-matter basis where there is potential to impact nuclear safety. Incremental reviews provide:

- A summary of the purpose of the L/R/C/S;

- Pertinent background information about the current revision of the L/R/C/S that is being considered;

- Identification of which Safety Factor(s) are applicable to the current revision of the L/R/C/S;

- A description of which version(s) of the L/R/C/S were assessed for Periodic Safety Review 1 (PSR1)[1] (i.e., Darlington ISR (for programmatic content), Pickering B ISR and PARTS code reviews);

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[1] A comprehensive Integrated Safety Review (ISR) was completed for Pickering Units 5-8 in 2009 in support of refurbishment and continued operation. Pickering Units 1,4 integrated safety assessments were also performed for Pickering A Return to Service (PARTS) in support of approval to restart Units 1 and 4. In addition to these Pickering-specific studies, the 2013 Darlington ISR performed extensive code and standard reviews that were updated in relation to the versions that were assessed in the 2009 Pickering B ISR. These previous ISRs are considered to constitute the first PSR completed for Pickering (referred to as "PSR1"). The current PSR (referred to as "PSR2") is a subsequent PSR building on the basis of earlier OPG integrated safety assessments through review of the various studies, assessments and licence renewals performed since PSR1. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [A-1].

- Identification of whether the current version of the L/R/C/S is an update of a previous version of the L/R/C/S that was assessed in PSR1 (and if so, a description of the major changes in the latest revision is provided as discussed below);

- An assessment of the applicability of PSR1 assessment findings (gaps and conclusions), including the implications of extending Pickering NGS operation beyond 2020 if any;

- An assessment of the applicability of assessment findings that address more recent (post-PSR1) editions of the L/R/C/S, including any implementation or transition plans that are already committed to by OPG; and

- Where PSR1 and post-PSR1 assessments are not sufficient to address changes in the latest edition of the L/R/C/S, an assessment of the changes from the previously assessed edition of the L/R/C/S (including identification of any safety significant PSR2 gaps which result).

High Level reviews provide the same information as above, where applicable, in a similar format. However, given that High Level L/R/C/Ss generally have not received past assessment during PSR1, the Incremental review content is augmented by a high level, section-by-section assessment of Pickering NGS compliance with the L/R/C/S (demonstrating, with supporting evidence, whether the intent of the requirements stipulated in the document are met).

The L/R/C/S reviews identify Compliances and Gaps as defined below:

- <u>Compliance</u>:
    - Where an Incremental review has been performed, Compliance indicates that the change in the safety requirement, per the topical review, is met.
    - Where a High Level review has been performed, Compliance indicates that the intent of the safety requirement is met.
- <u>Gap</u>:
    - Where an Incremental review has been performed, a Gap indicates that the change in the safety requirement, per the topical review, is not met.
    - Where a High Level review has been performed, a Gap indicates that the intent of the safety requirement is not met.

References

[A-1] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

A.1 CSA N286-12, “Management System Requirements for Nuclear Facilities”

A.1.1 Background

The following text paraphrased from CSA N286-12 [A.1-1] provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *CSA N286 provides overall direction to management to develop and implement sound management practices and controls, while the other CSA N-Series provide technical requirements and guidance that support the management system. This edition of the standard expands beyond nuclear power plants to include nuclear facilities.*
>
> *CSA N286 identifies management system requirements for Uranium mines and mills, Uranium processing and fuel manufacturing facilities, high energy reactors, research and isotope processing facilities, and radioactive waste management facilities. It integrates the requirements from management system standards for health, safety, environment, security, economics, and quality and is based on the following 12 principles:*
>
> 1. *Safety is the paramount consideration guiding decisions and actions.*
> 2. *The business is defined, planned, and controlled.*
> 3. *The organization is defined and understood.*
> 4. *Resources are managed.*
> 5. *Communication is effective.*
> 6. *Information is managed.*
> 7. *Work is managed.*
> 8. *Problems are identified and resolved.*
> 9. *Changes are controlled.*
> 10. *Assessments are performed.*
> 11. *Experience is sought, shared, and used.*
> 12. *The management system is continually improved.*

Sections 4 and 7 of CSA N286-12 are directly relevant to Safety Factors 10 (Organization, Management System and Culture) and 11 (Procedures). Various clauses of Sections 4 and 7 are also applicable to Safety Factors 5 and 6 (Probabilistic and Deterministic Safety Analysis) and 9 (Use of OPEX). Sections 0 to 3 are introductory. Sections 5, 6 and 8 are specific to other types of facilities and not relevant to PSR2. Section 9 is related to Waste Management, which is not in the scope of PSR2.

Compliance with CSA N286-05 (including Update No. 1) [A.1-2] is currently a licence requirement for Pickering NGS (per PROL 48.02/2018) as indicated in Section 2.1 of the R04 Pickering Licence Conditions Handbook (LCH) [A.1-3]. N286-12 is the second edition of this standard which supersedes the previous edition published in 2005 under the title "Management System Requirements for Nuclear Power Plants" [A.1-4], and its Updates No. 1 [A.1-2] and No. 2 [A.1-5]. The 2005 version of N286 superseded the 1992 version of N286.0 [A.1-6] and accompanying standards N286.1 to N286.6 [A.1-7] to [A.1-12].

The CSA Impact Statement notification of the 2012 edition of CSA N286 [A.1-13] provides a "Summary of Significant Changes from the Previous Edition" which identifies four primary changes to the Standard which are discussed in Section A.1.2 below. In addition to findings resulting from review of the CSA N286-12 Impact Statement, the results of PSR1 CSA N286 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington ISRs), as well as reviews performed since PSR1, have also been assessed for applicability to PSR2 in Section A.1.2.

As identified in Reference [A.1-14], the Pickering PSR2 review of CSA N286-12 is an Incremental Review. PSR2 Incremental Review includes an assessment of the intent of recent changes to the L/R/C/S on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.
- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

A.1.2 Compliance Assessment for Pickering PSR2

A.1.2.1 Application of PSR1 Reviews

The versions of N286 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

For the Pickering B ISR, different versions of N286 were assessed for the Plant Design, Safety Analysis, Safety Performance, Management System and Radiation Protection Safety Factors [A.1-15][A.1-16][A.1-17][A.1-18][A.1-19] as these reports were produced from 2007-2009. The N286 versions reviewed included the versions associated with the Pickering PROL at the time, CSA N286.0-92 [A.1-6] and the associated N286 series standards (References [A.1-7] to [A.1-12]), as well as CSA N286-05 [A.1-4] (both with and without Update No. 1 [A.1-2]). A clause-by-clause review of the applicable clauses of N286-05 (including Update No. 1) [A.1-2] was also performed for the Pickering B ISR by the OPG Performance Improvement & Nuclear Oversight (PINO) group to confirm that OPG Governance adequately addressed the requirements defined in the standard [A.1-20]. In all of these reviews, no compliance gaps were identified.

N-LIST-08130-10023, "CSA N286-05 to OPGN Governance Cross Matrix" [A.1-21] was also prepared in 2009 and last updated in 2012. This list is a controlled document cited in the current (R04) Pickering License Conditions Handbook [A.1-3]. Direct compliance with all applicable clauses of N286-05 was shown.

Pickering Units 1,4

A review of CSA N286 was not performed as part of the Pickering A Return to Service assessments on the basis that it "pertains mostly to quality assurance aspects" [A.1-22]. However, compliance with the N286-05 including Update No. 1 [A.1-2] is a license requirement for Pickering NGS per Section 2.1 of the Pickering License Condition Handbook [A.1-3]. Furthermore, Section 1.0 of Attachment 3 of the Pickering PROL Application P-CORR-00531-03719 R000 states [A.1-23]:

> *The OPG Management System is the framework which establishes the process and programs required to ensure that OPG and Pickering achieves its safety objectives and continuously monitors its performance against these objectives.*
>
> *The OPGN document N-CHAR-AS-0002 Nuclear Management System and referenced supporting documents establish the quality program. The Charter establishes the expectations regarding implementation of the nuclear management system.*
>
> *This Charter and reference documents fulfill the requirements of Canadian Standard Association (CSA) N285 and N286-05 standards, International Organization for Standardization (ISO) 14000 series of standards and American Society of Mechanical Engineers (ASME) NCA 4000. The Charter and the supporting documents identified in N-LIST-08130-10023, CSA N286-05 to OPGN Governance Cross Matrix, demonstrate compliance with CSA N286-05.*
>
> *…In addition nuclear oversight is proactively assessing changes to the CSA N286-05 standard upon release of the next revision to this standard and its impact to the nuclear management system.*
>
> *… Nuclear Organization Standard, N-STD-AS-0020, Nuclear Organization, describes the organization and responsibilities of OPG in support of its Management System and (CSA) N286-05, Management system requirements for nuclear power plants.*

Since compliance with N286-05 (including Update No. 1) [A.1-2] is currently a licence requirement for Pickering NGS (per PROL 48.02/2018), and this is confirmed by review of the Pickering PROL Application, Pickering Units 1,4 and 5-8 are in compliance with N286-05 (including Update No. 1). Further, assessments have been performed for Darlington which are programmatically applicable to Pickering, and which are used to demonstrate Pickering NGS compliance with the latest version of the Standard, as discussed below.

Darlington NGS

The Darlington ISR was originally performed using N286-05 (including Update No. 1) [A.1-2]. The review is documented in Reference [A.1-24] and concluded: "The review did not identify any ISR Gaps and found that Darlington NGS was compliant with CAN/CSA N286-05" [A.1-24].

A Code Refresh Review of the most recent version of the Standard, CSA N286-12 [A.1-1], was also performed for the Darlington ISR in NK38-REP-03680-10142-R000 [A.1-25]. This review concluded:

> *CSA N286-12 has been expanded and the scope broadened to include other nuclear facilities, i.e. uranium mines, mills, processing, fuel facilities and high energy reactor facilities. The overall requirements remain the same in principle but have been generalized to provide specific guiding principles applicable to many types of facilities and operation(s). The review of the updated clauses in this code refresh review report concludes that the Darlington station is in compliance with the updated requirements in CSA N286-12.*

The only recommendation resulting from the code-refresh review was administrative, i.e., "that the 2012 version of N286 be reflected in future updates of the OPG governing documents" [A.1-25].

Since compliance against CSA N286-12 is based on Governance, Programs and Procedures that apply across OPG's Nuclear operations, the Darlington ISR conclusions are applicable to Pickering PSR2.

A.1.2.2 Application of Post-PSR1 Reviews

The CSA Impact Statement notification of the 2012 edition of CSA N286 identifies the following changes from the previous edition (paraphrased) [A.1-13]:

1. *The scope of N286 is expanded to include all class 1 nuclear facilities and uranium mines and mills licensed by the Canadian Nuclear Safety Commission.*

2. *N286-12 was written with full knowledge of the strategy of nuclear power plant operators to move away from performing modification and refurbishment work in-house and contracting the work to Engineer, Procurement and Construction (EPC) suppliers. The wording of N286-12 was modified in many places to eliminate commonly used NPP operations jargon and to not repeat requirements in the licensed control areas that are adequately addressed by license and law (such as radiation protection and security). This makes the majority of N286-12 easier to apply to suppliers than N286-05.*

3. *Annex G was eliminated because it was not being used.*

4. *The standard explicitly integrates the requirements for management systems for health, safety, environment, security, economics and quality.*

5. *The standard addresses the direction from the Nuclear Standards Strategic Steering Committee to be technology neutral, internationally harmonized and reflect best*

industry practices: The standard is technology neutral, and is harmonized with IAEA – GS-R-3 and various international standards (i.e., ISO & OHSAS). The Standard explicitly states requirements for safety culture and includes a principle that "safety is the paramount consideration guiding decisions and actions."

As discussed in Section A.1.2.1 above, the most recent version of the Standard, CSA N286-12 [A.1-1], was assessed for the Darlington ISR (per NK38-REP-03680-10142 R000 [A.1-25]) and there were no gaps. That review is applicable to Pickering NGS. NK38-REP-03680-10142 R000 explicitly addressed the Impact Statement changes identified above, and therefore no additional assessment is required as part of PSR2.

Further, although there is no specific assessment against N286-12 for Pickering, N-LIST-08130-10025, "CSA N286-12 to OPGN Governance Cross-Matrix" [A.1-26] was produced in August 2015 to demonstrate OPG Nuclear fleet compliance with CSA N286-12. Similar to the controlled list prepared for N286-05 (N-LIST-08130-10023 [A.1-21]), this document contains a clause-by-clause listing of CSA N286-12 with cross-references to corresponding OPG governance documents. As shown by this compliance matrix, there are no gaps with respect to the applicable clauses of the CSA N286-12.

A.1.3 Compliance Summary for Pickering PSR2

There are no PSR2 gaps for CSA N286-12 [A.1-1]. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with N286-12.

A.1.4 References

[A.1-1] CSA Standard N286-12, *Management System Requirements for Nuclear Facilities*, 2012.

[A.1-2] CSA Standard N286-05 including Update No. 1 (R2010), *Management System Requirements for Nuclear Power Plants*, February 2005; Update No. 1: November 2007.

[A.1-3] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[A.1-4] CSA Standard N286-05, *Management System Requirements for Nuclear Power Plants*, February 2005.

[A.1-5] CSA Standard N286-05 including Update No. 1 and No. 2 (R2011), *Management System Requirements for Nuclear Power Plants*, February 2005; Update No. 1: November 2007; Update No. 2: December 2010.

[A.1-6] CSA Standard N286.0-92 (R2003), *Overall Quality Assurance Program Requirements for Nuclear Power Plants*, June 2003.

[A.1-7] CSA Standard N286.1-00, *Procurement Quality Assurance for Nuclear Power Plants*, March 2000.

[A.1-8] CSA Standard N286.2-00, *Design Quality Assurance for Nuclear Power Plants*, March 2000.

[A.1-9] CSA Standard N286.3-99 (R2004), *Construction Quality Assurance for Nuclear Power Plants*, September 1999.

[A.1-10] CAN/CSA Standard N286.4-M86 (R2000), *Commissioning Quality Assurance for Nuclear Power Plants*, September 1986.

[A.1-11] CSA Standard N286.5-95 (R2000), *Operations Quality Assurance for Nuclear Power Plants*, May 1995.

[A.1-12] CSA Standard N286.6-98 (R2003), *Decommissioning Quality Assurance for Nuclear Power Plans*, September 1998.

[A.1-13] CSA Impact Statement, *Notification of CSA N286 Management System Requirements for Nuclear Facilities; Product: New Edition; Product Designation: CSA N286-12*, Date not provided.

[A.1-14] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[A.1-15] OPG Report, NK30-REP-03680-00001 R000, *Pickering NGS-B - Report on Integrated Safety Review - Plant Design Safety Factor*, August 2007.

[A.1-16] OPG Report, NK30-REP-03680-00005 R000, *Pickering NGS-B - Integrated Safety Review - Safety Analysis Review,* June 2007.

[A.1-17] OPG Report, NK30-REP-03680-00006 R002, *Pickering NGS-B Integrated Safety Review: Review of Safety Performance,* March 2008.

[A.1-18] OPG Report, NK30-REP-03680-00008 R002, *Pickering NGS-B Integrated Safety Review: Management,* September 2009.

[A.1-19] OPG Report, NK30-REP-03680-00012 R001, *Pickering NGS-B Integrated Safety Review: Radiation Protection,* August 2009.

[A.1-20] OPG Report, NK30-REP-03680-00018 R000, *Pickering NGS-B Integrated Safety Review – Management Addendum #2*, September 2009.

[A.1-21] OPG Nuclear List, N-LIST-08130-10023 R003, *CSA N286-05 to OPGN Governance Cross Matrix*, October 2012.

[A.1-22] OPG Letter, NA44-CORR-00531-00381 R000, R. J. Strickert to J. S. C. Tong, *Pickering A – Updated Basis for Return to Service Document*, April 20, 2001.

[A.1-23] OPG Letter, P-CORR-00531-03719 R00, G. Jager to M. A. Leblanc, *Application for Renewal of Pickering Nuclear Generating Station Power Reactor Operating Licence*, July 4, 2012.

[A.1-24] OPG Report, NK38-REP-03680-10115 R000, *Review of CAN/CSA-N286-05, Management System Requirements for Nuclear Power Plants for Darlington Integrated Safety Review*, September 2011.

[A.1-25] OPG Report, NK38-REP-03680-10142 R000, *Code Refresh Review of CSA N286-12 (June 2012) Management System Requirements for Nuclear Power Plants*, February 2014.

[A.1-26] OPG Nuclear List, N-LIST-08130-10025 R000, *CSA N286-12 to OPGN Governance Cross-Matrix*, August 2015.

A.2 CNSC G-129 (2004), "Keeping Radiation Exposures and Doses 'As Low As Reasonably Achievable (ALARA)'"

A.2.1 Background

The following paraphrased from the purpose and scope of CNSC G-129 Revision 1 (2004) [A.2-1] provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *CNSC G-129 describes measures that regulated persons can take for the purpose of keeping the amount of exposure to radon progeny and the effective doses and equivalent doses received by and committed to persons as low as reasonable achievable, social, and economic factors being taken into account (ALARA).*

The standard is applicable in its entirety to Safety Factor 15 (Radiation Protection). Parts of Section 7 related to Operational Review (7.3.2), Documentation (7.5.1) and Radiological Performance Targets (7.5.2), as well as the parts of Section 8 related to Analysis and Rationale Required for Substantiation (8.2 (3), (4), (5) and (6)) are related to Safety Factor 8 (Safety Performance).

CNSC G-129 is identified in Appendix E.2 of the R04 Pickering Licence Conditions Handbook [A.2-2] as "Guidance or Criteria". As indicated in Section 8.1 of the LCH [A.2-2], G-129 is recommended for guidance in "developing, implementing and maintaining a radiation protection program to ensure that exposures will be ALARA".

CNSC G-129 revision 1 [A.2-1] is the second edition of this Regulatory Guide and supersedes the previous edition published in September 1997 under the title "Guidelines on How to Meet the Requirement to Keep All Exposures as Low as Reasonably Achievable" [A.2-3]. The results of PSR1 G-129 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington ISRs), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section A.2.2.

As identified in Reference [A.2-4], the Pickering PSR2 review of CNSC G-129 (2004) is an Incremental Review. PSR2 Incremental Review includes an assessment of the intent of recent changes to the L/R/C/S on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.
- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

A.2.2 Compliance Assessment for Pickering PSR2

A.2.2.1 Application of PSR1 Reviews

The versions of G-129 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

Revision 1 of G-129 [A.2-1] was reviewed as part of the Pickering B ISR, with the findings related to the Safety Performance Safety Factor presented in Reference [A.2-6] and with respect to the Radiation Protection Safety Factor presented in Reference [A.2-7]. Based on the review of the implementing program N-PROG-RA-0013 "Radiation Protection" and related OPG standards and procedures, it was concluded that "the overall OPG process for implementing a radiation protection program that keeps radiation exposure and doses ALARA, aligns with the intent of the process described in G-129 Rev. 01" [A.2-6]. No gaps were identified in either review.

Pickering Units 1,4

As part of the code review for Pickering A Return to Service, review of G-129 [A.2-3] was excluded from review on the basis that it "pertains mostly to operations aspect, or other aspects not having a direct or immediate effect on installed design features" [A.2-5]. However, the Pickering B ISR reviewed the latest version of G-129 and the conclusions of that work are applicable across the OPG Nuclear fleet (based on reference to Nuclear Programs such as N-PROG-RA-0013 "Radiation Protection"), including Pickering NGS (Units 1,4 and Units 5-8). Further, an assessment was also performed for Darlington which is programmatically applicable to Pickering, as discussed below.

Darlington NGS

Revision 1 of G-129 [A.2-1] was reviewed as part of the Darlington ISR. Similar to the Pickering B ISR review, it was concluded that "the Darlington Nuclear Generating Station is currently in compliance with CNSC Regulatory Guide G-129 Rev. 1" [A.2-8] based on the existence of suitable OPG Governance, Programs and Procedures. The findings were substantiated through the CNSC staff comments appended to the code review wherein the CNSC concurred with the "Compliant" conclusion based on the high level intent review combined with additional clarification provided in the comment dispositions [A.2-9].

Since compliance against CNSC G-129 is based on Governance, Programs and Procedures that apply across OPG's Nuclear operations, the Darlington ISR conclusions are applicable to Pickering PSR2.

Based on the above, OPG compliance with CNSC G-129 Revision 1 has been demonstrated through past assessments performed for the Pickering B and Darlington ISRs.

A.2.2.2 Application of Post PSR1 Reviews

The version of G-129 assessed in the Pickering B and Darlington ISRs is the most recent version of this guidance document. No additional code reviews have been performed as none were required.

A.2.3 Compliance Summary for Pickering PSR2

There are no PSR2 gaps for CNSC G-129 Revision 1 [A.2-1]. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CNSC G-129 Revision 1.

A.2.4 References

[A.2-1] CNSC Regulatory Guide G-129, Revision 1, *Keeping Radiation Exposure and Doses "As Low as Reasonably Achievable (ALARA)"*, October 2004.

[A.2-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[A.2-3] CNSC Regulatory Guide G-129, *Guidelines on How to Meet the Requirement to Keep All Exposures As Low As Reasonably Achievable*, September 1997.

[A.2-4] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[A.2-5] OPG Letter, NA44-CORR-00531-00381 R000, R. J. Strickert to J. S. C. Tong, *Pickering A – Updated Basis for Return to Service Document*, April 20, 2001.

[A.2-6] OPG Report, NK30-REP-03680-00006 R002, *Pickering NGS-B Integrated Safety Review: Review of Safety Performance*, March 2008.

[A.2-7] OPG Report, NK30-REP-03680-00012 R001, *Pickering NGS-B Integrated Safety Review – Radiation Protection*, August 2009.

[A.2-8] OPG Report, NK38-REP-03680-10052 R000, *Review of CNSC G-129 Rev. 1 (October 2004), Keeping Radiation Exposures and Doses "As Low As Reasonably Achievable (ALARA)" for Darlington Integrated Safety Review*, June 2011.

[A.2-9] OPG Report, NK38-REP-03680-10052-ADD-001 R000, *Addendum to the CNSC G-129 Code Review Report for Darlington ISR*, January 2014.

A.3 CNSC G-228 (2001), "Developing and Using Action Levels"

A.3.1 Background

The following paraphrased from the purpose and scope of CNSC G-228 (2001) [A.3-1] provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *The purpose of CNSC G-228 is to help applicants for Canadian Nuclear Safety Commission (CNSC) licences develop action levels in accordance with paragraph 3(1)(f) on the General Nuclear Safety and Controls Regulations (SOR/2000-202) and Section 6 of the Radiation Protection Regulations (SOR/2000-203).*

CNSC G-228 describes how the licence applicant can develop action levels that provide for the radiation protection of workers and the public during the conduct of activities licensed by the CNSC. G-228 is applicable for Safety Factors 8 (Safety Performance), 14 (Radiological Impact on the Environment) and 15 (Radiation Protection).

CNSC G-228 is identified in Appendix E.2 of the R04 Pickering Licence Conditions Handbook [A.3-2] as "Guidance or Criteria". As indicated in Section 8.1 and 10.2 of the LCH, G-228 "provides the licensees guidance for developing action levels in accordance with the General Nuclear Safety and Control Regulations and Section 6 of the Radiation Protection Regulations" [A.3-2]. CNSC G-228 (2001) is the first edition of this Regulatory Guide. The results of PSR1 G-228 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington ISRs), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section A.3.2.

As identified in Reference [A.3-3], the Pickering PSR2 review of CNSC G-228 (2001) is an Incremental Review. PSR2 Incremental Review includes an assessment of the intent of recent changes to the L/R/C/S on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.
- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

A.3.2 Compliance Assessment for Pickering PSR2

A.3.2.1 Application of PSR1 Reviews

The versions of G-228 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

For the Pickering B ISR, Sections 5 and 7 of the latest version of G-228 [A.3-1] were reviewed for the Safety Performance and Radiation Protection Safety Factors as documented in References [A.3-4] and [A.3-5]. These reviews concluded that:

> *The overall OPG process for developing and using action levels aligns with the intent of the process described in G-228. Therefore, Pickering B is judged to be in compliance with the intent of CNSC regulatory guide G-228.*

There is no evidence of a Pickering B ISR review of Section 6 which covers the development, use, revision and monitoring of Action Levels (ALs). However, as discussed further below, Pickering NGS compliance is demonstrated through statements made in the Pickering LCH [A.3-2]. Further, the Darlington ISR review of G-228 covers the full content of G-228 and the conclusions are programmatically applicable to Pickering NGS, as discussed below.

As discussed earlier, compliance with G-228 is not a license requirement for Pickering NGS. However, the document is referenced in the Pickering LCH as a guide to developing action levels. The LCH further states that "the licensee should conduct a documented review and, if necessary, revise the ALs specified above at least once per licence period in order to validate their effectiveness. The results of such reviews should be provided to CNSC staff" [A.3-2]. Pickering NGS adherence with the Environmental Action Limit requirements of G-228 is also confirmed through the following statement in Section 10.2 of the LCH:

> *OPG's EALs (Environmental Action Limits) are documented in the reports NA44-REP-03482-00001 "Derived Release Limits and Environmental Action Levels for Pickering Nuclear Generating Station A", NK30-REP-03482-00001 "Derived Release Limits and Environmental Action Levels for Pickering Nuclear Generating Station B", and P-REP-03482-00001 "Derived Release Limits and Environmental Action Levels for Pickering Nuclear Sewage Effluent", which are consistent with CNSC expectations set out in G-228.*

Pickering Units 1,4

No review was performed for Pickering Units 1,4 against any version of G-228. However, as discussed above, the Pickering B ISR reviewed the latest version of G-228 and the conclusions of that work are applicable to both Pickering Units 1,4 and Units 5-8. An assessment was also performed for Darlington which is programmatically applicable to Pickering, as discussed below.

Darlington NGS

The latest version of G-228 [A.3-1] (the same version reviewed for the Pickering B ISR) was reviewed as part of the Darlington ISR as documented in NK38-REP-03680-10053 [A.3-6]. NK38-REP-03680-10053 assessed Darlington compliance with G-228 by reference to two OPG reports: N-REP-03480-10003 R001, "Environmental Action Levels for Ontario Power Generation Nuclear Generating Stations" [A.3-7], and N-REP-03420-10001 R000, "Occupational Radiation Protection Action Levels for Power Reactor Operating Licences" [A.3-8]. Two observations were made:

1. N-REP-03420-10001 R000, which defines the OPG occupational action levels for inclusion in all OPG Power Reactor Operating Licence submissions, does not include procedures for revising occupational Action Levels [A.3-6].

2. A discrepancy between the reporting time in the as then current procedure in N-REP-03420-10001 and the Darlington Power Reactor Operating Licence [A.3-9] was found wherein the OPG report required notification to the CNSC within 7 days of becoming aware of reaching an Action Level, whereas references in the PROL required notification within three calendar days. This discrepancy was noted by the Health Physics Department and the Darlington PROL reporting requirements took precedence while N-REP-03420-10001 was revised to remove the discrepancy.

NK38-REP-03680-10053 [A.3-6] concluded that neither finding was a compliance gap with CNSC G-228 from a high level intent perspective and stated that "based on a programmatic high-level intent review, Darlington NGS is currently in compliance with CNSC G-228". This is supported by the appended CNSC staff comments [A.3-10]. The reporting time requirement was removed from the subsequent revision of N-REP-03420-10001. In addition, the discrepancy is no longer applicable to Darlington (and is not applicable to Pickering NGS) as the current versions of the Pickering and Darlington PROLs (Condition 8.1) require notification within seven days.

Based on the above, OPG compliance with CNSC G-228 [A.3-1] is demonstrated through the ISR assessments performed for both Pickering B and Darlington, as documented in References [A.3-4] to [A.3-10]. Since compliance against CNSC G-228 is based on Governance, Programs and Procedures that apply across OPG's Nuclear operations, the Darlington ISR conclusions are applicable to Pickering PSR2.

A.3.2.2 Application of Post PSR1 Reviews

The version of CNSC G-228 assessed in the Pickering B and Darlington ISRs is the most recent version of this guidance document. No additional code reviews have been performed as none were required.

A.3.3 Compliance Summary for Pickering PSR2

There are no PSR2 gaps for CNSC G-228 [A.3-1]. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CNSC G-228.

A.3.4 References

[A.3-1] CNSC Regulatory Guide G-228, *Developing and Using Action Levels*, March 2001.

[A.3-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[A.3-3] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[A.3-4] OPG Report, NK30-REP-03680-00006 R002, *Pickering NGS-B Integrated Safety Review: Review of Safety Performance*, March 2008.

[A.3-5] OPG Report, NK30-REP-03680-00012 R001, *Pickering NGS-B Integrated Safety Review – Radiation Protection*, August 2009.

[A.3-6] OPG Report, NK38-REP-03680-10053 R000, *Review of CNSC G-228 (March 2001), Developing and Using Action Levels for Darlington Integrated Safety Review*, June 2011.

[A.3-7] OPG Report, N-REP-03480-10003 R001, *Environmental Action Levels for Ontario Power Generation Nuclear Generating Stations*, March 2006.

[A.3-8] OPG Report, N-REP-03420-10001 R000, *Occupational Radiation Protection Action Levels for Power Reactor Operating Licences*, August 2002.

[A.3-9] *Darlington Nuclear Generating Station: Nuclear Power Reactor Operating License*, PROL 13.02/2013.

[A.3-10] OPG Report, NK38-REP-03680-10053-ADD-001 R000, *Addendum to the CNSC G-228 Code Review Report for Darlington ISR*, January 2014.

A.4 SOR/2000-202, "General Nuclear Safety and Control Regulations" (Amended in June 2015)

A.4.1 Background

The following outline of the General Nuclear Safety and Control Regulations (Amended in June 2015) [A.4-1], paraphrased from the CNSC List of Regulations Website [A.4-2], provides a brief overview of the purpose of the regulation and the information expressed therein:

> *The purpose of the General Nuclear Safety and Control Regulations is to provide general regulations with respect to licence applications and renewals, exemptions, obligations of licensees, prescribed nuclear facilities and equipment and information, contamination, record-keeping, and inspections.*

The General Nuclear Safety and Control Regulations are relevant to Safety Factor 10 (Organization, Management System and Safety Culture) and Safety Factor 15 (Radiation Protection). The General Nuclear Safety and Control Regulations were last amended in June, 2015. The changes made in 2015 include the following [A.4-3]:

1. An update to the referenced Packaging and Transport of Nuclear Substances Regulations, 2015 [A.4-4] with respect to General License Application Requirements (Section 3);

2. Changes to the exemption criteria for naturally occurring nuclear substances (Section 10); and

3. The definition of Prescribed Equipment (Section 20 (a)) with reference to the updated Packaging and Transport of Nuclear Substances Regulations, 2015 [A.4-4].

Other amendments that have been included as previous consolidated versions of the regulations were published and can be found on the Government of Canada Justice Laws Website [A.4-3].

Per Section 2.1 of the R04 Pickering Licence Conditions Handbook [A.4-5]:

> *Paragraphs 3(1)(k) and 15(a) and (b) of the General Nuclear Safety and Control Regulations require that a licence application contain information related to the organizational management structure and responsibilities… An adequately established and implemented management system provides CNSC staff confidence and evidence that the legal basis under which the Commission made its decision and had issued a licence, pursuant the Nuclear Safety and Control Act, remains valid.*

The results of PSR1 General Nuclear Safety and Control Regulations reviews (Pickering A Return to Service assessments, and Pickering B and Darlington ISRs), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section A.4.2. As identified in Reference [A.4-6], the Pickering PSR2 review of the General Nuclear Safety and Control Regulations (Amended in June 2015) is an Incremental Review. PSR2 Incremental Review includes an assessment of the intent of recent changes to the L/R/C/S on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.

- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

A.4.2 Compliance Assessment for Pickering PSR2

A.4.2.1 Application of PSR1 Reviews

The versions of the General Nuclear Safety and Control Regulations (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

The Pickering B ISR included a review of Radiation Protection in OPG Report NK30-REP-03680-00012 R001 [A.4-7]. That review included a clause-by-clause review of the SOR/2000-202 sections that are relevant to Radiation Protection. OPG was in compliance with all clauses.

Pickering Units 1,4

The General Nuclear Safety and Control Regulations were not reviewed as part of the Pickering A RTS assessments.

Section 2.1 of the R04 Pickering Licence Conditions Handbook [A.4-5] states:

> *Paragraphs 3(1)(k) and 15(a) and (b) of the General Nuclear Safety and Control Regulations require that a licence application contain information related to the organizational management structure and responsibilities…*
>
> *Safe and reliable operation requires a commitment and adherence to a set of management system principles and, consistent with those principles, the establishment and implementation of processes that achieve the expected results. The CNSC measures the activities carried out at a nuclear power plant throughout its life cycle against the CSA N286-05 standard, which contains requirements for a management system.*
>
> *A management system brings together in a planned and integrated manner the processes necessary to satisfy requirements and to carry out licensed activity in a safe manner. Management system requirements provide direction to management to develop and implement management practices and controls. The elements of a management system include areas such as organization structure and culture, resources, equipment, and information. Requirements in applicable codes and standards are addressed in the processes comprising the management system. Finally, the management system must satisfy the requirements set out in the NSCA, regulations made pursuant to the NSCA, the licence and the measures necessary to ensure that safety is of paramount consideration in implementation of the management system.*

An adequately established and implemented management system provides CNSC staff confidence and evidence that the legal basis under which the Commission made its decision and had issued a licence, pursuant the Nuclear Safety and Control Act, remains valid.

A compliance review against CSA N286 is assessed as part of PSR2, per Reference [A.4-6].

The Pickering NGS PROL Renewal Application [A.4-8] states:

Table 1 is included for convenience, to assist in locating specific information within the application corresponding to the requirements of the Nuclear Safety and Control Act and applicable Regulations.

Table 1 of [A.4-8] demonstrates where in the PROL application the applicable clauses of the General Nuclear Safety and Control Regulations are addressed. The PROL was issued in part on the basis of the PROL application, which demonstrates that the regulatory requirements have been achieved to the satisfaction of the regulator.

There are no PSR2 gaps.

Darlington NGS

An assessment of compliance with the General Nuclear Safety and Control Regulations (May 2000) was performed for the Darlington ISR in OPG Report NK38-REP-03680-10055 R000 [A.4-9]. The review is applicable to Pickering since the program requirements specified for Darlington are fleet-wide programs. No gaps were identified.

A.4.2.2 Application of Post-PSR1 Reviews

Subsequent to the 2012 Pickering Licence Renewal application [A.4-8], the General Nuclear Safety and Control Regulations were amended in 2015 [A.4-1]. The changes made in 2015 include the following:

1. An update to the referenced Packaging and Transport of Nuclear Substances Regulations, 2015 [A.4-4] with respect to General License Application Requirements (Section 3);

2. Changes to the exemption criteria for naturally occurring nuclear substances (Section 10); and

3. The definition of Prescribed Equipment (Section 20 (a)) with reference to the updated Packaging and Transport of Nuclear Substances Regulations, 2015 [A.4-4].

These amendments have no impact on nuclear safety. Further, the two amendments related to Packaging and Transportation are not part of the scope of PSR2 as described in the PSR2 Basis Document Appendix D [A.4-6].

There are no PSR2 gaps.

A.4.3 Compliance Summary for Pickering PSR2

There are no PSR2 gaps for the General Nuclear Safety and Control Regulations (Amended in June 2015) [A.4-1]. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with the General Nuclear Safety and Control Regulations (Amended in June 2015).

A.4.4 References

[A.4-1] Statutory Orders and Regulations, SOR/2000-202, *General Nuclear Safety and Control Regulations*, amended on June 12, 2015.

[A.4-2] CNSC Website, *List of regulations*, http://cnsc-ccsn.gc.ca/eng/acts-and-regulations/regulations/index.cfm, January 2016.

[A.4-3] Government of Canada, *Justice Laws Website - General Nuclear Safety and Control Regulations Table of Contents*, http://laws-lois.justice.gc.ca/eng/regulations/SOR-2000-202/index.html, June 2016.

[A.4-4] Statutory Orders and Regulations, SOR/2015-145, *Packaging and Transport of Nuclear Substances Regulations*, 2015, June 2016.

[A.4-5] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[A.4-6] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[A.4-7] OPG Report, NK30-REP-03680-00012 R001, *Pickering NGS-B Integrated Safety Review - Radiation Protection*, August 2009.

[A.4-8] OPG Letter, P-CORR-00531-03719 R000, G. Jager to M. A. Leblanc, *Application for Renewal of Pickering Nuclear Generating Station Power Reactor Operating Licence*, July 4, 2012.

[A.4-9] OPG Report, NK38-REP-03680-10055 R000, *Review of SOR/2000-202 (May 2000), General Nuclear Safety and Control Regulations for Darlington Integrated Safety Review*, June 2011.

A.5 SOR/2000-203, "Radiation Protection Regulations" (Amended in June 2015)

A.5.1 Background

The following outline of the Radiation Protection Regulations [A.5-1], paraphrased from the CNSC website [A.5-2], provides a brief overview of the purpose of the regulation and the information expressed therein:

> *The purpose of the regulation is to define the "as low as reasonably achievable" (ALARA) principle and regulations for radiation dose limits, action limits, and requirements for labeling and signage, and reports.*

The Radiation Protection Regulations are relevant to Safety Factors 8 (Safety Performance) and 15 (Radiation Protection). The Radiation Protection Regulations were last amended in June, 2015. Sections that have been amended include a citation to the amending Act or regulation. The changes made in 2015 include an update to the referenced Packaging and Transport of Nuclear Substances Regulations, 2015 [A.5-3] with respect to Labelling of Containers and Devices (Section 20, subsection 2). Other amendments from September 2007 have been included as previous consolidated versions of the regulations were published and can be found on the Government of Canada Justice Laws Website [A.5-4].

Per Section 8 of the R04 Pickering Licence Conditions Handbook [A.5-5]: "The Safety and Control Area 'Radiation Protection' covers the implementation of a radiation protection program in accordance with the *Radiation Protection Regulations*. This program must ensure that contamination and radiation doses received are monitored and controlled."

The results of PSR1 Radiation Protection Regulations reviews (Pickering A Return to Service assessments, and Pickering B and Darlington ISRs), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section A.5.2. As identified in Reference [A.5-6], the Pickering PSR2 review of the Radiation Protection Regulations (Amended in June 2015) is an Incremental Review. PSR2 Incremental Review includes an assessment of the intent of recent changes to the L/R/C/S on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.
- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

A.5.2 Compliance Assessment for Pickering PSR2

A.5.2.1 Application of PSR1 Reviews

The versions of the Radiation Protection Regulations (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

The Pickering B ISR included a review of Radiation Protection in OPG Report NK30-REP-03680-00012 R001 [A.5-7]. That review included a clause-by-clause review of the SOR/2000-203 (May 2000) sections that are relevant to Radiation Protection. OPG was in compliance with all clauses with the exception of the following two clauses, which were classed as acceptable deviations:

> *(1) SOR/2000-203 Clause 20 (1) (a): A container or device that contains a radioactive nuclear substance must be labeled with the words, "RAYONNEMENT – DANGER – RADIATION".*
>
> *(2) SOR/2000-203 Clause 21 (1): Hazard warning signs must contain the words, "RAYONNEMENT – DANGER – RADIATION".*

OPG container, device and hazard warning labels and signage at Pickering are not in compliance with the wording required in the Radiation Protection Regulations. Pickering labels and signage do not contain the words *"RAYONNEMENT – DANGER – RADIATION".* These signage wording differences were considered to be acceptable following the 2009 Pickering B ISR [A.5-7]. However as noted below, Darlington has committed to amending signage to align with the wording in the Radiation Protection Regulations which included actions for aligning Pickering U1,4 and U5-8 signage with the regulatory requirements [A.5-8]. All actions associated with the Regulatory Management Action # 28101238 open to track the commitment have been completed, as reported to the CNSC in status update N-CORR-00531-06042, "Radiation Labelling and Posting" [A.5-9]. As such, there are no outstanding gaps from the Pickering B ISR that are applicable to Pickering PSR2.

Pickering Units 1,4

The Radiation Protection Regulations were not part of the Pickering A Return to Service assessments. However, the Pickering NGS PROL Renewal Application [A.5-10] states:

> *Table 1 is included for convenience, to assist in locating specific information within the application corresponding to the requirements of the Nuclear Safety and Control Act and applicable Regulations.*

Table 1 of [A.5-10] demonstrates where in the PROL application the applicable clauses of the Radiation Protection Regulations are addressed. The PROL was issued in part on the basis of the PROL application, which demonstrates that the regulatory requirements have been achieved to the satisfaction of the regulator.

Therefore, there is no PSR2 gap.

Darlington NGS

A clause-by-clause review of the Radiation Protection Regulations (Amended September 2007)[2] was performed during the Darlington ISR in OPG Report NK38-REP-03680-10056 R000 [A.5-11]. Darlington was determined to be in compliance with the exception of gaps related to signage. The following is quoted from reference [A.5-11]:

1. *Clause 20 of SOR/2000-203 states "… No person shall possess a container or device that contains a radioactive nuclear substance unless the container or device is labelled with (a) the radiation warning symbol set out in Schedule 3 and the words "RAYONNEMENT - DANGER - RADIATION"; …"*

   Requirements for labelling of containers or devices are detailed in Section 1.4 of the OPG procedure N-PROC-RA-0024 R014, "Hazard Surveys, Posting, Labelling and Radiological Log."

   The labels referred to (OPG form N-FORM-10076 R003, "Radioactive Material Tag" and OPG form N-FORM-10017 R0001, "Licensed Radioisotope Source Inventory Control Label") indicate the appropriate radiation hazards. However, they do not currently have the words RAYONNEMENT - DANGER - RADIATION" as specified in Schedule 3 of SOR/2000-203.

2. *Clause 21 of SOR/2000-203 states "Every licensee shall post and keep posted, at the boundary of and at every point of access to an area, room or enclosure, a durable and legible sign that bears the radiation warning symbol set out in Schedule 3 and the words "RAYONNEMENT-DANGER-RADIATION", if;*

   (a) there is a radioactive nuclear substance in a quantity greater than 100 times its exemption quantity in the area, room or enclosure; or

   (b) there is a reasonable probability that a person in the area, room or enclosure will be exposed to an effective dose rate greater than 25 µSv/h."

   Requirements for posting of signs at boundaries and points of access are detailed in Section 1.3 of the OPG procedure N-PROC-RA-0024 R014, "Hazard Surveys, Posting, Labelling and Radiological Log". The signs used by OPG indicate the appropriate radiation hazards. However, the words "RAYONNEMENT-DANGER-RADIATION" are not currently specified as a requirement.

3. *Also in relation to clause 21, specifically sub-clause (a), Airborne and loose surface contamination levels are included; however relation to exemption quantities for radioactive nuclear substances is not specified.*

In the Darlington ISR Report [A.5-13], Issue D157 was assigned to the label and signage gaps (i.e. the fact that Darlington labels and signage do not include the words *"RAYONNEMENT-DANGER-RADIATION"* as required by the Radiation Protection Regulations). The gaps were

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[2] The clause-by-clause review states that SOR/2000-203 (May 2000) was reviewed. Although the latest amendment date of the regulations is not specified in the code review report, the reviewed clauses include the amendments from September 2007 [A.5-12].

addressed with OPG actions to update the Darlington Radiation signs and labels, and the associated procedures. The ISR issue was reclassified as an Acceptable Deviation. Regulatory Management Action # 28101238 was opened to track actions committed to in OPG Letter N-CORR-00531-04586, "Radiation Labelling and Posting" [A.5-8]. The AR included actions for aligning Pickering U1,4 and U5-8 labels to comply with the Radiation Protection Regulations. All actions to align the Pickering and Darlington labelling with the regulatory requirements have been completed, as reported to the CNSC in the status update N-CORR-00531-06042, "Radiation Labelling and Posting" [A.5-9]. Furthermore, the Darlington ISR gap related to Clause 21, sub-clause (a) was assigned to Issue D275, which was classified as Closed since the requirements were met by Section 1.6.2 of N-PROC-RA-0016, per the Darlington Final ISR Report [A.5-13]. As such, there are no outstanding gaps from the Darlington ISR that are applicable to Pickering PSR2.

Since compliance against SOR/2000-203 (Amended September 2007) is based on Governance, Programs and Procedures that apply across OPG's Nuclear operations, the Darlington ISR conclusions are applicable to Pickering PSR2.

A.5.2.2 Application of Post-PSR1 Reviews

Subsequent to the Darlington ISR review of SOR/2000-203 (Amended September 2007), a change was made in 2015 to update the referenced Packaging and Transport of Nuclear Substances Regulations [A.5-3] in Section 20, subsection 2 of SOR/2000-203, which relates to Labelling of Containers and Devices. This Amendment is not safety significant. Further, Packaging and Transportation are not part of PSR2 scope as described in PSR2 Basis Document Appendix D [A.5-6]. Therefore, there is no PSR2 gap.

A.5.3 Compliance Summary for Pickering PSR2

There are no PSR2 gaps for the Radiation Protection Regulations (Amended June 2015) [A.5-1]. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with the Radiation Protection Regulations (Amended June 2015).

A.5.4 References

[A.5-1] Statutory Orders and Regulations, SOR/2000-203, *Radiation Protection Regulations*, amended on June 12, 2015.

[A.5-2] CNSC Website, *List of regulations,* http://cnsc-ccsn.gc.ca/eng/acts-and-regulations/regulations/index.cfm, January 2016.

[A.5-3] Statutory Orders and Regulations, SOR/2015-145, *Packaging and Transport of Nuclear Substances Regulations, 2015*, June 2016.

[A.5-4] Government of Canada, *Justice Laws Website - Radiation Protection Regulations,* http://laws-lois.justice.gc.ca/eng/regulations/SOR-2000-203/index.html, June 2016.

[A.5-5] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[A.5-6] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[A.5-7] OPG Report, NK30-REP-03680-00012 R001, *Pickering NGS-B Integrated Safety Review - Radiation Protection*, August 2009.

[A.5-8] OPG Letter, N-CORR-00531-04586, P.F.Tremblay to T.E. Schaubel and P.A. Webster, *Radiation Labelling and Posting*, September 2, 2009.

[A.5-9] OPG Letter, N-CORR-00531-06042, L. Swami to M. Santini and F. Rinfret, *Radiation Labelling and Posting*, February 12, 2013.

[A.5-10] OPG Letter, P-CORR-00531-03719 R000, G. Jager to M. A. Leblanc, *Application for Renewal of Pickering Nuclear Generating Station Power Reactor Operating Licence*, July 4, 2012.

[A.5-11] OPG Report, NK38-REP-03680-10056 R000, *Review Of SOR/2000-203 (May 2000), Radiation Protection Regulations For Darlington Integrated Safety Review*, August 2011.

[A.5-12] Statutory Orders and Regulations, SOR/2000-203, *Radiation Protection Regulations*, amended on September 18, 2007.

[A.5-13] OPG Report, NK38-REP-03680-10104 R000, *Darlington NGS Integrated Safety Review (ISR) – Final ISR Report*, October 2011.

A.6 CNSC REGDOC-2.2.3 (2014), "Personnel Certification: Radiation Safety Officers"

A.6.1 Background

The following paraphrased from the purpose and scope of CNSC REGDOC-2.2.3 (2014) [A.6-1] provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *REGDOC-2.2.3 sets out guidance to assist applicants in completing an application for certification as a Radiation Safety Officer (RSO) pursuant to the Class II Nuclear Facilities and Prescribed Equipment Regulations.*
>
> *REGDOC-2.2.3 is based on the Nuclear Safety and Control Act (NSCA) and its regulations, which are administered by the CNSC. The document provides detailed information about the completion of an application and the process for the RSO certification. The document explains what is needed in an application form, in order to assess if the applicant has the qualifications necessary to be certified as a RSO. The information submitted will also help the CNSC plan and conduct the certification examination. Any information which is submitted may subsequently be referred to in the RSO certificate. It then becomes a requirement of the certificate, and is thus legally binding.*

REGDOC-2.2.3 is relevant to Safety Factor 15 (Radiation Protection). REGDOC-2.2.3 is not discussed in the R04 Pickering Licence Conditions Handbook [A.6-2]. As discussed in Sections A.6.2 and A.6.3 below, there is no Class II nuclear facility or prescribed equipment at Pickering NGS.

As identified in Reference [A.6-3], the Pickering PSR2 review of CNSC REGDOC-2.2.3 (2014) is a High Level review. For a PSR2 High Level review, the degree of conformance with clauses or groups of clauses in the L/R/C/S is demonstrated by supporting evidence stating whether the intent of the requirements stipulated in the requirement document is met. The review identifies Compliances and Gaps, where required, as defined below:

- <u>Compliance:</u> Compliance indicates that the intent of the safety requirement is met.
- <u>Gap</u>: A Gap indicates that the intent of the safety requirement is not met.

A.6.2 Compliance Assessment for Pickering PSR2

A.6.2.1 Application of PSR1 Reviews

Pickering NGS

There was no PSR1 compliance review done for Pickering NGS against REGDOC-2.2.3. Compliance with REGDOC-2.2.3 is only applicable if the facility operator has a Class II nuclear facility or prescribed equipment licence. There are no such licences issued for any Class II activities at the Pickering site.

Darlington NGS

There was no PSR1 compliance review done for Darlington ISR against REGDOC-2.2.3.

A.6.2.2 Application of the Post-PSR1 Reviews

Compliance with REGDOC-2.2.3 is a licence condition for operators of Class II facilities. OPG has one such facility located in the Common Service Area of the Darlington Plant [A.6-4]. Per REGDOC-2.2.3, if an individual has the responsibilities and authorization to be a Responsible Health Physicist (RHP) for a facility, the individual may also be authorized as a RSO per section 1.3 of [A.6-1]:

> *In accordance with section 15.12 of the Class II Nuclear Facilities and Prescribed Equipment Regulations, a licensee does not need to appoint a RSO for a Class II facility if a person who has duties equivalent to that of a RSO has been designated and that person is certified under subsection 9(2) of the Class I Nuclear Facilities Regulations.*

In addition to the above designation of RSOs, the appendix of Reference [A.6-4] identifies the documents in support of the licence. The primary reference relating to the Class II facility operation is [A.6-5]. The RSO accountabilities and responsibilities are detailed in Section 1.2.1 of [A.6-5] with further details provided in [A.6-6].

A.6.3 Compliance Assessment Summary for Pickering PSR2

There is no Class II nuclear facility or prescribed equipment at Pickering NGS and there is no requirement to have a Radiation Safety Officer. Hence, a high level review of REGDOC-2.2.3 is not required because REGDOC-2.2.3 is not applicable to Pickering PSR2.

A.6.4 References

[A.6-1] CNSC Regulatory Document REGDOC-2.2.3, *Personnel Certification: Radiation Safety Officers*, *Class II Nuclear Facilities and Prescribed Equipment Regulations*, June 2014.

[A.6-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[A.6-3] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[A.6-4] CNSC Letter to OPG, K. Murthy to J. Duhig, N-CORR-00531-07506 R000, *Class II Nuclear Facility and Prescribed Licence No. 12861-18-26.0,* January 28, 2016.

[A.6-5] OPG Manual, N-MAN-03420-10000 R001, *Class II Facility and Prescribed Equipment Program Manual,* November 2015.

[A.6-6] OPG Manual, N-MAN-08131-10000-CNSC-032 R00, *Corporate Radiation Safety Officer*, October 2014.

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ONTARIO POWER GENERATION	
ACCEPTED	✓
ACCEPTED AS NOTED	
REVISE AND RESUBMIT	
Signature: [signature]	Date: 13 Dec 2016
Name: Mike Ruffolo, Manager	
Dept: Pickering Engineering - Aging Management & Strategic Initiatives	
OPG Proprietary	
Doc No.: P-REP-03680-00021	**Rev: 000**
This acceptance does not relieve the contractor from responsibility for errors or omissions or from any obligations or liability under this contract.	

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Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 8, 10, 12, 13, and 14

PS112/RP/011 R00

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Revision Summary – For Amec Foster Wheeler Report PS112/RP/011

Rev	Date	Author	Comments
R00	December 12, 2016	Amec Foster Wheeler Staff	Initial issue of report addressing OPG comments on Law, Regulation, Code and Standard reviews.

EXECUTIVE SUMMARY

Ontario Power Generation (OPG) is performing a Periodic Safety Review (PSR) to support the possibility of continued operation of Pickering Nuclear Generating Station (NGS) beyond 2020. The PSR (referred to as "PSR2") is a subsequent PSR building on the review basis of earlier OPG Integrated Safety Reviews and other associated assessments. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

PSR2 will support and complement the licence renewal application for Pickering NGS going forward. Fifteen Safety Factors will be assessed as part of the PSR. The purpose of Safety Factor reviews is to confirm that the design, condition and operation of Pickering Units 1,4 and 5-8 (as well as common systems) will support continued safe operation for the period of PSR2 by:

- Assessing compliance against "Review Tasks" identified in the PSR2 Basis Document [1], which were derived from Canadian Nuclear Safety Commission (CNSC) REGDOC-2.3.3 [2] and International Atomic Energy Agency (IAEA) SSG-25 [3];

- Documenting assessments against applicable modern Laws, Regulations, Codes and Standards (L/R/C/Ss) (as defined in Reference [1]); and

- Reviewing the effectiveness of applicable OPG Nuclear Programs that support the above assessments, through review of audit and self-assessment results.

The process to identify the modern L/R/C/Ss that are applicable to the PSR2 Assessment Basis involved first creating a broad list from multiple sources (potential candidate L/R/C/Ss) and then filtering it to identify those that are most significant, and that are applicable to the PSR2 scope. The identification and selection criteria are detailed in the PSR2 Basis Document [1]. The result of the identification and selection process was a set of modern L/R/C/Ss that became part of the "PSR2 Assessment Basis". This report provides the reviews of L/R/C/Ss that are required to address PSR2 Safety Factors 8 (Safety Performance), 10 (Organization, Management System, and Safety Culture), 12 (Human Factors), 13 (Emergency Planning), and 14 (Radiological Impact on the Environment). As noted in Section 2.0, reviews of several L/R/C/Ss applicable to Safety Factors 8, 10 and 14 were provided in Reference [4] and findings from these reviews are not duplicated in this report. There is also some overlap with other Safety Factors for a number of L/R/C/Ss considered, as outlined in Table 1 in Section 2.0 of this report.

The summary of findings documented in Appendix B of this report is as follows:

- CSA N290.15-10, "Requirements for the Safe Operating Envelope of Nuclear Power Plants": No gaps.

- CSA N286.7-16, "Quality Assurance of Analytical, Scientific and Design Computer Programs": No gaps.

- CSA N288.1-14, "Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities": No gaps.

- CSA N288.4-10, “Environmental Monitoring Program at Class I Nuclear Facilities and Uranium Mines and Mills”: No gaps.

- CSA N293-12, “Fire Protection for Nuclear Power Plants”: Results are presented in the Pickering NGS PSR2 Plant Design Safety Factor Report.

- CNSC RD-204 (2008), “Certification of Persons Working at Nuclear Power Plants”: No gaps.

- CNSC REGDOC-3.1.1 (2014), “Reporting Requirements for Nuclear Power Plants”: No gaps.

- CNSC REGDOC-2.9.1 (2013), “Environmental Protection Policies, Programs and Procedures”: No gaps.

- CNSC REGDOC-2.10.1 (2014), “Nuclear Emergency Preparedness and Response”: There is one gap associated with Safety Factor 13.

- CNSC G-323 (2007), “Ensuring the Presence of Sufficiently Qualified Staff at Class I Nuclear Facilities-Minimum Shift Complement”: No gaps.

- CNSC G-278 (2003), “Human Factors Verification and Validation Plans”: No gaps.

- CNSC G-276 (2003), “Human Factors Engineering Program Plans”: No gaps.

- S.C. 1997, C.9 (Amended in February 2015), “Nuclear Safety and Control Act”: No gaps.

- CSA N1600-14, “General Requirements for Nuclear Emergency Management Programs”: No gaps.

- CSA N288.6-12, “Environment Risk Assessments at Class I Nuclear Facilities and Uranium Mines and Mills”: No gaps.

- CSA N288.5-11, “Effluent Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills”: No gaps.

- CNSC REGDOC-2.2.2 (2014), “Personnel Training”: No gaps.

- CNSC REGDOC-2.3.2 (2015), “Accident Management, Version 2”: There are no gaps associated with Safety Factors 8, 10, 12, 13 and 14. However, there is one gap associated with Safety Factor 1.

- CNSC REGDOC-2.3.3 (2015), “Periodic Safety Reviews”: No gaps.

- CSA N286.7.1-09, “Guideline for the Application of N286.7-99, Quality Assurance of Analytical, Scientific, and Design Computer Programs for Nuclear Power Plants”: An assessment of N286.7.1-09 was not performed. Relevant guidance from N286.7.1-09 has been amalgamated into CSA N286.7-16, which was reviewed for PSR2.

- CSA N290.12-14, “Human Factors in Design for Nuclear Power Plants”: No gaps.
- CSA N288.3.4-13, “Performance Testing of Nuclear Air-Cleaning Systems at Nuclear Facilities”: No gaps.
- CSA N288.7-15, “Groundwater Protection Programs at Class I Nuclear Facilities and Uranium Mines and Mills”: No gaps.

Details of the reviews can be found in Table 2 and Appendix B of this report.

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1.0 INTRODUCTION

Ontario Power Generation (OPG) is performing a Periodic Safety Review (PSR) to support the possibility of continued operation of Pickering Nuclear Generating Station (NGS) beyond 2020.[1] A comprehensive Integrated Safety Review (ISR) was completed for Pickering Units 5 through 8 in 2009 in support of refurbishment and continued operation. Pickering Units 1,4 integrated safety assessments were also performed for Pickering A Return to Service (PARTS) in support of approval to restart Units 1 and 4. In addition to these Pickering-specific studies, the 2013 Darlington ISR performed extensive code and standard reviews that were updated in relation to the versions that were assessed in the 2009 Pickering B ISR.[2] These previous ISRs are considered to constitute the first PSR completed for Pickering (referred to as "PSR1"). The current PSR (referred to as "PSR2") is a subsequent PSR building on the basis of earlier OPG integrated safety assessments through review of the various studies, assessments and licence renewals performed since PSR1. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

PSR2 will support and complement the licence renewal application for Pickering NGS going forward. Fifteen Safety Factors will be assessed as part of the PSR. The purpose of Safety Factor reviews is to confirm that the design, condition and operation of Pickering Units 1,4 and 5-8 (as well as common systems) will support continued safe operation for the period of PSR2 by:

- Assessing compliance against "Review Tasks" identified in the PSR2 Basis Document [1], which were derived from Canadian Nuclear Safety Commission (CNSC) REGDOC-2.3.3 [2] and International Atomic Energy Agency (IAEA) SSG-25 [3];

- Documenting assessments against applicable modern Laws, Regulations, Codes and Standards (L/R/C/Ss) (as defined in Reference [1]); and

- Reviewing the effectiveness of applicable OPG Nuclear Programs that support the above assessments, through review of audit and self-assessment results.

The process to identify the modern L/R/C/Ss that are applicable to the PSR2 Assessment Basis involved first creating a broad list from multiple sources (potential candidate L/R/C/Ss) and then filtering it to identify those that are most significant, and

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1 Currently, Pickering Units 5-8 are approved to operate to 247,000 Effective Full Power Hours. This operation limit is expected to be reached on some units in 2020. For the purposes of PSR2, OPG assumes operation of Pickering NGS for up to eight additional years, from 2020 until 2028. OPG will make a decision regarding the permanent shut down dates for the six reactors following the performance of a technical evaluation that will include PSR2, and will communicate it to the CNSC as required by the current Power Reactor Operating Licence (PROL).

2 Much of the compliance assessment and evaluation of Safety Factor health for the Darlington ISR is based on programs and practices that apply across OPG's nuclear operations. As a result, where Pickering is confirmed to follow the same nuclear programs and practices as were assessed for Darlington, the Darlington ISR programmatic conclusions are applicable to Pickering.

that are applicable to the PSR2 scope. The identification and selection criteria are detailed in the PSR2 Basis Document [1]. The result of the identification and selection process was a set of modern Laws, Regulations, Codes and Standards that became part of the "PSR2 Assessment Basis". The PSR2 Basis Document also identifies the modern version and date of the L/R/C/S and the type of review that will be completed in PSR2. The types of review are explained in Section 2.0 below.

This report provides the reviews of L/R/C/Ss with content applicable to Safety Factors 8 (Safety Performance), 10 (Organization, Management System and Safety Culture), 12 (Human Factors), 13 (Emergency Planning), and 14 (Radiological Impact on the Environment). As noted in Section 2.0, reviews of several L/R/C/Ss applicable to Safety Factors 8, 10 and 14 were provided in Reference [4] and findings from these reviews are not duplicated in this report. There is also some overlap with other Safety Factors for a number of L/R/C/Ss considered, as outlined in Table 1 in Section 2.0 of this report.

As outlined in IAEA SSG-25 [3], the objectives of these Safety Factor reviews are as follows:

- The objective of the review of Safety Factor 8 is to determine whether the plant's safety performance indicators and records of operating experience, including the evaluation of root causes of plant events, indicate any need for safety improvements.

- The objective of the review of Safety Factor 10 is to determine whether the organization, management system, and safety culture are adequate and effective for ensuring the safe operation of the plant.

- The objective of the review of Safety Factor 12 is to evaluate the various human factors that may affect the safe operation of the nuclear power plant and to seek to identify improvements that are reasonable and practicable.

- The objective of the review of Safety Factor 13 is to determine: (a) whether the operating organization has in place adequate plans, staff, facilities and equipment for dealing with emergencies; and (b) whether the operating organization's arrangements have been adequately coordinated with the arrangements of local and national authorities and are regularly exercised.

- The objective of the review of Safety Factor 14 is to determine whether the operating organization has an adequate and effective programme for monitoring the radiological impact of the plant on the environment, which ensures that emissions are properly controlled and are as low as reasonably achievable.

2.0 REVIEW SCOPE AND METHODOLOGY

PSR2 is focused on the extension of Pickering NGS operations beyond 2020. Thus, it is important that the methodology for PSR2 be focused on addressing aspects of the review that are likely to have material impact in terms of identifying enhancements that will be reasonable and practicable to implement during the remaining commercial life of the plant. PSR2 conducts reviews against a baseline of the PSR1 work. It is important to note that OPG conducts regular reviews of new and revised Codes and Standards, so a large amount of information is already available to assist in the Safety Factor reviews. In OPG letter N-CORR-00531-05661, W.M. Elliott to P.A Webster and M. Santini, "Design Codes and Standards Effective Dates for OPG Nuclear Fleet" [5], OPG stated:

> *OPG commits to completing a code-over-code review (i.e., review of changes) of subsequent editions, addendum and/or updates of the Codes and Standards listed in Attachment 1* [of the referenced document]*. Key emerging issues due to major changes in the codes will be addressed immediately, or as agreed with the CNSC on a case-by-case basis. Otherwise, OPG will confirm in a letter to the CNSC that these reviews have been completed and there are no significant technical issues...*

As a result, many of the updated codes and standards issued since PSR1 have already had gap assessments performed, to varying degrees of detail, which are utilized and cited in the present Pickering PSR2.

As a subsequent PSR, PSR2 focuses on changes in requirements, plant conditions, operating experience and new information. Since PSR2 is an update of previous ISRs, it incorporates reviews of L/R/C/Ss that have occurred as new versions have been issued. Therefore, clause-by-clause reviews of the majority of applicable L/R/C/Ss have already been completed and there is little value in repeating that process. If clause-by-clause reviews were to be undertaken in PSR2, a major portion of the review effort would be consumed by repackaging existing information that remains largely applicable and, therefore, is not contributing to the identification of new insights and enhancements. A more constructive approach is therefore applied that maximizes the value and usefulness of the work by focusing attention where it is most beneficial, i.e., on identifying new issues. The primary objective for this work, which is to identify safety significant enhancements that may be implemented during the limited remaining life of the station, is achieved using this process and is expected to result in the same (safety significant) Global Issues being identified as would result from a clause-by-clause assessment.

Since this assessment is a subsequent PSR, the focus is on identifying differences between what was previously assessed and what is now different within the current Pickering PSR2 Assessment Basis. In general, these differences relate to:

- More recent (new or revised) L/R/C/S versions than what was previously assessed;[3]

- Safety significant differences between Pickering and Darlington, if the Darlington ISR is the basis for the earlier assessment;

- Implications of extending Pickering NGS operation beyond 2020; and

- Safety significant differences between Pickering Units 1,4 and Units 5-8.

In most cases L/R/C/S reviews are incremental in nature and performed by topic or subject matter for revised requirements. The rationale for this is that new or updated requirements that need to be included in PSR2 are predominantly replacements for other L/R/C/S that were previously assessed, and specify requirements that can be readily mapped to existing OPG programs.

To align with the goals of a subsequent PSR, the following three tiers of reviews are applied for PSR2:

- Clause-by-Clause review: New L/R/C/S referenced in Pickering PROL 48.02/2018 (listed in Appendix C of the Licence Conditions Handbook) will be subjected to a clause-by-clause type review. In a clause-by-clause review, conformance with individual clauses is demonstrated by supporting evidence stating whether the requirements stipulated in the requirement document are met;

- High Level review: New L/R/C/S not referenced in Pickering PROL 48.02/2018 but which are in the PSR2 Assessment Basis will be subject to a high level review. In a high level review, the degree of conformance with clauses or groups of clauses in the L/R/C/S is demonstrated by supporting evidence stating whether the intent of the requirements stipulated in the requirement document are met; and

- Incremental review: For L/R/C/Ss that have been reviewed in PSR1 but have had revisions since the last review, a topical review will be performed of the changes.

Most of the L/R/C/Ss in the PSR2 Assessment Basis receive incremental reviews since PSR2 is an update of previous PSR1 assessments and clause-by-clause or high level reviews for the majority of the L/R/C/Ss in the PSR2 Assessment Basis have already been completed. Implementation plans (including gap analyses or code-over-code reviews) also exist for the latest editions of many L/R/C/Ss. As a result, an

[3] "New" refers to a code or standard that was not previously considered in the context of earlier assessments. "Revised" refers to an updated version of a code or standard that was previously considered in the context of earlier assessments. Where a document has a new number/type, but addresses the same topic from the same organization, it is a "revised", not "new", document (e.g., if a REGDOC replaces a CNSC G or RD document).

incremental review is also used in circumstances where a L/R/C/S in the PSR2 Assessment Basis was not assessed in previous PSR1 reviews but an implementation plan currently exists for compliance.

The PSR2 incremental reviews in this report include an assessment of the intent of recent changes to the L/R/C/Ss identified in Table 1 on a topic or subject-matter basis where there is potential to impact nuclear safety. Incremental reviews provide:

- A summary of the purpose of the L/R/C/S;
- Pertinent background information about the current revision of the L/R/C/S that is being considered;
- Identification of which Safety Factor(s) are applicable to the current revision of the L/R/C/S;
- A description of which version(s) of the L/R/C/S were assessed for PSR1 (i.e., Darlington ISR (where applicable), Pickering B ISR and PARTS code reviews);
- Identification of whether the current version of the L/R/C/S is an update of a previous version of the L/R/C/S that was assessed in PSR1 (and if so, a description of the major changes in the latest revision is provided as discussed below);
- An assessment of the applicability of PSR1 assessment findings (gaps and conclusions), including the implications of extending Pickering NGS operation beyond 2020 if any;
- An assessment of the applicability of assessment findings that address more recent (post-PSR1) editions of the L/R/C/S, including any implementation or transition plans that are already committed to by OPG; and
- Where PSR1 and post-PSR1 assessments are not sufficient to address changes in the latest edition of the L/R/C/S, an assessment of the changes from the previously assessed edition of the L/R/C/S (including identification of any safety significant PSR2 gaps which result).

High Level reviews provide the same information as above, where applicable, in a similar format. However, given that High Level L/R/C/Ss generally have not received past assessment during PSR1, the Incremental review content is augmented by a high level, section-by-section assessment of the degree of conformance of Pickering NGS with the L/R/C/S (demonstrating, with supporting evidence, whether the intent of the requirements stipulated in the document are met).

There are currently no L/R/C/S clause-by-clause reviews identified in the PSR2 Assessment Basis.

The Safety Factor 8, 10, 12, 13 and 14 L/R/C/S reviews identify Compliances and Gaps as defined below:

- Compliance:
    - Where an Incremental review has been performed, Compliance indicates that the change in the safety requirement, per the topical review, is met.
    - Where a High Level review has been performed, Compliance indicates that the intent of the safety requirement is met.
- Gap:
    - Where an Incremental review has been performed, a Gap indicates that the change in the safety requirement, per the topical review, is not met.
    - Where a High Level review has been performed, a Gap indicates that the intent of the safety requirement is not met.

The reviews assume that use of the word:

- "Shall" is used in an L/R/C/S to express a requirement, i.e., a provision that the licensee is obliged to satisfy in order to comply with the standard;
- "Should" is used to express a recommendation or that which is advised but not required;
- "May" is used to express an option or that which is permissible within the limits of the standard; and
- "Can" is used to express possibility or capability.

Table 1 identifies the L/R/C/Ss in the PSR2 Assessment Basis that are applicable to Safety Factors 8, 10, 12, 13, and 14, with several exceptions discussed below. Table 1 also identifies the modern version and date of each L/R/C/S to be considered, the Safety Factor(s) to which each document is applicable, and the type of review that will be completed in PSR2. Reviews for each L/R/C/S are provided in Appendix B, and results are summarized in Section 3.0.

Several L/R/C/Ss applicable to Safety Factors 8, 10 and 14 are excluded from Table 1 below. Specifically:

- CNSC G-129, "Keeping Radiation Exposures and Doses 'As Low As Reasonably Achievable (ALARA)'", and SOR/2000-203, "Radiation Protection Regulations", which are applicable to Safety Factor 8.

- CNSC G-228, “Developing and Using Action Levels”, which is applicable to Safety Factors 8 and 14.

- CSA N286-12, “Management System Requirements for Nuclear Facilities” and SOR/2000-202, “General Nuclear Safety and Control Regulations”, which are applicable to Safety Factor 10.

The reviews for the above L/R/C/Ss are provided in Reference [4] and findings are not duplicated in this report.

Table 1: Applicable L/R/C/Ss for Pickering PSR2 Safety Factors 8, 10, 12, 13, and 14

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
L/R/C/Ss Referenced in Pickering NGS PROL 48.02/2018						
1	CSA N290.15	Requirements for the Safe Operating Envelope of Nuclear Power Plants	N290.15-10	8	Incremental	N290.15 not addressed as part of Pickering B or Darlington ISRs, but gap analysis has been performed against OPG Governance and N290.15.
2	CSA N286.7	Quality Assurance Of Analytical, Scientific And Design Computer Programs	N286.7-16	1, 5, 6, 7, 10	Incremental	N286.7 addressed as part of Pickering B and Darlington ISRs.
3	CSA N288.1	Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities	N288.1-14	8, 14	Incremental	N288.1 addressed as part of Pickering B and Darlington ISRs.
4	CSA N288.4	Environmental Monitoring Program at Class I Nuclear Facilities and Uranium Mines and Mills	N288.4-10	8, 14	Incremental	N288.4 addressed as part of Pickering B and Darlington ISRs.
5	CSA N293 [4]	Fire Protection for Nuclear Power Plants	N293-12	1, 7, 13	Incremental	N293 addressed as part of Pickering B and Darlington ISRs, as well as PARTS code reviews.

[4] The PSR2 review of CSA N293-12 is in progress. Gaps identified from this review will be applicable to the Plant Design Safety Factor, and hence, results are presented in the Pickering NGS PSR2 Plant Design Safety Factor Report.

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
6	CNSC RD-204	Certification of Persons Working at Nuclear Power Plants	2008	10	Incremental	RD-204 addressed as part of Darlington ISR.
7	CNSC REGDOC-3.1.1	Reporting Requirements for Nuclear Power Plants	2014	10	Incremental	CNSC S-99 (precursor to REGDOC-3.1.1) addressed as part of Pickering B and Darlington ISRs.
8	CNSC REGDOC-2.9.1*	Environmental Protection Policies, Programs and Procedures	2013	8, 14	Incremental	REGDOC-2.9.1 addressed as part of Darlington ISR. S-296 also addressed as part of Pickering B and Darlington ISRs.
9	CNSC REGDOC-2.10.1*	Nuclear Emergency Preparedness and Response	2014	13	Incremental	Transition plan in place and gap assessment has been performed by OPG.
Additional L/R/C/Ss						
10	CNSC G-323	Ensuring the Presence of Sufficiently Qualified Staff at Class I Nuclear Facilities-Minimum Shift Complement	2007	10, 12	Incremental	G-323 addressed as part of Darlington ISR.
11	CNSC G-278	Human Factors Verification and Validation Plans	2003	1, 12	Incremental	G-278 addressed as part of Pickering B and Darlington ISRs.
12	CNSC G-276	Human Factors Engineering Program Plans	2003	1, 12	Incremental	G-276 addressed as part of Pickering B and Darlington ISRs.
13	S.C. 1997, C.9	Nuclear Safety and Control Act (NSCA) and its associated Regulations	Amended in February 2015	10	Incremental	S.C. 1997, C.9 addressed as part of Darlington ISR.
14	CSA N1600	General Requirements for Nuclear Emergency Management Programs	N1600-14	13	High Level	Not referenced in PROL 48.02/2018. Not reviewed as part of Pickering B or Darlington ISRs.

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
15	CSA N288.6	Environment Risk Assessments at Class I Nuclear Facilities and Uranium Mines and Mills	N288.6-12	8, 14	Incremental [5]	N288.6 not addressed as part of Pickering B or Darlington ISRs. Implementation Plan and clause-by-clause review have been prepared for Pickering Environmental Monitoring Program compliance with N288.6.
16	CSA N288.5	Effluent Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills	N288.5-11	8, 14	Incremental [5]	N288.5 not addressed as part of Pickering B or Darlington ISRs. OPG has performed a gap analysis and completed all actions in the implementation plan to satisfy mandatory requirements of N288.5.
17	CNSC REGDOC-2.2.2	Personnel Training	2014	10	Incremental	REGDOC-2.2.2 not addressed as part of Pickering B or Darlington ISRs. Transition Plan and gap analysis has been prepared for REGDOC-2.2.2.
18	CNSC REGDOC-2.3.2	Accident Management, Version 2	2015	1, 5, 6, 7, 8, 10	Incremental	REGDOC-2.3.2 addressed as part of Darlington ISR.
19	CNSC REGDOC-2.3.3	Periodic Safety Reviews	2015	8	High Level	REGDOC-2.3.3 not addressed as part of Pickering B or Darlington ISRs. New PSR methodology.

[5] Per Section 3.2.2 of the R02 PSR2 Basis Document [1]: "Table D1 identifies the review type to be applied to each of the Laws, Regulations, Codes and Standards in the PSR2 Assessment Basis. Following further assessment of past work, the review type of a listed modern Law, Regulation, Code or Standard may be changed from Clause-by-Clause or High Level to Incremental." Past assessments of CSA N288.3.4, N288.5 and N288.6 were reviewed and implementation plans with gap assessments were identified. As a result, the Review Type for these three L/R/C/Ss was changed from High Level to Incremental since "… implementation plans exist for many of the codes and standards not addressed in PSR1 and therefore an incremental review will be applied to these documents" [1].

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
20	CSA N286.7.1	Guideline for the Application of N286.7-99, Quality Assurance of Analytical, Scientific, and Design Computer Programs for Nuclear Power Plants	N286.7.1-09	1, 5, 6, 7, 10	N/A [6]	N286.7.1 not addressed as part of Pickering B or Darlington ISRs.
21	CSA N290.12	Human Factors in Design for Nuclear Power Plants	N290.12-14	1, 12	Incremental [7]	N290.12 not addressed as part of Pickering B or Darlington ISRs. OPG has completed a gap analysis against mandatory requirements of N290.12.
22	CSA N288.3.4	Performance Testing of Nuclear Air-Cleaning Systems at Nuclear Facilities	N288.3.4-13	8, 14	Incremental [5]	N288.3.4 addressed as part of Darlington ISR, but not addressed as part of Pickering B ISR. OPG has completed a gap analysis and is developing an implementation plan to satisfy mandatory requirements of N288.3.4.
23	CSA N288.7	Groundwater Protection Programs at Class I Nuclear Facilities and Uranium Mines and Mills	N288.7-15	14	High Level	First edition of N288.7 issued in 2015. Not addressed as part of Pickering B or Darlington ISRs. OPG is developing a gap analysis and implementation plan to satisfy mandatory requirements of N288.7.

* Superseding documents to those currently in Pickering NGS PROL 48.02/2018.

[6] The N286.7.1 guide has been amalgamated into the new (-16) edition of the N286.7 Standard. The N286.7 CSA Impact Statement states [6]: "The CSA N286.7.1 guide will no longer be maintained after this new edition of N286.7 is issued. Any relevant guidance has been put into the new edition of N286.7." As a result, only the review of N286.7-16 has been prepared for PSR2.

[7] Per CNSC's request in P-CORR-03680-0607223 R000, "Pickering PSR2 – Change to Review Type for CSA N290.12" [7], the Review Type for CSA N290.12-14 was changed from High Level to Incremental.

3.0 RESULTS AND CONCLUSIONS

The results of the PSR2 reviews of the L/R/C/Ss listed in Table 1 are summarized in Table 2 below. Additional background information and details regarding the gaps listed in Table 2 are provided in Appendix B of this report.

Table 2: PSR2 L/R/C/S Review Results for Safety Factors 8, 10, 12, 13 and 14

Appendix Section #	L/R/C/S Reviewed	L/R/C/S Review Results
B.1	CSA N290.15-10, "Requirements for the Safe Operating Envelope of Nuclear Power Plants"	There are no PSR2 gaps for CSA N290.15-10 (R2015). Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CSA N290.15-10 (R2015).
B.2	CSA N286.7-16, "Quality Assurance Of Analytical, Scientific And Design Computer Programs"	There are no PSR2 gaps for CSA N286.7-16. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CSA N286.7-16.
B.3	CSA N288.1-14, "Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities"	There are no PSR2 gaps for CSA N288.1-14. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CSA N288.1-14.
B.4	CSA N288.4-10, "Environmental Monitoring Program at Class I Nuclear Facilities and Uranium Mines and Mills"	There are no PSR2 gaps for CSA N288.4-10. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CSA N288.4-10.
B.5	CSA N293-12, "Fire Protection for Nuclear Power Plants"	The PSR2 review of CSA N293-12 is in progress. Gaps identified from this review will be applicable to the Plant Design Safety Factor, and hence, results are presented in the Pickering NGS PSR2 Plant Design Safety Factor Report.
B.6	CNSC RD-204 (2008), "Certification of Persons Working at Nuclear Power Plants"	There are no PSR2 gaps for CNSC RD-204 (2008). Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CNSC RD-204 (2008).
B.7	CNSC REGDOC-3.1.1 (2014), "Reporting Requirements for Nuclear Power Plants"	There are no PSR2 gaps for CNSC REGDOC-3.1.1 (2014). Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CNSC REGDOC-3.1.1 (2014).
B.8	CNSC REGDOC-2.9.1 (2013), "Environmental Protection Policies, Programs and Procedures"	There are no PSR2 gaps for CNSC REGDOC-2.9.1 (2013). Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CNSC REGDOC-2.9.1 (2013).

Appendix Section #	L/R/C/S Reviewed	L/R/C/S Review Results
B.9	CNSC REGDOC-2.10.1 (2014), "Nuclear Emergency Preparedness and Response"	There is one PSR2 REGDOC-2.10.1 (2014) gap which relates to Safety Factor 13 (Emergency Planning): 1. OPG has completed a gap analysis for transition to REGDOC-2.10.1 and has developed an action plan to achieve compliance. The transition plan that OPG has committed in order to bring Darlington into compliance with REGDOC-2.10.1 applies across the nuclear fleet and will also bring Pickering into compliance. Updating OPG governance to ensure that the Pickering Evacuation Time Estimate study is maintained and to define how the Potassium Iodide (KI) pill program will be sustained is in progress. As these two actions are not yet complete, this is identified as a PSR2 gap.
B.10	CNSC G-323 (2007), "Ensuring the Presence of Sufficiently Qualified Staff at Class I Nuclear Facilities-Minimum Shift Complement"	There are no PSR2 gaps for CNSC G-323 (2007). Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CNSC G-323 (2007).
B.11	CNSC G-278 (2003), "Human Factors Verification and Validation Plans"	There are no PSR2 gaps for CNSC G-278 (2003). Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CNSC G-278 (2003).
B.12	CNSC G-276 (2003), "Human Factors Engineering Program Plans"	There are no PSR2 gaps for CNSC G-276 (2003). Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CNSC G-276 (2003).
B.13	S.C. 1997, C.9 (Amended in February 2015), "Nuclear Safety and Control Act"	There are no PSR2 gaps for the Nuclear Safety and Control Act (Amended 2015). Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with the NSCA (Amended 2015).
B.14	CSA N1600-14, "General Requirements for Nuclear Emergency Management Programs"	There are no PSR2 gaps for CSA N1600-14. Per the definition of Compliance for a High Level review, Pickering has a PSR2 Compliance associated with CSA N1600-14.
B.15	CSA N288.6-12, "Environment Risk Assessments at Class I Nuclear Facilities and Uranium Mines and Mills"	There are no PSR2 gaps for CSA N288.6-12. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CSA N288.6-12.

Appendix Section #	L/R/C/S Reviewed	L/R/C/S Review Results
B.16	CSA N288.5-11, "Effluent Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills"	There are no PSR2 gaps for CSA N288.5-11. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CSA N288.5-11.
B.17	CNSC REGDOC-2.2.2 (2014), "Personnel Training"	There are no PSR2 gaps for CNSC REGDOC-2.2.2 (2014). Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CNSC REGDOC-2.2.2 (2014).
B.18	CNSC REGDOC-2.3.2 (2015), "Accident Management, Version 2"	There is one PSR2 CNSC REGDOC-2.3.2 (2015) gap which relates to Safety Factor 1 (Plant Design): 1. Full provision of Complementary Design Features for containment integrity as required by Clause 4.2.1 of REGDOC-2.3.2 will be addressed with the completion of Phase 2 Emergency Mitigating Equipment. This work is currently scheduled to be fully implemented by the end of 2017. Since this work is still in progress, it is identified as a PSR2 gap.
B.19	CNSC REGDOC-2.3.3 (2015), "Periodic Safety Reviews"	There are no PSR2 gaps for CNSC REGDOC-2.3.3 (2015). Per the definition of Compliance for a High Level review, Pickering has a PSR2 Compliance associated with CNSC REGDOC-2.3.3 (2015).
B.20	CSA N286.7.1-09, "Guideline for the Application of N286.7-99, Quality Assurance of Analytical, Scientific, and Design Computer Programs for Nuclear Power Plants"	The N286.7.1 guide has been amalgamated into the new (-16) edition of the N286.7 Standard. The N286.7 CSA Impact Statement states [6]: "The CSA N286.7.1 guide will no longer be maintained after this new edition of N286.7 is issued. Any relevant guidance has been put into the new edition of N286.7." As a result, only the review of N286.7-16 has been prepared for PSR2.
B.21	CSA N290.12-14, "Human Factors in Design for Nuclear Power Plants"	There are no PSR2 gaps for CSA N290.12-14. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CSA N290.12-14.
B.22	CSA N288.3.4-13, "Performance Testing of Nuclear Air-Cleaning Systems at Nuclear Facilities"	There are no PSR2 gaps for CSA N288.3.4-13. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CSA N288.3.4-13.
B.23	CSA N288.7-15, "Groundwater Protection Programs at Class I Nuclear Facilities and Uranium Mines and Mills"	There are no PSR2 gaps for CSA N288.7-15. Per the definition of Compliance for a High Level review, Pickering has a PSR2 Compliance associated with CSA N288.7-15.

4.0 REFERENCES

[1] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[2] CNSC REGDOC-2.3.3, *Periodic Safety Reviews*, April 2015.

[3] IAEA Safety Guide No. SSG-25, *Periodic Safety Review for Nuclear Power Plants*, 2013.

[4] OPG Report, P-REP-03680-0586480 R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 9, 11, and 15*, September 2016.

[5] OPG Correspondence, N-CORR-00531-05661, W. M. Elliott to P. A. Webster and M. Santini, *Design Codes and Standards Effective Dates for OPG Nuclear Fleet*, April 30, 2012.

[6] CSA Impact Statement, *Notification of CSA N286.7 on Quality Assurance of Analytical Scientific, and Design Computer Programs; Product: New Edition; Product Designation: CSA N286.7; Previous Edition Published: 1999, Reaffirmed 2007 and 2012*, Date not provided.

[7] OPG Correspondence, P-CORR-03680-0607223, E. Marczak to M. Ruffulo, *Pickering PSR2 – Change to Review Type for CSA N290.12*, July 25, 2016.

Appendix A: Nomenclature

AECB	Atomic Energy Control Board
AHJ	Authority Having Jurisdiction
AI	Action Item
AR	Action Request
AIM	Abnormal Incidents Manual
AM	Accident Management
ANO	Authorized Nuclear Operator
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
BDBA	Beyond Design Basis Accident
CANDU	CANadian Deuterium Uranium
CDF	Complementary Design Feature
CMCC	Crisis Management and Communications Centre
CNEP	Consolidated Nuclear Emergency Plan
CNSC	Canadian Nuclear Safety Commission
COG	CANDU Owners Group
COP	Continued Operations Plan
COTS	Commercial Off the Shelf
CPSM	Critical Safety Parameter Monitoring
CRSS	Control Room Shift Supervisor
CSA	Canadian Standards Association
DBA	Design Basis Accident
DECs	Design Extension Condition
DRL	Derived Release Limit
ECC	Engineering Change Control
EFADS	Emergency Filtered Air Discharge System
EITER	Equipment Important to Emergency Response
EM	Emergency Management
EME	Emergency Mitigating Equipment
EMP	Environmental Monitoring Program
EMS	Environmental Management System

EP	Emergency Preparedness
ERA	Environmental Risk Assessment
ERO	Emergency Response Organization
FADS	Filtered Air Discharge System
FAI	Fukushima Action Item
GAR	Global Assessment Report
GWMP	Groundwater Monitoring Program
GWPP	Groundwater Protection Program
HEPA	High Efficiency Particulate Air
HF	Human Factors
HFE	Human Factors Engineering
HFEPP	Human Factors Engineering Program Plan
HFESR	Human Factors Engineering Summary Report
IAEA	International Atomic Energy Agency
I&E	Instrumentation and Equipment
IFB	Irradiated Fuel Bay
IIP	Integrated Implementation Plan
ISO	International Organization for Standardization
ISR	Integrated Safety Review
ITR	Individual Training Record
KI	Potassium Iodide
L/R/C/Ss	Laws, Regulations, Codes and Standards
MAA	Mutual Aid Agreement
NEMP	Nuclear Emergency Management Program
NGS	Nuclear Generating Station
NPP	Nuclear Power Plant
NS	New Standard
NSCA	Nuclear Safety and Control Act
OPEX	Operating Experience
OPG	Ontario Power Generation
OPGN	Ontario Power Generation Nuclear
OSR	Operational Safety Requirements
PAR	Passive Autocatalytic Recombiner

PARMS	Post-Accident Radiological Monitoring System
PARTS	Pickering A Return to Service
P&G	Principles and Guidelines
PN	Pickering Nuclear
PPE	Personal Protective Equipment
PROL	Power Reactor Operating Licence
PSA	Probabilistic Safety Assessment
PSR	Periodic Safety Review
PSR1	Periodic Safety Review 1 (Earlier OPG PSR work and other associated assessments)
PSR2	Periodic Safety Review 2 (Subsequent PSR per CNSC REGDOC-2.3.3)
QA	Quality Assurance
QC	Quality Control
REMP	Radiological Environmental Monitoring Programs
RPO	Recovery Project Organization
SA	Severe Accident
SAT	Systematic Approach to Training
SAMG	Severe Accident Management Guidelines
SCA	Safety and Control Area
SCR	Station Condition Record
SOE	Safe Operating Envelope
SOP	Sustainable Operations Plan
SSC	Structures, Systems and Components
TIMS	Training Information Management System
TQD	Training and Qualification Description

Appendix B: L/R/C/S Reviews for Safety Factors 8, 10, 12, 13 and 14

B.1 CSA N290.15-10, "Requirements for the Safe Operating Envelope of Nuclear Power Plants"

B.1.1 Background

The following, paraphrased from CSA N290.15-10 (R2015) [B.1-1], provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *The licensing of a nuclear power plant requires a safety evaluation to demonstrate its safe operation. In the safety evaluation, the limits and conditions associated with the safety requirements form the safe operating envelope (SOE) and the operator is responsible to demonstrate compliance.*
>
> *CSA N290.15 provides the requirements for the definition, implementation, and maintenance of the safe operating envelope at nuclear power plants.*

All of CSA N290.15-10 (R2015) is directly relevant to Safety Factor 8 (Safety Performance).

Compliance with CSA N290.15-10 (R2015) is currently a licence requirement for Pickering NGS (per PROL 48.02/2018) as indicated in Appendix C.1 of the R04 Pickering Licence Conditions Handbook [B.1-2]. The current version of the Standard, N290.15-10 (R2015) is a reaffirmation of the initial standard introduced in 2010 without any changes.

CSA N290.15-10 is the first edition of this standard. The related impact statement and publication note [B.1-3] identifies the following significant features of the standard:

1. *The SOE, as expressed in terms of the limits and conditions which govern plant operation in compliance with the safety analysis are clearly, completely, and consistently defined, and fully reflected in the documentation that governs plant operation;*

2. *The SOE, and the basis for its derivation, are contained in a set of documentation that can be readily referenced by users requiring an understanding of the basis for safe plant operation;*

3. *A compliance framework has been established that avoids plant operation outside of the SOE, ensures timely detection of plant operation outside of the SOE, and specifies appropriate and timely corrective actions to restore plant operation to within the SOE; and*

4. *The SOE is maintained up to date within the context of other processes.*

The results of PSR1 CSA N290.15 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.1.2. As identified in Reference [B.1-4], the Pickering PSR2 review of CSA N290.15-10 is an Incremental review.

PSR2 Incremental review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.
- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.1.2 Compliance Assessment for Pickering PSR2

B.1.2.1 Application of PSR1 Reviews

Pickering and Darlington NGS

The first edition of this standard was issued in August 2010 after the completion of the code reviews for the Pickering B ISR and Pickering A Return to Service. Further, this standard was not identified as requiring consideration as part of the Darlington ISR. Therefore, no code review was performed for Pickering or Darlington NGS against CSA N290.15. However, the content of CSA N290.15 was extracted, with only minor changes, from the SOE Principles & Guidelines (P&G) document that the Canadian nuclear industry had prepared previously through the CANDU Owners Group (COG) [B.1-5]. The COG SOE P&G document basically described the SOE program that OPG had developed and was in the process of implementing at the time, as part of an Integrated Improvement Project initiated in the late 1990's. As such, when CSA N290.15-10 was issued for industry use, the expectation was that OPG would already be in compliance with its requirements.

The status of implementation of N290.15-10 is discussed further in Section B.1.2.2 below.

B.1.2.2 Application of Post-PSR1 Reviews

Subsequent to the issuance of CSA N290.15-10, the CNSC conducted a pilot Type 1 inspection of the OPG SOE program at Pickering Units 1,4 in 2011. As stated in Reference [B.1-6], the purpose of this inspection was to assess the implementation of the Safe Operating Envelope as defined by CSA standard N290.15-10. In preparation for this inspection, OPG performed a detailed review and assessment of all the SOE governing documents to ensure that the SOE documents meet the CSA N290.15 requirements. As summarized in Reference [B.1-7]:

> *The results show that the majority of the documented instructions clearly present the definition, implementation and maintenance of the SOE; consistent with the CSA N290.15 Standard. However, three potential gaps were identified during the process:*
>
> *(1) Criterion 9 of the CSA N290.15 states that the SOE shall be supported by a compliance framework that ensures training for affected SOE users and developers. However, authorized staff training is not considered to be part of the SOE as per N-ST-08131.02-10000.*

(2) As per N-INS-08131.02-10009, any major discrepancies between SOE and existing station-licensing documentation are dealt with by the discrepancy management process via the SOE Discrepancy Database and the SOE Implementation Gap Matrix. NSATD should establish and maintain both the database and the gap matrix to ensure all SOE discrepancies identified by SOE project staff are documented. However, neither of the database or the gap matrix has been established by NSATD.

(3) The SOE Compliance Table provides references to station operating documentation (SOD) for Required Actions and Action Times to ensure timely detection of instances of operation outside of the defined SOE. Pickering A has instructions on preparation of the OSR and IUC, but has no guidelines on preparing the Compliance Tables. This is a gap identified as non-compliance to CSA N290.15, Criterion 9. In the interim we have made reference to the Pickering B instruction, "NK30-INS-08131-00001".

The results of the CNSC Type I Inspection [B.1-8] were generally favourable, though the CNSC did identify five recommendations for consideration by OPG, in order to comply with CSA N290.15-10, before it was introduced into the Pickering A licence:

Recommendation 2011-SOE-PA-R-01: *CNSC recommends that OPG update its governing documents for SOE to ensure compliance with CSA N290.15-10.*

Recommendation 2011-SOE-PA-R-02: *CNSC recommends that OPG complete development of the SOE. Also, OPG should continue adequate maintenance of the SOE documentation through the existing processes.*

Recommendation 2011-SOE-PA-R-03: *CNSC staff recommends that OPG strengthen the adherence to the procedures and processes in place to develop and maintain adequately the SOE. CNSC staff also recommends that OPG staff be made aware of the importance of the SOE while the limits and conditions are being developed and maintained. Finally, CNSC staff recommends that OPG perform a complete review of the SOE limits and conditions while performing the next revision of the SOE documents.*

Recommendation 2011-SOE-PA-R-04: *CNSC staff recommends that OPG assess the completeness of the SOE limits and conditions for all systems. CNSC staff also recommends that OPG ensure that the SOE limits and conditions are stated in a manner that is useful by the operator. Finally, CNSC staff recommends that OPG clarify the interfaces between the systems to improve the horizontal linkage. This task should be performed while doing the next revision of the documentation.*

Recommendation 2011-SOE-PA-R-05: *CNSC recommends that OPG develop adequate training sessions for the affected users and developers of the SOE.*

As part of the process to complete implementation of the SOE program at Pickering and Darlington NGS, an OPG Nuclear standard was created and issued for implementation in 2012 [B.1-9]. This standard states the objectives of the OPG program and, in particular, states that these objectives will be achieved by meeting the intent of all of the SOE requirements specified in Section 4.0 of CSA N290.15-10. The OPG standard goes on to describe the OPG SOE program in detail and the manner in which it will be managed. The implementation date for all aspects of the standard was established as the end of 2012, with the exception of training for SOE users and developers, for which implementation was expected by the end of 2013.

In OPG's application for renewal of the Pickering NGS PROL in 2012 [B.1-10], issuance of the OPG Nuclear Standard for SOE was cited and it was indicated that completion of the SOE documentation for all SOE systems was expected by the end of 2012 for both Pickering Units 1,4 and Pickering Units 5-8. In addition, a transition plan for demonstrating compliance with CSA N290.15-10 was presented that addressed each of the gaps identified in Reference [B.1-7], all of which are programmatic in nature and applicable to all OPG Nuclear stations. Specifically:

1. *Review of existing SOE classroom training to determine whether an update is necessary (in progress).*

2. *The SOE Discrepancy Management instruction was made obsolete. SOE discrepancies are now managed by Station Condition Records (SCR) and Action Tracking database (complete).*

3. *Nuclear level instruction for preparation of SOE Compliance Tables was issued in Passport (complete).* ([B.1-11])

4. *N-STI-03602-10000, SOE Instrument Uncertainty and Allowable Value Calculations, has been updated to clarify that OSR documents, Compliance Tables and station operating documents need to be updated as required when an IUC has been updated (complete).*

Items 2 and 3 above eliminated the second and third gaps that had been identified in the OPG self-assessment [B.1-7]. With respect to the first gap identified in Reference [B.1-7], the following was committed to the CNSC in Reference [B.1-10]:

> *An action (tracked under CRC 2011-08) has been taken to provide SOE training for all SOE users and developers (i.e., Engineering, Operations and Maintenance) as per requirements in CSA N290.15. The action is expected to be completed by end of 2013.*

In addition it was noted in Reference [B.1-10] that a formal response to the findings of the CNSC Type I Inspection of the SOE implementation at Pickering A, including a corrective action plan, was committed for the end of 2012, with the expectation that all actions would be completed and/or implemented by the end of 2013.

OPG's response to the CNSC's Type I Inspection findings for Pickering A [B.1-12] outlined the actions and documentation updates undertaken to address the CNSC recommendations. The only CNSC recommendation left outstanding at the time was Recommendation 2011-SOE-PA-R-05 on the adequacy of SOE training for OPG staff. This finding relates to the same gap that OPG had identified previously in its self-assessment, and is covered by the corresponding regulatory commitment mentioned above in [B.1-10].

OPG communicated to the CNSC at the end of 2013 [B.1-13] that the one outstanding action to address SOE training requirements (as committed previously in [B.1-10] and [B.1-12]) had been completed and requested that the Pickering Licence Conditions Handbook be updated accordingly.

As part of the "Regulatory Oversight Report for Canadian Nuclear Power Plants" for the year 2014 [B.1-14], the CNSC stated:

> *All licensees are required to establish a safe operating envelope (SOE) program according to the requirements of N290.15-10, Requirements for the Safe Operating Envelope of Nuclear Power Plants. To date, Bruce Power, OPG and NB Power have completed the development and the baseline implementation of their SOEs, and continued to make improvements to their SOE programs. Program compliance assessments are being conducted through CNSC compliance monitoring activities, and CNSC staff were satisfied with the results from monitoring activities in 2014.*

Specific to Pickering, Reference [B.1-14] concluded that:

> *OPG's implementation of the safe operating envelope (SOE) ensured that the Pickering reactors operated in their analyzed states, thereby ensuring adequate safety at all times. The SOE implementation level was satisfactory at Pickering in 2014 and in compliance with N290.15, Requirements for the Safe Operating Envelope of Nuclear Power Plants.*

In Reference [B.1-15], the CNSC provided feedback in relation to industry responses to their earlier comments on COG report COG-02-901 [B.1-16]. This report outlined how the Canadian utilities had applied CSA N290.15-10 for the purpose of identifying which systems were included in the SOE program at each station. The expectation communicated by the CNSC in Reference [B.1-15] was that their remaining issues would be addressed by making changes to CSA N290.15. In 2016, an amendment to N290.15 was issued [B.1-17] which incorporated the clarifications requested by the CNSC. CNSC staff were part of the team that drafted the amendment, and provided concurrence that the changes made adequately met the CNSC's expectations. This latest code amendment was issued past the Freeze Date (January 15, 2016) for PSR2 and is therefore not reviewed in this assessment.

The above demonstrates that OPG and Pickering NGS have been assessed to now be fully compliant with CSA N290.15-10 (R2015) [B.1-1]. This standard is programmatic in nature and the status of OPG's compliance is not impacted by continued operation of Pickering NGS beyond 2020. Hence, there are no PSR2 gaps for CSA N290.15.

B.1.3 Compliance Assessment Summary for Pickering PSR2

There are no PSR2 gaps for CSA N290.15-10 (R2015) [B.1-1]. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CSA N290.15-10 (R2015).

B.1.4 References

[B.1-1] CSA Standard, N290.15-10 (R2015), *Requirements for the Safe Operating Envelope of Nuclear Power Plants*, August 2010.

[B.1-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.1-3] CSA Impact Statement and Publication Notice, *Product: New Standard; Product Designation CSA N290.15; Date of Release: 1st Release at Ballot (Jul-10); 2nd Release at Publication (Aug-10)*, Date not provided.

[B.1-4] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.1-5] OPG Letter, N-CORR-00531-02480, P.R. Charlebois to J.W. Blyth, *COG Working Group on Safe Operating Envelope (SOE)*, March 31, 2003.

[B.1-6] CNSC Letter, e-Doc # 3705941, OPG File No. NA44-CORR-00531-06704, T.E. Schaubel to G. Jager, *Type I Inspection of the Safe Operating Envelope*, April 8, 2010.

[B.1-7] OPG Letter, NA44-CORR-08131-0368201, K.L. Xu and K. Lemkay to C. Lorencez, *Gap Analysis of the OPG Governance and the CSA N290.15 Standard*, January 27, 2011.

[B.1-8] CNSC Letter, e-Doc # 3832814, OPG File No. NA44-CORR-00531-06823, M. Santini to G. Jager, *Pickering NGS A – Pilot Type I Compliance Inspection Report – Safe Operating Envelope, Report #PRPD-PICKAB-2010/2011-T17136-T2-121*, November 18, 2011.

[B.1-9] OPG Standard, N-STD-MP-0016 R002, *Safe Operating Envelope*, 2012.

[B.1-10] OPG Letter, P-CORR-00531-03719, G. Jager to M.A. Leblanc, *Application For Renewal of Pickering Nuclear Generating Station Power Reactor Operating Licence*, July 4, 2012.

[B.1-11] OPG Instruction, N-INS-03602-10001 R001, *Preparation of Safe Operating Envelope Compliance Tables*, February 2015.

[B.1-12] OPG Letter, P-CORR-00531-03745, G. Jager to M. Santini, *OPG Response to CNSC Pilot Type I Inspection of Pickering NGS A Safe Operating Envelope Implementation*, November 21, 2012.

[B.1-13] OPG Letter, P-CORR-00531-04026, B. Phillips to M. Santini, *Status Update on OPG SOE Training Implementation*, December 16, 2013.

[B.1-14] CNSC Letter, e-Doc # word 4778715 / pdf 4778718, OPG File No. P-CORR-00531-04495, L. Levert to B. McGee, *Presentation of the 2014 NPP Report*, June 16, 2015.

[B.1-15] CNSC Letter, e-Doc # 4571906, OPG File No. N-CORR-00531-06730, G. Rzentkowski to W.M. Elliott, *Darlington and Pickering NGS: Review of the Industry's Response to CNSC Staff Comments on the COG SOE Rationalization Report*, November 13, 2014.

[B.1-16] COG Report, COG 12-9031, *Report on the Industry Rationalization of Safe Operating Envelope Systems in Canadian CANDU Power Reactors*, March 2013.

[B.1-17] CSA Standard, N290.15-10 (R2015) including Update No.1, *Requirements for the Safe Operating Envelope of Nuclear Power Plants*, August 2010; Update No. 1: March 2016.

B.2 CSA N286.7-16, "Quality Assurance of Analytical, Scientific and Design Computer Programs"

B.2.1 Background

The following paraphrased from CSA N286.7-16 [B.2-1] provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *The Canadian nuclear industry has recognized the need to establish rigorous and effective requirements for application of quality assurance process to computer programs. This standard identifies the quality assurance requirements to support the management system for high energy reactor facilities where analytical tools are utilized in the life cycle of nuclear facilities.*
>
> *CSA Standard N286.7 specifies the requirements for the quality assurance program applicable to the design, development, maintenance, modification, acquisition, and use of analytical, scientific and design computer programs that are used in high energy reactor applications.*

Compliance with CSA N286.7-99 (R2007) [B.2-2] is currently a licensing requirement for Pickering NGS (per PROL 48.02/2018) as indicated in Appendix C.1 of the R04 Pickering Licence Conditions Handbook [B.2-3].

CSA N286.7-16 is the third edition of this standard, and supersedes previous editions published in 1999 and 1994. The CSA Group Impact Statement relating to the publication of this edition [B.2-4] provides the following summary of the changes from the previous edition (paraphrased):

1. *The scope has been expanded* [8] *and now includes: In-house developed computer programs, Third-party computer programs, Legacy computer programs, Programmed applications.*

2. *The N286.7.1 guide* [B.2-5] *has been amalgamated into the new edition of the N286.7 Standard and is no longer being maintained.*

3. *To provide the user with a clear breakdown of the requirements, they are divided into two parts: (a) Design, development and maintenance of software; and (b) Acquisition, qualification, control and use of software.*

4. *Overlap with CSA N286-12 has been removed. The new edition does not address qualification of staff or procurement requirements already covered by CSA N286-12.*

5. *Requirements were brought up-to-date with consideration of current technology.*

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[8] The scope was expanded to address programmed applications and third-party computer programs. In-house developed computer programs and legacy computer programs were previously addressed in the 1999 edition.

CSA N286.7 is relevant to Safety Factors 1 (Plant Design), 5 (Deterministic Safety Analysis), 6 (Probabilistic Safety Assessment), 7 (Hazard Analysis), and 10 (Organization, the Management System, and Safety Culture). The current version of the code is CSA N286.7-16, which was issued in January, 2016. The previous version of the code, CSA N286.7-99 which was reaffirmed in 2004 and in 2007, was the subject of reviews performed as part of PSR1 (Pickering B and Darlington Integrated Safety Reviews (ISRs)). The version of the code identified in the current license for Pickering [B.2-3] is CSA N286.7-99 (R2007) [B.2-2].

The results of PSR1 N286.7 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington ISRs), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.2.2. As identified in Reference [B.2-6], the Pickering PSR2 review of CSA N286.7-16 is an Incremental review. PSR2 Incremental review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- <u>Compliance:</u> Compliance indicates that the change in the safety requirement, per the topical review, is met.
- <u>Gap</u>: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.2.2 Compliance Review History

B.2.2.1 Application of PSR1 Reviews

The versions of N286.7 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

<u>Pickering Units 5-8</u>

OPG first implemented its software Quality Assurance programs for Nuclear Safety Analysis based on the requirements of CSA Standard N286.7-94. The version of the Standard reviewed as part of the Pickering B ISR was CSA N286.7-99, which was reaffirmed in 2003. A high level review of CSA N286.7-99 (R2003) [B.2-7] was performed as part of the Pickering B ISR for the Deterministic, Probabilistic Safety Analysis and Hazard Analysis Safety Factors in OPG Report NK30-REP-03680-00005 R000 [B.2-8]. It was confirmed through the review that OPG's governance and assessment processes adequately addressed each clause of the standard. The review concluded that Pickering B was in direct compliance with CSA N286.7-99 (R2003) [B.2-7].

OPG Report NK30-REP-03680-00001 R000 [B.2-9] documents the results of the Pickering B ISR based on the Plant Design Safety Factor. No gaps were identified.

The reviews performed as part of the Pickering B ISR did not identify any gaps against CSA N286.7-99. Hence, there are no PSR2 gaps which result from the Pickering B ISR.

Pickering Units 1,4

CSA N286.7 was not included in the list of codes and standards reviewed as part of Pickering A Return to Service. As noted in Reference [B.2-10], CSA N286.7-94 was not reviewed on the basis that it "pertains mostly to design support analysis" and it "pertains mostly to quality assurance aspects".

The subsequent review of CSA N286.7-99 against OPG software governance that was performed as part of the Pickering B ISR [B.2-8] identified no gaps. Since the review was performed against relevant OPG Governance, the conclusions of Reference [B.2-8] are equally applicable to Pickering Units 1,4 and Units 5-8.

Darlington NGS

A clause-by-clause review of OPG software governance was performed against CSA N286.7-99 March 1999 (R2007) [B.2-2] as part of the Darlington ISR. The results of this review are documented in References [B.2-11] and [B.2-12]. The review was conducted by comparing the requirements of the Standard against OPG software governance.

Reference [B.2-11] identified that all requirements of CSA N286.7-99 (R2007) were reflected in the OPG software governance with the exception of one gap. CSA N286.7-99 calls for an explanation of nomenclature and conventions appearing in the Problem Definition Document. At the time that the Darlington ISR review of CSA N286.7-99 was performed, the relevant OPG software governance did not specify this content requirement. This was identified as a licensing basis gap and Station Condition Record (SCR) #D-2010-11439 was raised against this issue. Applicable software governance was subsequently updated as part of Action Request (AR) # 28112428 assignment 01 and hence this gap no longer exists against CSA N286.7-99 (R2007). These requirements are identified in Table 2 of N-PROC-MP-0095 R002, "Qualification of Scientific, Engineering, and Safety Analysis Software" [B.2-13].

Reference [B.2-12] identified a second gap related to specification of library functions as part of the Design Description for Grade 1 software. At the time that the Darlington ISR review of CSA N286.7-99 was performed, the relevant OPG software governance did not specify this content requirement. This was identified as a gap and SCR #D-2012-12366 was raised against this issue. The current revision of N-PROC-MP-0095 [B.2-13] incorporates this requirement and hence this gap no longer exists against CSA N286.7-99.

As noted above, actions have been taken to address the two gaps against CSA N286.7-99 with OPG software governance identified as part of the Darlington ISR. The results of those actions are also applicable to Pickering. Hence, there are no PSR2 gaps which result from the Darlington ISR.

B.2.2.2 Application of Post-PSR1 Reviews

As discussed in Section B.2.1, CSA N286.7 was updated in January 2016 to CSA N286.7-16. The major changes relative to CSA N286.7-99 as documented in the CSA Impact Statement [B.2-4] are as follows:

1. The scope has been expanded [8] and now includes: in-house developed computer programs, third-party computer programs, legacy computer programs, Programmed applications.

2. The N286.7.1 guide [B.2-5] has now been amalgamated into the new edition of N286.7 Standard.

CNSC staff completed a review of OPG's Safety Analysis Program implementation in December, 2015. The results of the review are documented in Reference [B.2-14]. The review was intended to confirm CNSC staff's understanding of the effective implementation of OPG's Safety Analysis Program using the generic requirements of CSA N286-12, "Management System Requirements for Nuclear Facilities." The review covered the programmatic elements of the management of deterministic and probabilistic safety analysis as required by the Pickering and Darlington Licence Conditions Handbooks. This work was a high-level review of the Safety Analysis Program with a focus on the managed processes and interfaces of safety analysis activities and potential identification of gaps with REGDOC-2.4.1 and REGDOC-2.4.2. The CNSC review identified the following observation with regard to OPG's software governance and compliance to N286.7-16:

> *Note that the term "relaxation of requirements" has been removed from the recent version of N286.7-16. This new edition replaces N286.7-99 and its companion guidance document CSA N286.7.1-09. The OPG procedures in which the term "relaxation of requirements" is used (e.g., N-PROC-MP-0097 R003* ["Grading of Scientific Engineering and Safety Analysis Software"]*) should include a warning or otherwise make it clear that a relaxation of requirements is not applicable in performing a deterministic safety analysis updated as part of the scope of the REGDOC-2.4.1 implementation plan.*

This observation was identified as an area of improvement, not a safety significant finding, and has been addressed in the recent revisions of N-PROC-MP-0095, "Qualification of Scientific, Engineering, and Safety Analysis Software" [B.2-13], N-PROC-MP-0097, "Grading of Scientific, Engineering and Safety Analysis Software" [B.2-15], and N-STD-MP-0008, "Development, Qualification and Use of Scientific, Engineering, and Safety Analysis Software" [B.2-16]. Therefore, there is no gap for Pickering PSR2.

As part of a self assessment (NO16-001985-SA) conducted by the Design Engineering department, a clause-by-clause gap assessment of CSA N286.7-16 was performed against OPG governance in September 2016 [B.2-17]. The review identified two gaps relating to specific requirements from N286.7-16 that were not explicitly mentioned in the governance:

- N-STD-MP-0008, "Development, Qualification and Use of Scientific, Engineering, and Safety Analysis Software" [B.2-16], Section 1.7.1, Verification, does not discuss embedded models as required by Clause 8.2 (h) (ii); and

- N-PROC-MP-0095, "Qualification of Scientific, Engineering and Safety Analysis" [B.2-13], Section A.2.0, Design and Development Tasks – Requirements Specification, does not explicitly include a requirement for the computer program version number as required by Clause 13.2.2 (a).

These gaps were addressed in the latest revision of N-STD-MP-0008 [B.2-16] and N-PROC-MP-0095 [B.2-13]. Therefore, there are no gaps for Pickering PSR2.

B.2.3 Compliance Assessment Summary for Pickering PSR2

There are no PSR2 gaps for CSA N286.7-16. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CSA N286.7-16.

B.2.4 References

[B.2-1] CSA Standard N286.7-16, *Quality Assurance of Analytical, Scientific, and Design Computer Programs*, January 2016.

[B.2-2] CSA Standard N286.7-99 (R2007), *Quality Assurance of Analytical, Scientific, and Design Computer Programs for Nuclear Power Plants*, March 1999.

[B.2-3] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.2-4] CSA Impact Statement, *Notification of CSA N286.7 on Quality Assurance of Analytical Scientific, and Design Computer Programs; Product: New Edition; Product Designation: CSA N286.7; Previous Edition Published: 1999, Reaffirmed 2007 and 2012*, Date not provided.

[B.2-5] CSA Standard N286.7.1-09, *Guideline for the application of N286.7-99, Quality Assurance of Analytical, Scientific, and Design Computer Programs for Nuclear Power Plants*, November 2009.

[B.2-6] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.2-7] CSA Standard N286.7-99 (R2003), *Quality Assurance of Analytical, Scientific, and Design Computer Programs for Nuclear Power Plants*, March 1999.

[B.2-8] OPG Report, NK30-REP-03680-00005 R000, *Pickering NGS B – Integrated Safety Review - Safety Analysis Review*, June 2007.

[B.2-9] OPG Report, NK30-REP-03680-00001 R000, *Pickering NGS-B Integrated Safety Review - Plant Design Safety Factor*, August 2007.

[B.2-10] OPG Letter, R. Strickert to J. Tong, NA44-CORR-00531-00381 R000, *Pickering A – Updated Basis for Return to Service Document,* April 20, 2001.

[B.2-11] OPG Report, NK38-REP-03680-10001-R000, *Review of CAN/CSA-N286.7-99 (R2007) March 1999 Quality Assurance of Analytical, Scientific and Design Computer Programs for Nuclear Power Plants for Darlington Integrated Safety Review, August 16, 2011.*

[B.2-12] OPG Report, NK38-REP-03680-10001-ADD-001 R000, *Addendum to the CAN/CSA N286.7 Code Review Report for Darlington ISR,* January 2014.

[B.2-13] OPG Nuclear Procedure, N-PROC-MP-0095 R003, *Qualification of Scientific, Engineering, and Safety Analysis Software*, October 2016.

[B.2-14] CNSC Correspondence, e-Doc 4967475, OPG File No. N-CORR-00531-18039, *Darlington and Pickering NGS: Results of the CNSC Review of OPG's Safety Analysis Program Implementation*, April 15, 2016.

[B.2-15] OPG Nuclear Procedure, N-PROC-MP-0097 R004, *Grading of Scientific, Engineering, and Safety Analysis Software*, October 2016.

[B.2-16] OPG Nuclear Standard, N-STD-MP-0008 R005, *Development, Qualification and Use of Scientific, Engineering, and Safety Analysis Software,* October 2016.

[B.2-17] OPG Self Assessment, NO16-001985-SA, *Self Assessment: Annual High Level Review of Scientific, Engineering, Safety Analysis (SESA) Software Program Adherence*, September 2016.

B.3 CSA N288.1-14, "Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities"

B.3.1 Background

The following paraphrased from the preface and introduction of CSA N288.1-14 [B.3-1] provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *The derived release limit* [DRL] *for a given radionuclide is the release rate that would cause an individual of the most highly exposed group to receive and be committed to a dose equal to the regulatory annual dose limit due to release of the radionuclide to air or surface water during normal operation of a nuclear facility over the period of a calendar year. The DRL is derived using mathematical equations that describe the transfer of radioactive materials through the environment to humans. It takes into account all exposure pathways, including external exposure from immersion in contaminated air and water, external exposure to contaminated soil and beaches, and internal exposure from inhalation and ingestion of radioactivity.*
>
> *CSA N288.1 provides guidelines for calculating derived released limits for radioactive material in airborne and liquid effluents for normal operation of nuclear facilities. It is intended to apply primarily to CANDU nuclear power stations in Canada.*

CSA N288.1-14 [B.3-1] is relevant to Safety Factor 8 (Safety Performance) and Safety Factor 14 (Radiological Impact on the Environment).

Compliance with CSA N288.1 (R2008) [B.3-2] is currently a licence requirement for Pickering NGS (per PROL 48.02/2018) as indicated in Appendix C.1 of the R04 Pickering Licence Conditions Handbook (LCH) [B.3-3].

CSA N288.1-14 is the third edition of this standard, and supersedes the previous editions published in 2008 and 1987. The CSA Impact Statement for Final Publication [B.3-4] of the 2014 edition of CSA N288 provides a "Summary of Significant Changes" which are discussed in Section B.3.2.2 below.

The results of PSR1 CSA N288.1 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.3.2. As identified in Reference [B.3-5], the Pickering PSR2 review of CSA N288.1-14 is an Incremental review. PSR2 Incremental review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- <u>Compliance:</u> Compliance indicates that the change in the safety requirement, per the topical review, is met.

- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.3.2 Compliance Assessment for Pickering PSR2

B.3.2.1 Application of PSR1 Reviews

The versions of N288.1 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

For the Pickering B ISR in 2007 as part of the Environment Safety Factor [B.3-6] an intent review against the 1987 (CSA N288.1-M87) version of the code was conducted. The review confirmed that OPG programs and procedures were aligned with the practices in the standard.

Pickering Units 1,4

CSA N288.1 was not reviewed as part of Pickering A Return to Service because it was concluded that it "Pertains mostly to design support analysis" [B.3-7].

Section 10.1 of the R04 Pickering LCH [B.3-3] states:

> *The licensee shall establish the DRLs in accordance with CSA standard N288.1. The releases of nuclear substances to the environment from the Pickering NGS nuclear facility shall not exceed the DRLs listed below and the sum of all fractional DRL releases must remain less than unity.*
>
> *If any of the individual radionuclide DRLs are exceeded, or the sum of fractional DRL releases exceeds unity, it indicates that OPG is in non-compliance with the public dose limit of 1mSv/year as per the CNSC Radiation Protection Regulations.*

Further, Section 9.4 of the Pickering PROL Application P-CORR-00531-03719 R000 [B.3-8] states:

> *… changes were made to dose modeling parameters from the 2009 implementation of CSA N288.1-08, Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities.*

Since compliance with CSA N288.1 (R2008) is currently a licence requirement for Pickering NGS (per PROL 48.02/2018) as indicated in Appendix C.1 of the R04 Pickering LCH [B.3-3], and this is confirmed by review of the Pickering PROL Application [B.3-8], Pickering Units 1,4 (and Units 5-8) are also in compliance with N288.1 (R2008). Further, the Darlington ISR review of CSA N288.1 is also programmatically applicable to Pickering NGS, as discussed below.

Darlington NGS

The Darlington ISR conducted a high level intent review against CSA N288.1-M87 in 2011 per Reference [B.3-9]. One gap was initially identified relating to Clause 6.10. This gap was subsequently closed based on the justification provided in Appendix B of the referenced report.

Subsequent to this review, a code refresh review was performed in 2014 [B.3-10] against the 2008 update version of the code. A high level summary of the review follows:

> *The documents used in the review were based on the mapping of OPG Nuclear governance to CSA N288.1-2008 U2011 [R-1]. As the OPG governance is constantly revised, the documents reviewed were current as of January 10, 2013, and posted in Asset Suite. The changes in the requirements from CSA N288.1 M87 [R-2] are substantial. The review of all applicable clauses in this code review report confirms that OPG Nuclear governance is in compliance with the requirements of CSA N288.1-2008 [R-1].*
>
> *The majority of the assessment is a comparison with the COG report COG-06-3090 R2 [R-7], which outlines the methodology for calculating the permissible upper limit (the Derived Release Limit, DRL) for COG-member nuclear facilities. The methodology is used as input to compute Derived Release Limits per OPG Report, NK38-REP-03482-10001 [R-8].*

The review concluded that Darlington was in compliance with CSA N288.1-2008 U2011 and no gaps were identified. Since the Darlington review assessed compliance against the common OPG governance, its conclusions are considered to also be applicable to Pickering NGS.

B.3.2.2 Application of Post-PSR1 Reviews

CSA N288.1-14 is the third edition of this standard, and supersedes the previous editions published in 2008 and 1987. The CSA Impact Statement for CSA N288.1-14 identifies the following major changes (paraphrased) [B.3-4]:

> *This New Edition offers improved direction on the applicability of the Guideline.*
>
> *This New Edition contains updates to the following:*
>
> *a) Energy expenditures and dietary intake rates for humans;*
>
> *b) Half-lives, gamma energies, and photon yields for all radionuclides;*
>
> *c) Values for many parameters based largely on a new International Atomic Energy Agency (IAEA) handbook of parameter values (IAEA, 2010);*
>
> *d) Wind direction and precipitation data for use in the wet deposition model; and*
>
> *e) Specific activity model for tritium in animals;*

This New Edition includes the introduction of a model for wild waterfowl as an additional source of human exposure through ingestion, and the extension of the C-14 specific activity model to cover plant to animal transfer.

The R04 Pickering LCH references CSA N288.1-2008, which has an effective date of September 1, 2013. The current Pickering A and B "Derived Release Limit" reports [B.3-11] and [B.3-12] respectively, were issued in 2011. These were updates to the 2003 (R00) version of the DRL report to reflect: "…changes from the recommended calculations, transfer parameter values and exposure factors found in the Guidance for the Calculation of Derived Release Limits (OPG-2002) on which R001 of this report was based."

The Executive Summary of the Pickering B DRL report [B.3-12] states the following:

Upon receipt of the approval and license amendment from the Canadian Nuclear Safety Commission (CNSC), these limits will be implemented and will supersede the 2006 PNGS-B DRLs [R-1].

A routine review of PNGS-B DRLs in 2008 [R-2] indicated an update was required to:

- *Update methodology, pathway models, transfer factors, representative age groups, and dosimetry to conform to the recently updated Canadian Standards Association (CSA) N288.1-08 standard for the calculation of DRLs [R-3].*
- *Incorporate changes in the locations and characteristics of nearby members of the public as identified by the review of the Pickering Site Specific Survey [R-4] [R-5].*

DRL calculations were performed with version 5.4.0 of EcoMetrix's IMPACT software, which accurately represents the models contained in the CSA N288.1-08 standard. The software has undergone verification and validation according to OPG's Software Governance and meets the requirements of CSA Standard N286.7 [R-6]. This report and the DRL calculations were also independently verified by EcoMetrix Inc [R-7]…

The same Executive Summary is also included in the Pickering A DRL report.

As identified in the CSA Impact Statement [B.3-4] discussed above, the 2014 version of the standard introduces additional changes beyond those reflected in the 2008 standard. The "2015 Results of Environmental Monitoring Programs" [B.3-13], Section 6 identifies that:

Recommendations from these studies will be incorporated into the EMPs [Environmental Monitoring Programs] *following the revision of the station DRLs which will take place in 2016 and incorporation of N288.1-14 into the public dose calculations.*

In addition to this, there is an ongoing COG program that is refining modelling and tools to align with the new 2014 version of the Standard [B.3-14].

The above demonstrates that OPG and Pickering are fully compliant with the 2008 version of CSA N288.1, and are actively working towards implementing the 2014 version of CSA N288.1-14. Although the COG initiative is still in progress, the changes in CSA N288.1-14 are not expected to impact the physical plant design or safe operation of Pickering since: a) Pickering is compliant with the 2008 version reference in the LCH, and b) the incremental changes to the code largely reflect refinements in methodology. Therefore, there is no gap for Pickering PSR2.

B.3.3 Compliance Assessment Summary for Pickering PSR2

There are no PSR2 gaps for CSA N288.1-14 [B.3-1]. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CSA N288.1-14.

B.3.4 References

[B.3-1] CSA Standard N288.1-14, *Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities*, March 2014.

[B.3-2] CSA Standard N288.1-08, *Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities*, September 2008.

[B.3-3] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.3-4] CSA Impact Statement for Final Publication, *Product: New Edition; Product Designation: CSA N288.1; Product Title: Guidelines for calculating derived release limits for radioactive material in airborne and liquid effluents for normal operation of nuclear facilities; Date of release: 2014*, Date not provided.

[B.3-5] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.3-6] OPG Report, NK30-REP-03680-00010 R000, *Pickering NGS-B Integrated Safety Review – Environment*, May 2007.

[B.3-7] OPG Letter, R. Strickert to J. Tong, NA44-CORR-00531-00381 R000, *Pickering A – Updated Basis for Return to Service Document*, April 20, 2001.

[B.3-8] OPG Letter, P-CORR-00531-03719 R000, G. Jager to M. A. Leblanc, *Application for Renewal of Pickering Nuclear Generating Station Power Reactor Operating Licence*, July 4, 2012.

[B.3-9] OPG Report, NK38-REP-03680-10064 R000, Review of CAN/CSA-N288.1-M87 (R2008) (January 1987), *Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities for Darlington Integrated Safety Review*, August 2011.

[B.3-10] OPG Report, NK38-REP-03680-10146 R000, *Code Refresh Review of CAN/CSA-N288.1-2008 Update 1 for Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities for Darlington Integrated Safety Review*, January 2014.

[B.3-11] OPG Report, NA44-REP-03482-00001 R002, *Derived Release Limits and Environmental Action Levels for Pickering Nuclear Generating Station A*, January 2011.

[B.3-12] OPG Report, NK30-REP-03482-00001 R002, *Derived Release Limits and Environmental Action Levels for Pickering Nuclear Generating Station B*, January 2011.

[B.3-13] OPG Report, N-REP-03443-10015 R000, *2015 Results of Environmental Monitoring Programs*, April 2016.

[B.3-14] OPG Letter, R. Manley to M. Santini and F. Rinfret, N-CORR-00531-06905 R000, *REGDOC 3.1.1 Research and Development Annual Reporting*, June 16, 2015.

B.4 CSA N288.4-10, “Environmental Monitoring Program at Class I Nuclear Facilities and Uranium Mines and Mills”

B.4.1 Background

The following, paraphrased from the preface and introduction of CSA N288.4-10 (R2015) [B.4-1], provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *CSA N288.4 purpose is to protect the environment in conformance with the regulations under the Nuclear Safety and Control Act. This is done through the monitoring of radioactive and non-radioactive contaminants, physical stressors, potential biological effects, and pathways for both human and non-human biota.*
>
> *CSA N288.4 provides the requirements for the design and operation of environmental monitoring programs (EMPs) for Class I nuclear facilities and uranium mines and mills. This standard provides guidance and requirements on general objectives of an EMP, criteria for establishing and revising an EMP, design of an EMP, sampling and analytical procedures, interpretation of data, quality assurance and quality control, reporting and review, staff qualifications and training.*

CSA N288.4-10 is applicable to Safety Factor 8 (Safety Performance) and Safety Factor 14 (Radiological Impact on the Environment).

Compliance with CSA N288.4-10 is currently a licence requirement for Pickering NGS (per PROL 48.02/2018) as indicated in Appendix C.1 of the R04 Pickering Licence Conditions Handbook [B.4-2].

CSA N288.4-10, which was reaffirmed in 2015, is the second edition of this standard, and supersedes the first edition published in 1990 under the title “Guidelines for Radiological Monitoring of the Environment” [B.4-3]. The results of PSR1 CSA N288.4 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.4.2. As identified in Reference [B.4-4], the Pickering PSR2 review of CSA N288.4-10 is an Incremental review. PSR2 Incremental review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- <u>Compliance:</u> Compliance indicates that the change in the safety requirement, per the topical review, is met.
- <u>Gap</u>: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.4.2 Compliance Assessment for Pickering PSR2

B.4.2.1 Application of PSR1 Reviews

The versions of N288.4 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

A high level intent review was done against CSA N288.4-M90 [B.4-3] for Pickering B in 2007 [B.4-5], which identified the following:

> *Ontario Power Generation Pickering B-NGS Environmental monitoring program N-PROC-OP-0025, "Management of the Off-Site Radiological Environmental Monitoring Program at Ontario Power Generation", presents these quantitative methods in Section 1.4 Field Sampling, Section 1.5 Laboratory Analysis and Section 1.6 Review and Analysis of Data. It also discusses the QA standards recommended by N288.4-M90.*
>
> *N-STD-OP-0031, "Monitoring of Radioactivity in Effluents", further outlines the QA standards and reviews which are done in the effluent monitoring program.*

The review concluded that Pickering B and OPG Nuclear Programs documented in [B.4-6] and [B.4-7][9] were "Indirectly Compliant" with N288.4-M90.

Pickering Units 1,4

CSA N288.4 was not assessed as part of Pickering A Return to Service given it, "Pertains mostly to Operations aspects, or other aspects not having a direct or immediate effect on installed design features" [B.4-8]. However, compliance with N288.4-10 is a licence requirement for Pickering NGS per Section 10.1 of the R04 Pickering Licence Conditions Handbook [B.4-2]. Further, assessments have been performed for Darlington which are programmatically applicable to Pickering NGS, as discussed below.

Darlington NGS

As part of the Darlington ISR, a high level review against all relevant clauses of N288.4-M90 (R2008) [B.4-9] was conducted in 2009. The review concluded that:

> *The intents of all relevant clauses of CAN/CSA-N288.4-M90 (Reference 6) were met by Ontario Power Generation's Darlington Nuclear Generating Station as demonstrated in Management of the Off-site Radiological Environmental Monitoring Program at OPG Nuclear Site N-PROC-OP-0025 R06, Operating Standards for the Ontario Power*

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[9] Note, N-STD-OP-0031, "Monitoring of Nuclear and Hazardous Substances in Effluents" was formerly titled "Monitoring of Radioactivity in Effluents".

Generation Radiological Environmental Monitoring Program N-REP-03480-10001-R01 and associated documentation.

No gaps were identified in this review.

In addition, a Code Refresh review was conducted for Darlington in 2014 against the 2010 version of the standard [B.4-10]. CSA N288.4-10 [B.4-1], was the second edition of this standard, and superseded the 1990 version [B.4-3]. A synopsis and summary of the major changes to the standard are provided below [B.4-11]:

1. *The standard applies to all Class I nuclear facilities and uranium mines and mills (the previous version was only intended for CANDU nuclear power plants).*
2. *The standard addresses monitoring for the protection of both human and non-human biota (the previous version only addressed protection of humans).*
3. *Monitoring of nuclear substances, hazardous substances and physical stressors such as habitat loss and entrainment|impingement of aquatic organisms is addressed (the previous version only addressed monitoring of nuclear substances).*
4. *This standard requires a stricter adherence to a risk-based approach to the development of an Environmental Monitoring Program. This approach assumes that an Environmental Risk Assessment of the nuclear facility is available.*
5. *The standard requires that a systematic, informed planning process be followed during the development of an Environmental Monitoring Program.*
6. *Both Pathways Monitoring and Biological Effects Monitoring are addressed (the previous version only addressed Pathways Monitoring).*
7. *The standard requires a more formal periodic review of the need for, and adequacy of the Environmental Monitoring Program, which may result in either the addition of new program elements or the deletion of existing program elements.*
8. *Additional informative material, including examples of the application of the standard to hypothetical facilities, has been included to provide guidance to the user.*

These changes substantially increase the scope of the program; the number of facilities that fall within the scope of the standard; the number and types of potential effects that must be considered when developing a monitoring program and the types of monitoring that may be included in the program. The changes make the Standard consistent with current regulatory requirements, international guidance and accepted practice.

A summary of the review from [B.4-10] is provided below:

The changes made in the revised edition as CSA standard N288.4-2010, "Environmental Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills", relative to CSA standard N288.4-M90 (R2008), are substantial. It was changed from "requirements for radiological monitoring" to "requirements for monitoring of all environmental parameters, radiological and non-radiological." Due to *the extent of the changes, the entire CSA N288.4-2010 document has been assessed on a clause-by-clause basis, but comments are given generically at a High Level, as defined by N-PROC-LE-0005.*

This High Level review has determined that OPG Nuclear governance is in compliance with the requirements of CSA N288.4-2010. The review did not identify any gaps relative to its intent. The monitoring programs (existing and planned) comply with the intent of the Standard. The changes being implemented are not considered as gaps at the high level, but as improvements or supplements to the existing programs.

Since the Darlington review assessed compliance based on governance that applies across OPG's nuclear operations, the Darlington ISR conclusions are applicable to Pickering and Pickering PSR2.

B.4.2.2 Application of Post PSR1 Reviews

A gap analysis specific to Pickering Environmental programs against the 2010 version of the standard was completed in 2011 [B.4-12]. A summary of the gaps with either mandatory or recommended actions is included in Appendix B of Reference [B.4-12]. In support of the Pickering licence renewal application [B.4-13], a new/revised procedure for managing Environmental Monitoring Programs [B.4-14] was provided. As a follow-up to this, in November 2014, the CNSC requested a detailed implementation plan for both N288.4-10 and N288.5-11 [B.4-15] for both Darlington and Pickering, and specifically requested more information on supplemental studies. In January 2015 OPG communicated to the CNSC that all issues with respect to N288.4-10 implementation were complete [B.4-16] and only the supplementary studies in Attachment 1 of Reference [B.4-16] remained to be completed. These remaining studies are considered to be on-going monitoring programs under N288.4-10 and are not considered to be gaps for PSR2.

The 2015 version of N288.4 is a simple reaffirmation of the 2010 version of the standard. Hence, there are no changes incremental to the review already conducted for the 2010 version discussed in the section above.

B.4.3 Compliance Assessment Summary for Pickering PSR2

There are no PSR2 gaps for CSA N288.4-10 [B.4-1]. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CSA N288.4-10.

B.4.4 References

[B.4-1] CSA Standard, N288.4-10 (R2015), *Environmental Monitoring Program at Class I Nuclear Facilities and Uranium Mines and Mills*, May 2010.

[B.4-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.4-3] CSA Standard, N288.4-M90, *Guidelines for Radiological Monitoring of the Environment*, November 1990.

[B.4-4] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.4-5] OPG Report, NK30-REP-03680-00010 R000, *Pickering NGS-B Integrated Safety Review – Environment*, May 2007.

[B.4-6] OPG Program, N-PROG-OP-0006 R018, *Environmental Management*, April 2015.

[B.4-7] OPG Standard, N-STD-OP-0031 R006, *Monitoring of Nuclear and Hazardous Substances in Effluents*, October 2014.

[B.4-8] OPG Letter, NA44-CORR-00531-00381, R.J. Strickert to J.S.C. Tong, *Pickering A – Updated Basis for Return to Service Document*, April 20, 2001.

[B.4-9] OPG Report, NK38-REP-03680-10065 R000, *Review of CAN/CSA-N288.4-M90 (R2008) (November 1990), Guidelines for Radiological Monitoring of the Environment for Darlington Integrated Safety Review*, June 2011.

[B.4-10] OPG Report, NK38-REP-03680-10148 R000, *Code Refresh Review of CSA N288.4-2010 Guidelines for Radiological Monitoring of the Environment*, January 2014.

[B.4-11] CSA Impact Statement and Publication Notice, *Product: Revision; Product Designation CSA N288.4-10; Date of Release: 2010*, Date not provided.

[B.4-12] OPG Report, N-REP-03443-0443505 R001, *Gap Analysis for the Environmental Monitoring Programs for Pickering Nuclear against CSA N288.4-10*, August 2011.

[B.4-13] OPG Letter, P-CORR-00531-04193, G. Jager to M. Santini, *Pickering NGS – Pickering Licence Renewal Information Request pursuant to Application for Renewal – Environmental Monitoring Program and Implementation of CSA N288.4-10*, December 7, 2012.

[B.4-14] OPG Procedure, N-PROC-OP-0025 R011, *Management of the Environmental Monitoring Programs*, December 2015.

[B.4-15] CNSC Letter, N-CORR-00531-06735, M. Santini and F. Rinfret to B. McGee and B. Duncan, *Pickering and Darlington NGS: Request for Detailed Implementation Plans for Compliance with CSA Standard N288.4-10 and N288.5-11, New Action Item 2014-OPG-5550 and Closure of action Item 2014-48-3425*, November 6, 2014.

[B.4-16] OPG Letter, N-CORR-00531-06780, B. Reuber to M. Santini and F. Rinfret, *Implementation Plans for Compliance with CSA Standards N288.4-10 and N288.5-11, New Action Item 2014-OPG-5550*, January 22, 2015.

B.5 CSA N293-12, "Fire Protection for Nuclear Power Plants"

The PSR2 review of CSA N293-12 is in progress. Gaps identified from this review will be applicable to the Plant Design Safety Factor, and hence, results are presented in the Pickering NGS PSR2 Plant Design Safety Factor Report.

B.6 CNSC RD-204 (2008), "Certification of Persons Working at Nuclear Power Plants"

B.6.1 Background

The following paraphrased from the purpose and scope of CNSC RD-204 [B.6-1] provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *CNSC RD-204 purpose is to ensure that persons seeking a certification or renewal of a certification by the CNSC for a position referred to in the licence of a nuclear power plant are qualified to carry out the duties of that position in accordance with the Nuclear Safety and Control Act.*
>
> *CNSC RD-204 sets requirements on the programs and processes to train and examine persons seeking or renewing a certification, training and requalification tests that certified persons seeking a renewal of certifications must have completed, and the qualification required of persons seeking a certification.*

CNSC RD-204 is applicable to Safety Factor 10 (Organization, the Management System and Safety Culture).

Compliance with CNSC RD-204 is currently a licence requirement for Pickering NGS (per PROL 48.02/2018) as indicated in Appendix C.2 of the R04 Pickering Licence Conditions Handbook [B.6-2]. According to licence condition 3.3: "The license shall implement and maintain a training program that includes certification training, examinations and tests for positions requiring certified personnel in accordance with CNSC Regulatory Document RD-204, *Certification of Persons Working at Nuclear Power Plants*" [B.6-2]. The licence condition also specifies which positions require certified personnel. CNSC RD-204 (2008) is the first edition of this Regulatory Document.

The results of PSR1 RD-204 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington ISRs), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.6.2. As identified in Reference [B.6-3], the Pickering PSR2 review of CNSC RD-204 is an Incremental review. PSR2 Incremental review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.
- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.6.2 Compliance Assessment for Pickering PSR2

B.6.2.1 Application of PSR1 Reviews

The versions of RD-204 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

A review of RD-204 was not included as part of the Pickering B ISR or Pickering A Return to Service. As discussed below, the Darlington ISR prepared a clause-by-clause review against the current revision of RD-204, the findings of which are programmatically applicable to Pickering NGS.

Although no code review was performed for Pickering Units 1,4 or Units 5-8, compliance with RD-204 (2008) [B.6-1] is a licence requirement for Pickering NGS per Section 3.3 of the Pickering LCH [B.6-2]. Further, Section 2.3 of Attachment 3 of the Pickering PROL Renewal Application [B.6-4] states the following:

> *The ANO* [Authorized Nuclear Operator] *and CRSS* [Control Room Shift Supervisor] *training programs are based on a Systematic Approach to Training as required by RD-204. Recent improvements include development and implementation of revised On-the-Job Training Programs. Considerable work has been completed for the most recent CRSS Supplemental program by updating all the mentor guides and the development and implementation of a three day leadership workshop.*

Section 2.6 of Attachment 3 of the Pickering PROL Renewal Application states [B.6-4]:

> *For certified and security staff, Regulatory Documents RD-204 and RD-363 also outline specific fitness for duty requirements that are complied with by OPG.*

Darlington NGS

RD-204 (2008) [B.6-1] was reviewed as part of the Darlington ISR as documented in OPG Report NK38-REP-03680-10095 R000 [B.6-5]. The clause-by-clause review concluded that OPG governance meets the requirements of the Regulatory Document with the exception of 8 gaps (01489, 01498, 01504, 01505, 01509, 01510, 01514, and 01523). These were assigned to three issues in the Darlington Final ISR Report, NK38-REP-03680-10104 R000 [B.6-6] as outlined in the table below:

Issue	Gaps	Status
D272: "Reinstatement of a Person to the Duties of a Position Following Absence or Removal from Those Duties" *There is no documented process in place to reinstate a person who has been absent or removed from shift duties.*	01489 01498	Acceptable Deviation

Issue	Gaps	Status
D273: "Maintaining Qualification During Temporary Assignment to Other Positions" *There is no documented procedure identifying specific requirements for performing minimum shift duties during temporary assignment.*	01504 01505 01509 01510	Closed
D274: "Recertification Requirements after Decertification" *There is no documented process for re-certifying a person who has been decertified.*	01514 01523	Acceptable Deviation

The review noted that these gaps do not represent non-conformances with the Operating Licence since the review confirmed that there is governance in place at OPG Darlington consistent with the requirements of RD-204 for persons in the position of performing shift duties [B.6-5]. The above assessment is programmatically applicable to Pickering NGS.

Following the initial code review, issue D273 was closed with the issuance of D-GUID-09110-10004 [B.6-7] which was shown to meet the requirements of Clause 14 of CNSC RD-204 [B.6-1]. Subsequently, N-GUID-09110-10000, "Certification Requirements for Certified Staff on Rotation" [B.6-8] was issued. This OPG Nuclear document is identical in content to the Darlington Guide and applies to Pickering NGS.

The Darlington NGS final ISR Report summarized D272 as follows [B.6-6]:

> *This Issue is composed of Gaps # 01489 and 01498, which are both against sub-clauses of Clause 13 of CNSC RD-204 which defines requirements for reinstatement of a person to the duties of a position following absence or removal from those duties. In the Code Review Report, NK38-REP-03680-10095, it is concluded that since a documented process to reinstate certified shift staff who have been absent or removed from shift duties does not exist, all clauses related to reinstatement are identified as gaps.*

The justification for the classification of Acceptable Deviation for issue D272 was supported through the submission of additional documents to support the application to amend the Pickering and Darlington Power Reactor Operating Licences to implement RD-204. A list of the OPG implementing documents for RD-204 was provided to the CNSC in Reference [B.6-9]. These included N-TQD-103-00001 R007 "Nuclear Certified Shift Personnel Continuing Training and Qualification Description", N-INS-08920-100001 R003 "Requalification Testing of Certified Shift Personnel", N-INS-08920-10003 R002 "Independence And Security for Initial Certification Examinations and Requalification Testing of Certified Shift Personnel", and N-PROC-RA-0005 "Written Reporting to Regulatory Agency" (which calls up N-FORM-10824 R001 "Report on Performance and Status of Certified Personnel"). One reporting criterion in N-FORM-10824 requires the date of reinstatement to duties and any remedial actions taken prior to reinstatement be reported to the CNSC. These requirements demonstrate that the gap is addressed.

Issue D274 is related to the recertification of an individual who has been decertified and: "The gaps in this Issue point to there being no documented procedure at OPG for re-certifying a person who has been decertified" [B.6-6]. The justification for the classification of Acceptable Deviation for Issue D274 from the Final Darlington NGS ISR Report is as follows:

> *The reason OPG does not have governance for recertification of staff after decertification is that it is considered that a business need for such a procedure does not exist. Without a defined process for certification following decertification, individuals could become recertified after meeting the same requirements that are prescribed for initial certification, which are documented in the following three documents: N-TQD-101-00001 R005, "Authorized Nuclear Operator Initial Training and Qualification Description", N-TQD-102-00001 R007, "Nuclear Shift Manager/Control Room Shift Supervisor Initial Training and Qualification Description", or N-TQD-105-00001 R001, "Darlington Unit 0 Control Room Operator (CRO) Initial Training and Qualification Description".*
>
> *Alternatively, in absence of a defined process within OPG governance, OPG could, on a case-by-case basis, rely directly on the requirements of CNSC RD-204 Clause 34.*

Both D272 and D274 issues and the justifications as Acceptable Deviations are applicable to Pickering NGS given the programmatic nature of the gaps and their resolutions. These Acceptable Deviations are not impacted by operation beyond 2020. These issues are therefore not PSR2 gaps.

In summary, three issues related to CNSC RD-204 were identified as part of the Darlington ISR. Of these three, one (D273) was closed with the development of documentation (which was also produced for Pickering NGS), and the other two (D272 and D274) were Acceptable Deviations with justifications which are programmatically applicable to Pickering. There are therefore no PSR2 gaps.

B.6.2.2 Application of Post-PSR1 Reviews

OPG Compliance with RD-204 (2008) is demonstrated through the Pickering NGS PROL Application [B.6-4] and through the Darlington ISR review of RD-204 which is programmatically applicable to Pickering NGS. There have been no revisions to CNSC RD-204 since first issuance in 2008, and no other reviews have been performed as none were required.

B.6.3 Compliance Assessment Summary for Pickering PSR2

There are no PSR2 gaps for CNSC RD-204 (2008) [B.6-1]. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CNSC RD-204 (2008).

B.6.4 References

[B.6-1] CNSC Regulatory Document RD-204, *Certification of Persons Working at Nuclear Power Plants*, February 2008.

[B.6-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.6-3] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.6-4] OPG Letter, P-CORR-00531-03719 R000, G. Jager to M. A. Leblanc, *Application for Renewal of Pickering Nuclear Generating Station Power Reactor Operating Licence*, July 4, 2012.

[B.6-5] OPG Report, NK38-REP-03680-10095 R000, *Review of CNSC RD-204 (February 2008) Certification of Persons Working at Nuclear Power Plants for Darlington Integrated Safety Review*, August 2011.

[B.6-6] OPG Report, NK38-REP-03680-10104 R000, *Darlington NGS Integrated Safety Review (ISR) – Final ISR Report*, October 2011.

[B.6-7] OPG Guideline, D-GUID-09110-10004 R000, *Certification Maintenance Requirements for Certified Staff on Rotation*, December 2010.

[B.6-8] OPG Guideline, N-GUID-09110-10000 R000, *Certification Requirements for Certified Staff on Rotation*, May 2011.

[B.6-9] OPG Letter, N-CORR-00531-04314, W.R. Robinson to T.E. Schaubel and P.A. Webster, *Documents to be Submitted in Support of the Application to Amend a Power Reactor Operating Licence to Implement RD-204*, August 11, 2008.

B.7 CNSC REGDOC-3.1.1 (2014), "Reporting Requirements for Nuclear Power Plants"

B.7.1 Background

The following paraphrased from the preface and introduction of CNSC REGDOC-3.1.1 (2014) [B.7-1] provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *Nuclear power plants are required to report to the CNSC using event reports for situations or events of higher safety significance and that may require short-term action by the CNSC, and to submit routine scheduled reports on various topics that are required for longer-term compliance monitoring. Nuclear power plants are also required to provide notification of certain normal business activities and to file specific records with the CNSC in accordance with the Nuclear Safety and Control Act.*
>
> *Regulatory document REGDOC-3.1.1 sets out the timing and information that nuclear power plant licensees are required to report to the CNSC to support the conditions of applicable power reactor operating licences. REGDOC-3.1.1 presents the types of reports, their frequency and the applicable timeframe for reporting. It also contains guidance, explanatory information, forms and templates to assist users in meeting reporting requirements.*

REGDOC-3.1.1 (2014) is applicable to Safety Factor 10 (Organization, the Management System and Safety Culture).

Compliance with REGDOC-3.1.1 [B.7-1] is currently a licensing requirement for Pickering NGS (per PROL 48.02/2018) as indicated in Appendix C.2 of the R04 Pickering Licence Conditions Handbook [B.7-2]. Licence Condition 4.3 states: "the licensee shall notify and report in accordance with CNSC Regulatory Document REGDOC 3.1.1 *Reporting Requirements for Nuclear Power Plants"* [B.7-2].

REGDOC-3.1.1 is the result of the consolidation of two draft documents, RD-99.1, "Reporting Requirements for Operating Nuclear Power Plants" [B.7-3] and GD-99.1, "Guide to the Reporting Requirements for Operating Nuclear Power Plants" [B.7-4]. It also supersedes CNSC S-99, "Reporting Requirements for Operating Nuclear Power Plants" [B.7-5] which was published in 2003. It includes updates to the reporting requirements, including the Safety Performance Indicators.

The results of PSR1 S-99 and REGDOC-3.1.1 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.7.2. As identified in Reference [B.7-6], the Pickering PSR2 review of CNSC REGDOC-3.1.1 is an Incremental review. PSR2 Incremental review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.

- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.7.2 Compliance Assessment for Pickering PSR2

B.7.2.1 Application of PSR1 Reviews

The versions of REGDOC-3.1.1 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

A clause by clause review of CNSC S-99 [B.7-5] is documented in the OPG Report NK30-REP-03680-00006 R002 [B.7-7] for the Safety Performance Safety Factor, which demonstrated compliance with all relevant clauses of S-99.

The Environmental Program compliance with CNSC S-99 was assessed in OPG Report NK30-REP-03680-00010 R000 [B.7-8] for the Environment Safety Factor, wherein the OPG environmental procedures and standards which document the S-99 reporting requirements are documented. The report concluded that OPG complies with the intent of CNSC S-99.

S-99 was also reviewed with respect to the safeguards program in NK30-REP-03680-00011, "Pickering NGS-B Integrated Safety Review - Safeguards" [B.7-9]. The document states that "Pickering NGS-B has submitted the required safeguards reports to the CNSC and thereby is in compliance with Articles 59-69 and 74 of IAEA INFCIRC/164, S-99, AECB1049 and the safeguards requirement #13 of the PROL."

Based on the Pickering B ISR reviews, no PSR2 gaps were identified with respect to S-99 [B.7-5], which was the precursor to REGDOC-3.1.1 [B.7-1].

Pickering Units 1,4

Both S-99 [B.7-5] and REGDOC-3.1.1 [B.7-1] were issued subsequent to Pickering A Return to Service. AECB Regulatory Document R-99, "Reporting Requirement for Operating Nuclear Power Facilities", which preceded S-99, was omitted from review on the basis that it "pertains mostly to Operations aspects, or other aspects not having a direct or immediate effect on installed design features" [B.7-10]. However, the Pickering B ISR reviewed S-99 and the conclusions of that work are applicable across the OPG Nuclear fleet, including Pickering Units 1,4 and Units 5-8. Further, the Darlington ISR review of S-99 is programmatically applicable to Pickering as discussed below.

Darlington NGS

A clause-by-clause review of S-99 [B.7-5] was performed as part of the Darlington ISR, as documented in NK38-REP-03680-10045 R000 [B.7-11]. The report concluded that "OPG has implemented governance, programs and procedures to ensure compliance with the requirements of S-99. No gaps or major findings were found". Given that S-99 compliance is addressed by Governance, Program, Policies and Procedures that apply across OPG's Nuclear fleet, the results of the programmatic review are applicable to Pickering NGS. No PSR2 gaps are identified from the Darlington ISR review of S-99.

B.7.2.2 Application of Post-PSR1 Reviews

Compliance with CNSC REGDOC-3.1.1 is a license requirement per Pickering PROL 48.02/2018. The request from OPG to the CNSC for a licence amendment included a Transition Plan for REGDOC-3.1.1 which indicates that "Starting January 1, 2015, Pickering Nuclear will be in compliance with the requirements of Regulatory Document 3.1.1" [B.7-12]. Attachment 3 of [B.7-12] provides the details of the Transition Plan.

The CNSC Record of Proceedings for the Application to Amend Reporting Requirements in Power Reactor Operating Licence [B.7-13] describes the changes in REGDOC-3.1.1 [B.7-1] relative to S-99 [B.7-5] as follows:

> *REGDOC-3.1.1, which supersedes S-99, provides a modernized set of reporting requirements for NPPs, including an improved set of safety performance indicators. The included requirements and guidance would enable CNSC staff to effectively oversee NPP operations, while eliminating unnecessary or duplicate reporting.*

The CNSC Record of Proceedings also notes that several requirements introduced by REGDOC-3.1.1 were previously introduced as licence conditions in the PROL pending revision of S-99. Therefore some new requirements introduced by REGDOC-3.1.1 were already addressed at Pickering NGS through previous licence conditions. For Pickering, referencing REGDOC-3.1.1 as a licensing requirement included the removal of the following licence condition [B.7-2] related to reporting requirements now covered in REGDOC-3.1.1:

> *The licensee shall report any apparent non-compliance of applicable law at the federal, provincial or municipal level that pertains to the activities licensed under this licence;*

This represents a change in REGDOC-3.1.1 relative to S-99, but no change in the PROL reporting requirements for OPG since the new requirement was previously addressed through licence conditions.

Compliance with REGDOC-3.1.1 is further ensured through use of Action Requests to track scheduled submission requirements of REGDOC-3.1.1 compliance reports (e.g., effluent/emission reports, quarterly performance indicators, R&D annual reporting).

Based on the Transition Plan for REGDOC-3.1.1 [B.7-12] (which was accepted by the CNSC [B.7-2]), as well as ongoing compliance reports and compliance initiatives as discussed above, it is concluded that there are no PSR2 gaps for Pickering NGS compliance with REGDOC-3.1.1.

B.7.3 Compliance Assessment Summary for Pickering PSR2

There are no PSR2 gaps for CNSC REGDOC-3.1.1 (2014) [B.7-1]. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CNSC REGDOC-3.1.1 (2014).

B.7.4 References

[B.7-1] CNSC Regulatory Document REGDOC-3.1.1, *Reporting Requirements for Nuclear Power Plants*, May 2014.

[B.7-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.7-3] CNSC RD-99.1, *Reporting Requirements for Operating Nuclear Power Plants*, Draft, July 2012.

[B.7-4] CNSC GD-99.1, *Guide to the Reporting Requirements for Operating Nuclear Power Plants*, Draft, July 2012.

[B.7-5] CNSC Regulatory Standard S-99, *Reporting Requirements for Operating Nuclear Power Plants*, March 2003.

[B.7-6] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.7-7] OPG Report, NK30-REP-03680-00006 R002, *Pickering NGS-B Integrated Safety Review: Review of Safety Performance*, March 2008.

[B.7-8] OPG Report, NK30-REP-03680-00010 R000, *Pickering NGS-B Integrated Safety Review – Environment*, May 2007.

[B.7-9] OPG Report, NK30-REP-03680-00011 R000, *Pickering NGS-B Integrated Safety Review – Safeguards*, March 2007.

[B.7-10] OPG Letter, NA44-CORR-00531-00381 R000, R. J. Strickert to J. S. C. Tong, *Pickering A – Updated Basis for Return to Service Document*, April 20, 2001.

[B.7-11] OPG Report, NK38-REP-03680-10045 R000, *Review of CNSC S-99 (March 2003), Reporting Requirements for Operating Nuclear Power Plants for Darlington Integrated Safety Review*, June 2011.

[B.7-12] OPG Letter, P-CORR-00531-04346 R000, B. McGee to M. Leblanc, *Pickering Nuclear Generating Station – Request for a Licence Amendment to Incorporate CNSC Regulatory Document REGDOC-3.1.1*, October 15, 2014.

[B.7-13] CNSC Record of Proceedings, *Including Reasons for Decision, Application to Amend Reporting Requirements in Power Reactor Operating Licences*, http://nuclearsafety.gc.ca/eng/the-commission/pdf/2014-12-22-Decision-NPPsLicensees-CMD14-H119-e-edocs4603935.pdf, December 22, 2014.

B.8 CNSC REGDOC-2.9.1 (2013), "Environmental Protection Policies, Programs and Procedures"

B.8.1 Background

The following paraphrased from the preface and introduction of CNSC REGDOC-2.9.1 [B.8-1] provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *Environmental protection policies, programs, and procedures are an important component of the overall requirement for licensees to make adequate provision for protection of the environment. Licensees also have specific obligations to take all reasonable precautions to protect the environment and to control the releases of nuclear and hazardous substances. The respective regulations require submission of environmental protection policies and procedures at Class I nuclear facilities and submission of environmental protection policies and programs at uranium mines and mills.*
>
> *The purpose is to help assure that licensees implement adequate environmental protection policies, programs and procedures, other than for licences to abandon, at Class I nuclear facilities and uranium mines and mills, in accordance with the Nuclear Safety and Control Act (NSCA) and regulations.*
>
> *This document sets out the environmental protection policies, programs and procedures that licensees shall implement at Class I nuclear facilities and uranium mines and mills, when required by the applicable licence or other legally enforceable instrument.*

REGDOC-2.9.1 is directly relevant to Safety Factor 14 (Radiological Impact on the Environment). A number of clauses also relate to Safety Factor 8 (Safety Performance).

Although compliance with REGDOC-2.9.1 is not a licensing requirement, compliance with CNSC Regulatory Standard S-296 "Environmental Protection Policies, Programs and Procedures at Class I Nuclear Facilities and Uranium Mines and Mills" is currently a licence requirement for Pickering NGS (per PROL 48.02/2018) as indicated in Appendix C.2 of the R04 Pickering Licence Conditions Handbook [B.8-2]. REGDOC-2.9.1 supersedes S-296 [B.8-3] and combines the information from associated Regulatory Guide G-296, "Developing Environmental Protection Policies, Programs and Procedures at Class I Nuclear Facilities and Uranium Mines and Mills" [B.8-4].

On November 30, 2015, a revised Draft of REGDOC-2.9.1 was posted for an additional round of consultation [B.8-5]. Consultation closed on March 29, 2016, which is past the PSR2 freeze date. As a result, the 2013 version of REGDOC-2.9.1 [B.8-1] is reviewed as part of PSR2.

The results of PSR1 S-296 and CNSC REGDOC-2.9.1 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.8.2. As identified in Reference [B.8-6], the Pickering PSR2 review of CNSC REGDOC-2.9.1 is an Incremental review. PSR2 Incremental review includes an assessment of the intent of recent changes to the Law, Regulation, Code and Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.

- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.8.2 Compliance Assessment for Pickering PSR2

B.8.2.1 Application of PSR1 Reviews

The versions of REGDOC-2.9.1 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

CNSC S-296 [B.8-3] was reviewed as part of the Pickering B ISR, as documented in Environment Safety Factor Report NK30-REP-03680-00010 R000, "Pickering NGS-B Integrated Safety Review – Environment" [B.8-7]. The review included a clause by clause review of ISO 14001:2004, "Environmental Management Systems - Requirements with Guidance for Use" [B.8-8], taking into account the provisions stipulated in S-296 [B.8-3] and G-296 [B.8-4], in order to ensure compliance with ISO 14001:2004 and CNSC S-296. The review concluded that the OPG Environmental Management program, described in N-PROG-OP-0006 "Environmental Management" [B.8-9] is in compliance with the requirements and recommendations of S-296. There were no associated gaps. Since compliance is demonstrated based on Governance, Programs, Procedures and Practices that apply across OPG's Nuclear operations, this conclusion is applicable to all of Pickering NGS (Units 1,4 and Units 5-8).

Pickering Units 1,4

No review was performed for Pickering Units 1,4 against any version of REGDOC-2.9.1 [B.8-1] or CNSC S-296 [B.8-3]. However, Section 9.0 of Attachment 3 of the Pickering PROL Renewal Application [B.8-10] states:

> *The Environmental Program (N-PROG-OP-0006) and its supporting governing documents establish provisions for protection of the environment at OPG's nuclear facilities and continual improvement of environmental performance. This program provides Nuclear with a systems approach to managing environmental aspects in accordance with the elements of ISO 14001 Environmental Management Systems standard and the CNSC Standard S-296: Environmental Protection, Policies, Programs and Procedures at Class 1 Nuclear Facilities and Uranium Mines and Mills.*

Holding ISO 14001 (Environmental Management System) certification provides assurance that an environment policy is in place, compliance to legislation is effective, objectives are established, programs are in place to meet these objectives, and performance reviews against these objectives are conducted periodically to ensure overall continuous improvement.

Darlington NGS

Clause-by-clause reviews were performed for the Darlington ISR for both S-296 [B.8-3] and G-296 [B.8-4] with respect to the Radiological and Non-Radiological Impact on the Environment Safety Factor, as documented in References [B.8-11] and [B.8-12], respectively. The reviews examined OPG's Environmental Management System (EMS) as documented in N-PROG-OP-0006 "Environmental Management" [B.8-9]. The reviews did not identify any gaps and found that Darlington NGS was compliant with both CNSC S-296 [B.8-3] and CNSC G-296 [B.8-4]. Since compliance is demonstrated based on Governance, Programs, Procedures and Practices that apply across OPG's Nuclear operations, this conclusion is applicable to Pickering NGS (Units 1,4 and Units 5-8).

A high level intent review of REGDOC-2.9.1 was performed for Darlington, in the context of the Safety Factor for Radiological and Non-Radiological Impact on the Environment, as documented in OPG Report NK38-REP-03680-10208 R000 [B.8-13]. The review concluded that Darlington is in compliance with CNSC Regulatory Document REGDOC-2.9.1 and that no gaps were identified during the review. While the review is largely programmatic, several Darlington-specific documents were used in the assessment. One example is in measuring compliance with clause 3.2 of REGDOC-2.9.1 "EMS Scope". The review indicated that "Darlington has conducted an ecological risk assessment which concluded that radiological risks to non-human biota were not significant" [B.8-13]. This is not a gap for PSR2 since: a) similar ecological risk assessments have been performed for Pickering (e.g., References [B.8-14] and [B.8-15]), and b) the clause in question has not changed from CNSC G-296 [B.8-4], and as such, the previously assessed intent compliance for Pickering B [B.8-7] remains applicable. The same conclusion is applicable to the other Darlington-specific document examples that were used in the assessment (which relate to spill prevention, emergency preparedness, and environmental protection), since similar assessments have been performed for Pickering NGS and there have been no safety significant changes to any of the related clauses.

Section B.8.2.2 below discusses compliance for any clauses of CNSC S-296 that have changed in REGDOC-2.9.1.

B.8.2.2 Application of Post-PSR1 Reviews

The CNSC document history for REGDOC-2.9.1 [B.8-5] indicates that REGDOC-2.9.1 supersedes S-296 [B.8-3] and G-296 [B.8-4]. A comparison of REGDOC-2.9.1 (2013) to its predecessors reveals the following changes:

1. In the list of Relevant Regulations in Section 1.3 of REGDOC-2.9.1 [B.8-1], the following statement is incremental from the parallel clause 3.3 of S-296 [B.8-3]: "Other acts and regulations also apply to projects, to support environmental protection policies,

programs and procedures (refer to the References Section for details)." References added include:

a. *Migratory Birds Convention Act,* Ottawa, 1994.

b. *Fisheries Act*, Ottawa, 1985.

c. *CSA Group, Environmental Management Systems - General Guidelines on Principles, Systems and Support Techniques (Adopted ISO 14004:2004, second edition, 2004-11-15)*, CAN/CSA-ISO 14004-04 (R2009), Toronto, 2004.

d. Health Canada, Guidance documents related to Federal Contaminated Risk Assessment in Canada, Ottawa, 2010.

These changes are not safety significant and there is therefore no PSR2 gap.

2. The definition of 'licensing basis' was added to REGDOC-2.9.1, and used in the Preface to this standard. This change is not safety significant and does not result in a PSR2 gap.

Since REGDOC-2.9.1 (2013) [B.8-1] does not contain any new regulatory requirements in comparison to S-296 [B.8-3] or G-296 [B.8-4], Pickering NGS compliance is demonstrated through the original Pickering B ISR reviews [B.8-7] of S-296 [B.8-3] and G-296 [B.8-4]. This is further supported by the Darlington review in B.8.2.1 described above, where the review confirmed that there were no material changes from CNSC G-296 to REGDOC-2.9.1 and the programmatic requirements and conclusions are also applicable to Pickering.

B.8.3 Compliance Assessment Summary for Pickering PSR2

There are no PSR2 gaps for CNSC REGDOC-2.9.1 (2013) [B.8-1]. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CNSC REGDOC-2.9.1 (2013).

B.8.4 References

[B.8-1] CNSC Regulatory Document REGDOC-2.9.1, *Environmental Protection: Policies, Programs and Procedures*, September 2013.

[B.8-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.8-3] CNSC Regulatory Standard S-296, *Environmental Protection Policies, Programs and Procedures at Class I Nuclear Facilities and Uranium Mines and Mills,* March 2006.

[B.8-4] CNSC Regulatory Guide G-296, *Developing Environmental Protetion Policies, Programs and Procedures at Class I Nuclear Facilities and Uranium Mines and Mills*, March 2006.

[B.8-5] CNSC website, *Document History of REGDOC-2.9.1, Environmental Protection*, http://nuclearsafety.gc.ca/eng/acts-and-regulations/regulatory-documents/history/regdoc2-9-1.cfm, May 2016.

[B.8-6] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.8-7] OPG Report, NK30-REP-03680-00010 R000, *Pickering NGS-B Integrated Safety Review – Environment*, May 2007.

[B.8-8] National Standard of Canada CAN/CSA-ISO 14001:2004, *Environmental Management Systems – Requirements with Guidance for Use*, November 2004.

[B.8-9] OPG Nuclear Program, N-PROG-OP-0006 R018, *Environmental Management*, April 2015.

[B.8-10] OPG Letter, P-CORR-00531-03719 R000, G. Jager to M. A. Leblanc, *Application for Renewal of Pickering Nuclear Generating Station Power Reactor Operating Licence*, July 4th, 2012.

[B.8-11] OPG Report, NK38-REP-03680-10068 R000, *Evaluation of Darlington NGS Against Canadian Nuclear Safety Commission Regulatory Standard S-296, Environmental Protection Policies, Programs and Procedures at Class 1 Nuclear Facilities and Uranium Mines and Mills (March 2006)*, June 2011.

[B.8-12] OPG Report, NK38-REP-03680-10067 R000, *Evaluation of Darlington NGS Against CNSC Regulatory Guide G-296 (March 2006), Developing Environmental Protection Policies, Programs and Procedures at Class 1 Nuclear Facilities and Uranium Mines and Mills*, June 2011.

[B.8-13] OPG Report, NK38-REP-03680-10208 R000, *Code Refresh Review of CNSC REGDOC 2.9.1 (2013) Environmental Protection: Policies Programs and Procedures*, February 2014.

[B.8-14] OPG Report, NK30-REP-07701-00005, *Ecological Risk Assessment Technical Support Document, Refurbishment and Continued Operation of Pickering B Nuclear Generating Station Environmental Assessment*, June 2007.

[B.8-15] OPG Report, P-REP-07010-10012 R000, *Environmental Risk Assessment Report for Pickering Nuclear*, January 2014.

B.9 CNSC REGDOC-2.10.1 (2014), "Nuclear Emergency Preparedness and Response"

B.9.1 Background

The following paraphrase from the preface and introduction of CNSC REGDOC-2.10.1 [B.9-1], provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *Prevention of nuclear emergencies at Canadian nuclear facilities is the responsibility of the licensees. The CNSC regulates the Canadian nuclear industry in order to prevent unreasonable risk to the environment, the health and safety of persons, and national security. Mitigation of nuclear emergencies aims at ensuring that equipment, such as hydrogen recombiners, or procedures, such as emergency operating procedures, are put in place before a nuclear emergency to reduce the potential magnitude or impact of the hazard*
>
> *REGDOC-2.10.1 sets out the emergency preparedness requirements and guidance of the CNSC related to the development of emergency measures for licensees and licence applicants of Class I nuclear facilities and uranium mines and mills.*
>
> *REGDOC-2.10.1 lists and discusses the requirements and guidance that licence applicants and licensees shall implement and consider in the design of their emergency preparedness program (EP program)…*
>
> *For existing facilities: The requirements contained in this document do not apply unless they have been included, in whole or in part, in the licence or licensing basis.*

REGDOC-2.10.1 (2014) is directly relevant to Safety Factor 13 (Emergency Planning).

REGDOC-2.10.1 (2014) is the first version of this REGDOC and supersedes:

- G-225, "Emergency Planning at Class I Nuclear Facilities and Uranium Mines and Mills" (2001) [B.9-2].
- RD-353, "Testing the Implementation of Emergency Measures" [B.9-3].

Compliance with REGDOC-2.10.1 is not currently a licence requirement for Pickering NGS (in accordance with PROL 48.02/2018) per the R04 Pickering Licence Conditions Handbook (LCH) [B.9-4]. However, RD-353 (2008) is a licence requirement as indicated in Appendix C2 of the R04 Pickering LCH [B.9-4].

It is noted that "REGDOC-2.10.1, 'Nuclear Emergency Preparedness and Response,' version 2 was published in February 2016. While there will be no impact on any REGDOC-2.10.1 requirements and guidance, this update aligns with version 2 of REGDOC-2.3.2, '*Accident Management*'– which had been created to clarify requirements" [B.9-5]. Since Version 2 was issued after the PSR2 freeze date, the 2014 version has been assessed as part of PSR2.

According to Reference [B.9-6], "The new regulatory documents address lessons learned from the Fukushima nuclear accident and incorporate international post-Fukushima best practices and guidance for use by current and future Canadian licensees."

As identified in Reference [B.9-7], the Pickering PSR2 review of REGDOC-2.10.1 (2014) is an Incremental review. PSR2 Incremental review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.

- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.9.2 Compliance Assessment for Pickering PSR2

B.9.2.1 Application of PSR1 Reviews

The versions of REGDOC-2.10.1 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

The first edition of REGDOC-2.10.1 was issued in October 2014 after the completion of the Pickering B ISR. Therefore, no code review was performed for Pickering Units 5-8 against REGDOC-2.10.1.

CNSC G-225 [B.9-2] was reviewed as part of the Pickering B ISR, as documented in [B.9-8]. The review included a clause by clause review of CNSC G-225 in relation to the then current OPG Emergency Response Program [B.9-9] and its relevant implementing documentation. As documented in [B.9-8], the OPG program was found to be in direct compliance with all requirements, with only one exception. Gap 13-037 was raised against Clause 5.8 of G-225 associated with the Filtered Air Discharge System (FADS). Specifically, it was noted that there was an outstanding CNSC Action Item (AR# 28071493 – AI# 2006-8-02) related to Post-Accident Radiological Monitoring System (PARMS) iodine sampling not being consistent with design accuracy requirements, in terms of iodine activity measurement. In [B.9-10] (Gap 13-037 in Appendix F), OPG outlined a series of actions that were accepted by the CNSC in order to bring this issue to resolution. OPG also undertook that, upon completion of these actions, a determination would be made as to the validity of the resolution beyond the current design life of the station. The actions completed by OPG are documented in [B.9-11] in which closure of the Action Item was requested on the grounds that further action to address this issue was not feasible, and not necessary since it would be of limited value. The CNSC agreed and closed Action Item 2006-8-02 [B.9-12] and in so doing, stated that no further work is required. The technical basis for closure of this Action Item is not related to the design life of the station, hence, this resolution is considered applicable for extended operation of the Pickering station beyond 2020.

Although there is not a one-to-one correspondence, the subject matter of Clause 5.8 of CNSC G-225 maps generally to the content of Clauses 2.2.6 and 2.3.2 of REGDOC-2.10.1. As such, the previously identified gap relating to CNSC AI# 2006-08-02 is considered to be relevant in the context of REGDOC-2.10.1, as well as its resolution. Therefore, this does not represent a PSR2 gap.

No code review was performed for Pickering Units 5-8 against CNSC RD-353.

Pickering Units 1,4

The first edition of REGDOC-2.10.1 was issued in October 2014 after the completion of the Pickering A Return-to-Service. Therefore, no code review was performed for Pickering Units 1,4 against REGDOC-2.10.1.

No code review was performed for Pickering Units 1,4 against CNSC G-225. However, in the review done for Pickering Units 5-8 [B.9-8], compliance was assessed based on Governance, Programs, Procedures and Practices that apply either across OPG's Nuclear operations or for the Pickering site. Hence, the conclusions are applicable to all of Pickering NGS (Units 1,4 and Units 5-8). This includes the identified gap and its resolution relating to the FADS, which is a shared system between Pickering Units 1,4 and Units 5-8.

No code review was performed for Pickering Units 1,4 against CNSC RD-353.

Darlington NGS

The first edition of REGDOC-2.10.1 was issued in October 2014 after the completion of the code reviews for the Darlington ISR. Therefore, no code review was performed for Darlington against REGDOC-2.10.1.

CNSC G-225 [B.9-2] was reviewed as part of the Darlington ISR, as documented in [B.9-13] and [B.9-14]. The review included a high level review of CNSC G-225 in relation to the then current OPG Emergency Response Program [B.9-15] and its relevant implementing documentation. As documented in [B.9-14], the OPG program was found to comply with all requirements, with only one exception. Similar to the Pickering B ISR, the gap was related to Clause 5.8 of CNSC G-225 (iodine activity measurement). In the final Darlington ISR report [B.9-16], this was identified as ISR Issue D067. The proposed resolution was to monitor work under the outstanding CNSC AI# 2006-8-02 and to assess if the resolution is sufficient for refurbishment. As noted in [B.9-16], this Action Item is common for both Pickering and Darlington stations. Hence, the resolution of this issue as discussed above for Pickering Units 5-8 is applicable to Darlington. As such, ISR Issue D067 was reclassified as an Acceptable Deviation [B.9-17].

No code review was performed for Darlington against CNSC RD-353.

B.9.2.2 Application of Post PSR1 Reviews

As part of the 2012 PROL renewal application for Pickering [B.9-18], a report on the Nuclear Emergency Plan was presented, in which OPG stated the conclusion that:

> *Pickering NGS is in full compliance with CNSC RD-353, Testing and Implementation of Emergency Measures.*

This, in combination with the discussion of CNSC G-225 in the preceding section, provides confirmation that for Pickering, there are no gaps in relation to the CNSC documents that have now been superseded by REGDOC-2.10.1.

As part of the most recent PROL renewal application process for Darlington [B.9-19], OPG provided a detailed status report with respect to their Emergency Management Program, which included the following commitment:

> *OPG will submit a transition plan for compliance with REGDOC-2.10.1 to CNSC staff by September 30, 2015, and will be fully compliant by December 31, 2018.*

The transition plan was subsequently provided to the CNSC in [B.9-20], in which OPG stated that a gap analysis (documented in [B.9-21]) had been conducted to identify the required steps to transition to REGDOC-2.10.1. All requirements for which no gaps were identified are either generically applicable to the OPG emergency response program or, if station-specific aspects exist, these are met for both Pickering and Darlington. For the gaps identified, the specific actions to implement the transition plan are summarized in the table below. Regulatory Management Action Request AR # 28184526 was initiated to track completion of the transition plan by Q3 2017.

No.	REGDOC-2.10.1 Clause	OPG Action
1	2.1 (2)	OPG is undertaking to compile the applicable documentation of the planning basis considerations and to incorporate into EP governance in order to demonstrate full compliance
2	2.2.3 (5)	OPG is undertaking to review and develop a process for real time access to offsite monitoring data for the offsite authority and the CNSC
3	2.2.3 (7) (8)	OPG is undertaking to identify and revise applicable procedures such that the CNSC is included with the offsite authority in being provided this information
4	2.2.4 (2)	OPG is undertaking to formally compile existing agreements and reference in the emergency plan

No.	REGDOC-2.10.1 Clause	OPG Action
5	2.2.4 (additional requirement 2)	OPG is undertaking to update the Darlington Evacuation Time Estimate study. Subsequent to the target completion date, OPG will revise its governance to include the requirement to maintain the evacuation time estimates as required.
6	2.2.6 (11)	OPG is undertaking to formally compile existing arrangements and reference in EP governance
7	2.2.9 (2) (3)	Although changes to the emergency plan already follow a formal process to ensure continued effectiveness, OPG is undertaking to develop and document a validation process to demonstrate compliance with this requirement. OPG proposes to work with the CNSC to ensure that the applicable Emergency Response plans and procedures are reflected in the LCH.
8	2.3.4 (1) to (8)	OPG is in the process of stocking and distributing Potassium Iodide (KI) pills to meet this requirement by the end of 2015. OPG is undertaking to document associated processes, including lessons learned resulting from the stocking and distribution of KI pills.

Actions 1, 2, 3, 4, 6 and 7 relate entirely to Governance, Programs, Procedures and Practices that apply across OPG's nuclear operations. Once these actions are completed for the Darlington transition plan, Pickering will also be in compliance with these clauses of REGDOC-2.10.1 since the programs apply to both stations. Action 1 is to produce documentation that compiles material that already exists in other documents. Actions 2 and 3 relate to enhancing processes to share information with the CNSC during an accident. Actions 4, 6 and 7 are to produce documentation that describes arrangements or processes that OPG already follows. The absence of the above would not affect the manner in which OPG would manage the response to an emergency and is not safety significant. Therefore, the open status of these actions does not result in a PSR2 gap.

Actions 5 and 8 are safety significant and contain elements that are station-specific and apply to Pickering. In relation to Action 5, the Pickering Evacuation Time Estimate study has been updated as documented in [B.9-22] and made available to the public [B.9-23]. Updating OPG governance to ensure that the Pickering Evacuation Time Estimate study is maintained up to date is in progress. With respect to Action 8 dealing with public distribution of KI pills, in Section 2.3.3 of [B.9-24] the CNSC made note of actions underway at the time by OPG that would address this action for Pickering. Section 1.7 of the Licence Conditions Handbook for Pickering [B.9-4] also outlines the CNSC's expectations for the completion of these actions. The distribution of KI pills and associated public information providing instructions on their proper use was completed in 2015 for the primary distribution zone around the Pickering station. Details of the program are publically available in [B.9-25]. Updating OPG governance to define how the KI pill program will be sustained is in progress. The need to complete Actions 5 and 8 from the transition plan in [B.9-20], as they relate to Pickering and taking into account Pickering life extension, is identified as **PSR2 REGDOC-2.10.1 Gap #1.**

Note that REGDOC-2.10.1 is programmatic in nature and, except for the items noted under **PSR2 REGDOC-2.10.1 Gap #1**, the status of OPG's compliance with it is not impacted by the possibility of continued operation of Pickering beyond 2020.

B.9.3 Compliance Assessment Summary for Pickering PSR2

There is one PSR2 REGDOC-2.10.1 gap which relates to Safety Factor 13 (Emergency Planning):

1. OPG has completed a gap analysis for transition to REGDOC-2.10.1 and has developed an action plan to achieve compliance. The transition plan that OPG has committed in order to bring Darlington into compliance with REGDOC-2.10.1 applies across the nuclear fleet and will also bring Pickering into compliance. Updating OPG governance to ensure that the Pickering Evacuation Time Estimate study is maintained and to define how the Potassium Iodide (KI) pill program will be sustained is in progress. As these two actions are not yet complete, this is identified as a PSR2 gap.

B.9.4 References

[B.9-1] CNSC Regulatory Document REGDOC-2.10.1, *Nuclear Emergency Preparedness and Response*, 2014.

[B.9-2] CNSC Regulatory Guide G-225, *Emergency Planning at Class I Nuclear Facilities and Uranium Mines and Mills*, August 2001.

[B.9-3] CNSC Regulatory Document RD-353, *Testing the Implementation of Emergency Measures*, October 2008.

[B.9-4] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 23, 2015.

[B.9-5] CNSC Website, *Document History of REGDOC-2.10.1, Nuclear Emergency Preparedness and Response*. http://nuclearsafety.gc.ca/eng/acts-and-regulations/regulatory-documents/history/regdoc2-10-1.cfm, 22 Feb. 2016.

[B.9-6] CNSC Website, *Archived - CNSC Publishes Two Documents to Enhance Regulatory Requirements in Nuclear Emergency Management, including the Pre-distribution of KI Pills*. http://news.gc.ca/web/article-en.do?nid=892059, 10 Oct. 2014.

[B.9-7] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.9-8] OPG Report NK30-REP-03680-00009 R000, *Pickering NGS-B Integrated Safety Review – Safety Factor for Emergency Planning*, April 13, 2007.

[B.9-9] OPG Nuclear Program, N-PROG-RA-0001 R007, *Consolidated Nuclear Emergency Plan*, 2005.

[B.9-10] OPG Report, NK30-REP-03680-00016 R000, *OPG Response to CNSC Comments on Pickering NGS-B Integrated Safety Review – Plant Design, Safety Analysis, Safety Performance, Ageing and Equipment Qualification Safety Factors and Discrepancy Resolutions*, September 22, 2009.

[B.9-11] OPG Letter, G. Jager to M. Santini, NK30-CORR-00531-05978, *Pickering B – Type II System Inspection – Filtered Air Discharge System – Request to Close CNSC Action Item 2006-8-02*, November 23, 2011.

[B.9-12] CNSC Letter, M. Santini to G. Jager, NK30-CORR-00531-06381, *Pickering NGS-B – Type II System Inspection: Filtered Air Discharge System – Closure of CNSC Action Item 2006-8-02 (RIB #2413)*, August 24, 2012.

[B.9-13] OPG Report, NK38-REP-03680-10085 R000, *Review of CNSC G-225 (August 2001) Emergency Planning at Class I Nuclear Facilities and Uranium Mines and Mills*, March 2010.

[B.9-14] OPG Report, NK38-REP-03680-10082 R002, *Darlington NGS Integrated Safety Review – Emergency Planning Safety Factor Report*, October 5, 2011.

[B.9-15] OPG Nuclear Program, N-PROG-RA-0001 R009, *Consolidated Nuclear Emergency Plan*, 2010.

[B.9-16] OPG Report, NK38-REP-03680-10104 R000, *Darlington NGS Integrated Safety Review (ISR) - Final ISR Report*, October 18, 2011.

[B.9-17] OPG Report, NK38-REP-00770-0421412 R001, *Nuclear Refurbishment Issue Resolution Form – Darlington – Issue #D067 – Provisions for Post-Accident Sampling*, February 2013.

[B.9-18] OPG Letter, G. Jager to M.A. Leblanc, P-CORR-00531-03719, *Application for Renewal of Pickering Nuclear Generating Station Power Reactor Operating Licence*, July 4, 2012.

[B.9-19] OPG Letter, B. Duncan to M. Leblanc, NK38-CORR-00531-16796, *Darlington NGS – Notice of Participation for the CNSC Public Hearing 2015-H-04 – Application for the Renewal of Darlington Nuclear Generating Station Power Reactor Operating Licence*, July 2, 2015.

[B.9-20] OPG Letter, B. Duncan to F. Rinfret, NK38-CORR-00531-17593, *Darlington NGS – Transition Plan for Regulatory Document Nuclear Emergency Preparedness and Response (REGDOC – 2.10.1)*, September 30, 2015.

[B.9-21] OPG Self Assessment Report, NO15-001449-SA, *CNSC REGDOC 2.10.1 (October 2014) vs. N-PROG-RA-0001, CNEP*, February 23, 2016.

[B.9-22] OPG Report, P-REP-03490-00079 R000, *Pickering NGS Development of Evacuation Time Estimates*, April 22, 2016.

[B.9-23] OPG Website, *Safety At Pickering: Evacuation Time Estimates Report*. http://www.opg.com/generating-power/nuclear/stations/pickering-nuclear/Pages/safety-at-pickering.aspx, Accessed November 7, 2016.

[B.9-24] CNSC Letter, L. Levert to B. Phillips, P-CORR-00531-04237, *May 7, 2014 Public Hearing*, March 24, 2014.

[B.9-25] OPG Website, *Pickering Nuclear: KI Pill Distribution*. http://www.opg.com/generating-power/nuclear/stations/pickering-nuclear/Pages/pickering-nuclear.aspx, Accessed November 7, 2016.

B.10 CNSC G-323 (2007), "Ensuring the Presence of Sufficiently Qualified Staff at Class I Nuclear Facilities-Minimum Shift Complement"

B.10.1 Background

The following paraphrased from the purpose and scope of CNSC G-323 (2007) [B.10-1] provides a brief overview of the purpose of this Standard and the requirements expressed therein:

> *Adequate staffing levels are required to respond to the most resource-intensive conditions under all operating states, including normal operations, anticipated operational occurrences, design basis accidents, and emergencies. The purpose of CNSC G-323 is to assist Class I nuclear facility licensees and applicants for a Class I nuclear facility licence to demonstrate to the CNSC that they will ensure the presence of a sufficient number of qualified workers to carry on the licensed activity safely.*
>
> *CNSC G-323 sets out information related to the staffing of a Class I nuclear facility that should typically be included in an application for the issuance, renewal, amendment, or replacement of a licence to operate a facility. The guide sets out the key factors that CNSC staff will take into account when assessing whether the licensee has made, or the application will make, adequate provision for ensuring the presence of a sufficient number of qualified staff.*

CNSC G-323 is relevant to Safety Factor 10 (Organization, Management System and Culture) and Safety Factor 12 (Human Factors).

CNSC G-323 is identified in Appendix E.2 of the R04 Pickering Licence Conditions Handbook (LCH) [B.10-2] as "Guidance or Criteria". As indicated in Sections 3.1 and 3.2 of the LCH, G-278 provides the recommended approach for defining the minimum shift complement and sets out the key factors that CNSC staff will take into account when assessing whether there are adequate provisions for ensuring the presence of a sufficient number of qualified staff. G-323 (2007) is the first edition of this Regulatory Guide.

The results of PSR1 G-323 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.10.2. As identified in Reference [B.10-3], the Pickering PSR2 review of CNSC G-323 (2007) is an Incremental review. PSR2 Incremental review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- <u>Compliance:</u> Compliance indicates that the change in the safety requirement, per the topical review, is met.
- <u>Gap</u>: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.10.2 Compliance Assessment for Pickering PSR2

B.10.2.1 Application of PSR1 Reviews

The versions of G-323 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

G-323 was not reviewed as part of the Pickering B ISR. G-323 was also not reviewed for Pickering 1,4 Return to Service. As discussed below, Pickering NGS compliance with G-323 has been demonstrated through completion of a clause-by-clause review performed for the Darlington ISR review of G-323, the findings of which are programmatically applicable to Pickering NGS. Further, Pickering NGS compliance demonstration includes submission to, and acceptance by, the CNSC of P-INS-09100-00003, "Pickering Minimum Shift Complement" [B.10-4] as discussed below.

Darlington NGS

G-323 (2007) [B.10-1], which is the latest version of the Regulatory Guide, was reviewed as part of the Darlington ISR and documented in OPG Reports NK38-REP-03680-10094 R000 [B.10-5] and NK38-REP-03680-10094-ADD-001 R000 [B.10-6]. This clause-by-clause review against G-323 showed that OPG governing documents adequately address the intent of the majority of the clauses in the Regulatory Guide. Four gaps were identified which were assigned to Darlington ISR Issue D270 [B.10-7] and tracked under Action Request (AR) 28112704. The gaps are as follows:

1. *Clause 5.1.1 of CNSC Guide G-323 expects that the minimum staff complement is determined by the licensee through a systematic analysis, and the most resource-intensive condition for each operation state should be analyzed. Gap 01355 indicates that there is no explicit requirement in OPG governance requiring systematic analysis to determine minimum shift complement.*

2. *Clause 5.1.2 of CNSC Guide G-323 identifies the requirement of validating minimum staff requirement under all operating states including normal operation, Anticipated Operational Occurrences, Design Basis Accidents, and / or emergencies. Gap 01356 indicates that Manual N-MAN-06700-10001 R000 "Human Factors Validation Planning and Methods" does not provide details of the inputs required for validation such as operating states, range of scenarios or the objectives of the validation such as those identified in CNSC G-323.*

3. *Clause 5.1.2.1 of CNSC Guide G-323 identifies validation scenarios that should include the most resource-intensive events that could affect more than one unit, such as seismic events, loss of off-site power, steam line or feeder water line breaks. Gap 01357 indicates that Manual N-MAN-06700-10001 R000 does not provide details of the inputs required for validation such as operating states, range of scenarios or the objectives of the validation such as those identified in CNSC G-323.*

4. *Clause 5.1.2.2 of CNSC Guide G-323 specifies that the validation exercises should demonstrate that relevant procedures can be implemented in a timely manner, there is effective and timely response to Anticipated Operational Occurrences, Design Basis Accidents and emergencies, the facility can be effectively monitored, controlled and stabilized, there is effective communication and coordination, workers are able to maintain awareness of facility conditions, the physical and mental workload of minimum staff complement is achievable and all safety–critical human actions are achievable. Gap 01358 indicates that Manual N-MAN-06700-10001 R000 does not provide details of the inputs required for validation such as operating states, range of scenarios or the objectives of the validation such as those identified in CNSC G-323.*

Since the identified gaps relate to OPG governance, they are applicable to PSR2. Darlington ISR Issue D270 summarizes the gaps above as follows [B.10-7]:

> *The code requires the nuclear facility to use systematic analysis to determine minimum staff complement and validate minimum staff complement under all operating states. Reviews indicate that the current OPG governance does not have requirements for systematic analysis to determine minimum shift complement and that validation of minimum shift complement does not consider required inputs such as all operating states, the range of scenarios or the objectives to be achieved.*

AR # 28112704 was opened to address issue D270. Initiative Plan MA-08, "Days Based Maintenance", was developed to address requirements specified in G-323 at Darlington and at Pickering NGS and thus to address issue D270. Following completion of the actions associated with development of the validation plan, and submission of the minimum shift complement basis document to the CNSC, the issue was assigned an Acceptable Deviation on the basis that the minimum shift complement as specified in D-PROC-OP-0009 R008 "Station Shift Complement" [B.10-8], and D-INS-09260-10001 R002 "Duty Crew Minimum Complement Assurance" [B.10-9] were accepted by the CNSC.

As the Days Based Maintenance Initiative included Pickering, AR # 28112704 also included actions related to the preparation, verification and validation of the Pickering minimum shift complement document [B.10-4]. This AR has been closed with all items completed. Changes in the minimum shift complement as a result of these actions are identified in OPG letter to the CNSC, P-CORR-00531-03710 "Pickering A and B Request for Licence Amendments – Minimum Shift Complement" [B.10-10], wherein it is noted that: "The validation methodology has been documented and validation exercises have been observed by CNSC Staff."

B.10.2.2 Application of Post-PSR1 Reviews

Action Items 2004-4-09 and 2004-8-10 had been previously raised in 2004 to track OPG's progress in analyzing the minimum shift complement required for Pickering NGS. Closure of these action items was requested in OPG Letter P-CORR-00531-03585 [B.10-11], wherein the analysis in accordance with G-323 was submitted through an earlier revision (R005) of P-INS-09100-00003, "Pickering Minimum Shift Complement" [B.10-4]. The analysis methodology, analysis and validation reports for the minimum shift complement are listed as references in this document. As per the response from the CNSC [B.10-12]:

CNSC staff agrees with the process used to conduct the systematic analysis and integrated validation of the minimum shift complement at Pickering NGS.

P-INS-09100-00003 [B.10-4] is now referenced in Section 3.2 of the Pickering LCH [B.10-2] as the reference for compliance verification. Therefore, there is no gap for Pickering PSR2.

As discussed above, compliance with CNSC G-323 (2007) is demonstrated through the CNSC's acceptance of the process used to conduct the systematic analysis and integrated validation of the minimum shift complement at Pickering NGS [B.10-12]. There have been no revisions to CNSC G-323 since first issuance in 2007, and no other reviews performed since the Darlington ISR review as none were required. There are no gaps for Pickering NGS compliance with CNSC G-323.

B.10.3 Compliance Assessment Summary for Pickering PSR2

There are no PSR2 gaps for CNSC G-323 (2007) [B.10-1]. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CNSC G-323 (2007).

B.10.4 References

[B.10-1] CNSC Regulatory Guide G-323, *Ensuring the Presence of Sufficient Qualified Staff at Class I Nuclear Facilities – Minimum Staff Complement*, July 2007.

[B.10-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.10-3] OPG Report P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.10-4] OPG Nuclear Instruction, P-INS-09100-00003 R009, *Pickering Minimum Shift Complement*, December 2014.

[B.10-5] OPG Report, NK38-REP-03680-10094 R000, *Review of CNSC G-323 (July 2007), Ensuring the Presence of Sufficient Qualified Staff at Class I Nuclear Facilities – Minimum Staff Complement Guide for Darlington Integrated Safety Review*, August 2011.

[B.10-6] OPG Report, NK38-REP-03680-10094-ADD-001 R000, *Addendum to the CNSC G-323 Code Review Report for Darlington ISR*, January 2014.

[B.10-7] OPG Report, NK38-REP-03680-10104 R000, *Darlington NGS Integrated Safety Review (ISR) - Final ISR Report*, October 2011.

[B.10-8] OPG Procedure, D-PROC-OP-0009 R008, *Station Shift Complement*, March 2007.

[B.10-9] OPG Instruction, D-INS-09260-10001 R002, *Duty Crew Minimum Complement Assurance*, May 2006.

[B.10-10] OPG Letter, P-CORR-00531-03710 R000, G. Jager to M. Leblanc, *Pickering A and B - Request for Licence Amendments – Minimum Shift Complement*, October 16, 2012.

[B.10-11] OPG Letter, P-CORR-00531-03585 R000, G. Jager and P. Pasquet to T. E. Schaubel, *Pickering A and B – Request for Concurrence with Minimum Shift Complement Document, P-INS-09100-00003 Revision 5 – Action Items 2004-4-09, 2004-8-10 and 2006-4-01*, February 10, 2011.

[B.10-12] CNSC Letter, E-Docs # 3752906/4.01.02, OPG File No. P-CORR-00531-03640 R000, M. Santini to G. Jager, *OPG Request for Concurrence with Minimum Shift Complement documents, P-INS-09100-00003 and P-INS-09260-0008, Action Items 2004-4-09, 2004-8-10 and 2006-4-01*, August 9, 2011.

B.11 CNSC G-278 (2003), "Human Factors Verification and Validation Plans"

B.11.1 Background

The following paraphrased from the purpose and scope of CNSC G-278 (2003) [B.11-1] provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *The purpose of CNSC G-278 is to assist licensees and licence applicants in planning for human factors verification and validation activities. Such activities help satisfy certain regulatory requirements by demonstrating that licensees and applicants have made adequate provision for the protection of the environment and the health and safety of persons.*
>
> *CNSC G-278 describes the elements of effective human factors verification and validation planning for Class I nuclear facilities and uranium mines and mills. A suggested format for documenting these elements is presented in the guide as a Human Factors Validation and Verification Plan. However, equivalent documentation that meets the objectives and intent of the guide is also acceptable.*

CNSC G-278 (2003) is applicable to Safety Factor 1 (Plant Design) and Safety Factor 12 (Human Factors).

CNSC G-278 is identified in Appendix E.2 of the R04 Pickering Licence Conditions Handbook (LCH) [B.11-2] as "Guidance or Criteria". As indicated in Sections 3.1 and 3.2 of the LCH, G-278 provides additional recommendations and guidance for the development of human performance programs, for verification and validation of the minimum shift complement, and for considering human factors in design programs. CNSC G-278 (2003) is the first edition of this Regulatory Guide.

The results of PSR1 G-278 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.11.2. As identified in Reference [B.11-3], the Pickering PSR2 review of CNSC G-278 (2003) is an Incremental review. PSR2 Incremental review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard (L/R/C/S) on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.
- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.11.2 Compliance Assessment for Pickering PSR2

B.11.2.1 Application of PSR1 Reviews

The versions of G-278 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

A clause-by-clause review of CNSC G-278 (2003) [B.11-1] was performed for the Pickering B ISR. The results are documented in OPG Report NK30-REP-03680-00014 R000 [B.11-4] as an Addendum to the Management Safety Factor Report (which included Human Factors-related L/R/C/S reviews and Review Tasks), wherein indirect compliance with the majority of clauses was determined.

Two clauses were assessed as Acceptable Deviations on the basis of the technical assessment. The Acceptable Deviation findings are as follows:

1. Compliance with Section 6.3 and its subsections was assigned an Acceptable Deviation based on the exception for simple modifications where it was assessed that explicitly documenting each validation element is not judged necessary. Where appropriate, these elements are addressed in either a Human Factors Engineering Program Plan or a separate Verification & Validation plan, as recommended in COG-92-444, COG-92-445 and NUREG-0711.

2. When using the Human Factors (HF) Worksheet, HF Validation includes completion of two questions in Section 6: HF Validation. Sources of information used to complete this section are documented on the worksheet. Given that the HF Worksheet is used for simple straightforward modifications, the elements used in the validation process are limited to a recommended set of activities.

These Acceptable Deviations are not impacted by Pickering NGS operation beyond 2020. Further, an assessment of G-278 (2003) was also performed for the Darlington ISR, the findings of which are programmatically applicable to Pickering NGS, as discussed below.

Pickering Units 1,4

No code review was performed for Pickering Units 1,4 against any version of G-278. However, as discussed above, the Pickering B ISR reviewed the latest (and only) version of G-278 and the conclusions of that work are applicable across the OPG Nuclear fleet, including Pickering NGS (Units 1,4 and Units 5-8). Further, the Darlington ISR review of G-278 is also programmatically applicable to Pickering NGS, as discussed below.

Darlington NGS

CNSC G-278 (2003) [B.11-1] (the same version as used for the Pickering B ISR) was reviewed as part of the Darlington ISR and documented in Reference [B.11-5]. The review, which is programmatically applicable across OPG's Nuclear fleet and to Pickering NGS, did not identify any gaps and found that Darlington NGS was compliant with CNSC G-278 (2003) [B.11-1]. In reference to the Acceptable Deviation conclusions for Section 6.3 of G-278 as determined in the Pickering B ISR review [B.11-4], the review conducted for Darlington [B.11-5] states: "The content of the verification and validation activities in the worksheet addresses the requirements for verification and validation as stated in G-278. However, the verification and validation activities required during completion of the Human Factors Worksheet do not need to be completed by a qualified Human Factors Specialist as the worksheet is completed for minor or uncomplicated modifications where HF Specialist input is not required." The Darlington ISR Report concluded that based on a high-level intent review, OPG's Engineering Change Control process (and related HF engineering processes) comply with the intent of G-278. Since compliance against CNSC G-278 is based on Governance, Programs and Procedures that apply across OPG's Nuclear operations, the Darlington ISR conclusions are applicable to Pickering PSR2.

B.11.2.2 Application of Post-PSR1 Reviews

The version of CNSC G-278 assessed in the Pickering B and Darlington ISRs is the most recent version of this guidance document. No additional code reviews have been performed as none were required.

B.11.3 Compliance Assessment Summary for Pickering PSR2

There are no PSR2 gaps for CNSC G-278 (2003) [B.11-1]. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CNSC G-278 (2003).

B.11.4 References

[B.11-1] CNSC Regulatory Guide G-278, *Human Factors Verification and Validation Plans*, June 2003.

[B.11-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.11-3] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.11-4] OPG Report, NK30-REP-03680-00014 R000, *Pickering NGS-B Integrated Safety Review – Management Addendum*, August 2007.

[B.11-5] OPG Report, NK38-REP-03680-10044 R000, *Review of CNSC G-278 (June 2003), Human Factors Verification and Validation Plans for Darlington Integrated Safety Review*, June 2011.

B.12 CNSC G-276 (2003), "Human Factors Engineering Program Plans"

B.12.1 Background

The following paraphrased from the purpose and scope of CNSC G-276 (2003) [B.12-1] provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *The purpose of CNSC G-276 is to assist licensees and licence applicants in developing human factors engineering program planning documentation that demonstrates how human factors considerations are incorporated into activities licensed by the CNSC. Such considerations help satisfy certain regulatory requirements by demonstrating that licensees and applicants have made adequate provision for health, safety and protection of the environment.*
>
> *CNSC G-276 describes the elements of effective human factors engineering program planning documentation for Class I nuclear facilities and uranium mines and mills. A suggested documentation format is presented in the guide as a Human Factors Engineering Program Plan. However, equivalent documentation that meets the objectives and intent of the guide is also acceptable.*

CNSC G-276 is applicable to Safety Factor 1 (Plant Design) and Safety Factor 12 (Human Factors).

CNSC G-276 is identified in Appendix E.2 of the R04 Pickering Licence Conditions Handbook (LCH) [B.12-2] as "Guidance or Criteria". As indicated in Section 3.1, and Section 6.1 of the LCH, G-276 provides additional guidance for considering human factors in design programs, and for the preparation of human performance programs [B.12-2]. CNSC G-276 (2003) is the first edition of this Regulatory Guide.

The results of PSR1 G-276 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.12.2. As identified in Reference [B.12-3], the Pickering PSR2 review of CNSC G-276 (2003) is an Incremental review. PSR2 Incremental review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.
- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.12.2 Compliance Assessment for Pickering PSR2

B.12.2.1 Application of PSR1 Reviews

The versions of G-276 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

A clause-by-clause review of CNSC G-276 (2003) [B.12-1] was performed for the Pickering B ISR Management System Safety Factor, as discussed in OPG Report NK30-REP-03680-00008 [B.12-4]. The results of the review are documented in an Addendum to the Management System Safety Factor Report [B.12-5], wherein indirect compliance with each applicable clause was demonstrated through the Human Factors Engineering screening process within the Engineering Change Control process, and the subsequent preparation of Human Factors Engineering Program Plans and Human Factors Worksheets. The assessment concluded that compliance has been demonstrated with the intent of the code based on the high level intent review. No gaps were identified.

Pickering Units 1,4

No review was performed for Pickering Units 1,4 against G-276. However, as discussed above, the Pickering B ISR reviewed the latest (and only) version of G-276 and the conclusions of that work are applicable across the OPG Nuclear fleet, including Pickering Units 1,4 and Units 5-8. Further, the Darlington ISR review of G-276 is also programmatically applicable to Pickering NGS, as discussed below.

Darlington NGS

CNSC G-276 [B.12-1] (the same version used for the Pickering B ISR) was reviewed as part of the Darlington ISR and documented in Reference [B.12-6]. The review concluded that "based on a programmatic, high-level intent review, OPG's engineering change control process and related Human Factors Engineering processes comply with the intent of G-276". No gaps were identified and Darlington NGS was found to be compliant with CNSC G-276 (2003).

Since compliance against CNSC G-276 is based on Governance, Programs and Procedures that apply across OPG's Nuclear operations, the Darlington ISR conclusions are applicable to Pickering PSR2.

B.12.2.2 Application of Post PSR1 Reviews

The version of CNSC G-276 assessed in the Pickering B and Darlington ISRs is the most recent version of this guidance document. No additional code reviews have been performed as none were required.

B.12.3 Compliance Assessment Summary for Pickering PSR2

There are no PSR2 gaps for CNSC G-276 (2003) [B.12-1]. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CNSC G-276 (2003).

B.12.4 References

[B.12-1] CNSC Regulatory Guide G-276, *Human Factors Engineering Program Plans*, June 2003.

[B.12-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.12-3] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.12-4] OPG Report, NK30-REP-03680-00008 R002, *Pickering NGS-B Integrated Safety Review – Management*, September 2009.

[B.12-5] OPG Report, NK30-REP-03680-00014 R000, *Pickering NGS-B Integrated Safety Review – Management Addendum*, August 2007.

[B.12-6] OPG Report, NK38-REP-03680-10043 R000, *Review of CNSC G-276 (June 2003), Human Factors Engineering Program Plans for Darlington Integrated Safety Review*, June 2011.

B.13 S.C. 1997, C.9 (Amended in February 2015), "Nuclear Safety and Control Act"

B.13.1 Background

The following paraphrased from the Nuclear Safety and Control Act (NSCA) (Amended 2015) [B.13-1] provides a brief overview of the purpose of this Act and the information expressed therein:

> *The purpose of the act is to provide for the limitation of the risks to national security, the health and safety of persons and the environment that are associated with the development, production and use of nuclear energy and the production, possession and use of nuclear substances, prescribed equipment and prescribed information. In addition, the act is to provide for the implementation in Canada of measures to which Canada has agreed respecting international control of the development, production and use of nuclear energy, including the non-proliferation of nuclear weapons and nuclear explosive devices.*
>
> *The act does this by establishing the Canadian Nuclear Safety Commission (CNSC) and defines its objectives, members, employees, power, record and reporting requirements, offences, punishments, and penalties.*

The NSCA is relevant to Safety Factor 10 (Organization, Management System and Safety Culture). The NSCA was last amended on February 26, 2015. Sections that have been amended include a citation to the amending Act or regulation. Consolidation of the amendments was published in April 2016, which includes some amendments not in force. Section 80 of the Act was amended by S.C. 2015 c. 3, s.136 [B.13-2] to reference the 1996 AECB Cost Recovery Fees Regulations instead of the 1994 Regulations. Other amendments have been included as previous consolidated versions are published, and can be found on the Government of Canada Justice Laws Website [B.13-3].

Per Section 2.1 of the R04 Pickering Licence Conditions Handbook (LCH) [B.13-4]:

> *Paragraphs 3(1)(k) and 15(a) and (b) of the General Nuclear Safety and Control Regulations require that a licence application contain information related to the organizational management structure and responsibilities… An adequately established and implemented management system provides CNSC staff confidence and evidence that the legal basis under which the Commission made its decision and had issued a licence, pursuant the Nuclear Safety and Control Act, remains valid.*

The results of PSR1 NSCA reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.13.2. As identified in the Pickering PSR2 Basis Document [B.13-5], the Pickering PSR2 review of the NSCA (Amended in February 2015) is an Incremental review. PSR2 Incremental review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.

- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.13.2 Compliance Assessment for Pickering PSR2

B.13.2.1 Application of PSR1 Reviews

The versions of the NSCA (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

NSCA was not assessed in the Pickering B ISR.

Pickering Units 1,4

NSCA was not part of the Pickering A Return to Service assessments.

The Pickering LCH [B.13-4] confirms that the NSCA is the basis for the granting of the PROL. For example the LCH states:

> *An adequately established and implemented management system provides CNSC staff confidence and evidence that the legal basis under which the Commission made its decision and had issued a licence, pursuant the Nuclear Safety and Control Act, remains valid…*

The Pickering PROL Renewal Application [B.13-6] states:

> *Table 1 is included for convenience, to assist in locating specific information within the application corresponding to the requirements of the Nuclear Safety and Control Act and applicable Regulations."*

Table 1 of [B.13-6] demonstrates how compliance with regulations under the NSCA is achieved, and there are no PSR2 gaps. Further, the Darlington ISR review of the NSCA is programmatically applicable to Pickering NGS and is discussed further below.

Darlington NGS

A clause-by-clause review of the NSCA (1997) was performed for Darlington ISR in NK38-REP-03680-10089 R000 [B.13-7]. The majority of the Act is applicable only to the CNSC. The parts that are applicable to OPG were addressed with reference to fleet-wide documents. The findings were that OPG has implemented Governance, Programs and Procedures to ensure compliance with the requirements of the NSCA. The report did not identify any gaps. This conclusion is also applicable to Pickering and is not affected by operation past 2020.

A subsequent Darlington ISR Code Refresh clause-by-clause review of the 2014 Amendment of the NSCA was performed in OPG Report NK38-REP-03680-10216 R000 [B.13-8]. No gaps were found in this review. The compliance assessment refers to fleet-wide licences and programs with the exception of the compliance discussion on Section 30 which refers to Darlington correspondence to explain how OPG is committed to providing timely access to CNSC inspectors to enable them to perform their duties under the NSCA. Similar programs are in place at Pickering NGS. A search was performed of the Station Condition Records issued since 2009 to identify any issues with inspector access. None were found. As a result, there are no PSR2 gaps.

B.13.2.2 Application of Post-PSR1 Reviews

Subsequent to the 2012 Licence Renewal application [B.13-6], the NSCA was amended in 2015 [B.13-1]. The only change made to the NSCA is described in Miscellaneous Statute Law Amendment Act, 2014 (S.C. 2015, c. 3) clause 136 [B.13-2] as follows:

> *Section 80 of the* Nuclear Safety and Control Act *is replaced by the following:*
>
> *80. A licence that is issued pursuant to regulations made under paragraph 9(b) of the Atomic Energy Control Act and that is in force immediately before the commencement day is deemed to have been issued under section 24 of this Act and to be in force for the remainder of the period for which it was issued under the Atomic Energy Control Act and any fees paid or payable under the AECB Cost Recovery Fees Regulations, 1996 in respect of such a licence are deemed to be paid or payable, as the case may be, under this Act.*

This amendment has no impact on nuclear safety and is not applicable in the context of PSR2. There is no PSR2 gap.

B.13.3 Compliance Assessment Summary for Pickering PSR2

There are no PSR2 gaps for the Nuclear Safety and Control Act (Amended 2015) [B.13-1]. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with the NSCA (Amended 2015).

B.13.4 References

[B.13-1] Statutes of Canada, S.C. 1997, c. 9, *Nuclear Safety and Control Act*, amended on February 26, 2015.

[B.13-2] Statutes of Canada, S.C. 2015, c. 3, *Miscellaneous Statute Law Amendment Act*, assented to February 2015.

[B.13-3] Government of Canada, *Justice Laws Website –Nuclear Safety and Control Act Table of Contents*, http://laws-lois.justice.gc.ca/eng/acts/N-28.3/, July 2016.

[B.13-4] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.13-5] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.13-6] OPG Letter, P-CORR-00531-03719 R000, G. Jager to M. A. Leblanc, *Application for Renewal of Pickering Nuclear Generating Station Power Reactor Operating Licence*, July 4, 2012.

[B.13-7] OPG Report, NK38-REP-03680-10089 R000, Review of NSCA (1997), *Nuclear Safety and Control Act for Darlington Integrated Safety Review,* May 2010.

[B.13-8] OPG Report, NK38-REP-03680-10216 R000, *Code Refresh Review Of The Nuclear Safety And Control Act (2013 Amendment)*, January 2014.

B.14 CSA N1600-14, "General Requirements for Nuclear Emergency Management Programs"

B.14.1 Background

The following paraphrased from the purpose and scope of CSA N1600-14 [B.14-1] provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *The purpose of CSA N1600-14 is to establish criteria for the emergency management programs of on- and off-site organizations to address nuclear emergencies at Canadian reactor facilities.*
>
> *CSA N1600-14 provides requirements for a comprehensive nuclear emergency management program (NEMP) embracing the Emergency Management (EM) components (prevention/mitigation, preparedness, response, and recovery) in keeping with international EM practice, with a predominant focus on preparedness, response, and recovery. It establishes the elements of a continuous improvement process to develop, implement, maintain, and evaluate the EM functions of nuclear facilities and their surrounding communities.*

CSA N1600-14 is directly relevant to Safety Factor 13 (Emergency Planning). However, it is worth noting that CSA N1600 is intended to cover all aspects of emergency management, which includes requirements that do not apply to utility organizations (for example, the management of off-site response). For the purpose of the PSR2 review, only requirements that apply to OPG require assessment.

Compliance with CSA N1600-14 is not a licence requirement for Pickering NGS (per PROL 48.02/2018) and it is not referenced in the Pickering Licence Conditions Handbook [B.14-2].

CSA N1600-14 is the first edition of this standard, published in May 2014. The impact statement [B.14-3] for N1600-14 highlights the significant features of the standard as follows:

1. *This new Canadian Standard outlines the requirements for on-site and off-site emergency management programs to address nuclear emergencies at nuclear power plants (NPPs).*

2. *This Standard does not apply to nuclear emergencies at Class IB nuclear facilities, Class II nuclear facilities, and mines and mills; however, this Standard may provide guidance to nuclear facilities other than NPPs. The operators of these facilities may, together with the authority having jurisdiction, determine the applicability and suitability of the guidance provided by this Standard.*

3. *This Standard provides the unique requirements to develop, implement, evaluate, maintain, and continuously improve a nuclear emergency management program (NEMP) and reflects the Five Pillars/Components of Emergency Management (prevention, mitigation, preparedness, response, and recovery) in keeping with international emergency management practice.*

4. *Although this Standard reflects the Five Pillars/Components of Emergency Management, the predominant focus is on the preparedness for, the response to, and the recovery from a nuclear emergency at a NPP.*

5. *This Standard includes requirements pertaining to: i) planning basis, ii) communications, iii) program management, iv) nuclear emergency response plans, v) nuclear emergency recovery plans, vi) training, facilities and equipment maintenance, vii) public awareness and education, viii) exercises, ix) program evaluation, audit, and review, and x) management review.*

As identified in Reference [B.14-4], the Pickering PSR2 review of CSA N1600-14 is a High Level Review. For a PSR2 High Level Review, the degree of conformance with clauses or groups of clauses in the Law, Regulation, Code or Standard (L/R/C/S) is demonstrated by supporting evidence stating whether the intent of the requirements stipulated in the requirement document is met. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the intent of the safety requirement is met.
- Gap: A Gap indicates that the intent of the safety requirement is not met.

B.14.2 Compliance Assessment for Pickering PSR2

B.14.2.1 Application of PSR1 Reviews

Pickering NGS

The first edition of CSA N1600 was issued in May 2014 after the completion of the Pickering B Integrated Safety Review (ISR) and Pickering A Return-to-Service. Therefore, no code reviews were performed for Pickering Units 5-8 or Units 1,4 against CSA N1600.

Darlington NGS

The first edition of CSA N1600 was issued in May 2014 after the completion of the code reviews for the Darlington ISR. Therefore, no code review was performed for Darlington against CSA N1600.

B.14.2.2 Application of Post PSR1 Reviews

A compliance review against CSA N1600-14 [B.14-1] was not undertaken as part of previous PSR1 reviews and the following High Level assessment has been completed. In the review below, the degree of conformance with clauses or groups of clauses in the standard is assessed for Pickering NGS by reference to supporting evidence stating whether the intent of the requirements stipulated in the standard is met.

CSA N1600-14 Clause	PSR2 Review	Compliant or Gap
0 Introduction	This clause provides introductory background information and does not specify requirements. Assessment not required.	N/A
1 Scope	This clause defines the scope of N1600 and does not specify requirements. Assessment not required.	N/A
2 Reference publications	This clause identifies references used in N1600 and does not specify requirements. Assessment not required.	N/A
3 Definitions	This clause defines terms used in N1600 and does not specify requirements. Assessment not required.	N/A
4 Nuclear Emergency Management Program (NEMP)		
4.1 General	This clause contains sub-clauses specifying requirements related to the following subject areas: • NEMP overview • NEMP elements • Integration • Alternatives <u>Assessment</u> Clause 4.1 of N1600 establishes high level requirements for a NEMP that are further elaborated on in subsequent Clauses 4.2 to 4.10. These high level requirements are addressed by the content of N-PROG-RA-0001, "Consolidated Nuclear Emergency Plan" (CNEP) [B.14-5] and the various implementing documents that it references, which collectively define OPG's NEMP. The CNEP is integrated with OPG's overall governance framework, and receives its authority from N-CHAR-AS-0002, "Nuclear Management System" [B.14-6]. Integration in the form of collaboration and coordination with other stakeholders and external organizations occurs primarily through the conduct of drills and exercises (refer to discussion for Clause 4.10 of N1600).	Compliant

CSA N1600-14 Clause	PSR2 Review	Compliant or Gap
	Clause 4.1 of N1600 makes reference to event prevention and mitigation as being elements of the NEMP, but acknowledges that these are not addressed in N1600 (they are covered by other codes and standards). OPG also addresses these as part of other programs, which are outside of the NEMP. Hence, this is not a PSR2 gap. On this basis, the OPG Nuclear emergency management program meets the intent of the requirements specified in Clause 4.1 of CSA N1600-14. Since this applies across OPG Nuclear, this conclusion is applicable to the Pickering station.	
4.2 Planning basis	This clause contains sub-clauses specifying requirements related to the following subject areas: • Hazards identification • Risk assessment • Impact analysis • NPP planning basis requirements • Off-site planning basis requirements • Review frequency <u>Assessment</u> The emergency response planning basis is discussed in Section 1.1 of the CNEP, N-PROG-RA-0001 [B.14-5]. The need is specified to consider the range of Design Basis Accidents that could reasonably be postulated to occur (as identified, described and analyzed in the station's Safety Report). As well, potentially more severe Beyond Design Basis Accidents need to be considered, as reflected in new governance that OPG has established subsequent to the Fukushima accident, such as N-STD-MP-0019, "Beyond Design Basis Accident Management" [B.14-7]. Consideration is also given to events caused by conditions external to the site and initiating events that are non-nuclear in nature. This ensures that the full spectrum of potential hazards is considered. Clause 4.2.2 of N1600 specifies the requirement to perform a risk assessment to determine the potential impact of hazards. Requirements applicable to performing risk assessments are addressed in the Pickering PSR2 Safety Factor 6 Report – Probabilistic Safety Assessment (PSA). Impact analysis deals with the need to identify and ensure the availability of functions that are critical to supporting the emergency response. This requirement is addressed by N-PROC-RA-0133, "Management of	Compliant

CSA N1600-14 Clause	PSR2 Review	Compliant or Gap
	Equipment Important to Emergency Response" [B.14-8] (refer to the discussion for Clause 4.8 of N1600). Clause 4.2.5 relating to off-site planning requirements is specifically addressed to outside organizations (i.e., not OPG). Clause 4.2.6 of N1600 sets requirements for the emergency response planning basis to be documented (in accordance with the organization's document management process) and reviewed at least every five years. The planning basis is well understood; it is derived from sources such as the station's Safety Report and PSA. As part of the OPG transition plan for CNSC REGDOC-2.10.1 [B.14-9], an action is in progress to compile the planning basis references for incorporation into the CNEP, which is subject to an annual review frequency. This is a documentation issue that does not affect the manner in which OPG would manage the response to an emergency and is not safety significant. On this basis, the OPG Nuclear emergency management program meets the intent of the requirements specified in Clause 4.2 of CSA N1600-14. Since this applies across OPG Nuclear, this conclusion is applicable to the Pickering station.	
4.3 Communication	This clause contains sub-clauses specifying requirements related to the following subject areas: • Communication needs analysis • Process for external communication • Data and information needs for effective communication • Data and information transmittal requirements for NPPs • Communications plan establishment • Communications plan scope • Communication procedures • Testing of communication systems <u>Assessment</u> Requirements for external communication are specified throughout the CNEP, N-PROG-RA-0001 [B.14-5]. Specific details with respect to how communications are managed during a nuclear emergency are provided in:	Compliant

CSA N1600-14 Clause	PSR2 Review	Compliant or Gap
	• OPG-PROC-0028, "Crisis Management and Communications Centre (CMCC) Procedure" [B.14-10] • OPG-PROC-0112, "Corporate Relations and Communications Emergency Preparedness and Response Procedure" [B.14-11] • N-STD-AS-0010, "Nuclear Crisis Communications Standard" [B.14-12] Collectively and in conjunction with other implementing governing documents under the CNEP (e.g., Clause 4.8 discussion covers testing of communications equipment), these address the requirements specified in Clause 4.3 of N1600, with one exception. Clause 4.3.2 of N1600 requires that the approach to managing emergency communications be based on performing a communications needs analysis. The need for this is not specified in the OPG governing documents. However, post-Fukushima, OPG embarked on an Emergency Telecommunications Enhancement project for Pickering as documented in [B.14-13], [B.14-14] and [B.14-15]. Communications needs have been clearly defined for Pickering in [B.14-14] and used as the basis for assessing the adequacy of emergency communications at Pickering. Furthermore, communication protocols have been tested on several occasions over the years in exercises and drills involving external organizations, and will continue to be evaluated as part of future exercises. Considering the above, the absence of reference to a communications needs analysis in the governance is not safety significant, hence, this is not a PSR2 gap. On this basis, the OPG Nuclear emergency management program meets the intent of the requirements specified in Clause 4.3 of CSA N1600-14. Since this applies across OPG Nuclear, this conclusion is applicable to the Pickering station.	
4.4 Program management	This clause contains sub-clauses specifying requirements related to the following subject areas: • Planning cycle for the NEMP • Leadership and commitment • Program coordination • NEMP review committee • NEMP administration • Review of the NEMP • Financial management	Compliant

CSA N1600-14 Clause	PSR2 Review	Compliant or Gap
	• NEMP maintenance <u>Assessment</u> Planning, development, implementation and maintenance of the emergency management program are established as specified in the CNEP, N-PROG-RA-0001 [B.14-5]. Requirements with respect to leadership and commitment, program coordination, program administration, program maintenance and financial management are addressed as part of the roles and accountabilities specified in Section 2.0 of the CNEP. Clause 4.4.4 of N1600 specifies the requirement for a NEMP review committee to provide guidance and advice on the organization's NEMP. Each site has an Emergency Response Oversight Committee that encompasses and addresses any issues identified with the CNEP. Performance measures are addressed in Section 1.6.5 of the CNEP as well as N-GUID-03491-10008, "Emergency Preparedness Performance Measure System" [B.14-16]. Mutual aid agreements are addressed in Section 1.3 of the CNEP as well as in N-LEGL-03490-0413370 [B.14-17] documenting the arrangements between OPG, Bruce Power, AECL and New Brunswick Power, and N-GUID-03490-10001, "Mutual Aid Agreement Implementation" [B.14-18]. Documentation requirements are specified in Section 1.6.1 of the CNEP. Control and periodic review of Emergency Preparedness documents are governed by OPG-STD-0001, "Requirements for Administrative Governance Documents" [B.14-19] and OPG-PROC-0001, "Process Administrative Governance Documents" [B.14-20]. Records and documentation are managed in accordance with OPG-PROG-0001, "Information Management" [B.14-21] and OPG-PROC-0019, "Records and Document Management" [B.14-22]. Processes to ensure ongoing review and maintenance of the program are specified in Section 1.6.5 of the CNEP including monitoring of performance measures, drills and exercises, self-assessment, independent assessment, use of the corrective action program and OPEX. Validation of the program and changes to the program are not explicitly addressed in the CNEP, however, this is adequately covered by the range and frequency of drills conducted regularly per Table 1 in N-PROC-RA-0045, "Emergency Preparedness Drills and Exercises" [B.14-23]. Drills and exercises serve as validation of the program, therefore, this is not a PSR2 gap.	

CSA N1600-14 Clause	PSR2 Review	Compliant or Gap
	Clause 4.4.1.3 of N1600 specifies a five year planning cycle for the NMEP. This is considered to be redundant to the five year review cycle for the planning basis that is specified in Clause 4.2.6 of N1600. On this basis, the OPG Nuclear emergency management program meets the intent of the requirements specified in Clause 4.4 of CSA N1600-14. Since this applies across OPG Nuclear, this conclusion is applicable to the Pickering station.	
4.5 Nuclear emergency response plan	This clause contains sub-clauses specifying requirements related to the following subject areas: • Nuclear emergency response plan development • Response organization requirements • Nuclear emergency categorization and notification • Emergency assessment • Protective actions • Interface and support between response organizations • Emergency personnel protection • Critical facilities • Communication, information flow and public alerting systems • Continuity of nuclear emergency response operations • Deviation from the nuclear emergency response plan • Validation of the nuclear emergency response plan and procedures • Nuclear emergency response plan maintenance Assessment The OPG NEMP, as specified in the CNEP, N-PROG-RA-0001 [B.14-5] and its suite of implementing documents possesses the high level characteristics recommended in Clause 4.5.1 of N1600. The OPG NEMP as defined in these documents also meets the plan development requirements specified in Clause 4.5.2 of N1600. Clause 4.5.2.5 requires that information about the plan be readily accessible to the public. A summary of the NEMP to the extent it is relevant to the public is readily available on OPG's Emergency Preparedness website [B.14-24]. Requirements with respect to the Emergency Response Organization (ERO) responsible for managing an event at an OPG station are specified in Section 2.2 of the CNEP.	Compliant

CSA N1600-14 Clause	PSR2 Review	Compliant or Gap
	This includes roles and accountabilities for all identified positions within the ERO. This is further supported by a set of procedures that provide detailed execution instructions for all key ERO roles (e.g., N-INS-03491-10000, "CEOF Emergency Response Director" [B.14-25]). Staffing quorum requirements are also specified for each ERO facility to declare itself operational. The command and control structure is defined in Section 1.2.1 of the CNEP and further reinforced by the role-specific instruction documents that elaborate on interface requirements between roles, thereby ensuring an integrated response effort. Requirements are also specified within the CNEP for the identification, deployment and managing of required support resources. Processes governing activation of the ERO are defined, including reporting time requirements for key positions. Further details with respect to how ERO staffing is managed, including availability and notification provisions are specified in N-PROC-RA-0046, "Emergency Response Organization Staffing and Availability" [B.14-26]. Minimum staffing requirements in terms of on-site staff required to perform all essential event mitigation and response duties following limiting design basis accidents are defined for the station per P-INS-09100-0003, "Pickering Minimum Shift Complement" [B.14-27]. Provisions are also made for the possibility that emergency response may be extended over a period of time (i.e., over multiple shifts). As a result of actions taken subsequent to the Fukushima event, procedures and guidelines have been developed and implemented that also assure that emergency response to a multi-unit event is supported. Validation exercises have been performed to confirm this. Nuclear emergency categorization and notification requirements are defined in Section 1.2.2 of the CNEP. Requirements for emergency assessment, in particular the need to predict and monitor radiological conditions both on-site and off-site are addressed in the CNEP. This includes the designation of specific ERO roles to perform these functions in Section 2.2 of the CNEP. This is further supported by implementing procedures such as N-STD-RA-0004, "Emergency Off-Site Radiological Monitoring Process for Airborne Releases of Radioactive Materials" [B.14-28] and N-STD-RA-0005, "Emergency Dose Projection Process" [B.14-29]. Clause 4.5.6 of N1600 deals with off-site protective actions that are primarily the responsibility of the Province of Ontario. The CNEP addresses these considerations, to the extent required for OPG to support the identification	

CSA N1600-14 Clause	PSR2 Review	Compliant or Gap
	and implementation of protective actions (e.g., communication requirements, provision of dose projections). Clause 4.5.6.3.2 of N1600 deals with public evacuation time estimates. The Pickering Evacuation Time Estimates report has been recently updated [B.14-30] and is available to the public [B.14-31]. OPG governance is in the process of being updated to ensure that the report is maintained up to date on an as required basis. Clause 4.5.6.3.3 of N1600 deals with the distribution of iodine thyroid blocking agents (i.e., potassium iodide (KI) pills). The distribution of KI pills, including instructions on their proper use, was completed by OPG prior to the end of 2015 for the primary distribution zone around the Pickering station. Details regarding this program are available to the public [B.14-32]. OPG governance is in the process of being updated to sustain this program going forward. Completion of these actions to update governance for Evacuation Time Estimates and KI pills has been committed to the CNSC as part of the OPG transition plan to achieve compliance with CNSC REGDOC-2.10.1. As such, this is documented as a gap in the PSR2 REGDOC-2.10.1 assessment (PSR2 REGDOC-2.10.1 Gap#1) and is not identified as a gap for CSA N1600. N-STD-RA-0043, "Nuclear Recovery Planning" [B.14-33] addresses the need for communications with organizations outside of OPG but does not explicitly mention inter-organizational coordination. However, Table 2 of the standard identifies required interfaces with external stakeholders and assigns responsibilities within OPG for managing each of these. This is sufficient to meet the intent of the requirement. Mutual aid agreements are discussed as part of the discussion for Clause 4.6 of N1600. Clause 4.5.7.5 of N1600 provides requirements with respect to venting containment. The CNEP outlines how the potential need for containment venting is managed in accordance with these requirements, including the associated expectations for notification, consultation and approval authority. Instructions for physically implementing venting strategies are embedded in station emergency operating procedures. The CNEP and its implementing documents, in addition to station operating procedures, address the requirements for emergency personnel protection, including the provision and use of Personal Protective Equipment (PPE). Required equipment and facilities to support the emergency response are addressed in the CNEP, and managed in accordance with OPG's Equipment Important To Emergency Response (EITER) program (refer to discussion for Section 4.8 of N1600). Potential threats to	

CSA N1600-14 Clause	PSR2 Review	Compliant or Gap
	habitability are considered, and alternate facility locations are specified to allow essential emergency response functions to be performed in the event that primary facility locations are unavailable or uninhabitable. Requirements for communication are specified throughout the CNEP (refer to discussion for Clause 4.3 of N1600). Continuity of emergency response operations and the potential need to authorize deviations from plan are addressed in the CNEP. Validation of the program and changes to the program are not explicitly addressed in the CNEP, however, this is adequately covered by the range and frequency of drills conducted regularly per Table 1 in N-PROC-RA-0045, "Emergency Preparedness Drills and Exercises" [B.14-23]. Clause 4.5.13.3 of N1600 requires that the Authority Having Jurisdiction (AHJ) be notified of changes to the NEMP and that validation results be submitted to the AHJ at least 30 days prior to implementation. This is not addressed in OPG's governing documents, however, this is not safety significant so it is not a PSR2 gap. Documentation requirements and maintenance of the emergency management program are addressed as part of the discussion for Clause 4.4 of N1600. Clause 4.5.8.4 of N1600 requires that the types and quantities of PPE be identified and provided that would allow the station's emergency response to be self-sufficient (i.e., without outside assistance) for the first 72 hours following an event. This 72 hour window is consistent with provisions OPG has implemented following the Fukushima event to address Beyond Design Basis Accidents. As documented in [B.14-34], guidance has been implemented to ensure that adequate supplies are identified and maintained on-site for the station to be self-sufficient for 72 hours following an accident. In particular, [B.14-35] itemizes the radiation protection PPE supplies and their quantities for Pickering and Darlington to be self-sufficient for at least 72 hours, which meets the intent of the requirements. On this basis, the OPG Nuclear emergency management program meets the intent of the requirements specified in Clause 4.5 of CSA N1600-14. Since this applies across OPG Nuclear, this conclusion is applicable to the Pickering station.	
4.6 Nuclear emergency recovery plan	This clause contains sub-clauses specifying requirements related to the following subject areas:	Compliant

CSA N1600-14 Clause	PSR2 Review	Compliant or Gap
	• Nuclear emergency response plan development • Transition from response to recovery • Plan implementation authority • Technical resource needs • Scope of the nuclear emergency recovery plan • Recovery organizational requirements • Other resource requirements • Communications and information flow • Continuity of nuclear emergency recovery plan operations • Deviation from the nuclear emergency recovery plan • Nuclear emergency recovery plan accessibility • Validation of the nuclear emergency recovery plan and procedures • Nuclear emergency recovery plan maintenance <u>Assessment</u> Requirements related to emergency recovery are specified in Section 1.2.7 of the CNEP, N-PROG-RA-0001 [B.14-5]. N-STD-RA-0043, "Nuclear Recovery Planning" [B.14-33] specifies in more detail the processes by which OPG manages the emergency recovery phase. N-STD-RA-0043 [B.14-33] stipulates that the recovery effort by the utility will be managed as if it were a project and that a Recovery Project Organization (RPO) will be established to execute the project. N-STD-RA-0043 [B.14-33] cites N1600-14 as a Bases reference and has been developed taking into consideration the requirements in Clause 4.6 of N1600 that are applicable to NPPs. Consideration is given to the potential need to develop special procedures and business processes and to perform damage assessments, recognizing that the available recovery options will depend on the extent of damage. The potential need for recovery activities to commence while some emergency response activities are still ongoing is acknowledged, and roles and accountabilities are defined to manage the transition to emergency recovery. Mutual aid agreements are addressed in Section 1.3 of the CNEP as well as in N-LEGL-03490-0413370 [B.14-17] documenting the arrangements between OPG, Bruce Power, AECL and New Brunswick Power, and N-GUID-03490-10001, "Mutual Aid Agreement Implementation" [B.14-18]. Clause 4.6.1 of N1600 specifies the need to develop supporting procedures, but recognizes that detailed	

CSA N1600-14 Clause	PSR2 Review	Compliant or Gap
	procedures likely cannot be prepared until after the event has occurred, since they will depend on the extent of damage to the plant. This is acceptable since strategies and a basic framework have been put in place prior, including roles and responsibilities and expectations related to other organizations. These are covered in N-STD-RA-0043 [B.14-33]. Furthermore, by specifying that the recovery effort will be managed as a project, it is implicit that the RPO will recognize the need to create the procedures it will require to perform its functions, based on the known conditions at the time. Table 1 in N-STD-RA-0043 provides a list of project execution documents the RPO may need to generate. The transition from emergency response to emergency recovery is addressed in Sections 1.2.6 and 1.2.7 of the CNEP [B.14-5] and Sections 1.3.4 and 1.4.4 of N-STD-RA-0043 [B.14-33]. N-STD-RA-0043 addresses plan implementation, scope of the plan, resource needs, organizational requirements, communication requirements and continuity of operations. The potential need to authorize and manage deviations from plan is addressed in Sections 1.2.1 and 1.5, and is implicit in establishing a project organization to manage the recovery effort, with roles and accountabilities as specified in Section 2.0. Public accessibility of the recovery plan is not applicable to OPG, since this requirement is written relative to off-site requirements that are outside of OPG's responsibility to manage. Clause 4.6.12 of N1600 specifies requirements to perform and document validation of the recovery plan and changes to the plan. Clause 4.6.13 of N1600 specifies requirements to review the recovery plan at least every five years and to maintain it. The recovery plan has been validated through stakeholder review including the Fukushima project for Beyond Design Basis Accidents and the review cycle for N-STD-RA-0043 aligns with the five year timeframe specified. This meets the intent of the requirements. Clause 4.6.12 of N1600 also contains a requirement to submit validation results to the Authority Having Jurisdiction, however, this is not safety significant, and is not a PSR2 gap. On this basis, the OPG Nuclear emergency management program meets the intent of the requirements specified in Clause 4.6 of CSA N1600-14. Since this applies across OPG Nuclear, this conclusion is applicable to the Pickering station.	

CSA N1600-14 Clause	PSR2 Review	Compliant or Gap
4.7 Training	This clause contains sub-clauses specifying requirements related to the following subject areas: • Levels of training • Training requirements • Competencies • Training design • Systematic approach to training • Frequency and scope • Qualification maintenance • On-site and off-site emergency response • Record retention <u>Assessment</u> Training requirements are specified in Section 1.6.3 of the CNEP, N-PROG-RA-0001 [B.14-5]. Specific details, including the training requirements for all ERO roles, are provided in N-TQD-503-00001, "Nuclear Emergency Response Organization Training and Qualification Description" [B.14-36]. Per N-LIST-08920-10001, "Trained Performance Areas" [B.14-37], ERO roles are among those designated as being governed by N-PROG-TR-0005, "Training" [B.14-38]. This requires that training be analyzed (to identify required competencies and training requirements), designed, developed, implemented and evaluated in accordance with a Systematic Approach to Training (SAT), details of which are specified in N-PROC-TR-0008, "Systematic Approach to Training" [B.14-39]. Training documentation is maintained in accordance with N-PROC-TR-0012, "Records and Documentation" [B.14-40]. Detailed training and qualification information for individuals is managed using the Training Information Management System (TIMS), in accordance with N-PROC-TR-0041, "TIMS II Administration" [B.14-41]. This includes identification of requalification requirements and expiry dates for affected qualifications. Clause 4.7.9.1 of N1600 could be interpreted as specifying a requirement to support the emergency response training needs of emergency response agencies external to OPG. Sections 1.3.3.1 and 2.1.7.3 of the CNEP recognize this and OPG does provide training for emergency workers outside of OPG. Clause 4.7.9.2 of N1600 specifies a requirement to submit training program information to the Authority Having Jurisdiction at least 30 days prior to implementation. This is not addressed in	Compliant

CSA N1600-14 Clause	PSR2 Review	Compliant or Gap
	the OPG governing documents, however, this is not safety significant, and is not a PSR2 gap. On this basis, the OPG Nuclear emergency management program meets the intent of the requirements specified in Clause 4.7 of CSA N1600-14. Since this applies across OPG Nuclear, this conclusion is applicable to the Pickering station.	
4.8 Facilities and equipment maintenance	This clause contains sub-clauses specifying requirements related to the following subject areas: • Working condition assurance • Functionality inspection and tests • System testing • Gaps and limitations • Corrective action plans <u>Assessment</u> Requirements for the testing and maintenance of emergency facilities and equipment are specified in Sections 1.6.4 and 1.6.6 of the CNEP, N-PROG-RA-0001 [B.14-5]. OPG has implemented an EITER program that addresses all facilities and equipment that are essential for performing ERO functions following an actual emergency, as documented in N-PROC-RA-0133, "Management of Equipment Important to Emergency Response" [B.14-8]. Details with respect to testing and maintenance requirements are contained in N-PROC-RA-0040, "Maintenance and Testing of Emergency Preparedness Facilities and Equipment" [B.14-42]. This procedure refers to N-LIST-03490-10028, "Emergency Preparedness Facility Inventory and Check Documents" [B.14-43] that provides a listing of affected equipment, including required checks and frequencies. References are also made to other equipment-specific checks, such as for radiation protection instruments or emergency response vehicles. In the event that any EITER components are found to be unavailable to meet their performance requirements, required actions are specified in P-INS-03491-00050, "Unavailability of Emergency Important to Emergency Response – Pickering" [B.14-44] and N-INS-03491-10025, "Unavailability of Emergency Important to Emergency Response – Off-Site" [B.14-45]. Additional specific information related to the capability and operation of ERO equipment and facilities at Pickering is provided in P-MAN-	Compliant

CSA N1600-14 Clause	PSR2 Review	Compliant or Gap
	03490-00002, "Pickering ERO Equipment and Facility Manual" [B.14-46], including information for dealing with contingencies during an actual event (e.g., loss of power). On this basis, the OPG Nuclear emergency management program meets the intent of the requirements specified in Clause 4.8 of CSA N1600-14. Since this applies across OPG Nuclear, and covers equipment and facilities that are specific to Pickering, this conclusion is applicable to the Pickering station.	
4.9 Public awareness and education	This clause contains sub-clauses specifying requirements related to the following subject areas: • Program requirements • Capabilities • Accountability <u>Assessment</u> Requirements for public education related to the possibility of a nuclear emergency at an OPG nuclear power plant are outlined in Section 1.3.6 of the CNEP, N-PROG-RA-0001 [B.14-5]. This is primarily the responsibility of the Province of Ontario and Section 1.3.6 contains sufficient high level information to describe the role OPG plays to support this. Extensive information regarding the OPG emergency management program is made available to the public through OPG's Emergency Preparedness website [B.14-24]. On this basis, the OPG Nuclear emergency management program meets the intent of the requirements specified in Clause 4.9 of CSA N1600-14. Since this applies across OPG Nuclear, this conclusion is applicable to the Pickering station.	Compliant
4.10 Exercises	This clause contains sub-clauses specifying requirements related to the following subject areas: • Exercise program • Exercise schedule • Exercise design • Exercise facilitator qualifications • Exercise preparation • Exercise conduct • Post-exercise activities	Compliant

CSA N1600-14 Clause	PSR2 Review	Compliant or Gap
	• Exercise evaluation <u>Assessment</u> Section 1.6.5.2 of the CNEP, N-PROG-RA-0001 [B.14-5] specifies the requirements for drills and exercises. Specific details governing the performance of drills and exercises are outlined in N-PROC-RA-0045, "Emergency Preparedness Drills and Exercises" [B.14-23], and N-INS-03490-10002, "Conduct of Emergency Preparedness Drills and Exercises" [B.14-47]. Collectively, these documents lay out OPG's exercise program and comply with the requirements contained in Clause 4.10 of N1600-14. This includes identification of the types of drills and exercises required, their required frequency, and the detailed specification of performance objectives and evaluation criteria. Detailed requirements and instructions are provided for the design, preparation and conduct of drills and exercises. Details are also provided for post-exercise activities including debriefings, evaluation to identify areas for improvement and documentation (e.g., drill reports). Exercise controller and evaluator qualifications are addressed in N-TQD-503-00001, "Nuclear Emergency Response Organization Training and Qualification Description" [B.14-36]. Clause 4.10.2.3 specifies that a multi-jurisdictional operations-based exercise be conducted at least once every five years. This requirement is not specified in the OPG governing documents, although in practice exercises of this nature are conducted periodically. The frequency with which such exercises are conducted is not safety significant, so this is not a PSR2 gap. Clause 4.10.6.4 states that exercise controllers and evaluators should be separate individuals. This is generally not the case for on-site OPG exercises and is not specified in OPG governing documents. However, this is not safety significant, so this is not a PSR2 gap. On this basis, the OPG Nuclear emergency management program meets the intent of the requirements specified in Clause 4.10 of CSA N1600-14. Since this applies across OPG Nuclear, this conclusion is applicable to the Pickering station.	
5 Response	This clause contains sub-clauses specifying requirements related to the following subject areas: • Activation of the nuclear emergency response plan • Inter-organizational emergency response coordination • Assessment of the response needs	Compliant

CSA N1600-14 Clause	PSR2 Review	Compliant or Gap
	• Deviation from the nuclear emergency response plan • Termination of the emergency response phase • Response evaluation <u>Assessment</u> The CNEP, N-PROG-RA-0001 [B.14-5] and its suite of implementing documents specify the processes by which OPG manages the emergency response phase. The program addresses both activation and termination of the Emergency Response Organization, which manages the utility response to the event. Inter-organizational coordination is addressed extensively in Section 1.3. Assessment of response needs is not applicable to OPG, since this is written relative to off-site needs that are outside of OPG's responsibility to manage (aside from external communication which is dealt with extensively in the program). The potential need to authorize and manage deviations from plan is not addressed explicitly, but is implicit in the roles and accountabilities as specified in Section 2.0. Evaluation for the purpose of determining the root cause of the event and to identify actions to prevent recurrence is addressed in Section 1.2.7. Evaluation of response effectiveness is not specified in the standard, however, this is addressed per normal OPG Nuclear processes including: • N-PROG-RA-0003, "Corrective Action" [B.14-48] • N-PROG-RA-0010, "Independent Assessment" [B.14-49] • N-PROG-RA-0097, "Self-Assessment and Benchmarking" [B.14-50]. On this basis, the OPG Nuclear emergency management program meets the intent of the requirements specified in Clause 5 of CSA N1600-14. Since this applies across OPG Nuclear, this conclusion is applicable to the Pickering station.	
6 Recovery	This clause contains sub-clauses specifying requirements related to the following subject areas: • Activation of the nuclear emergency recovery plan • Inter-organizational emergency recovery coordination • Assessment of the recovery needs • Deviation from the nuclear emergency recovery plan • Termination of the recovery operation	Compliant

CSA N1600-14 Clause	PSR2 Review	Compliant or Gap
	• Recovery evaluation <u>Assessment</u> Requirements related to emergency recovery are specified in Section 1.2.7 of the CNEP, N-PROG-RA-0001 [B.14-5]. N-STD-RA-0043, "Nuclear Recovery Planning" [B.14-33] specifies in more detail the processes by which OPG manages the emergency recovery phase. N-STD-RA-0043 stipulates that the recovery effort by the utility will be managed as if it were a project and that a Recovery Project Organization (RPO) will be established to execute the project. Activation and termination of the RPO are outlined in Section 1.4.4 of N-STD-RA-0043. The standard addresses the need for communications with organizations outside of OPG but does not explicitly mention inter-organizational coordination. However, Table 2 of the standard identifies required interfaces with external stakeholders and assigns responsibilities within OPG for managing each of these. This meets the intent of the requirement. Assessment of recovery needs is not applicable to OPG, since this is written relative to off-site needs that are outside of OPG's responsibility to manage (aside from external communication which is dealt with extensively in the standard). The potential need to authorize and manage deviations from plan is addressed in Sections 1.2.1 and 1.5, and is implicit in establishing a project organization to manage the recovery effort, with roles and accountabilities as specified in Section 2.0. Evaluation of recovery effectiveness is not specified in the standard, however, this is addressed per normal OPG Nuclear processes including: • N-PROG-RA-0003, "Corrective Action" [B.14-48] • N-PROG-RA-0010, "Independent Assessment" [B.14-49] • N-PROG-RA-0097, "Self-Assessment and Benchmarking" [B.14-50]. On this basis, the OPG Nuclear emergency management program meets the intent of the requirements specified in Clause 6 of CSA N1600-14. Since this applies across OPG Nuclear, this conclusion is applicable to the Pickering station.	

CSA N1600-14 Clause	PSR2 Review	Compliant or Gap
7 NEMP evaluation, audit and review	This clause contains sub-clauses specifying requirements related to the following subject areas: • NEMP evaluation • NEMP audit • NEMP review <u>Assessment</u> Section 2.1 of the CNEP, N-PROG-RA-0001 [B.14-5] specifies senior management roles and accountabilities related to the program, including specific role assignments for conducting program assessments, measuring and reporting on the effectiveness of the program, and identifying and implementing corrective actions. Requirements to ensure the ongoing health of the program are specified in Section 1.6.5 of the CNEP including monitoring of performance measures, drills and exercises, self-assessment, independent assessment, use of the corrective action program and OPEX. These are managed per normal OPG Nuclear processes, including: • N-PROG-RA-0003, "Corrective Action" [B.14-48] • N-PROG-RA-0010, "Independent Assessment" [B.14-49] • N-PROG-RA-0097, "Self-Assessment and Benchmarking" [B.14-50]. Clause 7.1.1 of N1600 specifies the requirement for a NEMP review committee to provide guidance and advice on the organization's NEMP. Membership needs to include those responsible for managing the NEMP, and others with emergency management expertise, and organizational knowledge with the ability to identify needed resources. The roles identified in Section 2.1 of the CNEP encompass all of the above, including the need for senior managers to provide strategic and programmatic direction. Each site has an Emergency Response Oversight Committee that encompasses and addresses any issues identified with the CNEP. On this basis, the OPG Nuclear emergency management program meets the intent of the requirements specified in Clause 7 of CSA N1600-14. Since this applies across OPG Nuclear, this conclusion is applicable to the Pickering station.	Compliant

CSA N1600-14 Clause	PSR2 Review	Compliant or Gap
8 Management oversight	This clause contains sub-clauses specifying requirements related to the following subject areas: • Senior management oversight • Continual improvement <u>Assessment</u> Section 2.1 of the CNEP, N-PROG-RA-0001 [B.14-5] specifies senior management roles and accountabilities related to the program, including specific role assignments for oversight, corrective action and the use of OPEX. Each site has an Emergency Response Oversight Committee that encompasses and addresses any issues identified with the CNEP. Requirements to ensure continual improvement of the program are specified in Section 1.6.5 of the CNEP including monitoring of performance measures, drills and exercises, self-assessment, independent assessment, use of the corrective action program and OPEX. These are managed per normal OPG Nuclear processes, including: • N-PROG-RA-0003, "Corrective Action" [B.14-48] • N-PROG-RA-0010, "Independent Assessment" [B.14-49] • N-PROG-RA-0097, "Self-Assessment and Benchmarking" [B.14-50]. On this basis, the OPG Nuclear emergency management program meets the intent of the requirements specified in Clause 8 of CSA N1600-14. Since this applies across OPG Nuclear, this conclusion is applicable to the Pickering station.	Compliant
Annex A (informative) Introduction figures	This annex is non-mandatory and provides additional explanation for information contained in Clause 0, Introduction of N-1600. It does not specify requirements. Assessment not required.	N/A
Annex B (informative) Nuclear emergency management program overview	This annex is non-mandatory and illustrates the relationship between the various legislation, plans and organizations involved in emergency response. It does not specify requirements. Assessment not required.	N/A

CSA N1600-14 Clause	PSR2 Review	Compliant or Gap
Annex C (informative) Conversion equivalence table	This annex provides information for converting between units of radiation measurement and does not specify requirements. Assessment not required.	N/A

Although no previous reviews have been documented against CSA N1600, several reviews have been performed as part of PSR2 in relation to other L/R/C/S related to emergency management that are potentially relevant to the subject matter of CSA N1600-14. The following codes were reviewed as part of the Pickering PSR2.

Code or Standard Number	Code or Standard Title
CNSC G-323 (2007)	Ensuring Presence of Sufficiently Qualified Staff at Class I Nuclear Facilities-Minimum Shift Complement
CSA N288.2-14	Guidelines for calculating Radiological Consequences to the public from a release of airborne radioactive material for Nuclear Reactor Accidents
CNSC REGDOC-2.3.2 (2015)	Accident Management, Version 2
CNSC REGDOC-2.10.1 (2014)	Nuclear Emergency Preparedness and Response

Note that CSA N1600-14 is programmatic in nature and, as such, the status of OPG's compliance with it is not impacted by the possibility of continued operation of Pickering beyond 2020.

B.14.3 Compliance Assessment Summary for Pickering PSR2

There are no PSR2 gaps for CSA N1600-14 [B.14-1]. Per the definition of Compliance for a High Level review, Pickering has a PSR2 Compliance associated with CSA N1600-14.

B.14.4 References

[B.14-1] CSA Standard N1600-14, *General Requirements for Nuclear Emergency Management Programs*, May 2014.

[B.14-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.14-3] CSA Final Publication – Impact Statement, *Product: New Standard; Product Designation: CSA N1600-EN; Product Title: General Requirements for Nuclear Emergency Management Programs; Date of Release: May 2014*, Date not provided.

[B.14-4] OPG Report P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.14-5] OPG Program, N-PROG-RA-0001 R014, *Consolidated Nuclear Emergency Plan*, May 2015.

[B.14-6] OPG Charter, N-CHAR-AS-0002 R018, *Nuclear Management System*, March 2015.

[B.14-7] OPG Standard, N-STD-MP-0019 R001, *Beyond Design Basis Accident Management*, September 2014.

[B.14-8] OPG Procedure, N-PROC-RA-0133 R000, *Management of Equipment Important to Emergency Response*, December 2014.

[B.14-9] CNSC Regulatory Document REGDOC-2.10.1, *Nuclear Emergency Preparedness and Response*, 2014.

[B.14-10] OPG Procedure, OPG-PROC-0028 R005, *Crisis Management and Communications Centre (CMCC) Procedure*, February 2015.

[B.14-11] OPG Procedure, OPG-PROC-0112 R001, *Corporate Relations and Communications Emergency Preparedness and Response Procedure*, January 2016.

[B.14-12] OPG Standard, N-STD-AS-0010 R003, *Nuclear Crisis Communications Standard*, January 2011.

[B.14-13] OPG Project Charter, NA44-PCH-03490-00001 R000, *Pickering Emergency Telecommunications Enhancement Project*, September 14, 2011.

[B.14-14] OPG Report, NK30-REP-60200-00002 R000, *Feasibility Study for Fukushima Emergency Telecommunication Enhancements of Pickering NGS*, March 4, 2013.

[B.14-15] OPG Report, P-REP-60200-0483361 R000, *Design Report Pickering Fukushima Emergency Communication Enhancement Project*, November 18, 2013.

[B.14-16] OPG Guide, N-GUID-03491-10008 R003, *Emergency Preparedness Performance Measure System*, April 2015.

[B.14-17] OPG Document, N-LEGL-03490-0413370, *Mutual Aid Agreement for Nuclear Emergency Support*, November 2012.

[B.14-18] OPG Guide, N-GUID-03490-10001 R001, *Mutual Aid Agreement Implementation*, December 2013.

[B.14-19] OPG Standard, OPG-STD-0001 R006, *Requirements for Administrative Governance Documents*, June 2016.

[B.14-20] OPG Procedure, OPG-PROC-0001 R010, *Processing Administrative Governance Documents*, June 2016.

[B.14-21] OPG Program, OPG-PROG-0001 R009, *Information Management*, September 2015.

[B.14-22] OPG Procedure, OPG-PROC-0019 R007, *Records and Document Management*, March 2016.

[B.14-23] OPG Procedure, N-PROC-RA-0045 R010, *Emergency Preparedness Drills and Exercises*, December 2015.

[B.14-24] OPG Website, *Emergency Preparedness*, http://www.opg.com/about/safety/emergency-preparedness/Pages/emergency-preparedness.aspx, Accessed November 7, 2016.

[B.14-25] OPG Instruction, N-INS-03491-10000 R009, *CEOF Emergency Response Director*, June 2016.

[B.14-26] OPG Procedure, N-PROC-RA-0046 R008, *Emergency Response Organization Staffing and Availability*, April 2015.

[B.14-27] OPG Instruction, P-INS-09100-00003 R009, *Pickering Minimum Shift Complement*, August 2015.

[B.14-28] OPG Standard, N-STD-RA-0004 R002, *Emergency Off-Site Radiological Monitoring Process for Airborne Releases of Radioactive Materials*, January 2013.

[B.14-29] OPG Standard, N-STD-RA-0005 R005, *Emergency Dose Projection Process*, April 2015.

[B.14-30] OPG Report, P-REP-03490-00079 R000, *Pickering NGS Development of Evacuation Time Estimates*, April 22, 2016.

[B.14-31] OPG Website, *Safety At Pickering: Evacuation Time Estimates Report*. http://www.opg.com/generating-power/nuclear/stations/pickering-nuclear/Pages/safety-at-pickering.aspx, Accessed November 7, 2016.

[B.14-32] OPG Website, *Pickering Nuclear: KI Pill Distribution*. http://www.opg.com/generating-power/nuclear/stations/pickering-nuclear/Pages/pickering-nuclear.aspx, Accessed November 7, 2016.

[B.14-33] OPG Standard, N-STD-RA-0043 R000, *Nuclear Recovery Planning*, August 2015.

[B.14-34] OPG Guidance, N-GUID-09013-10002 R000, *Supply Chain Support – Nuclear Emergency Responders Guideline*, July 28, 2015.

[B.14-35] OPG Guidance, N-GUID-03490-10003 R000, *Emergency Preparedness 72 Hour Emergency Supplies*, July 3, 2015.

[B.14-36] OPG Training and Qualification Description, N-TQD-503-00001 R017, *Nuclear Emergency Response Organization Training and Qualification Description*, January 2016.

[B.14-37] OPG List, N-LIST-08920-10001 R008, *Trained Performance Areas*, March 2016.

[B.14-38] OPG Program, N-PROG-TR-0005 R016, *Training*, January 2016.

[B.14-39] OPG Procedure, N-PROC-TR-0008 R020, *Systematic Approach to Training*, March 2016.

[B.14-40] OPG Procedure, N-PROC-TR-0012 R012, *Records and Documentation*, October 2012.

[B.14-41] OPG Procedure, N-PROC-TR-0041 R011, *TIMS II Administration*, March 2013.

[B.14-42] OPG Procedure, N-PROC-RA-0040 R006, *Maintenance and Testing of Emergency Preparedness Facilities and Equipment*, April 2015.

[B.14-43] OPG List, N-LIST-03490-10028 R000, *Emergency Preparedness Facility Inventory and Check Documents*, April 2015.

[B.14-44] OPG Instruction, P-INS-03491-00050 R002, *Unavailability of Equipment Important to Emergency Response – Pickering*, November 2015.

[B.14-45] OPG Instruction, N-INS-03491-10025 R000, *Unavailability of Equipment Important to Emergency Response – Off-Site*, December 2014.

[B.14-46] OPG Manual, P-MAN-03490-00002 R004, *Pickering ERO Equipment and Facility Manual*, March 2015.

[B.14-47] OPG Instruction, N-INS-03490-10002 R005, *Conduct of Emergency Preparedness Drills and Exercises*, December 2014.

[B.14-48] OPG Program, N-PROG-RA-0003 R010, *Corrective Action*, January 2015.

[B.14-49] OPG Program, N-PROG-RA-0010 R014, *Independent Assessment*, April 2016.

[B.14-50] OPG Program, N-PROC-RA-0097 R008, *Self-Assessment and Benchmarking*, December 2014.

[B.14-51] OPG Letter, B. Duncan to F. Rinfret, NK38-CORR-00531-17593, *Darlington NGS – Transition Plan for Regulatory Document Nuclear Emergency Preparedness and Response (REGDOC – 2.10.1)*, September 30, 2015.

B.15 CSA N288.6-12, "Environment Risk Assessments at Class I Nuclear Facilities and Uranium Mines and Mills"

B.15.1 Background

The following paraphrased from the purpose and scope of CSA N288.6-12 [B.15-1] provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *Environmental risk assessment (ERA) of nuclear facilities is a systematic process used to identify, quantify, and characterize the risk posed by contaminants and physical stressors in the environment on biological receptors, including the magnitude and extent of the potential effects associated with a facility.*
>
> *CSA N288.6-12 addresses the design, implementation, and management of an environmental risk assessment program that incorporates best practices used in Canada and internationally.*

CSA N288.6 is relevant to Safety Factors 8 (Safety Performance) and 14 (Radiological Impact on the Environment). N288.6 is not identified in Appendix C of the R04 Licence Conditions Handbook [B.15-2], meaning it is not in Pickering PROL 48.02/2018. CSA N288.6-12 is the first edition of this standard.

As identified in Reference [B.15-3], the Pickering PSR2 review of CSA N288.6-12 is an Incremental review. PSR2 Incremental review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.
- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.15.2 Compliance Assessment for Pickering PSR2

B.15.2.1 Application of PSR1 Reviews

There have not been any previous PSR1 compliance reviews relating to N288.6.

Section 5 of the R04 Pickering License Conditions Handbook [B.15-2] states:

> *Environmental Risk Assessment (ERA) should be conducted in accordance with CSA standard N288.6-12 "Environment Risk Assessments at Class I Nuclear Facilities and Uranium Mines and Mills". The ERA provides the basis for the environmental monitoring program (CSA standard N288.4) and also the effluent monitoring program (CSA standard N288.5) (see section 10.1 for more details regarding these standards). The ERA should be updated periodically with the results from the environmental and effluent monitoring programs in order to confirm the effectiveness of proposed mitigation measures…*

CSA standard N288.6-12 "Environmental risk assessments at Class I nuclear facilities and uranium mines and mills", published in June 2012, specifies the need for an ERA. The results of ERA provide the basis for monitoring programs in accordance with CSA Standards such as N288.4 and N288.5, including Radiological Environmental Monitoring Programs (REMP) and Fish Impingement Monitoring Program. The ERA should be periodically updated with the results of monitoring to confirm the effectiveness of proposed mitigation measures.

B.15.2.2 Application of Post-PSR1 Reviews

A clause-by-clause review and gap analysis against CSA N288.6-12 was prepared on behalf of OPG Nuclear in 2012 and issued in 2014 [B.15-4]. Table 6.1 of [B.15-4] summarizes the review findings and recommended actions to eliminate the identified gaps. Appendix A of [B.15-4] provides the clause-by-clause review of N288.6. OPG subsequently completed and issued an Environmental Risk Assessment in January 2014 [B.15-5] which addressed the gaps identified in [B.15-4].

The 2015 OPG environmental monitoring programs results were reported to the CNSC in [B.15-6]. Relating to the Environmental Risk Assessment, the reference states:

> *The PN* [Pickering Nuclear] *ERA was updated in 2013 in accordance with the requirements of CSA N288.6-12, Environmental Risk Assessments at Class I Nuclear Facilities and Uranium Mines and Mills [R-9]. The results indicate that PN station operation does not present any radiological or physical stressor risks to human or non-human biota, however hydrazine in lake water was identified as a potential human health risk due to uncertainty in the lake water concentrations used in the assessment [R-10]. To clarify this potential risk, a supplementary study was conducted in 2014 which confirmed that there is no risk of human health or ecological effects from hydrazine in Lake Ontario near the PN facility [R-18].*

OPG has conducted a clause-by-clause review CSA N288.6-12 and has completed gap resolution for Pickering resulting in a revised ERA compliant with N288.6-12. Hence, there are no PSR2 gaps relating to CSA N288.6-12 compliance.

B.15.3 Compliance Assessment Summary for Pickering PSR2

There are no PSR2 gaps for CSA N288.6-12 [B.15-1]. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CSA N288.6-12.

B.15.4 References

[B.15-1] CSA Standard N288.6-12, *Environment Risk Assessments at Class I Nuclear Facilities and Uranium Mines and Mills*, June 2012.

[B.15-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.15-3] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.15-4] OPG Report, P-REP-07010-10011 R000, *Review of Pickering Nuclear Environmental Risk Assessment*, January 2014.

[B.15-5] OPG Report, P-REP-07010-10012 R000, *Environmental Risk Assessment Report for Pickering Nuclear*, January 2014.

[B.15-6] OPG Letter to CNSC, N-CORR-00531-07499 R000, *2015 Results of Environmental Monitoring Programs*, April 26, 2016.

B.16 CSA N288.5-11, "Effluent Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills"

B.16.1 Background

The following paraphrased from the purpose and scope of CSA N288.5-11 [B.16-1] provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *Nuclear facilities or licensed activities can release hazardous or nuclear substances to the surrounding environment. Various federal and provincial/territorial regulations require licensees to monitor and report on the characteristics of airborne and waterborne effluents (e.g., the quantity and concentration of nuclear and hazardous substances that are emitted to the environment).*
>
> *CSA N288.5-11 addresses the design, implementation, and management of an effluent monitoring program that meets legal and business requirements and incorporates current best practices and technologies used internationally.*

CSA N288.5 is relevant to Safety Factor 8 (Safety Performance) and Safety Factor 14 (Radiological Impact on the Environment). N288.5 is not identified in Appendix C of the R04 Licence Conditions Handbook [B.16-2], meaning it is not in Pickering PROL 48.02/2018. N288.5-11 is the first edition of this standard.

As identified in Reference [B.16-3], the Pickering PSR2 review of CSA N288.5-11 is an Incremental review. PSR2 Incremental review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.
- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.16.2 Compliance Assessment for Pickering PSR2

B.16.2.1 Application of PSR1 Reviews

There have not been any previous PSR1 compliance reviews relating to N288.5.

Section 10 of the R04 Pickering License Conditions Handbook [B.16-2] states:

> *CSA standard N288.5-11: "Effluent Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills" addresses the design, implementation, and management of an effluent monitoring program that meets legal and business requirements and incorporates current best practices and technologies used internationally.*

Implementation Strategy: As described in OPG letter N-CORR-00531-06608 (e-Doc 4463626), OPG has completed a gap analysis and developed a preliminary implementation plan to satisfy mandatory requirements of CSA N288.5. OPG will complete implementation of CSA N288.5 mandatory requirements by December 31, 2015. After implementation is achieved, compliance with N288.5 will be measured against OPG program documents.

This Implementation Plan is discussed further below.

B.16.2.2 Application of the Post-PSR1 Reviews

A clause-by-clause review and gap analysis against CSA N288.5-11 was performed for OPG Nuclear (OPGN) including Darlington and Pickering in 2012 [B.16-4]. A summary of the gap analysis results is provided below:

In general, the OPGN effluent monitoring program meets the requirements of the CSA N288.5-11; a number of gaps were identified. A number of gaps are administrative in nature as many of the requirements of CSA N288.5-11 are already performed at sites but they are not documented in the OPGN's effluent monitoring program documents, e.g., STD 31. The identified gaps are summarized as follows; certain aspects of the gaps require further assessment by each site:

1. *Assessment is required to ensure (a) monitoring is performed at the point of release or before dilution occurs, and (b) samples are representative (for permanent and alternative samplers).*
2. *Ensure an Environmental Risk Assessment (ERA) for each nuclear site is up to date. The results of the ERA shall be used to determine substances that need to be monitored.*
3. *Specify performance or acceptance criteria for each effluent stream. Acceptance or performance criteria may include limits on the results of quality control (QC) measurements (e.g., background, blank, or spike samples), sample availability (e.g., number of samples collected, percentage unavailability of continuous monitors), measurement performance (e.g., detection limits), treatment of uncertainties, statistical significance, etc.*
4. *(a) Conduct initial characterization for all effluent streams, and (b) verify the effluent stream characterizations periodically.*
5. *Document justifications or rationales when estimation is used instead of direct measurement.*
6. *Assess if the sampling locations are appropriate to the monitoring objectives and the characteristics of the effluent stream.*
7. *Re-evaluate the estimated airborne emissions annually and/or following any changes that may significantly alter emissions.*

8. *Develop a toxicity management plan for each site.*

9. *Include the requirement to use isokinetic sampling in the effluent monitoring program.*

10. *Determine uncertainty of the reported emissions.*

11. *Include the requirement of routine performance testing of the sample collection system in the effluent monitoring program (this test is currently in place only for radioactive airborne effluents).*

12. *Review the need for, and adequacy of, the effluent monitoring program for each site (a) after any change that has the potential to alter the nature or quantity of the effluent (b) following any update or revision of the ERA for the facility, and (c) every five years after the last review.*

13. *(a) Re-assess the frequency of QA* [quality assurance]*/QC activities, and (b) ensure QC/QA programs are defined for hazardous substances monitoring.*

14. *Annually, review the performance of the effluent monitoring program and the selection of the monitored effluent streams.*

The results of this gap analysis show that the elements of CSA N288.5-11 and OPGN effluent monitoring program are largely equivalent, however, CSA N288.5-11 is more stringent in several areas such as annual review of the program, effluent characterization, interpretation of data, uncertainty of data, quality assurance and quality control, documentation, and reporting.

To address the identified gaps, a three phase Implementation Plan was developed and documented in [B.16-5]. Table 1 of [B.16-5] assigned Asset Suite Action Tracking Assignment Numbers to the gaps under Action Request (AR) #28154791. All Assignments under this AR have subsequently been completed.

As part of the implementation, revised versions of the OPG nuclear standard [B.16-6] and the Pickering site specific emissions monitoring plan [B.16-7] were issued. OPG subsequently notified the CNSC that it was in compliance with CSA N288.5-11 in January 2016 [B.16-8].

It is concluded that there are no PSR2 gaps relating to Pickering NGS compliance with CSA N288.5-11 based on the previously conducted clause-by-clause review.

B.16.3 Compliance Assessment Summary for Pickering PSR2

There are no PSR2 gaps for CSA N288.5-11 [B.16-1]. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CSA N288.5-11.

B.16.4 References

[B.16-1] CSA Standard N288.5-11, *Effluent Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills*, April 2011.

[B.16-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.16-3] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.16-4] OPG Report, N-REP-03480-10012 R000, *Gap Analysis for Effluent Monitoring Program Against CSA N288.5-11*, February 2012.

[B.16-5] OPG Report, N-REP-03480-0460140 R000, *Implementation Plan for Shall Clauses of CSA N288.5-11*, April 2013.

[B.16-6] OPG Standard, N-STD-OP-0031 R006, *Monitoring of Nuclear and Hazardous Substances in Effluents,* 2014.

[B.16-7] OPG Plan, P-PLAN-03480-00001 R008, *Pickering Nuclear Radioactive and Hazardous Emissions Monitoring Plan*, November 2015.

[B.16-8] OPG Letter, N-CORR-00531-07456 R000, *Compliance with CSA N288.5-11: Effluent Monitoring Programs at Class 1 Nuclear Facilities and Uranium Mines and Mills*, January 28, 2016.

B.17 CNSC REGDOC-2.2.2 (2014), "Personnel Training"

B.17.1 Background

The following paraphrase from the introduction and purpose of CNSC REGDOC-2.2.2 (2014) [B.17-1] provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *The purpose of training in the nuclear industry is to ensure that workers are competent and qualified to perform the duties of their position. As required by the General Nuclear Safety and Control Regulations, workers shall be trained to carry on the licensed activity.*
>
> *REGDOC-2.2.2 sets out requirements and guidance for the analysis, design, development, implementation, evaluation, documentation and management of training at nuclear facilities within Canada, including the essential principles and elements of an effective training system.*

CNSC REGDOC-2.2.2 (2014) is directly relevant to Safety Factor 10 (Organization, Management System and Safety Culture).

CNSC REGDOC-2.2.2 (2014) is the first edition of this regulation. The publication notice for the 2014 publication of REGDOC-2.2.2 identifies the following key features [B.17-2]:

1. *The requirement and guidance align with IAEA recommendations on the use of Systematic Approach to Training (SAT) methodology, which is the nuclear industry standard for training development.*

2. *The document formalized the CNSC's existing oversight program for training in nuclear facilities, and provides the basis for assessing the acceptability of licensee training programs.*

Compliance with CNSC REGDOC-2.2.2 (2014) is not a licence requirement for Pickering NGS (per PROL 48.02/2018) and it is not referenced in the R04 Pickering Licence Conditions Handbook (LCH) [B.17-3]. However, the CNSC advised of the following in Reference [B.17-4]:

> *REGDOC-2.2.2, Personnel Training, which was published in August 2014, sets out the CNSC requirements for licensees regarding the development and implementation of a training system. REGDOC-2.2.2 also provides guidance on how these requirements should be met. REGDOC-2.2.2 has not yet been added to the licensing basis of the NPPs. However, each licensee will be expected to conduct a gap analysis of existing practices against REGDOC-2.2.2, and the estimated timeframe for implementation is between 2016 and 2018. At present, licensees continue to meet the SAT requirements as specified in RD-204, Certification of Persons Working at Nuclear Power Plants.*

As part of the recent license renewal process for Darlington NGS ([B.17-5]), the CNSC also noted the following:

> *REGDOC-2.2.2 does not represent a fundamental change to OPG's current training programs. OPG will perform a gap analysis by March 2017 and will take the appropriate actions to ensure that its training system meets the requirements set out in the regulatory document as per OPG's transition plan which will follow after the gap analysis is complete.*

As discussed below in Section B.17.2.2, OPG has completed this assessment with no gaps identified. Since OPG's training governance applies across OPG's nuclear operations, the results are also applicable for the Pickering station.

While REGDOC-2.2.2 may be new, it is important to recognize that the training of personnel is a critical and well-established aspect of the OPG Nuclear program. Of particular relevance is CNSC Regulatory Document RD-204 [B.17-6], for which compliance is a licensing requirement for Pickering as specified in Section 3.3 of Reference [B.17-3]. RD-204 specifies requirements that ensure personnel are sufficiently qualified to perform the duties associated with specific positions for which CNSC certification is required. This includes specification of the type of training required for applicable positions associated with nuclear power plant operation. Although RD-204 does not provide detailed requirements with respect to how training is to be managed, Section 4.3 of [B.17-6] states:

> *The licensee shall establish and implement the initial and continuing training programs specified in Part II and Part III of this regulatory document in accordance with the principles of a systematic approach to training.*

CNSC REGDOC-2.2.2 is complementary to RD-204 in that it specifies requirements for the analysis, design, development, implementation, evaluation, documentation and management of training for staff performing licensed activities, including guidance for the application of SAT principles.

As identified in Reference [B.17-7], the Pickering PSR2 review of REGDOC-2.2.2 (2014) is an Incremental review. PSR2 Incremental review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- <u>Compliance:</u> Compliance indicates that the change in the safety requirement, per the topical review, is met.

- <u>Gap</u>: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.17.2 Compliance Assessment for Pickering PSR2

B.17.2.1 Application of PSR1 Reviews

The versions of REGDOC-2.2.2 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

The first edition of REGDOC-2.2.2 was issued in August 2014 after the completion of the Pickering B Integrated Safety Review (ISR) and Pickering A Return-to-Service. Therefore, no code reviews were performed for Pickering Units 5-8 or Units 1,4 against REGDOC-2.2.2.

Darlington NGS

The first edition of REGDOC-2.2.2 was issued in August 2014 after the completion of the code reviews for the Darlington ISR. Therefore, no code review was performed for Darlington against REGDOC-2.2.2.

As part of the Darlington ISR, a code review was performed against RD-204 (2008) as documented in Reference [B.17-8]. Only three issues were identified, none of which are pertinent to the content of REGDOC-2.2.2.

B.17.2.2 Application of Post PSR1 Reviews

As discussed above, a review of CNSC REGDOC-2.2.2 was not undertaken as part of the Pickering or Darlington PSR1 reviews, as the document did not exist at the time.

As noted in Section B.17.1 above, REGDOC-2.2.2 provides more specific requirements with respect to training in relation to the more general requirements contained in RD-204 [B.17-6]. A PSR2 code review of RD-204 (2008) has been performed, which identified no gaps and concluded that Pickering is in compliance with RD-204 (2008).

OPG has completed a gap assessment of CNSC REGDOC 2.2.2 (2014), relative to the OPG Nuclear training governance, as documented in [B.17-9]. The assessment maps each of the requirements in CNSC REGDOC-2.2.2 (2014) to the corresponding sections of applicable OPG governing documents and in so doing, demonstrates that the OPG Nuclear Training Program is fully compliant. No gaps are identified.

As part of this PSR2 assessment, the following summary table has been prepared that provides additional information describing the degree of conformance with the clauses of CNSC REGDOC-2.2.2 (2014) [B.17-1] and supports the conclusion of [B.17-9] that there are no gaps and that the OPG Nuclear Training Program is fully compliant.

CNSC REGDOC-2.2.2 Section	PSR2 Review	Compliant or Gap
1. Introduction	This Section provides background information with respect to the purpose, scope and other relevant legislation. It does not provide specific requirements, but it does state the following high level requirements per the "General Nuclear Safety and Control Regulations" (paraphrased): • Workers shall be trained to carry on the licensed activity. • Every licensee shall ensure the presence of a sufficient number of qualified workers to carry on the licensed activity safely. • Every licensee shall train the workers to carry on the licensed activity. In addition, the regulatory document is identified as applying to workers in positions where the consequence of human error poses a risk to the environment, the health and safety of persons, or the security of the nuclear facilities and of nuclear substances, and states: • The licensees shall define these positions in their training governing documents. <u>Assessment</u> N-PROG-TR-0005, "Training" [B.17-10] describes the OPG Nuclear Training Program. Section 1 clearly states the OPG direction that training is used to develop and maintain competent personnel to safely operate, maintain and improve plant performance, to minimize the impact of plant operation on environment, health and public safety, and to drive human performance improvements in a cost-effective manner. N-PROG-TR-0005 and the extensive suite of performance references that support it clearly demonstrate OPG's commitment to ensuring effective training of all workers engaged in performing licensed activities such that the intent of the high level requirements embodied in the first three bullets noted above is fully met. Since these apply across OPG Nuclear, this conclusion is applicable to the Pickering station. N-PROG-TR-0005 is specified as being applicable to staff in roles that are identified in N-LIST-08920-10001, "Trained Performance Areas" [B.17-11]. Table 1 in N-LIST-08920-10001 provides a comprehensive listing of all OPG Nuclear positions to which N-PROG-TR-0005 applies and identifies the Training and Qualification Description (TQD) document that has been prepared for each position. In addition to including positions that are	Compliant

CNSC REGDOC-2.2.2 Section	PSR2 Review	Compliant or Gap
	directly involved in nuclear power plant Operations and Maintenance, the list includes staff in roles such as: • Emergency Response • Fleet Support Services (e.g., Chemistry Lab, Radiation Protection, Engineering, Environment, Supply Chain, Conventional Safety, Work Protection) • Fire Protection • Health Physics • Inspections • Leadership and Management • Trainer • Security • Waste Management • Audit Personnel Also identified is a TQD for Nuclear General Employee, which applies to all staff in OPG Nuclear. N-LIST-08920-10001 further distinguishes staff who belong to Major Technical Performance Areas that are deemed to have sufficient importance to require a higher level of oversight. This includes staff in Fleet Operations (i.e., Authorized Nuclear Operator, Shift Manager/Control Room Shift Supervisor, Nuclear Certified Personnel, Control Room Operator and Nuclear Operator) as well as staff in other safety-related roles (i.e., Chemistry Lab Technician, Radiation Protection Technician, Engineering, Trainer, and Maintenance). Even for staff who are not in roles identified in N-LIST-08920-10001, N-PROG-TR-0005 specifies that their training still comply with essential elements of the OPG nuclear training program, including: N-PROC-TR-0007, "On-the-Job Training, On-the-Job Evaluation, and Practical Evaluation Process" [B.17-12], N-PROC-TR-0008, "Systematic Approach to Training" [B.17-13], N-PROC-TR-0012, "Records and Documentation" [B.17-14], N-PROC-TR-0041, "TIMS II Administration" [B.17-15] and N-PROC-TR-0044, "Training Demand, Scheduling, and Cancellation Process" [B.17-16]. The positions identified in N-LIST-08920-10001 are comprehensive in terms of identifying workers in roles for which the consequence of human error may pose risks impacting on the environment, health and safety, and security. This fully meets the intent of the fourth bullet noted above. Since this applies across OPG Nuclear, this conclusion is applicable to the Pickering station.	

CNSC REGDOC-2.2.2 Section	PSR2 Review	Compliant or Gap
2. Principles	This Section states that a licensee's training system adhere to the following fundamental principles: • Performance-oriented: All instruction that is subject to this regulatory document shall focus on essential knowledge, skills and safety-related attributes required to meet job requirements and nuclear-safety-specific needs throughout the lifecycle of the facility. • Systematically developed: Training shall be defined, produced and maintained through an iterative and interactive series of steps, leading from the identification of a training requirement to the confirmation that the requirement has been satisfied. <u>Assessment</u> N-PROG-TR-0005, Training [B.17-10] describes the OPG Nuclear Training Program. With respect to the second principle noted above, Section 1.1.7 explicitly states the expectation that a systematic approach be used for the analysis, design, development, implementation and evaluation of training. N-PROC-TR-0008, Systematic Approach to Training [B.17-13] is cited as a performance reference, and it describes in detail the systematic approach to training that is followed in OPG Nuclear. With respect to the first principle noted above, Sections 1.1 through 1.7 of N-PROC-TR-0008 provide detailed instructions for the analysis, design, development, implementation and evaluation of training. This includes steps (e.g., training needs analysis, job and task analysis, development of terminal and enabling learning objectives, trainee performance evaluation) to ensure that essential knowledge and skills to perform each job are being identified, trained and confirmed to be learned. OPG's training governance does not make explicit reference to the concept of "safety-related attributes" as defined in REGDOC-2.2.2. However, this is considered to be implicitly addressed by OPG's implementation of programs and processes such as N-PROG-AS-0002 (Human Performance) [B.17-17], N-STD-OP-0002 (Procedure Use and Adherence) [B.17-18] and N-STD-OP-0012 (Conservative Decision-Making) [B.17-19] in conjunction with specific instructions for training development as stated in N-PROC-TR-0008, such as: • Section 1.3.2 indicating that Job and Task Analysis include consideration of "performance expectations", "attitudes", "consequences of inadequate performance" and "personnel and personal safety issues".	Compliant

CNSC REGDOC-2.2.2 Section	PSR2 Review	Compliant or Gap
	• Section 1.4 indicating that training should be designed to "develop required attitudes" and "reinforce conduct and performance expectations". • Section 1.5.3.1 indicating that Lesson Plans address "key management expectations for such concepts as personal safety, procedure use and adherence, radiation safety and conservative decision-making". With respect to training needs over the life cycle of the facility, it is noted that OPG's training program is focused on the operational phase of its plants and specific reference is not made to other phases such as Construction or Decommissioning. However, REGDOC-2.2.2 is clearly stated as being applicable to licensed activities, which in the context of PSR2 applies only to Pickering as an operating nuclear power plant. On this basis, the OPG Nuclear training program fully meets the intent of the two principles noted above. Since this applies across OPG Nuclear, this conclusion is applicable to the Pickering station.	
3. Training Systems for Nuclear Facilities	This Section states the following high level requirements : • *Licensees shall ensure workers who carry on licensed activities are qualified to do the work assigned to them through the use of a training system to systematically analyze, design, develop, implement evaluate, document and manage new training and the revision of existing training, including continuing training.* • *It [the training system] shall be used whether the training is defined, designed, developed, implemented, evaluated, recorded and managed internally by licensees or externally through vendors or contractors.* This Section goes on to state the following specific requirements that licensees shall: *1. Identify all performance requirements of a job or duty area relating to licensed activities by conducting a job analysis to determine all of the tasks involved.* *2. Define and document the necessary general worker training, initial job training and continuing training requirements for workers, based on a task analysis of the knowledge and skills required to perform each task and the safety-related attributes required to perform their duties.*	Compliant

CNSC REGDOC-2.2.2 Section	PSR2 Review	Compliant or Gap
	3. *Ensure that appropriate training is designed, developed and implemented to meet the qualification requirements.* 4. *Ensure that trainers meet and maintain documented qualification requirements, particularly in the areas of subject matter expertise and instructional skills.* 5. *Ensure that formal evaluations are used to confirm and document that all trained workers are qualified to perform their duties.* 6. *Implement a training change-management process that will systematically analyze procedural and equipment changes, changes in job descriptions, and operating experience feedback (including facility and industry-wide events), in order to identify changes to the tasks and task lists and to assess potential training implications leading to training modifications.* 7. *Ensure continuing training is provided to workers as deemed necessary through the job and task analyses processes, and that it includes updates to training programs stemming from the change-management process as identified through the training needs analysis process.* 8. *Evaluate training regularly and incorporate the results of the evaluations into a training improvement process.* 9. *Ensure that workers' records in support of training and qualifications are established and maintained.* 10. *Ensure that workers have a level of training related to nuclear safety corresponding to the duties of their position and employment, including but not limited to radiation safety, fire safety, onsite emergency arrangements and conventional health and safety.* <u>Assessment</u> For the first high level requirement noted above, refer to the assessment discussion for Sections 1, 2 and 5 of REGDOC-2.2.2. For the second high level requirement noted above, N-PROC-TR-0002, "Control of Vendor-Supplied Training" [B.17-20] specifies the processes by which training developed or delivered by organizations external to OPG is managed to ensure that the requirements of the OPG Nuclear training program (as specified in N-PROG-TR-0005, "Training" [B.17-10]) are met.	

CNSC REGDOC-2.2.2 Section	PSR2 Review	Compliant or Gap
	For the ten specific requirements noted above, all aspects are explicitly covered as part of the processes OPG follows for the analysis, design, development, implementation and evaluation of training, as specified in N-PROC-TR-0008, "Systematic Approach to Training" [B.17-13]. Specifically: 1. Compliance with this requirement is demonstrated in Sections 1.2 and 1.3 of N-PROC-TR-0008 and other governing documents such as N-PROC-TR-0021, "Training and Qualification Description Development and Approval Process" [B.17-26]. 2. Compliance with this requirement is demonstrated in Sections 1.2 and 1.3 of N-PROC-TR-0008 and other governing documents such as N-PROC-TR-0021, "Training and Qualification Description Development and Approval Process" [B.17-26]. 3. Compliance with this requirement is demonstrated in Sections 1.4, 1.5 and 1.6 of N-PROC-TR-0008 and other governing documents such as N-STD-TR-0001, "Conduct of Training" [B.17-25]. 4. Compliance with this requirement is demonstrated in Sections 1.2, 1.5.4, 1.6.2 and 1.7.1.3 of N-PROC-TR-0008 and other governing documents such as N-TQD-602-00001, "Nuclear Trainer Training and Qualification Description" [B.17-21] and N-STD-TR-0001, "Conduct of Training" [B.17-25]. 5. Compliance with this requirement is demonstrated in Sections 1.4, 1.5 and 1.6 of N-PROC-TR-0008 and other governing documents such as N-STD-TR-0001, "Conduct of Training" [B.17-25], N-PROC-TR-0007, "On-the-Job Training, On-the-Job Evaluation and Practical Evaluation Process" [B.17-12] and N-PROC-TR-0018, "Examination Security, Development, Approval and Implementation" [B.17-27]. 6. Compliance with this requirement is demonstrated in Sections 1.1 (particularly sub-sections 1.1.1.2, 1.1.1.3 and 1.1.1.4), 1.4.3, 1.5.5 and 1.7.1.4 of N-PROC-TR-0008 and other governing documents such as N-INS-08920-10037, "Training Change Control" [B.17-22]. 7. Compliance with this requirement is demonstrated in Section 1.3.2 of N-PROC-TR-0008 and other governing documents such as N-INS-08920-10021, "Continuing and Requalification Training – Curriculum Development and Implementation Process" [B.17-23] and N-PROC-TR-0021, "Training	

CNSC REGDOC-2.2.2 Section	PSR2 Review	Compliant or Gap
	and Qualification Description Development and Approval Process" [B.17-26]. 8. Compliance with this requirement is demonstrated in Sections 1.6.4, 1.6.7 and 1.7.1 of N-PROC-TR-0008 and other governing documents such as N-STD-TR-0001," Conduct of Training" [B.17-25] and N-INS-08920-10017, "Training Committees"[B.17-28]. 9. Compliance with this requirement is demonstrated in N-PROC-TR-0012," Records and Documentation" [B.17-14], N-PROC-TR-0041, "TIMS II Administration" [B.17-15] and other governing documents such as N-STD-TR-0001, "Conduct of Training" [B.17-25]. 10. Compliance with this requirement is demonstrated throughout N-PROC-TR-0008, as well as N-LIST-08920-10001, "Trained Performance Areas" [B.17-11], N-PROC-TR-0021, "Training and Qualification Description Development and Approval Process" [B.17-26] and N-STD-TR-0001, "Conduct of Training" [B.17-25]. On this basis, the OPG Nuclear training program fully meets the intent of the requirements specified in Section 3 of REGDOC-2.2.2. Since this applies across OPG Nuclear, this conclusion is applicable to the Pickering station.	
4. Records Management for a Training System	This Section states the following specific requirements (paraphrased): • Licensees shall develop and manage documentation related to all phases of their training including analysis, design, development, implementation and evaluation. • Licensees shall maintain records on the training and qualification of all workers and these records shall: o be managed and controlled, o have immediate, unencumbered and ready access by workers' supervisors and managers, o include all qualifications and certifications granted by or relied upon by the licensee, and o include expiration dates for time-sensitive qualifications and certifications, and all requalification or recertification requirements.	Compliant

CNSC REGDOC-2.2.2 Section	PSR2 Review	Compliant or Gap
	<u>Assessment</u> N-PROG-TR-0005, Training [B.17-10] describes the OPG Nuclear Training Program. Sections 1.17.1 and 1.17.2 explicitly state the expectations with respect to training records and documentation, and the Training Information Management System (TIMS). N-PROC-TR-0012, "Records and Documentation" [B.17-14] is cited as a performance reference, and it provides direction for completing, processing and maintaining training documents and records in OPG Nuclear. Two types of training records are recognized: • Training support materials (i.e., documents generated through the analysis, design, development, implementation and evaluation of training in accordance with N-PROC-TR-0008 [B.17-13]); and • Individual Training Records (ITRs) (i.e., documented evidence of a person's completion of training requirements). With respect to training support materials, Table 1 in Section 1.6.2 of N-PROC-TR-0012 identifies the documents that are considered to be essential records, and which of these are to be managed as Controlled Documents in accordance with OPG-PROC-0179, "Nuclear Quality Assurance Records" [B.17-24]. ITRs are maintained in TIMS for all staff in OPG Nuclear, in accordance with N-PROC-TR-0041, "TIMS II Administration" [B.17-15], which is also cited as a performance reference in N-PROC-TR-0012. N-PROC-TR-0041 describes the processes by which ITRs are managed and controlled using TIMS. An individual's TIMS information identifies all qualifications and certifications that they possess, including requalification requirements and expiration dates. Per Section 1.8 of N-PROC-TR-0012, ready access to an individual's ITRs is provided to the individual, their First Line Manager and Direct Line Manager, as well as others having a business need to access this information (e.g., training staff). Control and processing of documentation related to examination materials is managed in accordance with N-PROC-TR-0018, "Examination Security, Development, Approval and Implementation" [B.17-27]. The above demonstrates that the OPG training program fully meets the intent of Section 4 of REGDOC-2.2.2 and	

CNSC REGDOC-2.2.2 Section	PSR2 Review	Compliant or Gap
	since it applies across OPG Nuclear, this conclusion is applicable to the Pickering station.	
5. Guidance of the Systematic Approach to Training	This Section provides guidance only and describes the phases and recommended content of a training program that is developed in accordance with SAT principles. Five phases of a SAT-based program are outlined as follows: 1. Analysis, containing details related to: • Training needs analysis. • Job and task analysis. • Learning objectives (specifically terminal learning objectives). • Target audience analysis. 2. Design, containing details related to: • Trainee characteristics. • Instructional program design. • Enabling objectives. • Learning assessment plan. • Instruction strategies. • On-the-job training. • Training development plan. 3. Development, containing details related to: • Procurement/production of training materials. • Assessment tests. • Conduct of trials (pilot courses). 4. Implementation, containing details related to instructor preparation and the actual delivery of training. 5. Evaluation, containing details related to assessing the effectiveness and efficiency of training, as well as change management. <u>Assessment</u> N-PROG-TR-0005, "Training" [B.17-10] describes the OPG Nuclear Training Program. Section 1.1.7 explicitly states the expectation that a systematic approach be used for the analysis, design, development, implementation and evaluation of training. N-PROC-TR-0008, "Systematic	Compliant

CNSC REGDOC-2.2.2 Section	PSR2 Review	Compliant or Gap
	Approach to Training" [B.17-13] is cited as a performance reference, and it describes the systematic approach to training that is followed in OPG Nuclear. The structure of N-PROC-TR-0008 is fully aligned with the five SAT phases defined in Section 5 of REGDOC-2.2.2. Detailed instructions for each phase are provided which address all aspects of the guidance contained in Section 5 of REGDOC-2.2.2. Implementation of N-PROC-TR-0008 is supported in part by other aspects of the OPG Nuclear Training Program, including: N-LIST-08920-10001, "Trained Performance Areas" [B.17-11], N-PROC-TR-0007, "On-the-Job Training, On-the-Job Evaluation and Practical Evaluation Process" [B.17-12] and N-PROC-TR-0002, "Control of Vendor-Supplied Training" [B.17-20]. N-PROC-TR-0008 fully meets the intent of Section 5 of REGDOC-2.2.2 and since it applies across OPG Nuclear, this conclusion is applicable to the Pickering station.	

Note that REGDOC-2.2.2 is programmatic in nature and, as such, the status of OPG's compliance with it is not impacted by the possibility of continued operation of Pickering beyond 2020.

B.17.3 Compliance Assessment Summary for Pickering PSR2

There are no PSR2 gaps for CNSC REGDOC-2.2.2 (2014) [B.17-1]. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CNSC REGDOC-2.2.2 (2014).

B.17.4 References

[B.17-1] CNSC Regulatory Document REGDOC-2.2.2, *Personnel Training*, August 2014.

[B.17-2] CSNC Website, *Archived - CNSC publishes REGDOC-2.2.2, Personnel Training*, http://news.gc.ca/web/article-en.do?nid=874969, August 13, 2014.

[B.17-3] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633 R000, *Pickering NGS: Licence Conditions Handbook*, December 23, 2015.

[B.17-4] CNSC Letter, P-CORR-00531-04495 R000, L. Levert to B. McGee, *Presentation of the 2014 NPP Report*, June 16, 2015.

[B.17-5] CNSC Letter, NK38-CORR-00531-17515 R000, L. Levert to B. Duncan, *August 19, 2015 Public Hearing*, July 9, 2015.

[B.17-6] CNSC Regulatory Document, RD-204, *Certification of Persons Working at Nuclear Power Plants*, February 2008.

[B.17-7] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.17-8] OPG Report, NK38-REP-03680-10095 R000, *Review of CNSC Regulatory Document RD-204 (February 2008) Certification of Persons Working at Nuclear Power Plants for Darlington Integrated Safety Review*, August 2011.

[B.17-9] OPG List, N-LIST-08920-10002 R000, *CNSC REG DOC 2.2.2 To OPGN Training Governance Cross Matrix*, July 27, 2016.

[B.17-10] OPG Program, N-PROG-TR-0005 R016, *Training*, January 2016.

[B.17-11] OPG List, N-LIST-08920-10001 R008, *Trained Performance Areas*, March 2016.

[B.17-12] OPG Procedure, N-PROC-TR-0007 R011, *On-the-Job Training, On-the-Job Evaluation, and Practical Evaluation Process*, March 2016.

[B.17-13] OPG Procedure, N-PROC-TR-0008 R020, *Systematic Approach to Training*, March 2016.

[B.17-14] OPG Procedure, N-PROC-TR-0012 R012, *Records and Documentation*, October 2012.

[B.17-15] OPG Procedure, N-PROC-TR-0041 R011, *TIMS II Administration*, March 2013.

[B.17-16] OPG Procedure, N-PROC-TR-0044 R006, *Training Demand, Scheduling, and Cancellation Process*, October 2014.

[B.17-17] OPG Program, N-PROG-AS-0002 R016, *Human Performance*, May 2016.

[B.17-18] OPG Standard, N-STD-AS-0002 R015, *Procedure Use and Adherence*, September 2015.

[B.17-19] OPG Standard, N-STD-OP-0012 R004, *Conservative Decision-Making*, October 2012.

[B.17-20] OPG Procedure, N-PROC-TR-0002 R007, *Control of Vendor-Supplied Training*, November 2013.

[B.17-21] OPG Document, N-TQD-602-00001 R017, *Nuclear Trainer Training and Qualification Description*, April 2015.

[B.17-22] OPG Instruction, N-INS-08920-10037 R000, *Training Change Control*, June 2016.

[B.17-23] OPG Instruction, N-INS-08920-10021 R000, *Continuing and Requalification Training – Curriculum Development and Implementation Process*, March 2009.

[B.17-24] OPG Procedure, OPG-PROC-0179 R001, *Nuclear Quality Assurance Records*, March 2016.

[B.17-25] OPG Standard, N-STD-TR-0001 R018, *Conduct of Training*, May 2012.

[B.17-26] OPG Procedure, N-PROC-TR-0021 R011, *Training and Qualification Description Development and Approval Process*, June 2016.

[B.17-27] OPG Procedure, N-PROC-TR-0018 R008, *Examination Security, Development, Approval and Implementation*, June 2012.

[B.17-28] OPG Instruction, N-INS-08920-10017 R005, *Training Committees*, October 2015.

B.18 CNSC REGDOC-2.3.2 (2015), "Accident Management, Version 2"

B.18.1 Background

The following paraphrased from the purpose and scope of CNSC REGDOC-2.3.2 Version 2 (2015) [B.18-1] provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *REGDOC-2.3.2 sets out the requirements and guidance of the Canadian Nuclear Safety Commission (CNSC) for the development, implementation and validation of accident management programs for reactor facilities.*
>
> *Accident management is a commitment to the defence-in-depth approach and is an important component in the licensee's overall capabilities to ensure the risks from nuclear reactors remain low. Defence in depth is applied to all organizational, behavioural, and design-related safety and security activities to ensure they are subject to overlapping provisions. It is important for licensees to implement and maintain operational procedures, guidelines and adequate capabilities to deal with abnormal situations and accidents, including severe accidents. REGDOC 2.3.2 specifies safety principles, high-level requirements and supporting guidelines that allow licensees to develop, implement, and evaluate an integrated accident management program, which includes components that address severe accident management.*

REGDOC-2.3.2 Version 2 (2015) is applicable to Safety Factors 1 (Plant Design), 5 (Deterministic Safety Analysis), 6 (Probabilistic Safety Assessment), 7 (Hazard Analysis), 8 (Safety Performance), and 10 (Organization, the Management System and Safety Culture).

Compliance with REGDOC-2.3.2 Version 2 (2015) is not currently a licence requirement for Pickering NGS (in accordance with PROL 48.02/2018) per the R04 Pickering Licence Conditions Handbook [B.18-2].

REGDOC-2.3.2 Version 2 (2015) supersedes:

- REGDOC-2.3.2, "Accident Management (October 2014)" – Issued for Comment [B.18-3]
- REGDOC-2.3.2, "Severe Accident Management Programs for Nuclear Reactors" (September 2013) [B.18-4]
- G-306, "Severe Accident Management Programs for Nuclear Reactors" (2006) [B.18-5]

The updated version reflects lessons learned from the Fukushima nuclear event of March 2011, and addresses findings from the CNSC Fukushima Task Force Report [B.18-6].

The results of PSR1 REGDOC-2.3.2 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.18.2. As identified in Reference [B.18-7], the Pickering PSR2 review of CNSC REGDOC-2.3.2 (2015) is an Incremental review. PSR2 Incremental review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.

- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.18.2 Compliance Assessment for Pickering PSR2

B.18.2.1 Application of PSR1 Reviews

The versions of REGDOC-2.3.2 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

REGDOC-2.3.2 was not issued at the time the Pickering B ISR was performed. As part of the ISR, a high level intent review of a predecessor CNSC document G-306 was performed and documented in [B.18-8]. This review identified several gaps relating to Severe Accident Management Guidelines (SAMG) implementation and an Action Tracking request (AR #28064164) was created to address the gaps and was subsequently closed.

However, developments relating to the Fukushima event, Beyond Design Basis Accident (BDBA) response, SAMG and the issuance of REGDOC 2.3.2 Version 2, have superseded significant elements of the assessment performed for Pickering B.

Pickering Units 1,4

Neither REGDOC-2.3.2 nor G-306 were available when the Pickering A safety assessment was conducted and no standards relating to BDBA or Severe Accident Management requirements were addressed as part of Return-to Service.

Darlington NGS

For the Darlington ISR a clause-by-clause review against G-306 was performed in 2011 [B.18-9]. This review identified several gaps that were subsequently tracked to completion under an Action Request. Subsequently, a Code Refresh review of Darlington against the first version of REGDOC 2.3.2 [B.18-4] was conducted in February 2014 [B.18-10]. The major findings from the review were:

- *DNGS does not currently fully meet the intent of CNSC REGDOC-2.3.2.*

- *All clauses which were determined not to meet the full intent of the clause have a defined, on-going work scope for which compliance is expected upon completion of the work scope.*

- *It is expected that Darlington will meet the full intent of CNSC REGDOC-2.3.2 by the end of 2015, coinciding with the completion of OPG's SAMG Implementation Plan.*

The REGDOC-2.3.2 review identified 5 gaps related to:

- #02193 performance of instrumentation
- #02194 instrument and equipment survivability
- #02195 availability and accuracy of instrumentation
- #02196 drills and training for multi-unit events
- #02197 SAM validations of reviews not complete

All gaps were assigned a safety significance level of 4 and grouped under ISR Issue #D565 [B.18-11] to be addressed via ongoing work under AR #28153869. This review and the resultant gaps and Issue are generic/programmatic for OPG, and are hence fully applicable to Pickering. However, as detailed in the section below, all gaps have subsequently been addressed.

B.18.2.2 Application of Post PSR1 Reviews

The scope and structure of REGDOC-2.3.2 (October 2014) [B.18-3] and Version 2 (September 2015) [B.18-1] have significantly changed from the previous version (2013) [B.18-4]. Whereas the 2013 version of the document specifically dealt with Severe Accident Management Programs, the new version is broader in terms of an overall Accident Management (AM) program, which encompasses Anticipated Operational Occurrences (AOOs), Design Basis Accidents (DBAs) and BDBAs including Design Extension Conditions (DECs)[10] and Severe Accidents (SAs). Other significant changes relate to:

1. Providing guidance for an integrated AM framework (Section 2.0)
2. Documenting the AM framework as it relates to the levels of defense-in-depth (Section 3.1)
3. Recognition of the application of BDBA Emergency Mitigating Equipment (EME) and EME Procedures (throughout document)

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[10] DECs are a subset of BDBAs that are considered in the design process of the facility in accordance with best-estimate methodology to keep releases of radioactive material within acceptable limits. DECs could include severe accident conditions.

4. Identification and treatment of cliff-edge effects (Section 4.2.1)

5. The concept of Design Extension Conditions (DECs) and Complementary Design Features (CDFs) (Section 4.3.1)

6. CDFs for prevention of core melt at high pressure and hydrogen detonation (Section 4.3.1) and direct containment heating form corium (Section 4.2.1)

7. Consideration and treatment of extreme external hazards (Section 4.2.1) and incremental requirements on the plant and resources (Section 4.3.3)

8. Human interactions, including environmental conditions and timing of actions (Section 4.3.1)

9. Multi-unit failures and Irradiated Fuel Bay (IFB) failures (Section 4.2.5)

10. More specific details relating to suitability of instrumentation and equipment for accident mitigation (Section 4.3.1 and Section 4.3.2)

11. Assessment of external resources (Section 6.3)

The overriding objective of the REGDOC relates to ensuring the licensees have developed a comprehensive AM program. AOOs are generally upsets rather than accidents and are addressed by responding to the failure or condition in accordance with event based procedures (i.e., response to alarms, Operating Manuals, Abnormal Incidents Manuals (AIMs)). They are generally mitigated by process system action and human action and do not require AM activities beyond those of the normal plant Operations complement. For DBAs, Pickering has a fully mature AM program for all internal and external accidents. These events are fully assessed in the Safety Reports, risk assessments and supplementary assessments. Operating documents by way of AIMs have long been established to address DBAs, and in some cases BDBAs, where coincidental random or consequential failures have occurred. This framework has long been part of the licencing basis for Pickering and addresses all elements of AM, including:

- Identification and analysis of upsets and accidents in the Safety Report
- Established procedures to deal with upsets and accidents
- Validation and verification of procedures
- Qualification and availability of equipment and systems
- Training and qualification of staff to exercise response, including use of the simulator
- Assurance of resources (minimum shift complement), and response capability
- Interfaces with the Emergency Management organization

BDBA response, where Event Based response may not have been successful, or where multiple failures have occurred would be addressed via Symptom Based response, largely contained in the Critical Safety Parameter Monitoring (CSPM) procedures including Restoration Procedures. These Restoration Procedures were designed to restore critical parameters to within an acceptable range. Both the Event and Symptom Based procedures would result in accident unit(s) being in a controlled state with a long term heat sink, radioactive releases controlled and containment integrity maintained.

Hence, OPG Pickering is fully compliant with the Design Basis AM requirements in REGDOC-2.3.2, Version 2 (2015).

However, developments relating to a more structured response and AM framework for BDBA response are more recent, and have been significantly influenced by the Fukushima event in 2011. Therefore, this Incremental review is focused on the changes between the 2013 and 2015 version of the REGDOC as they pertain to BDBA AM and the lessons learned from Fukushima.

OPG undertook a significant initiative relating to obtaining insight into accident progression and accident management following the Fukushima event. As a result OPG has implemented several improvements relating to defence-in-depth and emergency response. In all, there were thirty-six issues/topics identified in Fukushima Action Items (FAIs) that were assessed and addressed. A summary of each of these FAIs and corresponding references is included in [B.18-12]. All of these FAIs have been closed, with one outstanding action relating to Emergency Response Prediction code improvements. This has been identified as a PSR2 gap in the PSR2 Safety Factor 13 report [B.18-13]. In addition to and in parallel with the FAI work, OPG as part of an industry initiative, participated in a project to enhance the overall SAMG to incorporate lessons learned from Fukushima and other industry insights/developments (COG-JP-4426). This project produced a series of topical reports relating to:

- Instrumentation and Equipment Survivability
- Multi-unit events
- Shutdown and Low Power states
- In-Vessel retention
- Habitability
- Containment Integrity
- Irradiated Fuel Bays
- Fukushima Lessons Learned
- Impact Assessment of Operating Experience (OPEX) on SAMG Documentation

Information, insights and recommendations from these generic reports were used to update OPG and Pickering specific documentation as part of the FAI process and BDBA (including SAMG) management program.

The significant changes identified in REGDOC-2.3.2, Version 2 (2015) are grouped under topical subheadings and assessed for Pickering under those headings below:

Accident Management Framework and Levels of Defense

OPG's AM framework for BDBAs, which is applicable to Pickering, is outlined in [B.18-14]. Appendix A of [B.18-14] illustrates the defense-in-depth as it relates to accident barriers. It details the interfaces between the design basis and beyond design basis response. Reference [B.18-20] summarizes the technical basis for the BDBA accident response while [B.18-15] details the transition from DBA into BDBA and the AM process once into BDBA response. BDBA accident response is either entered from Symptom Based procedures (CSPM) or certain Event Based procedures. Severe Accident Management response and SAMG entry occurs if SAMG entry conditions are met during the execution of the BDBA response (e.g., while executing EME Guidelines).

Therefore, OPG Pickering has an AM framework that is in compliance with REGDOC-2.3.2, Version 2 (2015), and there are no gaps relating to the first three incremental changes identified above, specifically:

1. *Providing guidance relating to an integrated AM framework (Section 2.0)*
2. *Documenting the Accident Management framework as it relates to the levels of defence-in-depth (Section 3.1)*
3. *Recognition of the application of BDBA Emergency Mitigating Equipment (EME) and EME Procedures (throughout document)*

Identification and Treatment of Cliff-Edge Effects

4. *Identification and treatment of cliff-edge effects (Section 4.2.1)*

Cliff-edge effects were implicitly addressed as part of the review of external hazards at Pickering for capability to manage external events [B.18-16].

Additionally, work done for instrument and equipment survivability explicitly considered the margin to confirm that incremental changes in conditions would not result in a step change in consequences. No specific instances were identified.

Potential cliff-edge effects were evaluated as part of the FAIs process [B.18-12]. The applicable FAIs are:

- FAI 1.1.1 – bleed condenser relief capacity
- FAI 1.2.1 – shield tank/calandria vault relief
- FAI 1.3.1 – adequacy of protection of containment integrity

These FAIs were completed and subsequently closed as detailed in [B.18-19]. It is therefore concluded that OPG Pickering has an AM framework that is in compliance with REGDOC-2.3.2, Version 2 (2015), and there are no gaps relating to the introduction of the requirement relating to cliff-edge effects.

<u>Design Extension Conditions and Complementary Design Features</u>

5. *The concept of Design Extension Conditions (DECs) and Complementary Design Features (CDFs) (Section 4.3.1)*

6. *CDFs for prevention of core melt at high pressure and hydrogen detonation (Section 4.3.1) and direct containment heating form corium (Section 4.2.1)*

CDFs are those modifications or features that are specifically provided to address DECs and are defined in [B.18-21]. Phase 1 EME is a CDF, which includes power, water and procedures, and has been fully implemented. Further detail relating to these CDFs is provided in [B.18-22] and [B.18-23]. These features provide additional capabilities for heat sinks (fuel cooling) and monitoring for BDBAs. In addition, Passive Autocatalytic Recombiners (PARs) have been installed in all Pickering units for BDBA hydrogen mitigation and this issue has been closed under FAI 1.4.1. EME Guidelines have been developed for the deployment, connection and operation of EME in the event it is required for a BDBA.

However, Clause 4.2.1 of REGDOC-2.3.2 states:

> *Challenges that are not considered in the reactor design envelope, but could threaten the integrity of the containment should be practically eliminated;*

Full provision of CDFs for containment integrity for BDBAs, as required by REGDOC-2.3.2, will be addressed with the completion of Phase 2 EME. As discussed in [B.18-24], Phase 2 EME will provide a longer term supply of AC power to restore cooling provisions (e.g., Reactor Building Air Conditioning Units) and the Filtered Air Discharge System for containment venting. Phase 2 EME is currently scheduled to be fully implemented by the end of 2017 [B.18-25]. Since this work is still in progress, it is identified as **<u>PSR2 REGDOC-2.3.2 Version 2 (2015) Gap #1</u>**.

<u>Treatment of External Hazards and Human Interactions</u>

7. *Consideration and treatment of extreme external hazards (Section 4.2.1) and incremental requirements on plant and resources (Section 4.3.3)*

8. *Human interactions, including environmental conditions and timing of actions (Section 4.3.1)*

External hazards have been addressed in [B.18-16] and as part of the DEC review discussed above. Hence, the incremental implication of this change is specific to the issues of resources and timing. The timing and resources for response to BDBA external events as evaluated in [B.18-26] and [B.18-27] provide an overall summary of EME deployment timing. The EME deployment action time Safety Limits for Pickering 1,4 and Pickering 5-8 are summarized in Tables A.2 of [B.18-22] and [B.18-23] respectively.

In addition to the above references, adequacy of timing and resources are evaluated in the emergency preparedness exercises and simulations. Reference [B.18-28] provides the details of

SAMG validation activities involving drills and exercises and provides a high level summary of the Human Factors evaluations conducted. This report was produced to support closure of FAI 3.1.4.

It is recognized that BDBA response to multi-unit events including irradiated fuel bay failures would pose additional challenges to resources. Depending on the nature of the BDBA external event and uncertainty relating to affected units and collateral damage (impeding on and off site access), station resources could be significantly challenged and off-site support delayed. For these reasons a BDBA decision making framework has been established to prioritize use and deployment of available resources [B.18-15]. Additionally, for a BDBA multi-unit event, if any unit progresses into a severe accident, there is a prioritization framework built into SAMG to protect containment integrity as the highest priority.

Therefore sufficient focus and assessment has been given to BDBA response in terms of evaluating the event timing and adequacy of resources. There are no gaps related to resourcing for BDBAs.

Multi-unit Events and Irradiated Fuel Bay Failures

9. *Multi-unit failures and Irradiated Fuel Bay (IFB) failures (Section 4.2.5)*

The BDBA and SAMG response has been upgraded to address response to multi-unit events and fuel bay failures. Actions relating to these events have been addressed as part of [B.18-12], specifically:

FAI 1.3.1 – adequacy of means to protect containment

FAI 3.1.2 – plans and schedule for addressing multi-unit events in SAMG

FAI 3.1.2 – plans and schedule for addressing IFB events

FAI 3.2.1 – modelling of multi-unit events

FAI 1.9.1 – habitability of control facilities (including multi-unit events)

FAI 4.2.1 – exercise improvement program (including multi-unit events)

All of the above FAIs have been completed and are closed. BDBA response (including SAMG) has been upgraded to address both multi-unit failures and IFB events and there are no gaps relating to treatment of these failures.

Instrumentation and Equipment Survivability

10. *More specific details relating to suitability of instrumentation and equipment for accident mitigation (Section 4.3.1 and Section 4.3.2)*

Instrumentation and equipment (I&E) survivability has been addressed for Pickering as part of the FAI process [B.18-12], under FAI 1.8.1. This FAI was specific to providing a plan and schedule to address I&E survivability, and has been closed per [B.18-12], which states:

OPG has prepared specific methodology to be used for the OPG stations and the assessments for Pickering and Darlington are now complete. Reference 2 summarizes the findings and details are available in References 3-6.

The assessment has demonstrated that there is reasonable assurance that sufficient equipment and instrumentation will be available to facilitate operator actions at both Pickering and Darlington under a wide range of BDBA conditions, and hence no further actions are required.

OPG has conducted both instrument and equipment survivability assessments for Pickering Units 1,4 and Pickering Units 5-8 [B.18-17] and [B.18-18]. These assessments focused on 'high value' I&E for BDBA including SAs. The assessment concluded that there is sufficient I&E with a Reasonable Level of Confidence of Survivability for harsh conditions, such that the stations can establish and maintain a stable shut down state.

There are no PSR2 gaps relating to BDBA I&E survivability for Pickering.

<u>External Resources</u>

11. Assessment of external resources (Section 6.3)

A confirmatory assessment of external resource protocols was conducted as part of the FAI process. Issues pertaining to external interfaces were addressed under:

FAI 5.2.1 - Identify the external support and resources that may be required during an emergency.

FAI 5.2.2 - Identify the external support and resource agreements that have been formalized and documented.

FAI 5.2.3 - Confirm if any undocumented arrangements can be formalized.

These FAIs were completed and closed per [B.18-19]. The "OPGN Emergency Preparedness Response to CNSC FAI 5.2.1, 5.2.2 and 5.2.3" report, N-REP-03490-10023 [B.18-29], identifies a list of existing external support agreements, which includes the Mutual Aid Agreement (MAA), N-LEGL-03490-0413370 [B.18-30]. The MAA identifies comprehensive support and resources available from four other major Canadian nuclear operators, implemented on November 30, 2012. An internal implementation document, N-GUID-03490-10001, "Mutual Aid Agreement Implementation" [B.18-31], outlines the process of how OPG will implement emergency support under the MAA. Direction to consult this guide is incorporated into the relevant Emergency Response Organization instructions. Both N-REP-03490-10023 [B.18-29] and the MAA [B.18-30] include a list of potential external support and resources that may be required in an emergency.

In addition to the agency agreements detailed above, the REGDOC also specifically identifies agreements relating to the procurement of resources in Section 6.3. These are not generally addressed under the AM framework, but rather they are addressed by the emergency organization under the Consolidated Nuclear Emergency Plan and the role of the Resource Deployment Manager [B.18-32]. The Resource Deployment Manager instruction, N-INS-03491-

10022 [B.18-33], as well as the Emergency Response Manager instruction, N-INS-03491-10015 [B.18-34] and the Corporate Emergency Operations Facility Emergency Recovery Director instruction, N-INS-03491-10000 [B.18-35], now include references to N-GUID-09013-10002, "Supply Chain Support – Nuclear Emergency Responders Guideline" [B.18-36]. This guideline identifies consumables and supplies for Operations and documents the strategic just-in-time sourcing that may be anticipated following a significant emergency. Therefore, Pickering is compliant with the requirements for external resources.

B.18.3 Compliance Assessment Summary for Pickering PSR2

There is one PSR2 CNSC REGDOC-2.3.2 (2015) gap which relates to Safety Factor 1 (Plant Design):

1. Full provision of Complementary Design Features for containment integrity as required by Clause 4.2.1 of REGDOC-2.3.2 will be addressed with the completion of Phase 2 Emergency Mitigating Equipment. This work is currently scheduled to be fully implemented by the end of 2017 [B.18-25]. Since this work is still in progress, it is identified as a PSR2 gap.

B.18.4 References

[B.18-1] CNSC Regulatory Document REGDOC-2.3.2, *Accident Management*, Version 2, September 2015

[B.18-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.18-3] CNSC Regulatory Document, REGDOC-2.3.2, *Accident Management*, October 2014.

[B.18-4] CNSC Regulatory Document, REGDOC-2.3.2, *Accident Management: Severe Accident Management Programs for Nuclear Reactors*, September 2013.

[B.18-5] CNSC Regulatory Guide, CNSC *G-306 Severe Accident Management Programs for Nuclear Reactors*, May 2006.

[B.18-6] CNSC Website, *Document History of REGDOC-2.3.2, Accident Management, Version 2,* http://nuclearsafety.gc.ca/eng/acts-and-regulations/regulatory-documents/history/regdoc2-3-2.cfm, September 29, 2015.

[B.18-7] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.18-8] OPG Report, NK30-REP-03680-00005 R000, *Pickering NGS B - Integrated Safety Review – Safety Analysis Review*, June 2007.

[B.18-9] OPG Report, NK38-REP-03680-10017 R000, *Review of CNSC G-306 (May 2006) Severe Accident Management Programs for Nuclear Reactors for Darlington Integrated Safety Review*, August 2011.

[B.18-10] OPG Report, NK38-REP-03680-10206 R000, *Code Refresh Review Of CNSC REGDOC 2.3.2 (2013) Accident Management: Severe Accident Management Programs For Nuclear Reactors*, February 2014.

[B.18-11] OPG Report, NK38-REP-03680-10207 R000, *Darlington NGS Integrated Safety Review Emerging Issues Report*, February 2014.

[B.18-12] OPG Report, N-REP-03600-10003 R007, *Fukushima Action Item Status Report*, November 2015.

[B.18-13] OPG Report, P-REP-03680-00017 R000, *Pickering NGS PSR2 Safety Factor 13 Report: Emergency Planning*, December 2016.

[B.18-14] OPG Standard, N-STD-MP-0019 R001, *Beyond Design Basis Accident Management*.

[B.18-15] OPG Report, N-REP-09013-10004 R001, *Beyond Design Basis Events: Prioritization and Decision-Making*, December 2014.

[B.18-16] OPG Report, N-REP-03500-0401509 R000, *Implications of the Fukushima Daiichi Event on OPG Nuclear Power Plants: A Summary Report*, July 2012.

[B.18-17] OPG Report, N-REP-09013-10007 R000, *Ontario Power Generation Severe Accident Management Guidance Instrumentation and Equipment Survivability – Summary Report*, December 2013.

[B.18-18] OPG Report, N-REP-09013-10013 R000, *Ontario Power Generation Severe Accident Management Guidance Instrumentation and Equipment Survivability for Review Level Conditions,* September 2014.

[B.18-19] OPG Letter to CNSC, W.S. Woods to M. Santini & F. Rinfret, N-CORR-00531-06906, *OPG Progress Report No. 7 on CNSC Action Plan – Fukushima Action Items*, November 30, 2015.

[B.18-20] OPG Manual, N-BDB-03600-00002 R000, *OPG Emergency Mitigating Equipment for Beyond Design Basis Accidents: Technical Basis Document*, October 2015.

[B.18-21] OPG Guideline, N-GUID-01130-10000 R001, *Modifications for Beyond Design Basis Accidents*, March 2015.

[B.18-22] OPG Guideline, NA44-GUID-03600-00001 R000, *Pickering 1-4 Beyond Design Basis Functional Safety Requirements*, October 2014.

[B.18-23] OPG Guideline, NK30-GUID-03600-00001 R000, *Pickering 5-8 Beyond Design Basis Functional Safety Requirements*, October 2014.

[B.18-24] OPG Report, P-REP-09013-00002 R001, *Pickering NGS – Beyond Design Basis Containment Integrity*, January 2014.

[B.18-25] OPG Letter, B. McGee to H. Khouaja, P-CORR-00531-04784, *Pickering NGS – CNSC Action Item 2016-48-7470 Status Update on Emergency Mitigating Equipment and Telecommunications Projects*, July 29, 2016.

[B.18-26] OPG Report, N-REP-03611-10021 R001, *Simplified Human Reliability Analysis Process for Emergency Mitigation Equipment (EME) Deployment*, September 2014.

[B.18-27] OPG Memorandum, N-CORR-09013-0490807, *Emergency Mitigating Equipment (EME) Deployment – Resources and Timelines*, July 16, 2014.

[B.18-28] OPG Report, N-REP-03490-10030 R000, *SAMG Effectiveness – Response to CNSC Fukushima Action Item 3.1.4*, December 2013.

[B.18-29] OPG Report, N-REP-03490-10023 R000, *OPGN Emergency Preparedness Response to CNSC FAI 5.2.1, 5.2.2 and 5.2.3*, January 2013.

[B.18-30] OPG File, N-LEGL-03490-0413370, *Mutual Aid Agreement for Nuclear Emergency Support*, November 30th, 2012.

[B.18-31] OPG Guideline, N-GUID-03490-10001 R001, *Mutual Aid Agreement Implementation*, December 2013.

[B.18-32] OPG Program, N-PROG-RA-0001 R014, *Consolidated Nuclear Emergency Plan*, June 2015.

[B.18-33] OPG Instruction, N-INS-03491-10022 R002, *Resource Deployment Manager*, July 2016.

[B.18-34] OPG Instruction, N-INS-03491-10015 R004, *Emergency Response Manager*, July 2016.

[B.18-35] OPG Instruction, N-INS-03491-10000 R009, *CEOF Emergency Recovery Director*, June 2016.

[B.18-36] OPG Guideline, N-GUID-09013-10002 R000, *Supply Chain Support – Nuclear Emergency Responders Guideline*, August 2015.

B.19 CNSC REGDOC-2.3.3 (2015), "Periodic Safety Reviews"

B.19.1 Background

The following paraphrased from the Purpose and Scope of CNSC REGDOC-2.3.3 (2015) [B.19-1] provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *REGDOC-2.3.3 sets out the CNSC's requirements for the conduct of a periodic safety review (PSR). A PSR is a comprehensive evaluation of the design, condition and operation of a nuclear power plant (NPP, plant). It is an effective way to obtain an overall view of plant safety and the quality of the safety documentation, and to determine reasonable and practical improvements to ensure safety until the next PSR or, where appropriate, until the end of commercial operation.*
>
> *PSRs have been effective in achieving improvements in safety. Adopting PSRs in support of licence renewal will ensure the continued improvement of NPP safety. Past experience with life-extension projects gives the CNSC and the Canadian nuclear industry a large degree of familiarity with the PSR process. As such, the application of a PSR in Canada represents an evolution of a current practice, as opposed to the adoption of a new one.*

All of CNSC REGDOC-2.3.3 (2015) is directly relevant to Safety Factor 8 (Safety Performance).

Compliance with REGDOC-2.3.3 is not currently a licence requirement for Pickering NGS (in accordance with PROL 48.02/2018) per the R04 Pickering Licence Conditions Handbook (LCH) [B.19-2]. REGDOC-2.3.3 superseded RD-360, "Life Extension of Nuclear Power Plants" [B.19-3] (also later referred to as "Life Management of Nuclear Power Plants", and "Long-Term Operation Management for Nuclear Power Plants").

Per Section 16.2 of the Pickering Licence Conditions Handbook [B.19-2]:

> *OPG conducted, between 2006 and 2009, an environmental assessment (EA) and an Integrated Safety Review (ISR) in anticipation of refurbishing Pickering B. In February 2010, OPG announced that Pickering B will not be refurbished, but would continue to operate for another decade (i.e. until 2020). Because RD-360, "Life Extension of Nuclear Power Plants" issued in February 2008 did not cover the aspect of continued operations, CNSC staff developed and established regulatory expectations for continued operation beyond the assumed design life of Pickering B, which were formally communicated to OPG on May 12, 2010 (e-Doc 3546506). From mid-2010 to mid-2012, two draft revisions were produced, the latest document published for public consultation is RD/GD-360 version 2 (July 2012), "Long-Term Operation Management for Nuclear Power Plants". On August 1, 2012, CNSC staff communicated a formal regulatory position and directions to OPG regarding the regulatory framework applicable to the End-of-Life approach for Pickering, to ensure regulatory stability and predictability (e-Doc 3973315).*
>
> *The continued operations plan (COP) is an integrated improvement plan to close issues identified in the EA and ISR for Pickering B, which contains the actions required to support the technical basis for operating the Pickering B units to the end of 2020.*

The sustainable operations plan (SOP) describes the arrangements and activities required to demonstrate that safe and reliable operation of Pickering will be maintained and sustained, for each of the fourteen safety and control areas (SCAs), for the period of operation up to until each reactor unit is permanently shutdown. This licence condition provides the regulatory requirement to implement and maintain the COP and SOP ensuring safe operation of Pickering, and request OPG to provide an end date by which the Pickering units will be shutdown.

The results of previous REGDOC-2.3.3 (2015) reviews have been assessed for applicability to PSR2 in Section B.19.2. As identified in Reference [B.19-4], the Pickering PSR2 review of CNSC REGDOC-2.3.3 is a High Level review. For a PSR2 High Level review, the degree of conformance with clauses or groups of clauses in the Law, Regulation, Code or Standard (L/R/C/S) is demonstrated by supporting evidence stating whether the intent of the requirements stipulated in the requirement document is met. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the intent of the safety requirement is met.
- Gap: A Gap indicates that the intent of the safety requirement is not met.

B.19.2 Compliance Assessment for Pickering PSR2

B.19.2.1 Application of PSR1 Reviews

The versions of REGDOC-2.3.3 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering and Darlington NGSs

The Pickering B and Darlington Integrated Safety Reviews (ISRs) were performed to meet draft CNSC Regulatory Document RD-360, "Life Extension of Nuclear Power Plants" [B.19-3], which established the CNSC's approach and requirements relating to Life Extension of Nuclear Power Plants (NPPs). In accordance with RD-360, specific guidance for performing the ISR review was identified in IAEA NS-G-2.10, "Periodic Safety Review of Nuclear Power Plants - Safety Guide" [B.19-5]. The IAEA Safety Guide was designed to deal specifically with the ISR of existing NPPs.

PSR2 is a subsequent PSR as defined in REGDOC-2.3.3 [B.19-1] and IAEA SSG-25 [B.19-6]. PSR2 is an update of the Pickering B ISR completed in 2009, the Darlington ISR completed in 2013 and integrated safety assessments performed for Pickering A Return to Service in 2000. The Pickering PSR must satisfy the requirements of CNSC REGDOC-2.3.3 [B.19-1], which supersedes RD-360. REGDOC-2.3.3, in turn, refers to IAEA document SSG-25 [B.19-6], which supersedes NS-G-2.10. Since REGDOC-2.3.3 and IAEA SSG-25 did not exist at the time that the previous Darlington and Pickering B ISRs were performed, they introduce potential additional requirements for PSR2. Neither document was reviewed as part of PSR1. REGDOC-2.3.3 or its predecessors were not assessed as part of the Pickering A Return to Service.

Both IAEA NS-G-2.10 and SSG-25 are consistent in that they recommend 14 PSR Safety Factors to facilitate the review. CNSC REGDOC-2.3.3 encompasses all of the PSR Safety Factors recommended by the IAEA in SSG-25 and expands upon it by adding Safety Factor 15, Radiation Protection. As discussed in REGDOC-2.3.3 [B.19-1]:

> *The PSR approach is outlined in SSG-25. The complex process of conducting a PSR can be facilitated by subdividing it into tasks that are identified as safety factors. These safety factors are intended to cover all aspects that are important to the safety of an operating nuclear power plant. The terms "safety factor" and "safety factor reports" are an adoption of the SSG-25 terms, with the addition of a safety factor for radiation protection…*
>
> *SSG-25 describes 14 safety factors that have been selected on the basis of international experience and are intended to cover all factors important to NPP safety. The scope, tasks and methodologies of these 14 safety factors are considered to meet the CNSC's expectations for corresponding safety factors 1–14 listed above. The CNSC has included an additional safety factor on radiation protection; the licensee should refer to Appendix A for guidance on the scope and tasks for the review of this safety factor.*

SSG-25 Section 5 contains review element guidance for Safety Factors 1 to 14. For Safety Factor 15, review elements are identified in Appendix A of REGDOC-2.3.3. A clause-by-clause review of Section 5 of SSG-25 (which identifies Review Tasks for PSR Safety Factors 1 to 14) is addressed in Reference [B.19-7].

PSR2 is a subsequent PSR as defined in REGDOC-2.3.3 and is an update of the ISRs that were completed for Pickering B in 2009 and Darlington in 2013. A high level review of REGDOC-2.3.3 is provided in Section B.19.2.2 below.

B.19.2.2 Application of Post-PSR1 Reviews

As discussed above, a review of CNSC REGDOC-2.3.3 was not undertaken as part of the Pickering B or Darlington ISRs as the document did not exist at the time.

OPG provided comments on a draft version of REGDOC-2.3.3 [B.19-8] and in doing so, a comprehensive review was performed to compare the ISR baseline processes and documents against the requirements specified in REGDOC-2.3.3. The review concluded that OPG's existing processes and programs can be credited with the implementation of recommended enhancements in order to meet REGDOC-2.3.3, and that the recommended enhancements can be easily integrated into the existing processes.

A compliance review against REGDOC-2.3.3 was not undertaken as part of previous PSR1 reviews and the following High Level assessment has been completed. In the review below, the degree of conformance with clauses or groups of clauses in the L/R/C/S is assessed for Pickering NGS by reference to supporting evidence stating whether the intent of the requirements stipulated in the L/R/C/S is met. The focus of the review is on PSR2. PSR2 is being conducted in accordance with the PSR2 Basis Document [B.19-4], which was accepted by the CNSC [B.19-9].

REDOC-2.3.3 Clause	PSR2 Review	Compliant or Gap
1. Introduction	There are no requirements specified. The introduction describes the purpose of REGDOC-2.3.3 and the purpose of a PSR.	Compliant
1.1 Purpose	There are no requirements specified. Sets context.	Compliant
1.2 Scope	There are no requirements specified. Sets applicability to nuclear power plants and other nuclear facilities.	Compliant
1.3 Relevant Regulations	There are no unique requirements specified. Sets the legal context for the REGDOC-2.3.3. Specified regulations are addressed separately.	Compliant
1.4 National and international standards	Establishes context. Identifies the alignment with IAEA SSG-25. Section 5 of SSG-25 is assessed in [B.19-7].	Compliant
2. General Requirements The licensee shall conduct a PSR in accordance with this regulatory document for the period until the next PSR or, if applicable, until the end of commercial operation of the plant. The PSR shall be conducted in four phases:	REGDOC-2.3.3 is the basis for PSR2 as documented in the PSR2 Basis Document [B.19-4]. The period addressed by PSR2 covers to the next license renewal period expected to be in 2028 which aligns with the period until the next PSR or encompasses the end of commercial operation of the plant. Section 2.2 of the PSR2 Basis Document identifies the four phases of the PSR process as identified in REGDOC-2.3.3 clause 2.	Compliant
Guidance The guidance describes the PSR objectives, the use of Safety Factors, and the scope of PSR Basis Document. The use of the Global Assessment Report and Integrated Implementation Plan are briefly described. Documentation to be submitted to CNSC is identified. The 10 year interval for the performance of PSRs is discussed. The expectation of less effort on a second PSR is explained. The fact that a PSR complements other regulatory activities is explained.	The PSR2 Basis Document identifies the use of Safety Factors, Global Assessment Report (GAR) and Integrated Implementation Plan (IIP) as identified in REGDOC-2.3.3. The scope of the PSR2 Basis Document is consistent with the guidance. The PSR2 Basis Document is consistent with REGDOC-2.3.3 requirements for CNSC submissions. The second PSR is aligned with the expected licence renewal period of 10 years, which is at the recommended 10 year interval. The use of previous PSR information (referred to as PSR1) is described in the PSR2 Basis Document. This guidance about the PSR2 being complementary to other regulatory activities establishes context only.	Compliant

REDOC-2.3.3 Clause	PSR2 Review	Compliant or Gap
3. Periodic Safety Review Basis Document The required elements of the PSR basis document are:		Compliant
1. statement of current licensing basis, including exemptions and acceptable deviations (also described in section 3.1)	Addressed in PSR2 Basis Document [B.19-4] Section 1.4.	
2. statement of the proposed operating strategy of the facility (also described in section 3.2)	Addressed in PSR2 Basis Document [B.19-4] Section 1.2.	
3. description of scope of the PSR (also described in section 3.3)	Addressed in PSR2 Basis Document [B.19-4] Section 2.	
4. description of the methodology for the performance of the PSR, including the period for which the PSR is valid	The methodology is addressed in PSR2 Basis Document [B.19-4] Section 3. The period for which the PSR is valid is described in Sections 1.2, and 2.4.	
5. statement of applicable modern codes, standards and practices	Addressed in PSR2 Basis Document [B.19-4] Appendix D, Table D1.	
6. description of the methodology for the identification, dispositioning and tracking of gaps	The definition of a gap is specified in the PSR2 Basis Document [B.19-4] Section 3.2.3. The handling of gaps during the Global Assessment is described in PSR2 Basis Document [B.19-4] Sections 3.3.2 and 3.3.3. The tracking of implementation of initiatives is discussed in PSR2 Basis Document [B.19-4] Sections 3.4.1 and 3.4.2.	
7. description of the methodology for the global assessment	Addressed in PSR2 Basis Document [B.19-4] Section 3.3 and subsections.	
8. PSR governance	Addressed in PSR2 Basis Document [B.19-4] Section 4.0.	
Guidance		Compliant
Fifteen Safety Factors are listed.	The PSR2 Basis Document applies the 15 Safety Factors listed in REGDOC-2.3.3.	
The use of SSG-25 for the first 14 Safety Factors is described. The 15th safety factor is described in Appendix A.	SSG-25 is the basis for Safety Factor 1-14 Review Tasks as described in PSR2 Basis Document [B.19-4] Section 2.6.1.2. Review Tasks for Safety Factor 15 are derived directly from REGDOC-2.3.3 Appendix A as described in PSR2 Basis Document Section 2.6.1.2.	
Use of earlier PSR conclusions in a subsequent PSR is discussed.	Addressed in PSR2 Basis Document [B.19-4] Section 2.4 and Appendix A.	

REGDOC-2.3.3 Clause	PSR2 Review	Compliant or Gap
3.1 Current Licensing Basis 3.2 Proposed Operating Strategy for the Nuclear Power Plant 3.3 Scope of the Periodic Safety Review	Compliance with Sections 3.1 to 3.3 of REGDOC-2.3.3 is addressed by the compliance discussion above for Section 3.	Compliant
3.4 Methodology for the performance of the periodic safety review. The requirements include specifying a methodology for conducting assessments to confirm that the plant will continue to meet its licensing basis until the end of commercial operation, conducting assessments against applicable modern codes, standards and practices, conducting a global assessment and identifying corrective actions and safety improvements. The guidance identifies IAEA SSG-25 as defining an appropriate approach. The use of appropriate internal documents based on a freeze date is recommended.	The PSR2 Basis Document was developed to address requirements defined in IAEA SSG-25 as outlined in the PSR2 Basis Document [B.19-4] Section 2.4. The PSR2 methodology is specified in Section 3 of the PSR2 Basis Document [B.19-4] and addresses all of the aspects identified in REGDOC-2.3.3 Section 3.4 as noted in PSR2 Basis Document Sections: - 3.2, Safety Factor Reviews. - 3.3, Global Assessment. - 3.4, Integrated Implementation Plan. A PSR2 freeze date is specified in Section 2.4 of the PSR2 Basis Document [B.19-4]. The period for PSR2 validity is defined in Section 2.4.	Compliant
3.5 Applicable modern codes, standards and practices The requirement is to list applicable codes, standards, and practices to be used in the PSR along with selection criteria, PSR cut-off date, and definition of review type. The guidance identifies that codes, standard and practices should be selected taking into consideration CNSC's regulatory documents and international experience. Documents listed in the license and regulatory documents should be considered. Clause by clause reviews are recommended for new versions of codes and standards and those referenced in the PROL or LCH. Guidance is provided on the type	The methodology and selection criteria used to select codes and standards are defined in the PSR2 Basis Document [B.19-4] Section 2.6.2 and Appendix D. Multiple sources were used including the Pickering PROL, Pickering LCH, and CNSC web sites. The resulting list is provided in Appendix D Table D1. A freeze date (which functions as a cut-off date) is specified in Section 2.4 of the PSR2 Basis Document [B.19-4]. The review type is defined for each code and standard in Table D1 based on a methodology outlined in Section 3.2.2 of the PSR2 Basis Document [B.19-4]. Clause-by-clause reviews are applied to new codes or standards listed in the Pickering PROL or LCH. High Level and Incremental reviews (as defined by the licensee) are also defined with criteria for their application.	Compliant

REDGOC-2.3.3 Clause	PSR2 Review	Compliant or Gap
of review and allows the licensee to propose review types.		
Guidance Expectations for the selection of modern codes and standards are specified. This includes establishing a list before any work is carried out. Selection criteria for Codes and Standards are recommended. Mandatory clauses are to be reviewed. Sub-tier referenced sections are also to be reviewed. New code versions referenced in the LCH should have a clause-by-clause review.	A list of codes and standards is identified in the PSR2 Basis Document Table D1. PSR2 was started on the basis of the Revision 0 of the PSR2 Basis Document and was amended to include changes that were incorporated into Revisions 1 and 2. The extensive basis for the selection of Codes and Standards, including use of the Pickering PROL and LCH, is documented in the PSR2 Basis Document [B.19-4] Appendix D.1.0. The PSR2 Basis Document [B.19-4] states: - Section 3.2.2: *As a subsequent PSR, PSR2 will focus on changes in requirements, plant conditions, operating experience and new information, rather than repeating the activities of previous reviews. Since PSR2 is an update of previous ISRs, it will incorporate reviews of Laws, Regulations, Codes and Standards that have occurred as new versions have been issued. Therefore, clause-by-clause reviews of the majority of applicable Laws, Regulations, Codes and Standards have already been completed and there is little value in repeating that process…* *New Laws, Regulations, Codes and Standards referenced in Pickering PROL 48.02/2018 (listed in Appendix C of the Licence Conditions Handbook) will be subjected to a clause-by-clause type review.* - Section D.1.0, bullet 3.2: *If a sub-tier Code or Standard is called up or cited as mandatory by a PROL Law, Regulation, Code or Standard (or a more recent version of a PROL Law, Regulation, Code or Standard as identified in Step 3.1), and either the sub-tier Code or Standard was not already assessed as part of PSR1 or the sub-tier Code or Standard has been updated since PSR1, then the applicable parts of the sub-tier Code or Standard are included in the PSR2 Assessment Basis if they are determined to be safety significant.*	Compliant

REGDOC-2.3.3 Clause	PSR2 Review	Compliant or Gap
3.6 Methodology for the identification, dispositioning and tracking of gaps The licensee shall describe the process and methodology for identifying, categorizing, prioritizing and dispositioning gaps. The licensee shall state what decision-making process will be used to evaluate and decide on the various alternatives to disposition the gaps. To the extent practicable, the licensee shall resolve identified gaps with respect to applicable modern codes, standards and practices. The licensee shall use established processes to resolve identified gaps with the current licensing basis. The licensee shall track dispositioning and resolution of all gaps identified during the PSR through to their resolution.	The PSR2 Basis Document [B.19-4] Section 3.3.3 defines the consolidation of strengths and gaps from the 15 Safety Factor Reports, the definition of Global Issues, and the development of resolutions. The PSR2 Basis Document [B.19-4] Section 3.3.3 describes the use of a risk informed decision making process considering the overall safety significance of Global Issues. The PSR2 Basis Document [B.19-4] Section 3.3.3 identifies that the Global Issues will be tabularized, tracking sources of the issues, to facilitate further review and assessment. The PSR2 Basis Document [B.19-4] Section 3.3.3 describes the process to resolve identified global issues and gaps. Section 3.4.2 describes the implementation of tracking and reporting, and the change management process. .	Compliant
Guidance Findings are to be identified as strengths and gaps. The rationale is to be provided. Gaps are to be prioritized according to safety significance, considering deterministic and probabilistic safety analysis and engineering judgement.	The PSR2 Basis Document [B.19-4] Section 3.2.3 describes the criteria for compliances and gaps with respect to the review elements in the PSR2 Assessment Basis. Section 3.3.3 outlines how the Strengths and Gaps from the 15 individual Safety Factor Reports will be consolidated and grouped by topic area to support the Global Assessment. With the assembly of Global Issues and Strengths, and considering the recommendations from the Component Condition Assessments, the aggregate impact of the Global Issues will be assessed. The PSR2 Basis Document [B.19-4] Section 3.3.3 outlines the use of deterministic criteria (PSR2 Basis Document Appendix E) and probabilistic criteria (PSR2 Basis Document Appendix F).	Compliant

REDOC-2.3.3 Clause	PSR2 Review	Compliant or Gap
The overall priority of a gap should inform the course of action to be taken to establish its recommended disposition. The licensee should establish and maintain a database of all gaps identified during the PSR.	The PSR2 Basis Document [B.19-4] Section 3.3.3 outlines how the Safety Significance level will consider deterministic and probabilistic safety analysis impact, as appropriate. The assignment of Safety Significance values for prioritization was derived based on OPG experience and takes into account the priority values from the OPG guidelines for evaluating and prioritizing Safety Report Issues, the COG Benefit-Cost Analysis processes, and the OPG Station Condition Record categorization process. Safety Significance will be derived primarily through the thresholds defined in PSR2 Basis Document Appendix E Table E1 and per the risk assessment criteria in Appendix F. The PSR2 Basis Document [B.19-4] Section 3.3.3 also outlines how the resolution of global issues and gaps (i.e., "course of action" per the SSG-25 Guidance wording) is dependent on the significance level. The PSR2 Basis Document Section [B.19-4] 3.3.3 identifies that the Global Issues will be tabularized, tracking sources of the issues, to facilitate further review and assessment.	
3.7 Methodology for the global assessment The methodology for performing the global assessment shall be described in the PSR basis document. The methodology shall address and include: 1. results of the safety factor reviews, in particular, the findings (both gaps and strengths) of NPP design and operation 2. the interdependencies between gaps and the significance of their aggregate effects 3. recommended corrective actions and safety improvements to address individual and consolidated gaps	The methodology for performing the Global Assessment is described in Section 3 of the PSR2 Basis Document [B.19-4]. As noted in Section 3.3.3 of the PSR2 Basis Document [B.19-4]: *the Strengths and Gaps from the 15 individual Safety Factor Reports will be consolidated and grouped by topic area to support the Global Assessment*. *The consolidation of Gaps into Global Issues will provide a means to assemble Gaps of a common nature, facilitating the assessment of safety impact and identifying and assessing practical and effective resolutions… The aggregate impact of the Global Issues will be assessed.* *Resolution options will be developed and assessed using risk informed decision making techniques.*	Compliant

REGDOC-2.3.3 Clause	PSR2 Review	Compliant or Gap
4. the extent to which the safety requirements of defence in depth are fulfilled 5. an estimate of global risk associated with facility operation with any unresolved gaps The results from the global assessment shall be documented in the global assessment report.	*An important element of the development of proposed recommendations will be to assess the overall defence-in-depth and aggregate impact of the residual Global Issues / Acceptable Deviations. After evaluating a range of resolutions for Global Issues, and determining a recommended resolution to be selected, the impact on defence-in-depth, considering both deterministic and probabilistic elements, will be evaluated to assess the aggregate impact on overall safety.* *As a final step in the assessment process, the team will assess the overall acceptability of operation of the plant over the period considered in PSR2. This will entail a review of the results of the Safety Factor Reviews, a consideration of enhancements planned (both newly identified in PSR2 and from other station plans), and a consideration of plant performance and initiatives underway.* As noted in PSR2 Basis Document [B.19-4] Section 3.3.4, the results of the Global Assessment will be documented in the Global Assessment Report, presenting the results, assessing the overall defence-in-depth of the plant, and documenting the conclusions, corrective actions, and enhancements to be considered.	
Guidance The guidance elaborates on the requirements listed above.	Addressed in the compliance assessment above concerning clause 3.7.	Compliant
3.8 Periodic safety review governance In the PSR basis document, the licensee shall establish, and describe governance for the conduct of the PSR.	The governance for the conduct of the PSR is described in PSR2 Basis Document [B.19-4] Section 4.	Compliant
Guidance The licensee's governance for the conduct of PSR should address that: the PSR team is qualified to carry out the review	As described in the PSR2 Basis Document [B.19-4]: Section 4.0: *PSR2 work will be conducted under OPG's quality management program (compliant with CSA N286-05). Where external contractors are engaged in performing portions of PSR2, they will either work under OPG's quality program, or under a quality program that has been accepted by OPG as meeting the quality requirements for the contracted work scope.* Training and qualification are implicit in the quality management program.	Compliant

REGDOC-2.3.3 Clause	PSR2 Review	Compliant or Gap
provisions have been made for peer or independent review of work done controls are in place to ensure that information and data are used consistently across the review requirements for the preparation and verification of documentation are satisfied results are recorded in a systematic and auditable manner The licensee should develop a project plan for the conduct of the PSR that includes established project management processes and quality management provisions.	Section 3.3.1: *The Global Assessment will be conducted by an interdisciplinary team, with appropriate expertise in operations, design and safety at the plant, including appropriate participants from the Safety Factor reviews, and members who are independent from the Safety Factor review teams…* Section 4.0: *Effective communications practices will be used throughout the performance of PSR2. Interfaces between the work of different external contractors and OPG staff will be managed to ensure consistency between related deliverables and accuracy of information.* Section 4.0: *PSR2 work will be conducted under OPG's quality management program (compliant with CSA N286-05).* Preparation and verification of documentation are implicit in the quality management program. Section 3.3.3: *The Global Issues will be tabularized, tracking sources of the issues, to facilitate further review and assessment.* Section 4.0: *PSR2 work will be conducted under OPG's quality management program (compliant with CSA N286-05).* Auditable records are implicit in the quality management program. Section 4 of the PSR2 Basis Document [B.19-4] describes the application of project management principles, and that PSR2 work will be conducted under OPG's quality management program (compliant with CSA N286-05).	
4. Performance of the Periodic Safety Review The licensee shall conduct the PSR in accordance with the accepted PSR basis document following its acceptance by CNSC staff.	PSR2 has been initiated based on the R02 PSR2 Basis Document [B.19-4] which has been reviewed and accepted by the CNSC.	Compliant

REDGOC-2.3.3 Clause	PSR2 Review	Compliant or Gap
Guidance It is recommended that the licensee does not undertake substantive work on the PSR until such time as CNSC staff has accepted the PSR basis document.	PSR2 has been initiated based on the R02 PSR2 Basis Document [B.19-4] which has been reviewed and accepted by the CNSC.	Compliant
4.1 Safety Factor reports The licensee shall ensure that each safety factor report documents: 1. objective, scope, tasks and methodology for the review 2. applicable codes , standards and practices 3. overview of applicable facility programs and processes 4. findings of the review which identify gaps and strengths 5. categorized and prioritized gaps 6. interfaces with other safety factor report findings 7. options for corrective actions for each gap	The content of Safety Factor reports is specified in PSR2 Basis Document Section 3.2.3 [B.19-4]. These reports will include: • The scope of the review; • Review methodology; • Applicable elements of the PSR2 Assessment Basis (Review Tasks and applicable Laws, Regulations, Codes and Standards); • Effectiveness review of OPG programs supporting compliance assessments; • Review findings (Compliances and Gaps); • Assessment of compliance with Review Tasks; • Overall assessment of the Safety Factor. The PSR2 Basis Document Section 3.3.2 specifies that the categorization and prioritization of gaps (at the level of Global issues) is done as part of the Global Assessment and is not done within each Safety Factor Report; • Impacts on other Safety Factor reviews. Options for corrective actions for each gap are done as part of the Global Assessment and are not done within each Safety Factor Report.	Compliant
Guidance The guidance elaborates on the requirements listed above.	Addressed in the discussion above concerning clauses 4 and 4.1 requirements.	Compliant

REDGOC-2.3.3 Clause	PSR2 Review	Compliant or Gap
5. Global Assessment Report		Compliant
The licensee shall prepare a report that documents the results of the global assessment.	PSR2 Basis Document [B.19-4] Section 3.3.4 states: *The results of the Global Assessment will be documented in a Global Assessment Report, presenting the results, assessing the overall defence-in-depth of the plant, and documenting the conclusions, corrective actions, and enhancements to be considered.*	
The global assessment report (GAR) shall present the findings of the PSR, both strengths and gaps, to provide an overall assessment of the safety of plant.	PSR2 Basis Document [B.19-4] Section 3.3.3 states: *The Strengths and Gaps from the 15 individual Safety Factor Reports will be consolidated and grouped by topic area to support the Global Assessment.*	
The GAR shall document the overall conclusions, corrective actions and safety improvements to be considered.	PSR2 Basis Document [B.19-4] Section 3.3.3 states: *After evaluating a range of resolutions for Global Issues, and determining a recommended resolution to be selected, the impact on defence-in-depth, considering both deterministic and probabilistic elements, will be evaluated to assess the aggregate impact on overall safety.*	
The GAR shall be submitted to CNSC staff for review.	PSR2 Basis Document [B.19-4] Section 3.3.4 states: *The Global Assessment Report will be submitted to CNSC staff for review.*	
Guidance		Compliant
The GAR should provide a living database that captures the current state of the gaps. The database should be fully traceable so that a change in a gap, or in the assessment of a gap, can be easily tracked to its resolution.	PSR2 Basis Document [B.19-4] Section 3.4.1 states: *The IIP listing of enhancements will include those resulting from the Global Assessment Report, including both new modifications proposed as part of the resolution of Global Issues, and also considering the existing planned station process and physical modifications that were integral to the overall assessment of safety… The initiatives will be tabularized with owners assigned and planned implementation dates… The listing will include the priority and the basis for the priority. The implementation of the initiatives will be tracked and reported… The Integrated Implementation Plan will be tracked and progress will be regularly reported throughout the implementation period.*	

REGDOC-2.3.3 Clause	PSR2 Review	Compliant or Gap
The GAR should include the following elements:	PSR2 Basis Document [B.19-4] states:	
1. summaries of the safety factor reports and identified gaps and strengths	- Section 3.3.2: *The Global Assessment Process consists of … Identification and consolidation of Strengths and Gaps from the Safety Factor Reports.*	
2. overlaps, omissions, and interface issues of the findings from the safety factor reports	- Section 3.3.2: *The Global Assessment Process consists of … Assessment of interfaces between the various Safety Factors, Aggregate Impact of Global Issues.* - Section 3.3.3: *With the assembly of Global Issues and Strengths, and considering the recommendations from the Component Condition Assessments, the aggregate impact of the Global Issues will be assessed. In this way, the interaction between issues will be identified. New Global Issues may be identified as part of this consolidation review.*	
3. consolidation of gaps into global issues where appropriate	- Section 3.3.3: *The Strengths and Gaps from the 15 individual Safety Factor Reports will be consolidated and grouped by topic area to support the Global Assessment.*	
4. safety significance and risk ranking of all gaps (individual and consolidated)	- Section 3.3.2: *The Global Assessment Process consists of … prioritization of Global Issues… Ranking of Global Issues with identified actions.* - Section 3.3.3: *The Safety Significance level will consider deterministic and probabilistic safety analysis impact, as appropriate. The assignment of Safety Significance values for prioritization in Appendices E and F was derived based on OPG experience and takes into account the priority values from the OPG guidelines for evaluating and prioritizing Safety Report Issues, the COG Benefit-Cost Analysis processes, and the OPG Station Condition Record categorization process. Probability levels selected for delineation between categories are based on significance and engineering judgement, and are as used in previous Integrated Safety Reviews. These values account for overall safety impact and align, where appropriate, with requirements and limits in relevant safety standards… All Global Issues whose resolution involves identified actions will be ranked from 1 through N, where N is the total number, in accordance with overall safety significance.*	

REGDOC-2.3.3 Clause	PSR2 Review	Compliant or Gap
5. corrective actions, safety improvements and appropriate dispositions proposed for all gaps and global issues	- As identified in PSR2 Basis Document [B.19-4] Section 3.3.3: For Global Issue resolution – the process will be: o Evaluate the Global Issue to understand safety basis, and intent of requirement. o Consider possible options for resolution/mitigation. Consider safety significance and defence-in-depth elements. o Evaluate options with respect to effectiveness, cost, schedule, practicality. For potential plant modifications, this may require an evaluation of the safety impact, both deterministic and probabilistic. If it is not practicable to fully resolve a Global Issue, other mitigation options will be considered for enhancements. o Practicality of a proposed resolution will be evaluated in terms of cost, resources, schedule, and considered in relation to the overall safety impact. o Some proposed resolutions will be dependent on whether plant operation is assumed to continue into the 2025-2028 time period. These proposed resolutions will be distinguished as such. o Propose recommended resolution/mitigation. o Document the decision making process.	
6. a global assessment based on the aggregate effect of the findings resulting from all safety factor reports, taking the proposed corrective actions and safety improvements into account, and defence in depth	- PSR2 Basis Document [B.19-4] Section 3.3.3: *An important element of the development of proposed recommendations will be to assess the overall defence-in-depth and aggregate impact of the residual Global Issues / Acceptable Deviations. After evaluating a range of resolutions for Global Issues, and determining a recommended resolution to be selected, the impact on defence-in-depth, considering both deterministic and probabilistic elements, will be evaluated to assess the aggregate impact on overall safety.*	
7. statement of the licensee's assessment of the overall acceptability of operation of the NPP	- PSR2 Basis Document [B.19-4] Section 3.3.3: *As a final step in the assessment process, the team will assess the overall acceptability of operation of the plant over the period considered in PSR2. This will entail a review of the results of the Safety Factor Reviews, a consideration of enhancements planned (both newly identified in PSR2 and from other station plans), and a consideration of plant performance and initiatives underway.*	

REDOC-2.3.3 Clause	PSR2 Review	Compliant or Gap
6. Integrated Implementation Plan The licensee shall develop an Integrated Implementation Plan (IIP) that addresses the results of the global assessment. The IIP shall be submitted to CNSC staff for acceptance.	PSR2 Basis Document [B.19-4] Section 3.4 states: *The proposed enhancements resulting from the Global Assessment will be documented in the Integrated Implementation Plan.* PSR2 Basis Document [B.19-4] Section 3.4.2 states: *The Integrated Implementation Plan Report will be submitted to CNSC staff for acceptance, per CNSC REGDOC-2.3.3.*	Compliant
In the IIP, the licensee shall:		
1. list the corrective actions and safety improvements (including necessary physical NPP modifications) that will address all gaps identified in the PSR, and findings	PSR2 Basis Document [B.19-4] Section 3.4.1 states: *The initiatives will be tabularized with owners assigned and planned implementation dates. Existing initiatives integral to the overall assessment of safety during the Global Assessment will also be included in this listing. The listing will include the priority and the basis for the priority. The implementation of the initiatives will be tracked and reported.*	
2. specify the schedule for implementing the corrective actions and safety improvements	PSR2 Basis Document [B.19-4] Section 3.4 states: *The IIP will provide the proposed timeline for the implementation of the enhancements.* PSR2 Basis Document [B.19-4] Section 3.4.1 states: *… a change management process will be implemented to manage evolution of the resolution details and implementation schedules.*	
Guidance An overview of the acceptability of safe operation of plant in view of the proposed changes should be included in the IIP, to demonstrate that the outcome of safety improvements serves the intended purpose of the PSR.	The PSR2 Basis Document [B.19-4] states: - Section 3.3.4: *The Global Assessment Report will include a statement of OPG's assessment of the overall acceptability of operation of the plant.*	Compliant
In the IIP, the licensee should:		
1. demonstrate traceability and provide references to the GAR	- Section 3.4.2: *The processes will allow tracking of initiatives to completion or resolution in an auditable manner, consistent with OPG's Management System.*	

REGDOC-2.3.3 Clause	PSR2 Review	Compliant or Gap
2. specify the processes used for determining the detailed scope, including prioritization and scheduling of corrective actions and safety improvements	The PSR2 Basis Document [B.19-4] Section 3.4.1 defines the logistics for the IIP.	
3. schedule and implement corrective actions and safety improvements commensurate with their safety significance	The PSR2 Basis Document [B.19-4] Section 3.4.1 states: *The listing* [of IIP initiatives] *will include the priority and the basis for the priority. The implementation of the initiatives will be tracked and reported.*	
4. specify processes for identification and management of project risks and controls	As defined in the PSR2 Basis Document [B.19-4] Section 4.0: *OPG project management principles will be applied in performing PSR2.* Management of project risks and controls is an element of OPG project management practice.	
5. specify the process to be used to track the progress and completion of the corrective actions and safety improvements	The PSR2 Basis Document [B.19-4] Section 3.4.2 states: *The [IIP] report will also summarize the implementation tracking and reporting process, and the change management process for the IIP. The processes will allow tracking of initiatives to completion or resolution in an auditable manner, consistent with OPG's Management System.*	
The IIP should be organized according to the CNSC's safety and control areas	The PSR2 Basis Document [B.19-4] Section 3.4.2 states: *To facilitate the CNSC review of the Integrated Implementation Plan, the plan will be presented in a manner aligned with the CNSC Safety and Control Areas.*	
The licensee should have the following in place:		
1. a project organization, structured to execute the IIP	The PSR2 Basis Document [B.19-4] Section 3.4.1 states: *The initiatives will be tabularized with owners assigned and planned implementation dates.* The PSR2 Basis Document [B.19-4] Section 3.4.2 states: *The report will also summarize the implementation tracking and reporting process, and the change management process for the IIP. The processes will allow tracking of initiatives to completion or resolution in an auditable manner, consistent with OPG's Management System.*	

REDGOC-2.3.3 Clause	PSR2 Review	Compliant or Gap
2. governance for IIP delivery 3. scope, schedules and dependencies, at least for the earlier tasks 4. definition of resources and a resourcing plan 5. a mechanism for overall integration, peer or independent review and oversight	- Section 4.0 Governance [B.19-4]: *PSR2 work will be conducted under OPG's quality management program (compliant with CSA N286-05).* - The PSR2 Basis Document [B.19-4] Section 3.4 states: *The proposed enhancements resulting from the Global Assessment will be documented in the Integrated Implementation Plan (IIP). The IIP will provide the proposed timeline for the implementation of the enhancements.* - The PSR2 Basis Document [B.19-4] Section 3.4.1 states: *The initiatives will be tabularized with owners assigned and planned implementation dates.* - Section 3.4.1 [B.19-4]: *A review will be conducted with program owners and appropriate managers to derive plans for implementation based on priority and resources.* - Section 3.3.4 [B.19-4]: *Reviews and approval of the report will be conducted as required under the OPG Management System.*	
Appendix A: Safety Factor for Radiation Protection	Assessed in Appendix C of OPG Report, P-REP-03680-00003 R000, "Pickering NGS Periodic Safety Review 2 (PSR2) Definition of Safety Factor Review Tasks" [B.19-7].	Compliant
Appendix B: CNSC Safety and Control Areas	Provides information and does not establish any requirements.	Compliant
Glossary	Provides information and does not establish any requirements.	Compliant
References	Provides information and does not establish any requirements.	Compliant
Additional Information	Provides information and does not establish any requirements.	Compliant

B.19.3 Compliance Assessment Summary for Pickering PSR2

There are no PSR2 gaps for CNSC REGDOC-2.3.3 (2015) [B.19-1]. Per the definition of Compliance for a High Level review, Pickering has a PSR2 Compliance associated with CNSC REGDOC-2.3.3 (2015).

B.19.4 References

[B.19-1] CNSC Regulatory Document REGDOC-2.3.3, *Periodic Safety Reviews*, April 2015.

[B.19-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.19-3] CNSC Regulatory Document RD-360, *Life Extension of Nuclear Power Plants,* February 2008.

[B.19-4] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.19-5] IAEA Safety Standards Series No. NS-G-2.10, *Periodic Safety Review of Nuclear Power Plants - Safety Guide*, 2003.

[B.19-6] IAEA Safety Guide No. SSG-25, *Periodic Safety Review for Nuclear Power Plants*, March 2013.

[B.19-7] OPG Report, P-REP-03680-00003 R000, *Pickering NGS PSR2: Definition of Safety Factor Review Tasks*, May 2016.

[B.19-8] OPG Report, NK38-REP-09701-0523075-RN/A LOF, *Review of CNSC Draft REGDOC 2.3.3,* December 2014.

[B.19-9] CNSC Correspondence, e-Doc 5037314, OPG File No. P-CORR-00531-04789 R000, H. Khouaja to B. McGee, *Pickering NGS: CNSC Staff Acceptance of Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, July 8, 2016.

B.20 CSA N286.7.1-09, "Guideline for the Application of N286.7-99, Quality Assurance of Analytical, Scientific, and Design Computer Programs for Nuclear Power Plants"

B.20.1 Background

The following paraphrased from the purpose and scope of CSA N286.7.1 [B.20-1] provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *CSA N286.7.1 provides guidance on the application of CSA N286.7, based on industry experience. It is intended to assist owner organizations and participants in the preparation and implementation of software quality assurance processes in compliance with CSA N286.7. CSA N286.7 contains requirements that need interpretation or expansion in order to be implemented. CSA N286.7.1 provides guidance on graded implementation and draws from software quality assurance/management processes currently in use by owner and participant organizations. CSA N286.7.1 is based on the AECL document, Guideline for the Application of CSA N286.7-99, and has been provided to CSA as a supporting document for CSA N286.7.*

CSA N286.7.1 is relevant to Safety Factors 1 (Plant Design), 5 (Deterministic Safety Analysis), 6 (Probabilistic Safety Assessment), 7 (Hazard Analysis), and 10 (Organization, the Management System and Safety Culture). CSA N286.7.1 is not discussed in the R04 Pickering Licence Conditions Handbook [B.20-2].

The N286.7.1 guide has been amalgamated into the new (2016) edition of the N286.7 Standard. The N286.7 CSA Impact Statement states [B.20-3]: "The CSA N286.7.1 guide will no longer be maintained after this new edition of N286.7 is issued. Any relevant guidance has been put into the new edition of N286.7." As a result, only the review of N286.7-16 has been prepared for PSR2.

B.20.2 Compliance Assessment Summary for Pickering PSR2

As discussed in Section B.20.1, an assessment of N286.7.1 has not been performed as any relevant guidance from N286.7.1 has been amalgamated into CSA N286.7-16. A review of CSA N286.7-16 was performed as part of PSR2.

B.20.3 References

[B.20-1] CSA Standard N286.7.1-09, *Guideline for the application of N286.7-99, Quality Assurance of Analytical, Scientific, and Design Computer Programs for Nuclear Power Plants*, November 2009.

[B.20-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.20-3] CSA Impact Statement, *Notification of CSA N286.7 on Quality Assurance of Analytical Scientific, and Design Computer Programs; Product: New Edition; Product Designation: CSA N286.7; Previous Edition Published: 1999, Reaffirmed 2007 and 2012*, Date not provided.

B.21 CSA N290.12-14, "Human Factors in Design for Nuclear Power Plants"

B.21.1 Background

The following paraphrased from the purpose and scope of CSA N290.12-14 [B.21-1] provides an overview of the purpose of this standard and the requirements expressed therein:

> *Human Factors (HF) in Design for Nuclear Power Plants applies to nuclear safety, protection of the environment, health and safety of persons, security, productivity, and economics. The goal of HF in design is to apply theory, principles, data, and other methods to the design of structures, systems, and components (SSCs) to optimize human and system performance.*
>
> *CSA N290.12 covers HF in design for existing and new NPPs and covers HF in design activities related to construction, commissioning, operation, maintenance, inspection, testing, and decommissioning.*

CSA N290.12-14 is relevant to Safety Factors 1 (Plant Design) and 12 (Human Factors).

Compliance with CSA N290.12 is not currently a licence requirement for Pickering NGS (in accordance with PROL 48.02/2018) per the R04 Pickering Licence Conditions Handbook [B.21-2].

CSA N290.12-14 is the first edition of this standard. N290.12-14 was issued in December 2014, with a revision errata update made in February 2015. However, there was no change in technical content in this errata update [B.21-1].

The following "Significant Features" were obtained from the CSA Impact Statement for public review related to the release of N290.12-14 [B.21-3]:

1. *This new standard establishes:*
    a. *the requirements for Human factors in design for water-cooled nuclear power plants*
    b. *criteria for determining the level of effort (grading)*
    c. *how HF in design is considered from conceptual design through installation and commissioning*
    d. *the need to consider, planning, interfacing with other organizations, analysis and evaluation.*
2. *This standard identifies the purpose and expected outcomes (e.g. for validation) but does not attempt to describe detailed methodology.*

The following are the "Impacts of Standard" from the CSA Impact Statement:

1. *This standard will contribute towards the integration of HF into the design process and consistent application.*
2. *This standard is aligned with the CNSC's expectations for HF in design.*
3. *This standard is expected to be a reference when updating CNSC's regulatory documents concerning HF in design.*
4. *Some organizations may require minor changes to procedures to match N290.12:*
   a) *to ensure the grading criteria from the standard are reflected in the procedures (criteria include safety as well as others). The grading may result in an intermediate level of effort depending on the procedures in place at the various organizations.*
   b) *to ensure changes initiated during implementation are reviewed for HF in design considerations.*
   c) *to ensure HF in design has the opportunity to input to commissioning and installation.*
   d) *to ensure the HF aspects of the as-build are evaluated.*
   e) *to ensure application of the Appendix on Control Centres.*
   f) *to ensure evaluation of Commercial Off the Shelf (COTS) equipment.*
5. *Some organization's procurement may be impacted to ensure contracts and item equivalencies reflect HF approaches.*
6. *Some organizations may feel the need to produce a program level document to complement their procedures. This may be an economic consideration more than a technical consideration.*
7. *Some organizations may have to advance the HF planning to an earlier stage of design.*

The results of PSR1 CSA N290.12 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.21.2. As identified in Reference [B.21-4], the Pickering PSR2 review of CSA N290.12-14 is an Incremental review. PSR2 Incremental review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.

- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.21.2 Compliance Assessment for Pickering PSR2

B.21.2.1 Application of PSR1 Reviews

The versions of N290.12 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering and Darlington NGSs

As discussed earlier, CSA N290.12-14 is the first edition of this standard. As a result, N290.12 was not assessed during the Pickering B or Darlington ISRs, or for Pickering A Return to Service.

With respect to the evolution of CSA N290.12, Darlington ISR Report NK38-REP-03680-10211 R000 [B.21-5] states the following:

> *Around 1996 a draft CSA standard was under consideration (N290.12, "Control Room Design for CANDU Nuclear Power Plants"), but N290.12 was never issued. OPG's Engineering Change Control (ECC) process ensures that all modifications to the station are designed, installed, commissioned, and placed into service within the Safe Operating Envelope, design basis, and plant licensing conditions. The program and supporting procedures ensure proper reviews and approvals, including human factors, are achieved before modifications are implemented. It is also noted that under the current ECC process for new designs, there is systematic consideration of human factors as part of the design process.*

N290.12-14 was subsequently issued in 2014 (with revision errata update in 2015). Per Section 6.1 of the R04 Pickering Licence Conditions Handbook [B.21-2]:

> *The CSA standard N290.12 Human factors in design for nuclear power plants, was published in 2014. CNSC intends to add it to the licensing basis of NPPs in the future. The CNSC recommends that the licensee complete its gap analysis/implementation plan, currently scheduled for 2016.*

The status of implementation of N290.12-14 is discussed in Section B.21.2.2 below.

B.21.2.2 Application of Post PSR1 Reviews

OPG completed a review in October 2016 of OPG's Human Factors Engineering (HFE) practices against the fundamentals of N290.12-14, per OPG Memorandum N-REF-06700-0615412 [B.21-6], which had the following objectives:

- Identify governance requiring revision in order to become compliant with the mandatory clauses of N290.12-14;

- Identify governance in which N290.12-14 should be identified as a performance reference; and

- Provide insight to potential areas of weakness in the OPG HFE process (i.e. non-compliances) for possible consideration in future divisional self-assessments.

The assessment showed OPG to be largely compliant with CSA N290.12-14 and stated [B.21-6]:

> *Rather than by a unique HFE program, compliance is achieved primarily under the auspices of N-PROG-MP-0001 Engineering Change Control, N-PROG-MP-0009 Design Management and associated governance as well as some dependence on interfacing programs and governance.*
>
> *Additionally N-MAN-06700-10002* [B.21-7] *specifies the following:*
>
> *(a) OPG's HFE processes and approach to the conduct of HFE activities*
>
> *(b) OPG's expectations for performing HFE activities.*

Two partially non-compliant clauses were identified as follows [B.21-6]:

> ***Clause 4.4 b):*** *OPG is not fully compliant with Clause 4.4 b) for the HF in design plan to define the Project and Regulatory interfaces with Human Factors Engineering (HFE). There are a number of OPG programs and procedures that define and bound the communication interfaces between the Engineering (in general), Project and Regulatory organizations. For modifications categorized as "Full" HFE, N-FORM-10580 "Identifying Human Factors Level of Activity" and N-MAN-06700-10002 "Guide to OPG Human Factors Engineering Process" requires a Human Factors Engineering Program Plan (HFEPP) meeting certain CNSC guidelines and industry standards which require the definition of roles, authorities, resources and interfaces. Modifications categorized as "Basic" HFE utilize N-FORM-10221 as a pre-defined HFEPP and to document the HFE analysis and evaluations. Neither N-FORM-10221 nor the associated instruction, N-INS-06700-10000, define these interfaces. To be fully compliant with this clause it is recommended to revise N-MAN-06700-10002 and expand N-INS-06700-10000 1.4.14 to include the interfaces identified in N290.12-14 clause 4.4 b) notes and to provide the cross references to the OPG governance that mandate these interdepartmental interfaces and interaction. The implementation of this recommendation is tracked through assignment 28194048-01.*
>
> ***Clause 5.1.1 a) through d):*** *OPG is not fully compliant with the Clause 5.1.1 requirement that HF in design "shall consider" interfaces with other organizations, namely, a) procedure development, b) training development, c) safety analysis and d) staffing. Compliance with the clause requires documentation of the consideration of interface between HFE and these organizations. A number of OPG programs and governance prompt the interaction and interface between engineering (in general)*

and stakeholders in other departments. Neither N-MAN-06700-10002 "Guide to OPG Human Factors Engineering Process" nor N-INS-06700-10000 "Preparation of the Human Factors Engineering Worksheet" requires documentation of the considerations of these inter-departmental interfaces. N-INS-06700-10000 1.4.14 discusses the nature of interfaces with other departments including: Nuclear Safety, Conventional Safety, Training, Procedure Development, Installation and Commissioning. It does not include consideration of staffing. In order to fully comply with this clause, it is recommended to revise both of these documents to explain the requirement to record the specific consideration of these areas in the HFEPP, HFESR [Human Factors Engineering Summary Report] *and N-FORM-10221 and revision of N-MAN-06700-10002 to include staffing in addition to the other areas. The implementation of this recommendation is tracked through assignment 28194048-02.*

Work is in progress to address these minor issues, with interim mitigation in place. Given that the partially non-compliant clauses identified above do not have a nuclear safety impact, there is no PSR2 gap.

B.21.3 Compliance Assessment Summary for Pickering PSR2

There are no PSR2 gaps for CSA N290.12-14 [B.21-1]. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CSA N290.12-14.

B.21.4 References

[B.21-1] CSA Standard N290.12-14, *Human Factors in Design for Nuclear Power Plants,* December 2014; Errata: February 2015.

[B.21-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.21-3] CSA Impact Statement for Public Review, *Product: New Standard; Product Designation: CSA N290.12-14*, Date not provided.

[B.21-4] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.21-5] OPG Report, NK38-REP-03680-10211 R000, *Code Refresh Review of CSA N290.0-11 General Requirements for Safety Systems of Nuclear Power Plants*, January 2014.

[B.21-6] OPG Memorandum, N-REF-06700-0615412, *Subject: N290.12-14 Human Factors in Design for Nuclear Power Plants Compliance Assessment - Summary of Results and Implementation Recommendations (AR 28183796-06)*, October14, 2016.

[B.21-7] OPG Manual, N-MAN-06700-10002 R004, *Guide for OPG Human Factors Engineering Process*, December 2015.

B.22 CSA N288.3.4-13, "Performance Testing of Nuclear Air-Cleaning Systems at Nuclear Facilities"

B.22.1 Background

The following paraphrased from the preface and scope of CSA N288.3.4-13 [B.22-1] provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *Nuclear facilities or licensed activities can release radioactive, airborne contaminants to the surrounding environment. Therefore, in order to mitigate the airborne release of these substances, nuclear air-cleaning systems are often installed in the ventilation systems of nuclear facilities. Regular testing is required to confirm that nuclear air-cleaning systems are performing within their design basis. The design basis approach allows systems to be tested to their individual performance requirements. CSA N288.3.4 provides guidance for pre-operational acceptance testing of new and refurbished nuclear air-cleaning systems and for in-service performance testing of nuclear air-cleaning systems.*
>
> *CSA N288.3.4 addresses the design, implementation, and management of a nuclear air-cleaning system testing program that meets legal and business requirements and incorporates current best practices and technologies used internationally.*

CSA N288.3.4 is relevant to Safety Factor 8 (Safety Performance) and Safety Factor 14 (Radiological Impact on the Environment). CSA N288.3.4 is not discussed in the R04 Pickering Licence Conditions Handbook [B.22-2].

CSA N288.3.4-13 is the first edition of this standard. The CSA Impact Statement and public review notice for this edition identifies the following significant features [B.22-3]:

1. *The Standard addresses pre-operational acceptance testing of new and refurbished nuclear air-cleaning systems and routine in-service performance testing of nuclear air-cleaning systems at nuclear facilities.*

2. *The Standard establishes minimum performance requirements for the following system components: moisture separators, heaters, pre-filters, high efficiency carbon adsorbers, HEPA* [High Efficiency Particulate Air] *filters, filter housings, ductwork, dampers, system monitoring equipment and performance testing equipment.*

3. *The Standard identifies what routine tests are required during commissioning and normal operation and the recommended testing schedule. Criteria are also established for non-routine testing after a system upset (e.g. caused by filter media disturbance or replacement, suspected wetting, condensation or flooding of the filters or adsorbers, system modifications or repair to the housing, ductwork, damper or monitoring systems, etc.).*

4. *The Standard provides guidance for the design, implementation, execution and management of a nuclear air cleaning system testing program. It incorporates current*

best practices and available technologies to verify nuclear air cleaning system performance within its design basis.

5. *Additional informative material is provided. Included are examples of a visual inspection check list to detect obvious deficiencies, a typical HEPA filter bypass test procedure, a typical carbon adsorber bypass test procedure, and required staff qualifications.*

The results of PSR1 CSA N288.3.4 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.22.2. As identified in Reference [B.22-4], the Pickering PSR2 review of CSA N288.3.4-13 is an Incremental review. PSR2 Incremental review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.
- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

Per the PSR2 basis [B.22-4], the only system that has nuclear safety credited functions for filtration is containment, for which FADS (Filtered Air Discharge System) is the credited filtration system. Hence the focus of this review will be the performance testing of the FADS filters.

B.22.2 Compliance Assessment for Pickering PSR2

B.22.2.1 Application of PSR1 Reviews

The versions of N288.3.4 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

The first edition of this standard was issued in March 2013 after the completion of the Pickering Unit 5 to 8 ISR. Therefore, it was not reviewed for the Pickering B ISR.

Pickering Units 1,4

The first edition of this standard was issued in March 2013 after the completion of the Pickering A Return to Service. Therefore, N288.3.4 was not reviewed for Pickering A Return to Service. CSA N288.3.4 is not mentioned in the R04 Pickering Licence Conditions Handbook [B.22-2] or Pickering PROL Renewal Application [B.22-5].

Darlington NGS

An assessment of testing of air cleaning systems against the ASME requirements in ASME N-510-2007, "Testing of Nuclear Air Treatment Systems" was completed in 2011 as part of the Darlington ISR [B.22-6]. As part of this review, several compliance gaps were identified. These gaps were combined under Darlington ISR Issue D247 [B.22-7] and the Issue was determined to have low (level 4) safety significance. The resolution of the Issue was to adopt the N288.3.4 standard, given that it was considered to be more applicable to Darlington and superseded ASME N510. As part of the ISR, a Code Refresh review was subsequently performed against N288.3.4 in March 2014 and documented in Reference [B.22-8]. For this review two systems were selected for review as being representative of other ventilation systems, the Emergency Filtered Air Discharge System (EFADS) and the Powerhouse Ventilation System. This review did not identify any gaps against the standard for Darlington and concluded the following:

> *The changes made in CSA N288.3.4 [R-1] relative to ASME N510 [R-5] included substantial changes in structure and changes in intent of some of the clauses. The review of the clauses in this code refresh report confirms that Darlington NGS design and OPG Nuclear governance meets the intent of the requirements of CSA N288.3.4 [R-1].*

Given the similarities between the Pickering FADS and Darlington EFADS designs, and the common Nuclear testing governance (as identified in References [B.22-9] and [B.22-10]), the same conclusions are applicable to Pickering NGS and PSR2.

B.22.2.2 Application of the Post-PSR1 Reviews

A code review assessment against CSA N288.3.4-13 for OPG Nuclear facilities including Pickering NGS, Pickering Waste Management Facility, Darlington Waste Management Facility and Western Waste Management Facility was issued in January 2016 [B.22-11]. The scope of this review was much broader than just FADS and the clause-by-clause review identified thirty-five gaps [B.22-11] relating to multiple systems filters. Given the scope of the PSR2 review specifically relates to nuclear safety, FADS (as part of Containment) is the only filtered air or air cleaning system that needs to be considered per [B.22-4]. Therefore, the assessment in [B.22-11] has been reviewed to determine if any of the identified gaps could be applicable to Pickering FADS.

The review of [B.22-11] did not identify any gaps specifically related to FADS. The identified gaps relate to the generic test program design, documentation issues (e.g., inability to obtain historical records), treatment of uncertainties, and basis requirements. The recommendation from the review was,

> *Based on an assessment of documents and records that have been provided, there are numerous gaps in compliance of the OPG filter test program with the requirements and recommendations of CSA Standard N288.3.4.*
>
> *It is recommended that the main document describing the program, N-PROC-OP-0042, be expanded to more fully address the clauses of the Standard, either by adding description and explanation to the program document or by including explicit references*

to documents and records that provide the necessary evidence of compliance with the Standard. It may be helpful to construct a mapping of the sections of N-PROC-OP-0042 onto the clauses of the Standard to more clearly show compliance.

Because FADS is a Containment sub-system and part of a Special Safety System, the filter system has explicit design basis requirements to ensure that the nuclear safety credits are satisfied [B.22-12]. These are documented in the Safe Operating Envelope (SOE) for Pickering 1,4 and Pickering 5-8 in the Containment Operational Safety Requirements (OSRs) [B.22-13],[B.22-14], and in the associated OSR Compliance Tables. The Compliance Tables identify the routine operational tests and periodic surveillances specified to provide assurance of filter availability and effectiveness.

N-PROC-OP-0042 [B.22-9], Section 3.2.2 identifies the required FADS filter test frequency and Section 3.4 identifies the test acceptance limits. Entry A.7 of the Pickering B OSR Compliance Table [B.22-15] identifies the specified surveillances on the FADS filters. Iodine removal effectiveness is addressed by periodic replacement of charcoal. Report P-REP-03480-00042 [B.22-16] presents the test results of filter testing from 2002 to 2014. Appendix B and C of [B.22-16] present historical FADS bypass test results which demonstrate that the PROC [B.22-9] requirements of <0.05% and SOE requirement of <0.1% are consistently achieved. In addition to these surveillance requirements, there is a System Performance Monitoring Plan for the FADS system [B.22-17]. This plan specifies additional inspection and trending requirements for the filters.

Hence, it is concluded that Pickering FADS filter performance satisfies the requirements of N288.3.4-13 and there are no PSR2 gaps. The gaps identified in [B.22-11] do not have any safety significance relating to FADS performance and therefore, are not applicable to PSR2.

B.22.3 Compliance Assessment Summary for Pickering PSR2

There are no PSR2 gaps for CSA N288.3.4-13 [B.22-1]. Per the definition of Compliance for an Incremental review, Pickering has a PSR2 Compliance associated with CSA N288.3.4-13.

B.22.4 References

[B.22-1] CSA Standard N288.3.4-13, *Performance Testing of Nuclear Air-Cleaning Systems at Nuclear Facilities*, March 2013.

[B.22-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.22-3] CSA Impact Statement and Public Review Notice, *Product: New Standards; Product Designation: CSA N288.3.4; Date of release: 2013*, Date not provided.

[B.22-4] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.22-5] OPG Letter, P-CORR-00531-03719 R000, G. Jager to M. A. Leblanc, *Application for Renewal of Pickering Nuclear Generating Station Power Reactor Operating Licence*, July 4, 2012.

[B.22-6] OPG Report, NK38-REP-03680-10107 R001, *Review of ASME N510-2007 (December 2007) Testing on Nuclear Air Treatment Systems for Darlington Integrated Safety Review*, October 2011.

[B.22-7] OPG Report, NK38-REP-00770-0425825-R002, *In-service Testing of Air Treatment Systems,* June 2011.

[B.22-8] OPG Report, NK38-REP-03680-10147 R000, *Code Review of CSA N288.3.4-13 Performance Testing of Nuclear Air-Cleaning Systems at Nuclear Facilities*, March 2014.

[B.22-9] OPG Procedure, N-PROC-OP-0042 R004, *Contaminated Exhaust Ventilation Control Filter Testing,* October 2015.

[B.22-10] OPG Instruction, N-INS-73700-10000 R002, *Performance Testing of Contaminated Air Filtration Systems*, December 2011.

[B.22-11] OPG Report, N-REP-03480-0601454, *Code Review of CSA N288.3.4 for PN, PWMF, DWMF and WWMF High Efficiency Air Cleaning Assemblies*, January 2016.

[B.22-12] OPG Manual, System Design Requirements, NK30-DR-34230-10001 R000, *Filtered Air Discharge System*, March 2004.

[B.22-13] OPG Operational Safety Requirements, NA44-OSR-08131.02-00002 R003, *Pickering 1-4 Operational Safety Requirements: Negative Pressure Containment*, March 2015.

[B.22-14] OPG Operational Safety Requirements, NK30-OSR-08131.02-00003 R004, *Pickering 5-8 Operational Safety Requirements: Negative Pressure Containment*, March 2015.

[B.22-15] OPG Operational Safety Requirements, NK30-OSR-08131.02-00003 Table-01 R002, *Pickering NGS B Operational Safety Requirements: Negative Pressure Containment Compliance Table*, September 2014.

[B.22-16] OPG Report, P-REP-03480-00042 R000, *2014 Pickering Nuclear Air Emission Control Filter Performance Verification Report*, May 2015.

[B.22-17] OPG Plan, NK30-SPM-34230-00001 R011, *Pickering Nuclear System Performance Monitoring Plan, Filtered Air Discharge System*, February 2016.

B.23 CSA N288.7-15, "Groundwater Protection Programs at Class I Nuclear Facilities and Uranium Mines and Mills"

B.23.1 Background

The following text paraphrased from the introduction of CSA N288.7-15 [B.23-1] provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *The purpose of this Standard is to provide requirements and guidance which facilitate groundwater protection at Class 1 nuclear facilities and uranium mines and mills. Compliance with the Standard will allow facilities to demonstrate that they will not pose an unreasonable risk to the environment or the health and safety of humans and non-human biota from groundwater.*
>
> *The CSA N-Series Standards provide an interlinked set of requirements for the management of nuclear facilities and activities. The CSA N286 Standard provides overall direction to management to develop and implement sound management practices and controls while the other CSA nuclear Standards provide specific technical requirements and guidance that support the management system. This Standard works in harmony with CSA N286 and does not duplicate the generic requirements of CSA N286; however, it may provide more specific direction for meeting those requirements.*
>
> *This Standard addresses the design, implementation, and management of a groundwater protection program that incorporates best practices in Canada and internationally.*

CSA N288.7-15 is relevant to Safety Factor 14 (Radiological Impact on the Environment).

CSA N288.7-15 is the first edition of this standard. According to the N288.7-15 CSA Impact Statement and Public Review Notice [B.23-2]:

> *This NS* [New Standard] *is intended to provide requirements and guidance on designing and operating a GWPP* [Groundwater Protection Program] *throughout all phases of the facility lifecycle, with consideration given to baseline characterization, and facility construction, operation, decommissioning, and post-decommissioning phases. The focus is on radiological and non-radiological contaminants that have the potential to enter groundwater and that can impact humans and non-human biota. The Standard acknowledges the site-specificity of GWPP requirements, and gives consideration to monitoring approaches which are commensurate with the level of risk based on a designated groundwater end-use.*

Compliance with CSA N288.7 is not currently a licence requirement for Pickering NGS (in accordance with PROL 48.02/2018) per the R04 Pickering Licence Conditions Handbook [B.23-3].

As identified in Reference [B.23-4], the Pickering PSR2 review of CSA N288.7-15 is a High Level review. For a PSR2 High Level review, the degree of conformance with clauses or groups of clauses in the Law, Regulation, Code or Standard (L/R/C/S) is demonstrated by supporting evidence stating whether the intent of the requirements stipulated in the requirement document is met. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the intent of the safety requirement is met.
- Gap: A Gap indicates that the intent of the safety requirement is not met.

B.23.2 Compliance Assessment for Pickering PSR2

B.23.2.1 Application of PSR1 Reviews

CSA N288.7 was not reviewed as part of PSR1 as the document did not exist at the time that the previous Darlington and Pickering PSR1 reviews were performed. A high level review of N288.7-15 is provided in Section B.23.2.2 below.

B.23.2.2 Application of Post PSR1 Reviews

According to the N288.7-15 CSA Impact Statement and Public Review Notice [B.23-2], the following is a "Summary of Significant Features" of N288.7-15:

- *Feature 1: This is the first edition of CSA N288.7. This New Standard (NS) addresses the design and operation of groundwater protection programs (GWPPs) for Class I nuclear facilities and uranium mines and mills.*
- *Feature 2: This NS addresses design and implementation of groundwater monitoring programs (GWMP) for: i) pre-licensing baseline characterization; ii) site preparation, construction, and commissioning; iii) operations; iv) refurbishment or restarting after a prolonged shut-down; v) decommissioning; and vi) post-decommissioning prior to abandonment and/or institutional control.*
- *Feature 3: The scope of this NS includes: i) hazardous substances; ii) nuclear substances; and, iii) geochemical and physical characteristics of groundwater.*
- *Feature 4: This NS addresses review of systems, structures, and components (SSCs), and sentinel groundwater monitoring to provide early warning of any potential groundwater contamination issues, notwithstanding technical limitations.*
- *Feature 5: This NS has the following scope exclusions: a) groundwater monitoring of releases during accident scenarios; b) groundwater that has been discharged to, or mixed with, an effluent stream (Note: This is addressed in CSA N288.5); c) selecting or implementing risk management or remediation options; and d) defining dose assessment methods (Note: This is addressed in CSA N288.6.)* [11]

[11] PSR2 L/R/C/S reviews for CSA N288.5 and N288.6 are addressed separately.

OPG governance does not identify CSA N288.7-15 as a requirement. However, CSA N288.7 is not currently listed in the Pickering PROL or Licence Conditions Handbook. Therefore, this is not a PSR2 gap.

A compliance review against CSA N288.7-15 was not undertaken as part of previous PSR1 reviews and the following High Level assessment has been completed. In the review below, the degree of conformance with clauses or groups of clauses in the Standard is assessed for Pickering NGS by reference to supporting evidence stating whether the intent of the requirements stipulated in the Standard is met.

CSA N288.7-15 Clause	**PSR2 Review**	**Compliant or Gap**
0. Introduction	There are no requirements specified. The introduction describes the purpose of N288.7-15 and groundwater protection and monitoring programs.	N/A
1. Scope	There are no requirements specified. Sets context.	N/A
2. Reference Publications	There are no requirements specified. Describes the publications that N288.7-15 refers to.	N/A
3. Definitions and Abbreviations	There are no requirements specified. Defines various words or phrases, or acronyms, used in N288.7-15.	N/A
4. Goals and objectives of groundwater protection and groundwater monitoring programs *The overall goal of a groundwater protection program shall be to protect the quality and quantity of groundwater by minimizing interactions with the environment from activities associated with a nuclear facility, allowing for effective management of a groundwater resource.*	N-PROG-OP-0006, "Environmental Management" [B.23-5] gives authority to N-PROC-OP-0044, "Contaminated Lands and Groundwater Management" [B.23-6], which outlines directions and accountabilities for the groundwater monitoring program. Section 1.2.2 of [B.23-6] gives the requirement for a groundwater sampling and analysis plan (P-PLAN-10120-00001 for 2016) [B.23-7]. The sampling and analysis plan is updated yearly, and is based on P-REP-10120-10037, "Pickering Nuclear Generating Station Groundwater Monitoring Program Design" [B.23-8]. This document develops the groundwater monitoring program.	Compliant
4.1 Groundwater protection programs	There is no existing GWPP for Pickering NGS. However, the objective of the Pickering site GWMP is "to monitor the groundwater conditions on the site, including quality and quantity, and to identify any potential off-site impacts so that appropriate remedial actions can be taken" [B.23-8]. The GWMP design [B.23-8] generally follows CSA N288.4-10 [B.23-9], and is compliant with EPRI 1015118, "Groundwater Protection Guidelines for Nuclear Power Plants" [B.23-10]. This document is sufficiently comprehensive to also include	

CSA N288.7-15 Clause	PSR2 Review	Compliant or Gap
	the goals and objectives of a GWPP [B.23-8], and therefore Pickering meets the intent of Clause 4.1.	
4.2 Groundwater monitoring programs general objectives	The general objective of the GWMP is "to monitor the groundwater conditions on the site, including quality and quantity, and to identify any potential off-site impacts so that appropriate remedial actions can be taken" [B.23-8], as has been stated for Clause 4.1 above. The specific objectives of the GWMP are given in P-REP-10120-00041, "2015 Pickering Nuclear Groundwater Monitoring Program Results" [B.23-11]: • *Confirm predominant on-site groundwater flow characteristics of the PNGS site;* • *Monitor changes to on-site groundwater quality to ensure timely detection of inadvertent releases of nuclear and hazardous substances to groundwater; and,* • *Ensure that that there are no adverse off-site impacts from contaminants in groundwater.*	
5. Criteria for establishing groundwater protection and groundwater monitoring programs This section sets out the criteria for which a groundwater protection program and/or monitoring program must be implemented, as well as criteria for when a program should be considered. If the criteria for establishing a program are not met, there is a requirement for this to be documented.	See reviews of Clauses 5.1 and 5.2 below.	Compliant
5.1 Groundwater protection programs	Although there is no existing documentation specific to a GWPP, the GWMP design [B.23-8] includes the goals of a GWPP as discussed above in Clause 4.1, and the information required for a GWPP exists in other documents as discussed in the assessment for the Clauses in Section 6. Therefore, the lack of documentation specifically related to a GWPP is not safety significant. This finding is administrative and therefore is not a PSR2 gap.	

CSA N288.7-15 Clause	PSR2 Review	Compliant or Gap
5.2 Groundwater monitoring programs	There is an existing CSA N288.4-10-compliant GWMP for Pickering nuclear [B.23-8] which is currently in use [B.23-11]. Based on the assessment of Clauses 6-13 performed below, the existing GWMP for Pickering meets the intent of Clause 5.2.	
6. Design of a groundwater protection program This section sets out the requirements for the design of a groundwater protection program, including:	The existence of a GWPP for Pickering NGS has been discussed in Clauses 4.1 and 5.1 above, and is not repeated for the Clauses in Section 6. OPG is in compliance with the intent of CSA N288.7 section 6 [B.23-1] as a number of documents exist which contain information which is to be included in the design of a GWPP demonstrating that Clause 6 is met, including:	Compliant
6.1 General 6.2 Establishing of a conceptual site model 6.2.1 General 6.2.2 Evaluation of sources and potential releases to the subsurface 6.2.3 Characterization of the groundwater flow systems and contaminant migration 6.2.4 Groundwater end-use including receptors	• The "Pickering Nuclear Generating Station Groundwater Monitoring Program Design" [B.23-8] contains the groundwater protection goals, conceptual site model, a review of contaminants of potential concern based on previous assessments, and potential sources of contaminants. • Annual GWMP result reports (such as [B.23-11]) include a high-level discussion of the groundwater flow pattern, and contaminant sources. • NK30-REP-07701-00006, the "Refurbishment and Continued Operation of Pickering B Nuclear Generating Station Environmental Assessment Geology, Hydrogeology and Seismicity Technical Support Document" [B.23-12] contains a detailed description of the geology, hydrogeology and groundwater quality at the regional, local, and site level. It also summarizes the potential areas of groundwater contamination, groundwater/surface water interactions, end uses, and some receptors. • P-REP-07010-10012, the 2014 "Environmental Risk Assessment Report for Pickering Nuclear" [B.23-13] includes a description of the hydrogeology, including previous hydrogeological investigations. It also includes discussions on meteorology, soil quality, potential receptors, exposure pathways, and contaminants of potential concern. • P-REP-10120-10003, "2010 Pickering Nuclear Groundwater Monitoring System Report" [B.23-14] includes information on the groundwater flow characteristics of the site, the horizontal and vertical groundwater flow, the groundwater quality, and potential sources of contamination. This document	

CSA N288.7-15 Clause	PSR2 Review	Compliant or Gap
	indicates that hydrogeological investigations have been conducted at Pickering NGS since 1997. • The annual natural attenuation reports (such as P-REP-10120-0588548 [B.23-15]) for fuel oil impacted areas discuss the biodegradation of hydrocarbons by bacteria. • A site wide study of tritium in groundwater was performed in 2000 (NA44-REP-07010-10001, "Tritium in Groundwater Study Volume 1") [B.23-16] which details the regional land use and geology, site geology, hydrogeology, and groundwater flow regime. It examines the extent of the tritium contamination in the groundwater and investigates the sources of this contamination, and determines the impact to the environment.	
6.2.5 Groundwater vulnerability	No explicit assessment of groundwater vulnerability could be found; however, consideration of the hydrogeology, including water flow paths, recharge areas, and the penetrability and hydraulic conductivity of the overlying geology, can be found in the following documents: • The "Pickering Nuclear Generating Station Groundwater Monitoring Program Design" [B.23-8]; • The "Refurbishment and Continued Operation of Pickering B Nuclear Generating Station Environmental Assessment Geology, Hydrogeology and Seismicity Technical Support Document" [B.23-12]; • The 2014 "Environmental Risk Assessment Report for Pickering Nuclear" [B.23-13]; • "2010 Pickering Nuclear Groundwater Monitoring System Report" [B.23-14], and; • The tritium in groundwater study performed in 2000 [B.23-16]. Therefore Pickering meets the intent of Clause 6.2 relating to groundwater conceptual site model development.	
6.3 Prevention or minimization of potential releases	Prevention or minimization of potential releases is discussed in the following documents: • The tritium in groundwater study performed in 2000 [B.23-16] makes recommendations for mitigating the potential for future releases to groundwater, and;	

CSA N288.7-15 Clause	PSR2 Review	Compliant or Gap
	• Prevention or minimization of potential releases would be provided through equipment monitoring, inspection, and maintenance programs including the buried piping program (N-PROC-MA-0088, "Buried Piping Program Requirements") [B.23-17] and P-INS-07290-00001, the "Spill Prevention and Contingency Plan" [B.23-18]. Clause 6.3.2 states that "New (modern) facility designs and operations should, commensurate with the level of risk, include specific provisions for secondary containment and effective in-facility leak detection." • With respect to secondary/spill containment: o N-REP-07292-0255802, "Spill Risk Assessment for Pickering and Darlington Nuclear Power Stations" [B.23-29], lists the containment structures and measures in place at Pickering NGS for spills from storage facilities, operation units, and transportation activities. Per N-REP-07292-0255802, secondary containment is provided for most areas of the station via inactive and active drainage sumps, tanks and lagoons. o Per N-STD-OP-0026, "Spill Management" [B.23-30], applicable codes and design standards are to be applied to spill containment for any new equipment, storage tanks, drums, and containers of liquids. Appendix F provides guidance on spill containment requirements, including physical containment for interim and long-term storage as well as process systems at the station. • With respect to in-facility leak detection, various systems are available at Pickering Units 1,4 and 5-8 for this purpose (e.g., Heavy Water Transfer and Storage leak detection, tritium leak detection, Reactor Building D2O leak detection via beetles, etc.). In addition, the buried piping program [B.23-17] is in place to govern pipe condition monitoring and leak detection. Therefore, the intent of clause 6.3 is met and there is no PSR2 gap.	
6.4 Development of specific groundwater protection goals	The "Pickering Nuclear Generating Station Groundwater Monitoring Program Design" [B.23-8] discusses the goals of the GWMP, which include identifying potential off-site impacts and demonstrating there are no adverse impacts from hazardous substances to the receiving environment. The GWMP also develops the conceptual site model and discusses existing groundwater quality and flow and exposure pathways to potential receptors.	

CSA N288.7-15 Clause	PSR2 Review	Compliant or Gap
	The annual GWMP result reports (such as [B.23-11]) also discuss the monitoring goals and the results of the monitoring program. Impact on groundwater quality, groundwater flow, and impacts to receptors are discussed in the report. Therefore Pickering meets the intent of Clause 6.4 relating to GWPP goals.	
6.5 Development of a groundwater monitoring program	There is an existing GWMP, which is enacted under [B.23-6]. The program was designed in [B.23-8] and is performed under the groundwater monitoring plan which is updated each year [B.23-7]. Therefore Pickering meets the intent of Clause 6.5.	
6.6 Development of other programs associated with groundwater protection	Other programs associated with groundwater protection include programs such as: • Groundwater well inspection, as discussed in N-GUID-10120-10001, the guide for "Inspection of Groundwater Monitoring Wells" [B.23-19]; • The buried piping program [B.23-17]; • The "Spill Prevention and Contingency Plan" [B.23-18]; • N-STD-OP-0031, "Monitoring of Nuclear and Hazardous Substances in Effluents" [B.23-20]. Therefore Pickering meets the intent of Clause 6.5. Hence, Pickering meets the intent of all requirements in Clause 6 and is PSR2 compliant.	
7. Design of a groundwater monitoring program This section sets out the requirements for the design of a groundwater monitoring program, including:	The existing GWMP was designed in 2012 [B.23-8]. CSA N288.7 was published in 2015 [B.23-1]. OPG is in compliance with the intent of CSA N288.7 Clause 7 based on the following existing information:	Compliant
7.1 General 7.2 Systematic planning processes for the development of a groundwater monitoring program 7.2.1 General 7.2.2 Systematic planning process	• The "Pickering Nuclear Generating Station Groundwater Monitoring Program Design" [B.23-8] defines the objectives of the GWMP, and outlines the monitoring program including requirements for: o The program boundary and areas of concern; o Monitoring locations; o Sampling frequency and the sampling protocol;	

CSA N288.7-15 Clause	PSR2 Review	Compliant or Gap
7.2.3 Definition of GWMP objectives 7.2.4 Information required to meet groundwater monitoring program objectives 7.2.5 Spatial boundaries of the GWMP 7.2.6 Data evaluation methods 7.2.7 Data quality 7.2.8 Definition and establishment of groundwater evaluation criteria 7.2.9 Development of a process to address exceedances of groundwater evaluation criteria 7.2.10 Detailed design of GWMP 7.2.11 Review of GWMP	○ Data interpretation and results, including recommendations based on the program; ○ Data quality is discussed, with reference to blind duplicates, field blanks, and trip blanks; ○ The monitoring program was designed in accordance with CSA N288.4-10 [B.23-9]; ○ Groundwater evaluation criteria are given for contaminants of concern in Appendix A; and, ○ Audit and review of the program. • P-CLP-10120-00001, the "Chemistry Laboratory Procedure: Sampling Groundwater Monitoring System and Site Drainage" [B.23-21] includes objectives for the GWMP, and sample collection criteria including blind and blank samples. • The 2016 "Pickering Groundwater Sampling and Analysis Plan" [B.23-7] identifies the objectives of the monitoring plan, wells to be monitored, sample availability targets, total number of samples to be taken, and method detection limits. Therefore Pickering meets the intent of Clause 7.2 relating to planning and design of a GWMP.	
7.3 Selection of monitoring strategy 7.3.1 General 7.3.2 Monitoring in proximity of a potential release point 7.3.3 Monitoring along the down gradient perimeter of a site 7.3.4 Monitoring along the groundwater flow path of a contaminant plume	• The "Pickering Nuclear Generating Station Groundwater Monitoring Program Design" [B.23-8] identifies requirements for monitoring locations, including source monitoring, upgradient and downgradient monitoring along groundwater pathways, and perimeter monitoring at the site boundary. • The "2016 Pickering Groundwater Sampling and Analysis Plan" [B.23-7] identifies the wells to be monitored. Therefore Pickering meets the intent of Clause 7.3 relating to selection of a monitoring strategy for a GWMP.	

CSA N288.7-15 Clause	PSR2 Review	Compliant or Gap
7.4 Nuclear and hazardous substances to be monitored 7.4.1 General 7.4.2 Gross Parameters 7.4.3 Surrogate Parameters 7.4.4 Indicator Parameters	• The “Pickering Nuclear Generating Station Groundwater Monitoring Program Design” [B.23-8] defines the nuclear and hazardous substances to be monitored based on previous groundwater monitoring. Field chemistry and physical parameters measurement requirements are also discussed. • The “Chemistry Laboratory Procedure: Sampling Groundwater Monitoring System and Site Drainage” [B.23-21] discusses sampling and analysis for various nuclear and hazardous substances defined in Appendix A. • The “2016 Pickering Groundwater Sampling and Analysis Plan” [B.23-7] identifies substances to be monitored, as well as their method detection limits. Therefore Pickering meets the intent of Clause 7.4 relating to substances to be monitored as part of a GWMP.	
7.5 Boreholes and monitoring wells 7.5.1 General 7.5.2 Borehole drilling 7.5.3 Monitoring well design and installation 7.5.4 Borehole logs 7.5.5 Retention of borehole logs 7.5.6 Monitoring well assessment	The monitoring well network was established prior to the design of the GWMP. Existing boreholes were used in the “Groundwater Monitoring Program Design” [B.23-8]. The installation of some of the existing groundwater monitoring wells is discussed in NA44-REP-10130-10001, the “Groundwater Quality Investigation South of the Irradiated Fuel Bay at Pickering NGS ‘A’ Monitoring Well Network Installation” report [B.23-22]. This report discusses the subsurface stratigraphy of the area, and the methodology for choosing the well locations and for well construction. Detailed information on each borehole and well was recorded, including location, elevation, and borehole logs.	
7.5.7 Maintenance and inspection of monitoring wells	• The guide for “Inspection of Groundwater Monitoring Wells” [B.23-19] includes information on the frequency of well inspection as well as guidance on well maintenance. The guide also discusses physical protection and access control for wells. o The associated form N-FORM-11445, “Inspection of Groundwater Monitoring Wells” [B.23-23] gives detailed instructions for well inspection. • The “Sampling and Analysis Plan” [B.23-7] identifies requirements for the frequency of well inspection and maintenance in section 4.2.	

CSA N288.7-15 Clause	PSR2 Review	Compliant or Gap
	• The "Groundwater Monitoring Program Design" [B.23-8] considers the wells which would be required to meet the monitoring program objectives, and discusses deactivation of wells that are not required in section 4.6. • The "Chemistry Laboratory Procedure: Sampling Groundwater Monitoring System and Site Drainage" [B.23-21] states the requirement for well inspection prior to sampling in section 4.5.1. Therefore Pickering meets the intent of Clause 7.5 relating to boreholes and monitoring wells used as part of a GWMP.	
7.6 Sampling locations 7.6.1 General considerations 7.6.2 monitoring well placement for a new facility or in an unaffected setting 7.6.3 Monitoring well placement for assessing an existing plume	• The "Groundwater Monitoring Program Design" [B.23-8] determines sampling locations by considering the location of existing wells in the context of previous groundwater monitoring results. Monitoring locations are chosen to incorporate source monitoring, upgradient and downgradient monitoring along groundwater pathways, and perimeter monitoring at the site boundary. • The "Sampling and Analysis Plan" [B.23-7] includes a list of wells at which the groundwater level should be measured to confirm the groundwater flow characteristics. Therefore Pickering meets the intent of Clause 7.6 relating to boreholes and monitoring wells used in the GWMP.	
7.7 Sampling frequency 7.7.1 General considerations 7.7.2 Sampling frequency for routine groundwater monitoring 7.7.3 Sampling frequency for plume monitoring	• The "Groundwater Monitoring Program Design" [B.23-8] specifies the sampling frequency at each well, and identifies the rationale for the suggested frequency and circumstances that may result in a change to the sampling frequency. • The "Sampling and Analysis Plan" [B.23-7] identifies the schedules for the water level snapshots and hazardous substance sampling. The plan also lists any changes to the sampling schedule along with the rationale for the change. • Annual GWMP result reports (such as [B.23-11]) make recommendations on changes to the monitoring program, including changes to sampling frequencies, to be implemented in the following year's plan. Therefore Pickering meets the intent of Clause 7.7 relating to the sampling frequency used in the GWMP.	

CSA N288.7-15 Clause	PSR2 Review	Compliant or Gap
7.8 Supplementary studies and other monitoring activities 7.8.1 General 7.8.2 Refinement of conceptual site model 7.8.3 Characterization of potential effects from plume discharge	• Annual GWMP result reports (such as [B.23-11]) make recommendations for supplementary studies and changes to the monitoring program based on the monitoring results. • The annual natural attenuation reports (such as [B.23-15]) for fuel oil impacted areas discuss the biodegradation of hydrocarbons by bacteria. Therefore Pickering meets the intent of Clause 7.8 relating to studies to supplement or that relate to the GWMP.	
8. Sampling and analytical procedures This section sets out the requirements for the sampling and analysis procedures for the groundwater monitoring program, including: 8.1 General 8.2 Sampling equipment 8.3 Sample collection 8.4 Sample volume, containers, and preservatives	• The "Chemistry Laboratory Procedure: Sampling Groundwater Monitoring System and Site Drainage" [B.23-21] thoroughly describes the steps for groundwater sampling and monitoring, including sampling for water levels and for hazardous substances. The procedure includes instructions for well water purging, sample handling and storage, sampling equipment and materials, sample collection, management of waste water, blank and duplicate samples, and instructions for field measurements. • The "Sampling and Analysis Plan" [B.23-7] identifies the method detection limits for the substances to be monitored, and includes a discussion on duplicate and blank samples, as well as the sample availability target. Lab qualification requirements are also given. • The "Pickering Nuclear Generating Station Groundwater Monitoring Program Design" [B.23-8] summarizes the sampling protocol in section 4.4. This section references [B.23-21] and discusses well purging, sample collection method, field measurements, and storage and transportation. The design also includes the detection limits and benchmark values for the hazardous substances. • Annual GWMP result reports (such as [B.23-11]) discuss elements of the sampling procedures used in the preceding year, including lab qualification. • The annual natural attenuation reports (such as [B.23-15]) discuss elements of the sampling procedures used in the preceding year for hydrocarbons, including lab qualification and quality control requirements and limits. • N-PROC-OP-0014, the "Chemistry Measurement and Analysis" procedure [B.23-24] identifies "the minimum expectations to ensure chemistry measurements are	Compliant

CSA N288.7-15 Clause	PSR2 Review	Compliant or Gap
	accurate and representative," including field sampling requirements and contamination control.	
9. Interpretation of data This section sets out the requirements relating to the interpretation of data for the groundwater monitoring program, including: 9.1 Objectives of data 9.2 Data evaluation 9.3 Parameters 9.4 Comparison to groundwater evaluation criteria 9.5 Statistical analysis 9.5.1 General 9.5.2 Descriptive statistics 9.5.3 Outliers 9.5.4 Non-detectable results 9.5.5 Hypothesis testing 9.5.6 Trend analysis 9.5.7 Contour maps 9.6 Contextual considerations when interpreting results 9.7 Data management 9.8 Significant figures	• The "Pickering Nuclear Generating Station Groundwater Monitoring Program Design" [B.23-8] identifies general requirements related to data interpretation and reporting in Section 4.8. This section requires annual reporting, with data interpretation, evaluation of data quality, and a discussion of the implication of the data to be included in the report. There is a requirement to compare groundwater concentrations to administrative limits. • Annual GWMP result reports (such as [B.23-11]) contain the discussion of the interpretation of groundwater data as per the GWMP objectives. These reports include interpretation of the data for all contaminants measured (as per [B.23-8], [B.23-21], and [B.23-7]), comparison to the criteria, a statistical analysis and summary, a graphical representation of the monitoring data, and contour maps. • N-GUID-10120-10000, the "Annual Groundwater Monitoring Program Roles and Responsibilities" guide [B.23-25] discusses the responsibilities for the site-wide groundwater database in section 6.0. • The "Chemistry Measurement and Analysis" procedure [B.23-24] identifies the requirements and procedures for determining the appropriate analytical method. Requirements procedures for determining measurement accuracy, the use of statistics and data management are also discussed. • N-PROC-OP-0017, the "Laboratory Work and Data Management" procedure [B.23-26] describes procedures and requirements for management of data from laboratory results and for the Chemistry and Environment Management System.	Compliant
10. Quality assurance and quality control This section sets out the quality assistance and quality control requirements for the groundwater monitoring program, including: 10.1 General	• The "Pickering Nuclear Generating Station Groundwater Monitoring Program Design" [B.23-8] identifies quality assurance and quality control requirements for the program in section 4.7. Requirements for blank and duplicate samples, as well as sample precision, are discussed. • The "Sampling and Analysis Plan" [B.23-7] describes the requirements and process for quality assurance and quality control for field collection and lab analysis. The	Compliant

CSA N288.7-15 Clause	PSR2 Review	Compliant or Gap
10.2 Roles and responsibilities 10.3 Measurements and quality control 10.3.1 Equipment maintenance 10.3.2 Non-conformance 10.3.3 Performance verification 10.3.4 Procedures verification 10.4 Records	plan also includes the requirement for ISO 17025 certification for the analytical lab. • The "Chemistry Laboratory Procedure: Sampling Groundwater Monitoring System and Site Drainage" [B.23-21] identifies the quality assurance and quality control measures to be taken while performing groundwater sampling, including blank and duplicate samples. Instructions for instrument checks and sample storage and transportation are also given. • The "Contaminated Lands and Groundwater Management" procedure [B.23-6] outlines personnel accountable for all aspects of the groundwater management program and their responsibilities, including quality assurance and quality control. Requirements for records are also given, including a retention period. • The "Annual Groundwater Monitoring Program Roles and Responsibilities" guide [B.23-25] describes the training and qualification requirements, as well as responsibilities for GWMP personnel. • Annual GWMP result reports (such as [B.23-11]) discusses the quality assurance and quality control measures used in the GWMP, as well as the quality control results. The report discusses the results and interpretation of the monitoring data. • The "Chemistry Measurement and Analysis" procedure [B.23-24] identifies the requirements and procedures for laboratory quality control, including quality control samples, contamination control, measurement accuracy, blind sample assessments, and routine and long-term quality control monitoring. Instrument maintenance and calibration requirements are also described.	

CSA N288.7-15 Clause	PSR2 Review	Compliant or Gap

11. Reporting, review and audit
 	
This section sets out the reporting, review, and audit requirements for the groundwater monitoring program, including:
 	
11.1 Preparation of monitoring reports documenting the GWMP
 	
11.2 Periodic review of the groundwater protection program and groundwater monitoring program
 	
11.3 Annual assessment of the groundwater monitoring program
 	
11.4 Audits
	• The "Pickering Nuclear Generating Station Groundwater Monitoring Program Design" [B.23-8] sets out reporting, review, and audit requirements for the GWMP in sections 4.8 and 4.9.
	
	    o Reporting requirements include an annual report with interpretation of the monitoring data and evaluation of data quality.
	
	    o An audit of the GWMP is required every five years, or more frequently if site conditions change substantially. Audit contents, auditor qualifications, and documentation requirements are also outlined. Additionally, the audit of the GWMP meets the intent of the audit of the GWPP given in CSA N288.7 section 11.2 [B.23-1].
	
	    o An annual internal assessment of the GWMP is required each year. The scope of the assessment, staff qualifications, and documentation requirements are given.
	
	• A GWMP result report (such as [B.23-11]) is prepared annually as per the program design [B.23-8]. The report includes the results of the monitoring program, relevant groundwater and hydrogeological characteristics, an assessment of the QA results, and recommendation of supplementary studies. The report also includes the results and recommendations of the annual assessment of the GWMP.
	
	• A natural attenuation report (such as [B.23-15]) is prepared annually to document the results of the hydrocarbon natural attenuation monitoring as per the program design [B.23-8].
	
	• The "Contaminated Lands and Groundwater Management" procedure [B.23-6] outlines personnel accountable for the GWMP self-assessments and audits.
		Compliant

CSA N288.7-15 Clause	PSR2 Review	Compliant or Gap
12. Staff qualifications and training This section sets out the staff qualification and training requirements for the groundwater protection program and monitoring program, including: 12.1 Personnel qualifications 12.2 Training 12.3 Maintenance of training records	• The “Annual Groundwater Monitoring Program Roles and Responsibilities” guide [B.23-25] documents roles and responsibilities for the operation of the GWMP, including training and qualification requirements. • The “Pickering Nuclear Generating Station Groundwater Monitoring Program Design” [B.23-8] summarizes the qualification and training requirements for staff performing work related to the groundwater management program, as well as requirements for documentation and ongoing assessment of qualifications. • The “Contaminated Lands and Groundwater Management” procedure [B.23-6] gives a high-level description of the directions and accountabilities for establishing and maintaining the GWMP, including staff qualification and training requirements. • N-TQD-419-00001, the “Environment Professional Training and Qualification Description” [B.23-27] establishes training and qualification requirements for Environment Professionals, including groundwater management personnel, at OPG. • The “Chemistry Measurement and Analysis” procedure [B.23-24] specifies the qualification and training requirements for chemistry laboratory staff. • N-PROC-TR-0008, the “Systematic Approach to Training” procedure [B.23-28] identifies document retention periods for training records.	Compliant
13. Documentation This section sets out the documentation requirements for the groundwater protection program and monitoring program, including: 13.1 Groundwater protection program documentation 13.2 Groundwater monitoring program documentation	• The “Pickering Nuclear Generating Station Groundwater Monitoring Program Design” [B.23-8] documents the detailed design of the monitoring program, as described above. This includes identification of hazardous substances to be measured, sampling locations and frequency, data interpretation, evaluation criteria, quality assurance and quality control requirements, staff qualification and training requirements, audit and review of the program, and groundwater monitoring site maps. • The “Sampling and Analysis Plan” [B.23-7] is updated each year, and documents quality control requirements, hazardous substances to be measured, sampling locations and frequency, detection limits, sampling schedule, and well maintenance.	Compliant

CSA N288.7-15 Clause	PSR2 Review	Compliant or Gap
	• The "Chemistry Laboratory Procedure: Sampling Groundwater Monitoring System and Site Drainage" [B.23-21] documents the instructions for sampling, quality control requirements, substances to be measured, and sample requirements. • The guide for "Inspection of Groundwater Monitoring Wells" [B.23-19] documents inspection and maintenance requirements for monitoring wells. • Annual GWMP result reports (such as [B.23-11]) include a site map of the monitoring stations sampled. • The "Annual Groundwater Monitoring Program Roles and Responsibilities" guide [B.23-25] documents roles and responsibilities for the operation of the GWMP, including training and qualification requirements. • The "Contaminated Lands and Groundwater Management" procedure [B.23-6] outlines directions and accountabilities for establishing and maintaining the GWMP, and includes the document retention period.	
Annex A – Additional guidance for conceptual site models	There are no requirements specified. Informative.	N/A
Annex B – Development of groundwater evaluation criteria	There are no requirements specified. Informative.	N/A
Annex C - Uncertainty	There are no requirements specified. Informative.	N/A

B.23.3 Compliance Assessment Summary for Pickering PSR2

There are no PSR2 gaps for CSA N288.7-15 [B.23-1]. Per the definition of Compliance for a High Level review, Pickering has a PSR2 Compliance associated with CSA N288.7-15.

B.23.4 References

[B.23-1] CSA Standard N288.7-15, *Groundwater Protection Programs at Class I Nuclear Facilities and Uranium Mines and Mills*, June 2015.

[B.23-2] CSA Communication, *Impact Statement and Public Notice for CSA N288.7-15, Groundwater Protection Programs at Class I Nuclear Facilities and Uranium Mines and Mills*, Date not provided.

[B.23-3] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.23-4] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.23-5] OPG Program, N-PROG-OP-0006 R018, *Environmental Management*, April 2015.

[B.23-6] OPG Procedure, N-PROC-OP-0044 R003, *Contaminated Lands and Groundwater Management*, May 2014.

[B.23-7] OPG Report, P-PLAN-10120-00001 R002, *Pickering Nuclear Groundwater Sampling and Analysis Plan*, December 2015.

[B.23-8] OPG Report, P-REP-10120-10037 R000, *Pickering Nuclear Generating Station Groundwater Monitoring Program Design*, September 2012.

[B.23-9] CSA Standard N288.4-10, *Environmental Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills*, 2010.

[B.23-10] EPRI Report, Report No. 1015118, *Groundwater Protection Guidelines for Nuclear Power Plants*, November 2007.

[B.23-11] OPG Report, P-REP-10120-00041 R000, *2015 Pickering Nuclear Groundwater Monitoring Program Results*, March 2016.

[B.23-12] OPG Report, NK30-REP-07701-00006 R000, *Geology, Hydrogeology and Seismicity Technical Support Document: Refurbishment and Continued Operation of Pickering B Nuclear Generating Station Environmental Assessment*, March 2007.

[B.23-13] OPG Report, P-REP-07010-10012 R000, *Environmental Risk Assessment Report for Pickering Nuclear*, January 2014.

[B.23-14] OPG Report, P-REP-10120-10003 R000, *2010 Pickering Nuclear Groundwater Monitoring System Report*, April 2011.

[B.23-15] OPG Report, P-REP-10120-0588548 R000, *2015 Annual Report: Monitored Natural Attenuation*, March 2016.

[B.23-16] OPG Report, NA44-REP-07010-10001 R000, *Tritium in Groundwater Study Volume 1*, September 2000.

[B.23-17] OPG Procedure, N-PROC-MA-0088 R003, *Buried Piping Program Requirements*, April 2015.

[B.23-18] OPG Instruction, P-INS-07290-00001 R013, *Spill Prevention and Contingency Plan*, March 2015.

[B.23-19] OPG Guide, N-GUID-10120-10001 R000, *Inspection of Groundwater Monitoring Wells*, October 2011.

[B.23-20] OPG Standard, N-STD-OP-0031 R006, *Monitoring of Nuclear and Hazardous Substances in Effluents*, October 2014.

[B.23-21] OPG Procedure, P-CLP-10120-00001 R008, *Chemistry Laboratory Procedure: Sampling Groundwater Monitoring System and Site Drainage*, November 2015.

[B.23-22] Ontario Hydro Report, NA44-REP-10130-10001 R000, *Groundwater Quality Investigation South of the Irradiated Fuel Bay at Pickering NGS 'A' Monitoring Well Network Installation*, November 1998.

[B.23-23] OPG Form, N-FORM-11445 R000, *Inspection of Groundwater Monitoring Wells*, November 2011.

[B.23-24] OPG Procedure, N-PROC-OP-0014 R006, *Chemistry Measurement and Analysis*, November 2015.

[B.23-25] OPG Guide, N-GUID-10120-10000 R001, *Annual Groundwater Monitoring Program Roles and Responsibilities*, September 2012.

[B.23-26] OPG Procedure, N-PROC-OP-0017 R006, *Laboratory Work and Data Management*, July 2015.

[B.23-27] OPG Training and Qualification Description, N-TQD-419-00001 R009, *Environment Professional Training and Qualification Description*, January 2016.

[B.23-28] OPG Procedure, N-PROC-TR-0008 R019, *Systematic Approach to Training*, January 2014.

[B.23-29] OPG Report, N-REP-07292-0255802, *Spill Risk Assessment for Pickering and Darlington Nuclear Power Stations*, June 2008.

[B.23-30] OPG Standard, N-STD-OP-0026 R008, *Spill Management*, March 2014.

ONTARIOPOWER GENERATION	
ACCEPTED	✓
ACCEPTED AS NOTED	
REVISE AND RESUBMIT	
Signature	Date 03 MAR 2017
Name: Mike Ruffolo, Manager	
Dept: Pickering Engineering - Aging Management & Strategic Initiatives	
OPG Proprietary	
Doc No.: P-REP-03680-00029	Rev: 000
This acceptance does not relieve the contractor from responsibility for errors or omissions or from any obligations or liability under this contract.	

amec foster wheeler

Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 1, 5, 6, and 7

PS112/RP/020 R00

March 2, 2017

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Revision Summary – For Amec Foster Wheeler Report PS112/RP/020

Rev	Date	Author	Comments
R00	March 2, 2017	Amec Foster Wheeler Staff	Initial issue of report addressing OPG comments on PSR2 Law, Regulation, Code and Standard reviews.

EXECUTIVE SUMMARY

Ontario Power Generation (OPG) is performing a Periodic Safety Review (PSR) to support the possibility of continued operation of Pickering Nuclear Generating Station (NGS) beyond 2020. The PSR (referred to as "PSR2") is a subsequent PSR building on the review basis of earlier OPG Integrated Safety Reviews and other associated assessments. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

PSR2 will support and complement the licence renewal application for Pickering NGS going forward. Fifteen Safety Factors will be assessed as part of the PSR. The purpose of Safety Factor reviews is to confirm that the design, condition and operation of Pickering Units 1,4 and 5-8 (as well as common systems) will support continued safe operation for the period of PSR2 by:

- Assessing compliance against "Review Tasks" identified in the PSR2 Basis Document [1], which were derived from Canadian Nuclear Safety Commission (CNSC) REGDOC-2.3.3 [2] and International Atomic Energy Agency (IAEA) SSG-25 [3];

- Documenting assessments against applicable modern Laws, Regulations, Codes and Standards (L/R/C/Ss) (as defined in Reference [1]); and

- Reviewing the effectiveness of applicable OPG Nuclear Programs that support the above assessments, through review of audit and self-assessment results.

The process to identify the modern L/R/C/Ss that are applicable to the PSR2 Assessment Basis involved first creating a broad list from multiple sources (potential candidate L/R/C/Ss) and then filtering it to identify those that are most significant, and that are applicable to the PSR2 scope. The identification and selection criteria are detailed in the PSR2 Basis Document [1]. The result of the identification and selection process was a set of modern L/R/C/Ss that became part of the "PSR2 Assessment Basis". This report provides the reviews of L/R/C/Ss that are required to address PSR2 Safety Factors 1 (Plant Design), 5 (Deterministic Safety Analysis), 6 (Probabilistic Safety Assessment) and 7 (Hazard Analysis). As noted in Section 2.0, reviews of several L/R/C/Ss applicable to other Safety Factors were provided in References [4], [5] and [6] and findings from these reviews are not duplicated in this report. There is also some overlap with other Safety Factors for a number of L/R/C/Ss considered, as outlined in Table 1 in Section 2.0 of this report.

The summary of findings documented in Appendix B of this report is as follows:

- CSA N285.0-12, "General Requirements For Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants": There are two gaps associated with Safety Factor 1.

- CSA N293-12, "Fire Protection for Nuclear Power Plants": There are three gaps associated with Safety Factor 1.

- CNSC REGDOC-2.4.1 (2014), "Deterministic Safety Analysis": There are two gaps associated with Safety Factor 5.

- CNSC REGDOC-2.4.2 (2014), "Probabilistic Safety Assessment (PSA) for Nuclear Power Plants": There is one gap associated with Safety Factor 6.

- CSA N287.1-14, "General Requirements for Concrete Containment Structures for Nuclear Power Plants": No gaps.

- CSA N287.3-14, "Design Requirements for Concrete Containment Structures for Nuclear Power Plants": No gaps.

- CSA N287.5-11, "Examination and Testing Requirements for Concrete Containment Structures for Nuclear Power Plants": There is one gap associated with Safety Factor 1.

- CSA N290.0-11, "General requirements for Safety Systems of Nuclear Power Plants": There are three gaps associated with Safety Factor 1.

- CSA N290.1-13, "Requirements for the Shutdown Systems of Nuclear Power Plants": There is one gap associated with Safety Factor 1.

- CSA N290.2-11, "Requirements for Emergency Core Cooling Systems of Nuclear Power Plants": There are two gaps associated with Safety Factor 1.

- CSA N290.3-11, "Requirements for the Containment System of Nuclear Power Plants": There is one gap associated with Safety Factor 1.

- CSA N290.4-11, "Requirements for Reactor Control Systems of Nuclear Power Plants": There is one gap associated with Safety Factor 1.

- CSA N290.5-06, "Requirements for Electrical Power and Instrument Air Systems of CANDU Nuclear Power Plants": There is one gap associated with Safety Factor 1.

- CSA N290.6-09, "Requirements for Monitoring and Display of Nuclear Power Plant Safety Functions in the Event of an Accident": No gaps.

- CSA N290.11-13, "Requirements for Reactor Heat Removal Capability during Outage of Nuclear Power Plants": There are three gaps associated with Safety Factor 1.

- CSA N290.14-15, "Qualification of Digital Hardware and Software for Use in Instrumentation and Control Applications for Nuclear Power Plants": There is one gap associated with Safety Factor 1.

- CSA N291-15, "Requirements for Safety-related Structures for Nuclear Power Plants": There are three CSA N291-15 gaps associated with Safety Factor 1. There is also one PSR2 gap for CSA N291-15 associated with Safety Factor 4 which is identified as a PSR2 gap in OPG Report P-REP-03680-00024 R000, "Pickering 5-8 Continued Operations Plan Review in Support of PNGS Periodic Safety Review 2 (PSR2)" [7]. Therefore, a duplicate gap has not been created under CSA N291-15.

- CSA N285.6 Series-12, "Material Standards for Reactor Components for CANDU Nuclear Power Plants": No gaps.

- ASME B31.1 (2014), "Power Piping": No gaps.

- ASME BPVC (2015), "Boiler and Pressure Vessel Code": No gaps.

- CSA B51-14, "Boiler, Pressure Vessel, and Pressure Piping Code": No gaps.

- NFPA 20 (2016), "Standard for the Installation of Stationary Pumps for Fire Protection": No gaps.

- NFPA 24 (2016), "Standard for the Installation of Private Fire Service Mains and Their Appurtenances": There are two gaps associated with Safety Factor 1.

- CNSC REGDOC-2.5.2 (2014), "Design of Reactor Facilities: Nuclear Power Plants": There are eight gaps associated with Safety Factor 1, one gap associated with Safety Factor 5, and one gap associated with Safety Factor 6.

- CNSC G-144 (2006), "Trip Parameter Acceptance Criteria for the Safety Analysis of CANDU Nuclear Power Plants": No gaps.

- CNSC G-149 (2000), "Computer Programs Used in Design and Safety Analyses of Nuclear Power Plants and Research Reactors": No gaps.

- CNSC R-77 (1987), "Overpressure Protection Requirements for Primary Heat Transport Systems in CANDU Power Reactors Fitted with Two Shutdown Systems": No gaps.

- CSA N288.2-14, "Guidelines for Calculating Radiological Consequences to the Public from a Release of Airborne Radioactive Material for Nuclear Reactor Accidents": There is one gap associated with Safety Factor 5.

- CSA N290.7-14, "Cyber-Security for Nuclear Power Plants and Small Reactor Facilities": The gap analysis, N-REP-69000-10003 R000, "Gap Analysis Between CSA N290.7-14 Cyber Security Requirements for Nuclear Power Plants and Small Reactor Facilities" [8] and implementation plan for N290.7-14 was accepted by the CNSC. For reasons of security and confidentiality, the findings of the gap analysis for N290.7-14 will not be discussed in PSR2.

- NBCC (2010), "National Building Code of Canada": No gaps.

- NFCC (2010), "National Fire Code of Canada": There is one gap associated with Safety Factor 1. This issue is identified as a PSR2 gap in OPG Report P-REP-03680-00024 R000, "Pickering 5-8 Continued Operations Plan Review in Support of PNGS Periodic Safety Review 2" [7]. Therefore, a duplicate gap under NFCC (2010) has not been created.

- CSA N290.8-15, "Technical Specification Requirements for Nuclear Power Plant Components": There is one gap associated with Safety Factor 1.

Details of the reviews can be found in Table 2 and Appendix B of this report.

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1.0 INTRODUCTION

Ontario Power Generation (OPG) is performing a Periodic Safety Review (PSR) to support the possibility of continued operation of Pickering Nuclear Generating Station (NGS) beyond 2020.[1] A comprehensive Integrated Safety Review (ISR) was completed for Pickering Units 5 through 8 in 2009 in support of refurbishment and continued operation. Pickering Units 1,4 integrated safety assessments were also performed for Pickering A Return to Service (PARTS) in support of approval to restart Units 1 and 4. In addition to these Pickering-specific studies, the 2013 Darlington ISR performed extensive code and standard reviews that were updated in relation to the versions that were assessed in the 2009 Pickering B ISR.[2] These previous ISRs are considered to constitute the first PSR completed for Pickering (referred to as "PSR1"). The current PSR (referred to as "PSR2") is a subsequent PSR building on the basis of earlier OPG integrated safety assessments through review of the various studies, assessments and licence renewals performed since PSR1. The PSR2 scope and methodology are described in the Pickering PSR2 Basis Document [1].

PSR2 will support and complement the licence renewal application for Pickering NGS going forward. Fifteen Safety Factors will be assessed as part of the PSR. The purpose of Safety Factor reviews is to confirm that the design, condition and operation of Pickering Units 1,4 and 5-8 (as well as common systems) will support continued safe operation for the period of PSR2 by:

- Assessing compliance against "Review Tasks" identified in the PSR2 Basis Document [1], which were derived from Canadian Nuclear Safety Commission (CNSC) REGDOC-2.3.3 [2] and International Atomic Energy Agency (IAEA) SSG-25 [3];

- Documenting assessments against applicable modern Laws, Regulations, Codes and Standards (L/R/C/Ss) (as defined in Reference [1]); and

- Reviewing the effectiveness of applicable OPG Nuclear Programs that support the above assessments, through review of audit and self-assessment results.

The process to identify the modern L/R/C/Ss that are applicable to the PSR2 Assessment Basis involved first creating a broad list from multiple sources (potential candidate L/R/C/Ss) and then filtering it to identify those that are most significant, and

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1 Currently, Pickering Units 5-8 are approved to operate to 247,000 Effective Full Power Hours. This operation limit is expected to be reached on some units in 2020. For the purposes of PSR2, OPG assumes operation of Pickering NGS for up to eight additional years, from 2020 until 2028. OPG will make a decision regarding the permanent shut down dates for the six reactors following the performance of a technical evaluation that will include PSR2, and will communicate it to the CNSC as required by the current Power Reactor Operating Licence (PROL).

2 Much of the compliance assessment and evaluation of Safety Factor health for the Darlington ISR is based on programs and practices that apply across OPG's nuclear operations. As a result, where Pickering is confirmed to follow the same nuclear programs and practices as were assessed for Darlington, the Darlington ISR programmatic conclusions are applicable to Pickering.

that are applicable to the PSR2 scope. The identification and selection criteria are detailed in the PSR2 Basis Document [1]. The result of the identification and selection process was a set of modern Laws, Regulations, Codes and Standards that became part of the "PSR2 Assessment Basis". The PSR2 Basis Document also identifies the modern version and date of the L/R/C/S and the type of review that will be completed in PSR2. The types of review are explained in Section 2.0 below.

This report provides the reviews of L/R/C/Ss with content applicable to Safety Factors 1 (Plant Design), 5 (Deterministic Safety Analysis), 6 (Probabilistic Safety Assessment) and 7 (Hazard Analysis). As noted in Section 2.0, reviews of several L/R/C/Ss applicable to other Safety Factors were provided in References [4], [5] and [6] and findings from these reviews are not duplicated in this report. There is also some overlap with other Safety Factors for a number of L/R/C/Ss considered, as outlined in Table 1 in Section 2.0 of this report.

As outlined in IAEA SSG-25 [3], the objectives of these Safety Factor reviews are as follows:

- The objective of the review of Safety Factor 1 is to determine the adequacy of the design of the nuclear power plant and its documentation by assessment against the current licensing basis and national and international standards, requirements and practices.

- The objective of the review of Safety Factor 5 is to determine to what extent the existing Deterministic Safety Analysis (DSA) is complete and remains valid when the following aspects have been taken into account:

    - The actual plant design, including all modifications of Structures, Systems and Components (SSCs) since the last update of the safety analysis report or PSR1;

    - Current operating modes and fuel management;

    - The actual condition of SSCs important to safety and their predicted state at the end of the period covered by the PSR2;

    - The use of modern validated computer codes;

    - Current deterministic methods;

    - Current safety standards and knowledge (including research and development outcomes); and

    - The existence and adequacy of safety margins.

- The objective of the review of Safety Factor 6 is to determine:
    - The extent to which the existing Probabilistic Safety Assessment (PSA) study remains valid as a representative model of the plant;
    - Whether the results of the PSA show that the risks are sufficiently low and well balanced for all postulated initiating events and operational states;
    - Whether the scope (which should include all operational states and identified internal and external hazards), methodologies and extent (i.e., Level 1, 2 or 3) of the PSA are in accordance with current national and international standards and good practices;
    - Whether the existing scope and application of PSA are sufficient.
- The objective of the review of Safety Factor 7 is to determine the adequacy of protection of the plant against internal and external hazards, with account taken of the plant design, site characteristics, the actual condition of the SSCs important to safety and their predicted state at the end of the period covered by PSR2, and current analytical methods, safety standards and knowledge.

2.0 REVIEW SCOPE AND METHODOLOGY

PSR2 is focused on the extension of Pickering NGS operations beyond 2020. Thus, it is important that the methodology for PSR2 be focused on addressing aspects of the review that are likely to have material impact in terms of identifying enhancements that will be reasonable and practicable to implement during the remaining commercial life of the plant. PSR2 conducts reviews against a baseline of the PSR1 work. It is important to note that OPG conducts regular reviews of new and revised Codes and Standards, so a large amount of information is already available to assist in the Safety Factor reviews. In OPG letter N-CORR-00531-05661, W.M. Elliott to P.A Webster and M. Santini, "Design Codes and Standards Effective Dates for OPG Nuclear Fleet" [9], OPG stated:

> *OPG commits to completing a code-over-code review (i.e., review of changes) of subsequent editions, addendum and/or updates of the Codes and Standards listed in Attachment 1* [of the referenced document]*. Key emerging issues due to major changes in the codes will be addressed immediately, or as agreed with the CNSC on a case-by-case basis. Otherwise, OPG will confirm in a letter to the CNSC that these reviews have been completed and there are no significant technical issues...*

As a result, many of the updated codes and standards issued since PSR1 have already had gap assessments performed, to varying degrees of detail, which are utilized and cited in the present Pickering PSR2.

As a subsequent PSR, PSR2 focuses on changes in requirements, plant conditions, operating experience and new information. Since PSR2 is an update of previous ISRs, it incorporates reviews of L/R/C/Ss that have occurred as new versions have been issued. Therefore, clause-by-clause reviews of the majority of applicable L/R/C/Ss have already been completed and there is little value in repeating that process. If clause-by-clause reviews were to be undertaken in PSR2, a major portion of the review effort would be consumed by repackaging existing information that remains largely applicable and, therefore, is not contributing to the identification of new insights and enhancements. A more constructive approach is therefore applied that maximizes the value and usefulness of the work by focusing attention where it is most beneficial, i.e., on identifying new issues. The primary objective for this work, which is to identify safety significant enhancements that may be implemented during the limited remaining life of the station, is achieved using this process and is expected to result in the same (safety significant) Global Issues being identified as would result from a clause-by-clause assessment.

Since this assessment is a subsequent PSR, the focus is on identifying differences between what was previously assessed and what is now different within the current Pickering PSR2 Assessment Basis. In general, these differences relate to:

- More recent (new or revised) L/R/C/S versions than what was previously assessed;[3]

- Safety significant differences between Pickering and Darlington, if the Darlington ISR is the basis for the earlier assessment;

- Implications of extending Pickering NGS operation beyond 2020; and

- Safety significant differences between Pickering Units 1,4 and Units 5-8.

In most cases L/R/C/S reviews are incremental in nature and performed by topic or subject matter for revised requirements. The rationale for this is that new or updated requirements that need to be included in PSR2 are predominantly replacements for other L/R/C/S that were previously assessed, and specify requirements that can be readily mapped to existing OPG programs.

To align with the goals of a subsequent PSR, the following three tiers of reviews are applied for PSR2:

- Clause-by-Clause review: New L/R/C/S referenced in Pickering PROL 48.02/2018 (listed in Appendix C of the Licence Conditions Handbook) will be subjected to a clause-by-clause type review. In a clause-by-clause review, conformance with individual clauses is demonstrated by supporting evidence stating whether the requirements stipulated in the requirement document are met;

- High Level review: New L/R/C/S not referenced in Pickering PROL 48.02/2018 but which are in the PSR2 Assessment Basis will be subject to a high level review. In a high level review, the degree of conformance with clauses or groups of clauses in the L/R/C/S is demonstrated by supporting evidence stating whether the intent of the requirements stipulated in the requirement document are met; and

- Incremental review: For L/R/C/Ss that have been reviewed in PSR1 but have had revisions since the last review, a topical review will be performed of the changes. (Note: Incremental reviews may also include high level review elements if required, e.g., where potentially safety significant L/R/C/S clauses were not addressed in PSR1 (due to significant structural or technical changes to an L/R/C/S since previous PSR1 reviews), or where past Pickering NGS L/R/C/S reviews do not exist and application of Darlington ISR conclusions is

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[3] "New" refers to a code or standard that was not previously considered in the context of earlier assessments. "Revised" refers to an updated version of a code or standard that was previously considered in the context of earlier assessments. Where a document has a new number/type, but addresses the same topic from the same organization, it is a "revised", not "new", document (e.g., if a REGDOC replaces a CNSC G or RD document).

not fully sufficient (based on the need for station-specific supporting evidence).)

Most of the L/R/C/Ss in the PSR2 Assessment Basis receive incremental reviews since PSR2 is an update of previous PSR1 assessments and clause-by-clause or high level reviews for the majority of the L/R/C/Ss in the PSR2 Assessment Basis have already been completed. Implementation plans (including gap analyses or code-over-code reviews) also exist for the latest editions of many L/R/C/Ss. As a result, an incremental review is also used in circumstances where a L/R/C/S in the PSR2 Assessment Basis was not assessed in previous PSR1 reviews but an implementation plan currently exists for compliance.

The PSR2 incremental reviews in this report include an assessment of the intent of recent changes to the L/R/C/Ss identified in Table 1 on a topic or subject-matter basis where there is potential to impact nuclear safety. Incremental reviews provide:

- A summary of the purpose of the L/R/C/S;
- Pertinent background information about the current revision of the L/R/C/S that is being considered;
- Identification of which Safety Factor(s) are applicable to the current revision of the L/R/C/S;
- A description of which version(s) of the L/R/C/S were assessed for PSR1 (i.e., Darlington ISR (where applicable), Pickering B ISR and PARTS code reviews);
- Identification of whether the current version of the L/R/C/S is an update of a previous version of the L/R/C/S that was assessed in PSR1 (and if so, a description of the major changes in the latest revision is provided as discussed below);
- An assessment of the applicability of PSR1 assessment findings (gaps and conclusions), including the implications of extending Pickering NGS operation beyond 2020 if any;
- An assessment of the applicability of assessment findings that address more recent (post-PSR1) editions of the L/R/C/S, including any implementation or transition plans that are already committed to by OPG; and
- Where PSR1 and post-PSR1 assessments are not sufficient to address changes in the latest edition of the L/R/C/S, an assessment of the changes from the previously assessed edition of the L/R/C/S (including identification of any safety significant PSR2 gaps which result).

High Level reviews provide the same information as above, where applicable, in a similar format. However, given that High Level L/R/C/Ss generally have not received past assessment during PSR1, the Incremental review content is augmented by a high level, section-by-section assessment of the degree of conformance of Pickering NGS with the L/R/C/S (demonstrating, with supporting evidence, whether the intent of the requirements stipulated in the document are met).

There are currently no L/R/C/S clause-by-clause reviews identified in the PSR2 Assessment Basis.

The Safety Factor 1, 5, 6, and 7 L/R/C/S reviews identify Compliances and Gaps as defined below:

- Compliance:
    - Where an Incremental review has been performed, Compliance indicates that the change in the safety requirement, per the topical review, is met.
    - Where a High Level review has been performed, Compliance indicates that the intent of the safety requirement is met.
- Gap:
    - Where an Incremental review has been performed, a Gap indicates that the change in the safety requirement, per the topical review, is not met.
    - Where a High Level review has been performed, a Gap indicates that the intent of the safety requirement is not met.

The reviews assume that use of the word:

- "Shall" is used in an L/R/C/S to express a requirement, i.e., a provision that the licensee is obliged to satisfy in order to comply with the standard;
- "Should" is used to express a recommendation or that which is advised but not required;
- "May" is used to express an option or that which is permissible within the limits of the standard; and
- "Can" is used to express possibility or capability.

Table 1 identifies the L/R/C/Ss in the PSR2 Assessment Basis that are applicable to Safety Factors 1, 5, 6, and 7, with several exceptions discussed below. Table 1 also identifies the modern version and date of each L/R/C/S to be considered, the Safety Factor(s) to which each document is applicable, and the type of review that was completed in PSR2. Reviews for each L/R/C/S are provided in Appendix B, and results are summarized in Section 3.0.

The following L/R/C/Ss applicable to Safety Factors 1, 5, 6 and 7 are excluded from Table 1 below. Specifically:

- CSA N286-12, "Management System Requirements for Nuclear Facilities", which is applicable to Safety Factors 5 and 6;
- CSA N286.7-16, "Quality Assurance of Analytical, Scientific and Design Computer Programs", which is applicable to Safety Factors 1, 5, 6, and 7;
- CSA N285.4-14, "Periodic Inspection of CANDU Nuclear Power Plant Components", which is applicable to Safety Factor 1;
- CSA N285.5-13, "Periodic Inspection of CANDU Nuclear Power Plant Containment Components", which is applicable to Safety Factor 1;
- CSA N287.2-08, "Material Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants", which is applicable to Safety Factor 1;
- CSA N289.1-08, "General Requirements for Seismic Design and Qualification of CANDU Nuclear Power Plants", which is applicable to Safety Factor 1;
- CSA N289.2-10, "Ground Motion Determination for Seismic Qualification of Nuclear Power Plants", which is applicable to Safety Factor 1;
- CSA N289.3-10, "Design Procedures for Seismic Qualification of Nuclear Power Plants", which is applicable to Safety Factor 1;
- CSA N289.4-12, "Seismic Instrumentation Requirements for Nuclear Power Plants and Nuclear Facilities", which is applicable to Safety Factor 1;
- CSA N289.5-12, "General Requirements for Seismic Design and Qualification of CANDU Nuclear Power Plants", which is applicable to Safety Factor 1;
- CNSC G-278 (2003), "Human Factors Verification and Validation Plans", which is applicable to Safety Factor 1;
- CNSC G-276 (2003), "Human Factors Engineering Program Plans", which is applicable to Safety Factor 1;
- CNSC REGDOC-2.3.2 (2015), "Accident Management, Version 2", which is applicable to Safety Factors 1, 5, 6, and 7;
- CSA N286.7.1-09, "Guideline for the Application of N286.7-99, Quality Assurance of Analytical, Scientific, and Design Computer Programs for Nuclear Power Plants", which is applicable to Safety Factors 1, 5, 6 and 7; and

- CSA N290.12-14, "Human Factors in Design for Nuclear Power Plants", which is applicable to Safety Factor 1.

The reviews for the above L/R/C/Ss are provided in References [4], [5] and [6] and findings are not duplicated in this report.

Table 1: Applicable L/R/C/Ss for Pickering PSR2 Safety Factors 1, 5, 6, and 7

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
L/R/C/Ss Referenced in Pickering NGS PROL 48.02/2018						
1	CSA N285.0	General Requirements For Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants	N285.0-12	1	Incremental	N285.0 addressed as part of Pickering B and Darlington ISRs and PARTS.
2	CSA N293	Fire Protection for Nuclear Power Plants	N293-12	1, 7, 13	Incremental	N293 addressed as part of Pickering B and Darlington ISRs and PARTS.
3	CNSC REGDOC-2.4.1	Deterministic Safety Analysis	2014	5, 7	Incremental	C-6 addressed as part of the Pickering B ISR and PARTS. S-310 and RD-310 addressed as part of the Pickering B and Darlington ISRs, respectively. Implementation plan in place and gap assessment between REGDOC-2.4.1 and OPG Safety Analysis Program already performed.
4	CNSC REGDOC-2.4.2	Probabilistic Safety Assessment (PSA) for Nuclear Power Plants	2014	6, 7	Incremental	S-294 addressed as part of Pickering B and Darlington ISRs. Implementation plan in place.
Additional L/R/C/Ss						
5	CSA N287.1	General Requirements for Concrete Containment Structures for Nuclear Power Plants	N287.1-14	1	Incremental	N287.1 addressed as part of Pickering B and Darlington ISRs and PARTS.
6	CSA N287.3	Design Requirements for Concrete Containment Structures for Nuclear Power Plants	N287.3-14	1	Incremental	N287.3 addressed as part of Pickering B and Darlington ISRs and PARTS.

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
7	CSA N287.5	Examination and Testing Requirements for Concrete Containment Structures for Nuclear Power Plants	N287.5-11	1, 2	Incremental	N287.5 addressed as part of Darlington ISR.
8	CSA N290.0	General requirements for Safety Systems of Nuclear Power Plants	N290.0-11	1	Incremental	N290.0 addressed as part of Darlington ISR.
9	CSA N290.1	Requirements for the Shutdown Systems of Nuclear Power Plants	N290.1-13	1	Incremental	N290.1 addressed as part of Pickering B and Darlington ISRs and PARTS.
10	CSA N290.2	Requirements for Emergency Core Cooling Systems of Nuclear Power Plants	N290.2-11	1	Incremental	N290.2 addressed as part of Darlington ISR. CNSC R-9 (precursor to N290.2) addressed as part of Pickering B and Darlington ISRs and PARTS.
11	CSA N290.3	Requirements for the Containment System of Nuclear Power Plants	N290.3-11	1	Incremental	N290.3 addressed as part of Darlington ISR. CNSC R-7 (precursor to N290.3) addressed as part of Pickering B and Darlington ISRs and PARTS.
12	CSA N290.4	Requirements for Reactor Control Systems of Nuclear Power Plants	N290.4-11	1	Incremental	N290.4 addressed as part of Pickering B and Darlington ISRs and PARTS.
13	CSA N290.5	Requirements for Electrical Power and Instrument Air Systems of CANDU Nuclear Power Plants	N290.5-06	1	Incremental	N290.5 addressed as part of Pickering B and Darlington ISRs and PARTS.
14	CSA N290.6	Requirements for Monitoring and Display of Nuclear Power Plant Safety Functions in the Event of an Accident	N290.6-09	1	Incremental	N290.6 addressed as part of Pickering B and Darlington ISRs and PARTS.
15	CSA N290.11	Requirements for Reactor Heat Removal Capability During Outage of Nuclear Power Plants	N290.11-13	1	High Level	N290.11 not addressed as part of Pickering B or Darlington ISRs.

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
16	CSA N290.14	Qualification of Digital Hardware and Software for Use in Instrumentation and Control Applications for Nuclear Power Plants	N290.14-15	1	High Level	N290.14 not addressed as part of Pickering B or Darlington ISRs.
17	CSA N291	Requirements for Safety-related Structures for Nuclear Power Plants	N291-15	1, 2, 4	Incremental	N291 addressed as part of Darlington ISR.
18	CSA N285.6 Series-12	Material Standards for Reactor Components for CANDU Nuclear Power Plants	N285.6 Series-12	1	Incremental	N285.6 addressed as part of Pickering B and Darlington ISRs and PARTS.
19	ASME B31.1	Power Piping	B31.1-14	1	Incremental	B31.1 addressed as part of Darlington ISR and PARTS.
20	ASME BPVC	Boiler and Pressure Vessel Code	BPVC 2015	1	Incremental	BPVC addressed as part of Pickering B and Darlington ISRs and PARTS.
21	CSA B51	Boiler, Pressure Vessel, and Pressure Piping Code	B51-14	1	Incremental	B51 addressed as part of Pickering B and Darlington ISRs and PARTS.
22	NFPA 20	Standard for the Installation of Stationary Pumps for Fire Protection	NFPA-20 (2016)	1	Incremental	NFPA 20 addressed as part of Darlington ISR.
23	NFPA 24	Standard for the Installation of Private Fire Service Mains and Their Appurtenances	NFPA-24 (2016)	1	Incremental	NFPA 24 addressed as part of Darlington ISR.
24	CNSC REGDOC-2.5.2	Design of Reactor Facilities: Nuclear Power Plants	2014	1, 5, 6, 7	Incremental	RD-337 and NS-R-1 (precursors to REGDOC-2.5.2) addressed as part of Darlington ISR. NS-R-1 also addressed as part of Pickering B ISR.
25	CNSC G-144	Trip Parameter Acceptance Criteria for the Safety Analysis of CANDU Nuclear Power Plants	2006	5	Incremental	G-144 addressed as part of Pickering B and Darlington ISRs.

#	Document Number	Document Title	Modern Version for PSR2	Applicable Safety Factors	Type of Review	Review Type Basis
26	CNSC G-149	Computer Programs Used in Design and Safety Analyses of Nuclear Power Plants and Research Reactors	2000	1, 5, 6, 7	Incremental	G-149 addressed as part of Pickering B and Darlington ISRs.
27	CNSC R-77	Overpressure Protection Requirements for Primary Heat Transport Systems in CANDU Power Reactors Fitted with Two Shutdown Systems	1987	1	Incremental	R-77 addressed as part of Pickering B and Darlington ISRs and PARTS.
28	CSA N288.2	Guidelines for Calculating Radiological Consequences to the Public from a Release of Airborne Radioactive Material for Nuclear Reactor Accidents	N288.2-14	5	Incremental	N288.2 addressed as part of Pickering B and Darlington ISRs.
29	CSA N290.7	Cyber-Security for Nuclear Power Plants and Small Reactor Facilities	N290.7-14	1	High Level [4]	N290.7 not addressed as part of Pickering B or Darlington ISRs.
30	NBCC	National Building Code of Canada	NBCC 2010	1	Incremental	NBCC addressed as part of Pickering B and Darlington ISRs and PARTS.
31	NFCC	National Fire Code of Canada	NFCC 2010	1	Incremental	NFCC addressed as part of Pickering B and Darlington ISRs and PARTS.
32	CSA N290.8	Technical Specification Requirements for Nuclear Power Plant Components	N290.8-15	1	High Level	N290.8 not addressed as part of Pickering B or Darlington ISRs.

[4] As discussed in Appendix B.29, a gap analysis for N290.7-14 has been completed by OPG and satisfies the intent of this PSR2 High Level Review. For reasons of security and confidentiality, the findings of this gap analysis will not be discussed in PSR2.

3.0 RESULTS AND CONCLUSIONS

The results of the PSR2 reviews of the L/R/C/Ss listed in Table 1 are summarized in Table 2 below. Additional background information and details regarding the gaps listed in Table 2 are provided in Appendix B of this report.

Table 2: PSR2 L/R/C/S Review Results for Safety Factors 1, 5, 6, and 7

Appendix Section #	L/R/C/S Reviewed	L/R/C/S Review Results
B.1	CSA N285.0-12, "General Requirements For Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants"	There are two PSR2 CSA N285.0-12 (including Updates No. 1 and No. 2) gaps which relate to Safety Factor 1 (Plant Design): 1. Clause A.2.3.1 of CSA N285.0-06 identifies that for Shutdown Systems, pressure-retaining portions shall be classified as Class 1, except for three listed exceptions. It was identified during the Pickering B ISR that a limited number of Liquid Injection Shutdown System (LISS) components, which should have been Class 1, were purchased and installed as Class 3. In follow-up, OPG proposed four actions to address the deficiency. When refurbishment was not pursued, a code classification concession was accepted for continued operations. This code classification concession and the four actions identified in the Pickering B ISR gap resolution need to be reconsidered in the context of operation of Pickering NGS beyond 2020. Therefore, this has been identified as a PSR2 gap. 2. The PARTS review against CSA-N285.0-95 identified two Acceptable Deviations relating to Clause 7.0 requiring confirmation that the allowable cycles for fatigue would not be exceeded. For Pickering Units 1,4 and Units 5-8 operation beyond 2020, further confirmation is required that the allowable cycles for fatigue will continue to bound current service limits for extended operation. Therefore, this has been identified as a PSR2 gap.
B.2	CSA N293-12, "Fire Protection for Nuclear Power Plants"	There are three PSR2 gaps for CSA N293-12 which relate to Safety Factor 1 (Plant Design): 1. Clause 7.2.1.10.1 of CSA N293-12 states: "A display and control centre shall be located in the MCR [Main Control Room]… capable of providing detailed information on the location and nature of the signal. In addition, the panel operator shall be able to control the fire alarm system without having to leave his or her station." Pickering 014 Display Annunciation Station 014-67140-WS2342 in the Emergency Operating Centre is capable of providing annunciation only, and there is no Display Annunciation Station in the Pickering 014 MCR (although there is limited annunciation). Therefore, this has been identified as a PSR2 gap.

Appendix Section #	L/R/C/S Reviewed	L/R/C/S Review Results
		2. Clause 7.2.1.13 of CSA N293-12 states: "Electrical conductors that are installed in service spaces containing other combustible materials and that are used in connection with fire alarm systems and emergency equipment, including fire alarm cables… shall be capable of performing their intended functions for not less than 1 hour after the start of a fire." Modifications to the Fire Protection System meet the requirements of CAN/ULC-S524 which mandates a 1 hour fire rating as described in Section 2.5 of NA44-DM-71400.2-00001 R001, Section A.2 of NA44-DM-71400-00002 R000 and Section 2 of NK30-DM-71400-00001 R006. This is achieved by the use of Edwards System Technology (EST) that connects the fire alarm control panels via a data communication link with dual redundant circuit wiring paths. However, existing Pyrotronics fire alarm control panels are not similarly connected and, hence, may be susceptible to loss of alarm signal due to spot burning of a cable. While measures such as lack of combustible material in service spaces, combustible transient material control practices, and inherent protection afforded by Pickering NGS cable routing practices used in the Fire Protection systems mitigate the lack of such a feature, it could not be confirmed based on existing documentation that all essential fire alarm cables are capable of performing their intended functions for not less than 1 hour after the start of a fire to meet the requirement of N293-12 sub-clause 7.2.1.13. As a result, this has been identified as a PSR2 gap. 3. Clause 7.3.2.2 (d) of CSA N293-12 states: "At a minimum, the fire protection water pumping system shall consist of at least one diesel-engine-driven fire pump and one electric-motor-driven fire pump set, with each pump set being capable of providing, the flow rate and pressure specified in Item (a)". This Clause is met at Pickering Units 1,4 with the provision of diesel-driven firewater pumps, backed up by supplies from the High Pressure Service Water (HPSW) system (as noted in the Pickering A Safety Report NA44-SR-01320-00001 R015, Section 11.5.1.1). It is not met at Pickering Units 5-8, where the Fire Protection System is comprised of the HPSW supplies from the four units only. As a result, Pickering Units 5-8 does not comply with Clause 7.3.2.2 (d) of CSA N293-12 and this has been identified as a PSR2 gap.

Appendix Section #	L/R/C/S Reviewed	L/R/C/S Review Results
B.3	CNSC REGDOC-2.4.1 (2014), "Deterministic Safety Analysis"	There are two PSR2 CNSC REGDOC-2.4.1 (2014) gaps which relate to Safety Factor 5 (Deterministic Safety Analysis): 1. The REGDOC-2.4.1 Implementation Plan and associated gap assessments capture all gaps related to REGDOC-2.4.1 and incorporate a systematic selection of the scope of work to address the most pertinent gaps in accordance with the graded approach to upgrading existing analyses. REGDOC-2.4.1 compliant analysis activities and progress related to REGDOC-2.4.1 implementation in the Pickering Licence Conditions Handbook are tracked according to the CNSC Compliance Verification Criteria. Since the implementation is in progress, this has been identified as a PSR2 gap for Pickering NGS REGDOC-2.4.1 compliance. 2. As described in the REGDOC-2.4.1 Implementation Plan: "Limited upgrades are proposed in the Pickering A and B Plan, which has been developed with consideration for demonstration of continued safe operation while accounting for the limited remaining operating life of the Pickering Units". The REGDOC-2.4.1 Implementation Plan for Pickering did not consider operation past 2020 and therefore the need for review and update of the Implementation Plan in the context of operation of Pickering NGS beyond 2020 is identified as a PSR2 gap. This will be informed by the timeline of the Darlington REGDOC-2.4.1 Implementation Plan, and the limited additional years of Pickering NGS operation.
B.4	CNSC REGDOC-2.4.2 (2014), "Probabilistic Safety Assessment (PSA) for Nuclear Power Plants"	There is one PSR2 CNSC REGDOC-2.4.2 (2014) gap which relates to Safety Factor 6 (Probabilistic Safety Assessment): 1. The REGDOC-2.4.2 Pickering Implementation Plan agreed to with the CNSC did not consider operation beyond 2020 and therefore, the review and update of the Implementation Plan in the context of operation of Pickering NGS beyond 2020 is required. Therefore, this has been identified as a PSR2 gap.
B.5	CSA N287.1-14, "General Requirements for Concrete Containment Structures for Nuclear Power Plants"	There are no PSR2 gaps for CSA N287.1-14. Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with CSA N287.1-14.
B.6	CSA N287.3-14, "Design Requirements for Concrete Containment Structures for Nuclear Power Plants"	There are no PSR2 gaps for CSA N287.3-14. Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with CSA N287.3-14.

Appendix Section #	L/R/C/S Reviewed	L/R/C/S Review Results
B.7	CSA N287.5-11, "Examination and Testing Requirements for Concrete Containment Structures for Nuclear Power Plants"	There is one PSR2 CSA N287.5-11 gap which relates to Safety Factor 1 (Plant Design): 1. The Concrete Containment Structures (CCSs) at Pickering A and B were built and tested to meet the 1965 and 1970 National Building Code of Canada requirements, respectively, prior to the initial issuance of CSA N287.5. No assessments exist which demonstrate that the requirements in effect during construction of Pickering NGS CCSs comply with the requirements of CSA N287.5. Ongoing confirmation that the Pickering NGS CCSs remain fit for service is demonstrated via periodic and in-service inspections conducted in accordance with the requirements of CSA N285.5 and N287.7, and the resultant inspection reports attest to the quality of the design. In addition, the Engineering Change Control (ECC) process ensures that any design changes made to the Pickering CCSs will comply with N287.5 going forward, as applicable. The original Pickering construction included requirements for tests and quality control procedures which generally meet the intent of N287.5. Furthermore, retroactive application of N287.5 to the as-built design of CCSs cannot be practically achieved without rebuilding them. Nevertheless, there is a PSR2 gap for Pickering NGS given that compliance with the specific requirements of N287.5 has not been demonstrated.
B.8	CSA N290.0-11, "General requirements for Safety Systems of Nuclear Power Plants"	There are three PSR2 N290.0-11 gaps which relate to Safety Factor 1 (Plant Design): 1. The Darlington ISR identified a gap against Clause 4.14.10 of N290.0-11 as a result of the lack of design standards related to Human Factors Engineering (HFE) or HFE activities being formally documented when the control rooms were originally designed and constructed. Pickering NGS has many years of successful Special Safety System (SSS) operation and the absence of formal HFE in the original design is not expected to have any nuclear safety significance relating to SSSs. However, the Darlington gap is also applicable to Pickering NGS and is therefore identified as a PSR2 gap. 2. Clause 4.2 of N290.0-11 requires that Plant States be grouped into several categories, including Anticipated Operational Occurrences (AOOs). This is consistent with clauses of REGDOC-2.4.1 and REGDOC-2.5.2 related to identification and classification of initiating events. Since AOOs have not been identified and analyzed in the current Pickering Safety Reports, the requirements and credits attributed to the Special Safety Systems for AOOs, if any, cannot be readily ascertained. This

Appendix Section #	L/R/C/S Reviewed	L/R/C/S Review Results
		issue has therefore been identified as a PSR2 gap. It is being addressed as part of REGDOC-2.4.1 implementation. 3. OPG is currently in the process of completing the High Energy Line Break Assessment (HELBA) for Pickering NGS. Preliminary results show that there would be no consequential damage caused by the rupture of high energy pipes inside containment to safety related equipment, beyond that already accounted for in the Safety Reports. The final HELBA reports for Pickering Units 5-8 have been completed, while Pickering Units 1,4 are expected to be completed in 2017. Since this work has not been completed for Pickering 1,4, this is identified as a PSR2 gap.
B.9	CSA N290.1-13, "Requirements for the Shutdown Systems of Nuclear Power Plants"	There is one PSR2 CSA N290.1-13 gap which relates to Safety Factor 1 (Plant Design): 1. Clause 4.1.8.2 of CSA N290.1-13 is for a new plant and requires remote tripping and monitoring capability for both Shutdown Systems. Pickering Units 1,4 only have one Shutdown System with tripping capability from separate logic (SDSA and SDSE). Remote tripping capability is available for Pickering 5-8 SDS2 and Pickering 1,4 SDSE. However, Pickering Units 5-8 and 1,4 do not have remote tripping and monitoring capability for SDS1 or SDSA respectively. Therefore, this has been identified as a PSR2 gap.
B.10	CSA N290.2-11, "General Requirements for Emergency Core Cooling systems of Nuclear Power Plants"	There are two PSR2 CSA N290.2-11 gaps which relate to Safety Factor 1 (Plant Design): 1. Clause 5.2.1.2 of CSA N290.2-11 requires that Emergency Coolant Injection System (ECIS) design requirements be based on the assumption that the least effective of the Shutdown Systems has operated successfully. The Pickering Units 5-8 Safety Report analysis does address this requirement and the requirement is also contained in the Pickering Units 5-8 Design Requirements. However, this requirement cannot be met for Pickering Units 1,4 since there is only one Shutdown System (albeit with tripping capability from separate SDSA and SDSE logic). Therefore, this has been identified as a PSR2 gap. 2. Clause 5.14.11 of CSA N290.2-11 requires instrumentation to be available to monitor post-accident effectiveness and to determine the extent of plugging of ECIS debris interceptors (strainers). While relative health of a strainer can be inferred by a combination of ECIS recovery pump performance and reactor building water level, there is no direct correlation between these conditions and debris loading available.

Appendix Section #	L/R/C/S Reviewed	L/R/C/S Review Results
		Therefore, this has been identified as a PSR2 gap (which is applicable to both Pickering Units 5-8 and 1,4).
B.11	CSA N290.3-11, "Requirements for the Containment System of Nuclear Power Plants"	There is one PSR2 CSA N290.3-11 gap which relates to Safety Factor 1 (Plant Design): 1. Per CSA N290.3-11, a Containment Energy Management System (EMS) and Radionuclide Management System (RMS) are required to protect containment and minimize radiological releases for Beyond Design Basis Accidents (BDBAs). The Pickering EMS and RMS use the Filtered Air Discharge System (FADS) and Reactor Building Air Cooling Units (ACUs). Enhancements to the AC power supplies to these systems and related loads are being provided by Phase 2 Emergency Mitigating Equipment (EME), which is not yet fully implemented. This PSR2 gap has been identified to track the implementation of Phase 2 EME such that it can be used to support the EMS and RMS.
B.12	CSA N290.4-11, "Requirements for Reactor Control Systems of Nuclear Power Plants"	There is one PSR2 CSA N290.4-11 gap which relates to Safety Factor 1 (Plant Design): 1. Clause 4.2 and Clause 5.19 of CSA N290.4-11 require the capability of the Reactor Regulating System (RRS) to be assessed to deal with AOOs, by preventing them from escalating into Design Basis Accidents (DBAs) that would require Shutdown System action. In general, the setback function (and stepback in Pickering Units 5-8) addresses this requirement; however, AOOs have not been identified and analyzed in the current Pickering Safety Reports. Therefore, this has been identified as a PSR2 gap. It is being addressed as part of REGDOC-2.4.1 implementation. Note: There are also additional clauses which refer to requirements of RRS during AOOs (Clauses 5.6.2, 5.19, 5.16.1); however, for convenience, all issues related to AOO requirements for RRS in N290.4-11 are captured under this one PSR2 gap.
B.13	CSA N290.5-06 (R2011) including Update No. 1, "Requirements for Electrical Power and Instrument Air Systems of CANDU Nuclear Power Plants"	There is one PSR2 CSA N290.5-06 (R2011) including Update No. 1 gap which relates to Safety Factor 1 (Plant Design): 1. A gap exists for the Pickering Units 1,4 and 5-8 Instrument Air and Electrical Systems on Clauses 7.1 and 7.4.2 of N290.5-06 (R2011) including Update No. 1 dealing with requirements for AOOs. These clauses introduce the requirement for components to be qualified to perform their required functions during normal operation and AOOs. Only the portion of this clause on AOOs is pertinent to nuclear safety. It is likely that AOOs, due to their nature, do not result in a challenge to the qualification of systems, including Instrument Air and Electrical

Appendix Section #	L/R/C/S Reviewed	L/R/C/S Review Results
		systems. However, AOOs have not been identified and analyzed in the current Pickering Safety Reports. This issue has therefore been identified as a PSR2 gap. It is being addressed as part of REGDOC-2.4.1 implementation.
B.14	CSA N290.6-09, "Requirements for Monitoring and Display of Nuclear Power Plant Safety Functions in the Event of an Accident"	There are no PSR2 gaps for CSA N290.6-09. Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with CSA N290.6-09.
B.15	CSA N290.11-13, "Requirements for Reactor Heat Removal Capability During Outage of Nuclear Power Plants"	There are three PSR2 CSA N290.11-13 gaps which relate to Safety Factor 1 (Plant Design): 1. The CSA N290.11-13 Clause 5.1.1 to 5.1.5 requirement for back-up heat sinks to mitigate the conditions following an AOO is not specified in governance/procedures. Loss of a division of power, a single component failure, etc., which are likely to be in the set of AOOs, are accounted for in the specification of heat sinks. However, AOOs have not been identified and analyzed in the current Pickering Safety Reports. This issue is therefore a PSR2 gap. It is being addressed as part of REGDOC-2.4.1 implementation. 2. Clause 5.6.1 of CSA N290.11-13 requires design reliability to be established for outage heat sinks. Although some emergency heat sinks (e.g., Emergency Boiler Water Supply and Emergency Water Supply) have design reliability requirements, design reliability requirements have not been established for all normal and back-up heat sinks used at Pickering. Reliability of all outage heat sinks (including those without explicit targets) is managed under the Risk & Reliability Program (both through unavailability models as well as through Probabilistic Safety Assessment), hence reactor safety impact is assessed and monitored. However, there is a PSR2 gap with respect to establishment of design reliability requirements for Pickering Units 1,4 and 5-8 outage heat sinks. 3. Clause 5.6.1 of CSA N290.11-13 requires that the designed reliability for process heat sinks be consistent with AOO frequency limits, such that an emergency heat sink does not need to be used for an AOO. AOOs have not been identified and analyzed in the current Pickering Safety Reports. This issue is therefore a PSR2 gap and is being addressed as part of REGDOC-2.4.1 implementation.

Appendix Section #	L/R/C/S Reviewed	L/R/C/S Review Results
B.16	CSA N290.14-15, "Qualification of Digital Hardware and Software for Use in Instrumentation and Control Applications for Nuclear Power Plants"	There is one PSR2 CSA N290.14-15 gap relating to Safety Factor 1 (Plant Design): 1. Correspondence with the CNSC identifies all of the software application qualifications for software Categories 1, 2 and 3 from January 1, 2007 to the time of the correspondence (June 2016). However, an evaluation of legacy Real-Time Process Computing applications with respect to the requirements of N290.14-15 for Categories 1, 2 and 3 software has not been performed. Therefore, this has been identified as a PSR2 gap.
B.17	CSA N291-15, "Requirements for Safety-related Structures for Nuclear Power Plants"	There are three PSR2 CSA N291-15 gaps which relate to Safety Factor 1 (Plant Design): 1. Clause 6.5.2.2 of CSA N291-15 imposes new requirements for bolted connections in members that are part of the seismic load resisting system. Pickering NGS structures were not explicitly designed to meet these requirements and this is therefore identified as a PSR2 gap. 2. Clause 9 of CSA N291-15 contains new requirements related to aging management (including design provisions to account for aging) that are not in CSA N291-08 and that may have significance for operation of Pickering beyond 2020. Pickering structures were not explicitly designed to meet these requirements and this is therefore identified as a PSR2 gap. 3. Clauses 6.1.1(b) and 6.9.2.1.4 of CSA N291-15 state requirements for aspects of the design that are specifically based on the plant service life. Pickering structures were not explicitly designed or assessed in relation to the requirements of these clauses for operation beyond 2020. This is identified as a PSR2 gap. There is also one PSR2 gap for CSA N291 related to submission of Periodic Inspection Plans and Life Cycle Management Plans for a number of safety-significant civil structures to address fitness for service "to end of mission time" (which will need to be extended for Pickering operation beyond 2020). The gap is related to Safety Factor 4 (Aging). This issue is identified as a PSR2 gap in OPG Report P-REP-03680-00024 R000, "Pickering 5-8 Continued Operations Plan Review in Support of PNGS Periodic Safety Review 2 (PSR2)". Therefore, a duplicate gap has not been created under CSA N291-15.
B.18	CSA N285.6 Series-12, "Material Standards for Reactor Components for CANDU Nuclear Power Plants"	There are no PSR2 gaps for CSA N285.6 Series-12. Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with CSA N285.6 Series-12.

Appendix Section #	L/R/C/S Reviewed	L/R/C/S Review Results
B.19	ASME B31.1-14, "Power Piping"	There are no PSR2 gaps for ASME B31.1-14. Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with ASME B31.1-14.
B.20	ASME BPVC (2015), "Boiler and Pressure Vessel Code"	There are no PSR2 gaps for ASME BPVC (2015). Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with ASME BPVC (2015).
B.21	CSA B51-14 (including Update No. 1), "Boiler, Pressure Vessel, and Pressure Piping Code"	There are no PSR2 gaps for CSA B51-14 (including Update No. 1). Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with CSA B51-14 (including Update No. 1).
B.22	NFPA 20 (2016), "Standard for the Installation of Stationary Pumps for Fire Protection"	There are no PSR2 gaps for NFPA 20 (2016). Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with NFPA 20 (2016).
B.23	NFPA 24 (2016), "Standard for the Installation of Private Fire Service Mains and Their Appurtenances"	There are two PSR2 gaps for NFPA 24 (2016) which relate to Safety Factor 1 (Plant Design): 1. For OPG Report NK30-REP-71400-10001 R001, "Fire Protection Code Compliance Review Pickering Nuclear Generating Station B", there is an outstanding issue (Deviation # 13301) which relates to NFPA 24 1970 Section 3601: "Yard post indicator valves at PNGS B are not secured in the open position as required by code" (and which applies to Pickering Units 1,4 as well as Units 5-8). Work to resolve this deviation is currently in progress with locks installed on the majority of the affected valves. Based on OPG List P-LIST-71400-00001 R000, there are a number of SSCs in the yard which directly support plant operation and which are defined as being "related to nuclear safety". As a result, fire water supply to these SSCs is a credited safety function. Deviation # 13301 is not yet complete. Therefore, this has been identified as a PSR2 gap. 2. For Pickering Units 5-8 the baseline for NFPA 24 compliance is the 1970 version of the standard. Pickering Units 1,4 have not been previously assessed against NFPA 24. Although recent changes to the 2013 and 2016 versions of NFPA 24 will be addressed in any firewater system design changes going forward (as a result of Code-over-Code reviews performed for NFPA 24), compliance has not been formally documented for Pickering Units 1,4 or Units 5-8 against the most recent versions of NFPA 24. Furthermore, there have been a large number of significant changes to NFPA 24 since 1970, including the 2002 edition which "represented a complete revision of NFPA 24". Since Pickering NGS has not demonstrated compliance with the 2016 version of NFPA 24, this has been

Appendix Section #	L/R/C/S Reviewed	L/R/C/S Review Results
		identified as a PSR2 gap. It is noted that OPG is proactively replacing portions of the firewater piping in accordance with NFPA 24, under the Pickering A Firewater Pipe Replacement Project 13-80069.
B.24	CNSC REGDOC-2.5.2 (2014), "Design of Reactor Facilities: Nuclear Power Plants"	There are ten PSR2 gaps for CNSC REGDOC-2.5.2 (2014). The gaps and their associated Safety Factor are identified below: 1. Safety Factor 5 (Deterministic Safety Analysis) – Clauses 4.2.1, 6.4 and 7.3 of REGDOC-2.5.2 introduce new requirements and limits for AOOs, DBAs and BDBAs and include specific dose limits for AOOs and DBAs. Current Pickering Safety Report analyses do not identify and classify events into these categories. Dose limits currently used in Pickering are aligned with the single failure / dual failure limits in accordance with the Pickering Licence Conditions Handbook. This issue has therefore been identified as a PSR2 gap. It is being addressed as part of REGDOC-2.4.1 implementation. 2. Safety Factor 6 (Probabilistic Safety Assessment) – Clause 4.2.2 of REGDOC-2.5.2 introduces new requirements and limits for probabilistic analysis risk limits, such as a core damage frequency limit of $<10^{-5}$ yrs/yr. It has not been demonstrated that these requirements can be achieved. Therefore, this has been identified as a PSR2 gap. 3. Safety Factor 1 (Plant Design) – Containment Leak Tightness for Design Extension Conditions (DECs): Clauses 7.3 and 8.6.12 of REGDOC-2.5.2 require containment to provide a leak tight barrier following DECs with severe core damage for a period sufficient to implement off-site emergency measures. REGDOC-2.5.2 guidance suggests this period be at least 24 hours. Such a requirement does not exist in BDBA/Severe Accident (SA) mitigation, so this represents a PSR2 gap. 4. Safety Factor 1 (Plant Design) – On-Demand Reliability of Safety Systems: Clause 7.6 of REGDOC-2.5.2 requires all SSCs important to safety (SIS) to meet an on-demand failure rate of $<10^{-3}$ yrs/yr. This requirement is not met for several systems including Pickering 1,4 ECI and is therefore identified as a PSR2 gap. 5. Safety Factor 1 (Plant Design) – Sharing of Safety Systems and Turbine Hall: Clause 7.6.5 of REGDOC-2.5.2 has a new requirement that sharing of safety systems and the turbine generator building not be permitted. Pickering Units share Emergency Coolant Injection (ECI) and Negative Pressure

Appendix Section #	L/R/C/S Reviewed	L/R/C/S Review Results
		Containment (NPC), as well as the turbine hall; therefore, this has been identified as a PSR2 gap. 6. Safety Factor 1 (Plant Design) – Allowable Times for Crediting On-Site Operator Actions: Clauses 7.10 and 8.10.4 of REGDOC-2.5.2 establish new time limits for crediting operator actions, i.e., 30 minutes for MCR actions and 1 hour for field actions. Pickering NGS has not demonstrated that deterministic safety analysis consequences are acceptable if MCR and field action are not credited for these times respectively. Therefore, this has been identified as a PSR2 gap. 7. Safety Factor 1 (Plant Design) – Seismic Qualification and Design: Clause 7.13.1 of REGDOC-2.5.2 requires that Beyond Design Basis (BDB) Earthquake seismic margin be a factor of 1.67 beyond that required for the new plant Design Basis Earthquake (DBE). Fragility evaluations were completed for seismic mitigating SSCs, however, based on available information it could not be confirmed that the new plant BDB Earthquake margin of 1.67 would be achieved. Therefore, this has been identified as a PSR2 gap. 8. Safety Factor 1 (Plant Design) – Human Factors in Design: Clauses 7.21 and 8.10.1 of REGDOC-2.5.2 introduce new requirements for the systematic application of HFE principles to plant design. Many years of safe and reliable operating experience indicate that the design and processes for integration of human interactions with the plant were and remain robust. However, Pickering plant design predates the current requirements for incorporating HFE into the design and the existing plant has not been systematically demonstrated to meet the requirements for a new plant. Therefore, this has been identified as a PSR2 gap. 9. Safety Factor 1 (Plant Design) – Detection/Isolation of ECI Heat Exchanger Tube Leak: Clause 8.5 of REGDOC-2.5.2 requires ECI recovery heat exchanger tube leak detection capability. Pickering Units 5-8 ECI recovery heat exchangers do not have leak detection capability on the cooling water side. Therefore, this has been identified as a PSR2 gap. 10. Safety Factor 1 (Plant Design) – Safety Parameter Display System Qualification for DECs: Clause 8.10.1.1 of REGDOC-2.5.2 requires the MCR to contain a Safety Parameter Display System (SPDS) that presents sufficient information on safety-critical parameters for the diagnosis and mitigation of DBAs and DECs. The SPDSs are to be qualified for DEC and have parameters available in both the MCR and Secondary Control Areas (SCA), per Clause 8.10.2. Pickering SPDSs are not Review Level Condition (RLC) qualified or available in all

Appendix Section #	L/R/C/S Reviewed	L/R/C/S Review Results
		locations. As part of the Fukushima follow-up, instrumentation to support critical parameters required to function for DECs has been evaluated for survivability. The instrument loops associated with these parameters have been identified for use in Critical Safety Parameter Monitoring (CSPM) and BDBA procedures. However, the indications from these loops are not in one central location and, in some cases, require field action (e.g., power) to obtain data. This does not fully satisfy the requirements to have these parameters available from a SPDS in the MCR and SCA. Therefore, this has been identified as a PSR2 gap relating to the new plant requirement to have SPDS that is DEC qualified and with parameters available in the MCR and SCA.
B.25	CNSC G-144 (2006), “Trip Parameter Acceptance Criteria for the Safety Analysis of CANDU Nuclear Power Plants”	There are no PSR2 gaps for CNSC G-144 (2006). Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with G-144 (2006).
B.26	CNSC G-149 (2000), “Computer Programs Used in Design and Safety Analyses of Nuclear Power Plants and Research Reactors”	There are no PSR2 gaps for CNSC G-149 (2000). Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with G-149 (2000).
B.27	CNSC R-77 (1987), “Overpressure Protection Requirements for Primary Heat Transport Systems in CANDU Power Reactors Fitted with Two Shutdown Systems”	There are no PSR2 gaps for CNSC R-77 (1987). Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with CNSC R-77 (1987).
B.28	CSA N288.2-14, “Guidelines for Calculating Radiological Consequences to the Public from a Release of Airborne Radioactive Material for Nuclear Reactor Accidents”	There is one PSR2 CSA N288.2-14 gap which relates to Safety Factor 5 (Deterministic Safety Analysis): 1. Safety Report upgrades currently underway for Pickering as part of REGDOC-2.4.1 implementation for the period of 2017-2021 will utilize methods consistent with N288.2-14. The REGDOC-2.4.1 Implementation Plan update will consider the incremental implications of Pickering operation beyond 2020, including any considerations of N288.2 revisions. This issue has therefore been identified as a PSR2 gap. It is being addressed as part of REGDOC-2.4.1 implementation.

Appendix Section #	L/R/C/S Reviewed	L/R/C/S Review Results
B.29	CSA N290.7-14, "Cyber-Security for Nuclear Power Plants and Small Reactor Facilities"	The gap analysis, N-REP-69000-10003 R000, "Gap Analysis Between CSA N290.7-14 Cyber Security Requirements for Nuclear Power Plants and Small Reactor Facilities" and implementation plan for N290.7-14 was accepted by the CNSC. For reasons of security and confidentiality, the findings of the gap analysis for N290.7-14 will not be discussed in PSR2.
B.30	NBCC (2010), "National Building Code of Canada"	There are no PSR2 gaps for NBCC (2010). Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with NBCC (2010).
B.31	NFCC (2010), "National Fire Code of Canada"	There is one PSR2 gap for NFCC (2010), related to piping for flammable or combustible liquids at building entrances. The gap is related to Safety Factor 1 (Plant Design). This issue is identified as a PSR2 gap in OPG Report P-REP-03680-00024 R000, "Pickering 5-8 Continued Operations Plan Review in Support of PNGS Periodic Safety Review 2". Therefore, a duplicate gap under NFCC (2010) has not been created.
B.32	CSA N290.8-15, "Technical Specification Requirements for Nuclear Power Plant Components"	There is one PSR2 CSA N290.8-15 gap which relates to Safety Factor 1 (Plant Design): 1. Clause 4.7 of CSA N290.8-15 mandates that the technical specification requires the supplier to identify and describe all digital items included in their equipment. In the event that the use of digital items is identified by OPG in advance of issuing a Request for Proposal (RFP) or Request for Quotation (RFQ), existing OPG procedures are adequate for ensuring that requirements related to digital items are documented in the technical specification. However, a requirement for a supplier to self-identify whether their product contains any digital items is not reflected in OPG governing documents. This has therefore been identified as a PSR2 gap.

4.0 REFERENCES

[1] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[2] CNSC REGDOC-2.3.3, *Periodic Safety Reviews*, April 2015.

[3] IAEA Safety Guide No. SSG-25, *Periodic Safety Review for Nuclear Power Plants*, 2013.

[4] OPG Report, P-REP-03680-00004 R000, *Pickering NGS Periodic Safety Review 2: Code and Standard Reviews for Safety Factors 2 (Actual Condition of SSCs), 3 (Equipment Qualification) and 4 (Aging)*, July 2016.

[5] OPG Report, P-REP-03680-0586480 R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 9, 11, and 15*, September 2016.

[6] OPG Report, P-REP-03680-00021 R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 8, 10, 12, 13, and 14,* December 2016.

[7] OPG Report, P-REP-03680-00024 R000, *Pickering 5-8 Continued Operations Plan Review in Support of PNGS Periodic Safety Review 2 (PSR2),* January 2017.

[8] OPG Report, N-REP-69000-10003 R000, *Gap Analysis Between CSA N290.7-14 "Cyber Security Requirements for Nuclear Power Plants and Small Reactor Facilities"*, March 2016.

[9] OPG Correspondence, N-CORR-00531-05661, W. M. Elliott to P. A. Webster and M. Santini, *Design Codes and Standards Effective Dates for OPG Nuclear Fleet*, April 30, 2012.

Appendix A: Nomenclature

ACU	Air Cooling Unit
AD	Acceptable Deviation
ADDAM	Atmospheric Dispersion and Dose Analysis Method
AECB	Atomic Energy Control Board
AECL	Atomic Energy of Canada Limited
AFS	Available for Service
AHJ	Authority Having Jurisdiction
AI	Action Item
AIM	Abnormal Incidents Manual
ALARA	As Low As Reasonably Achievable
ANI	Authorized Nuclear Inspector
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
AR	Action Request
ARM	Annunciation Response Manual
ASIC	Application-Specific Integrated Circuit
ASME	American Society of Mechanical Engineers
ASTGMS	Automated Source Term Gamma Monitoring System
ASTM	American Society for Testing of Materials
BATEA	Best Available Technology and Techniques Economically Achievable
BDBA	Beyond Design Basis Accident
BPVC	Boiler Pressure Vessel Code
CANDU	CANada Deuterium Uranium
CAP	Corrective Action Plan
CCF	Common Cause Failure
CCI	Core-Concrete Interaction
CCR	Code Compliance Review
CCS	Concrete Containment Structure
CDF	Complementary Design Feature
CER	Control Equipment Room
CNE	Chief Nuclear Engineer

CNSC	Canadian Nuclear Safety Commission
C of A	Certificate of Authorization
COG	CANDU Owners Group
COMS	Constructability, Operability, Maintainability and Safety
COP	Continued Operations Plan
COTS	Commercial Off The Shelf
CPLD	Complex Programmable Logic Device
CSA	Canadian Standards Association
CSCA	Common Secondary Control Area
CSD	Computer System Design
CSPM	Critical Safety Parameter Monitoring
CSR	Computer System Requirements
CSSP	Critical Safety Support Parameters
DARA	Darlington Risk Assessment
DAS	Display Annunciation Station
DBA	Design Basis Accident
DBE	Design Basis Earthquake
DCR	Document Change Request
DEC	Design Extension Condition
DIF	Dynamic Increase Factor
DN	Darlington Nuclear
DSA	Deterministic Safety Analysis
EBWS	Emergency Boiler Water Supply
ECC	Engineering Change Control
ECCS	Emergency Core Cooling System
ECI	Emergency Coolant Injection
ECIS	Emergency Coolant Injection System
EHRS	Emergency Heat Removal System
EITER	Equipment Important to Emergency Response
EME	Emergency Mitigating Equipment
EMS	Energy Management System
EOC	Emergency Operating Centre
EPC	Engineering, Procurement and Construction

EPG	Emergency Power Generator
EPRC5	Ex-Plant Release Category 5
EPRI	Electrical Power Research Institute
EPS	Emergency Power Supply
EQ	Environmental Qualification
ERT	Emergency Response Team
EST	Edwards System Technology
ETAP	Electrical Transient Analysis Program
EWS	Emergency Water Supply
FADS	Filtered Air Discharge System
FAI	Fukushima Action Item
FDC	Fire Department Connection
FEA	Finite Elements Analysis
FHA	Fire Hazard Assessment
FLORC	Fast Loss of Reactivity Control
FME	Foreign Material Exclusion
FMEA	Failure Mode and Effects Analysis
FPA	Fire Protection Assessment
FPGA	Field Programmable Gate Arrays
FSA	Fire Safety Assessment
FSAR	Final Safety Analysis Report
FSSA	Fire Safety Shutdown Analysis
FTA	Fault Tree Analysis
GAI	Generic Action Item
GSS	Guaranteed Shutdown State
GT	Guide Tube
HA	Hazard Analysis
HAZOP	Hazard and Operability Analysis
HCLPF	High Confidence of Low Probability of Failure
HELBA	High Energy Line Break Assessment
HFE	Human Factors Engineering
HP	High Pressure
HPECI	High Pressure Emergency Coolant Injection

HPSW	High Pressure Service Water
HTS	Heat Transport System
HVAC	Heating, Ventilation and Air Conditioning
HX	Heat Exchanger
IA	Instrument Air
IAEA	International Atomic Energy Agency
I&C	Instrumentation & Control
ICRP	International Commission on Radiological Protection
IEC	International Electrotechnical Commission
IESO	Independent Electricity System Operator
IFB	Irradiated Fuel Bay
IIP	Integrated Implementation Plan
IPRV	Instrumented Pressure Relief Valves
ISO	International Organization for Standardization
ISR	Integrated Safety Review
IST	Industry Standard Toolset
ISTB	Inter-Station Transfer Bus
ITM	Inspection, Testing, and Maintenance
IUC	Instrument Uncertainty Calculation
IVR	In-Vessel Retention
LBLOCA	Large Break Loss of Coolant Accident
LISS	Liquid Injection Shutdown System
LLOCA	Large Loss of Coolant Accident
LCH	Licence Conditions Handbook
LCMP	Life Cycle Management Plan
LOCA	Loss of Coolant Accident
LOE	Limit of Operating Envelope
LOECI	Loss of Emergency Coolant Injection
LP	Low Pressure
LPSW	Low Pressure Service Water
L/R/C/Ss	Laws, Regulations, Codes and Standards
LRT	Leakage Rate Testing
MCR	Main Control Room

MDR	Modification Design Requirements
MG	Motor Generator
MOTS	Modifiable Off The Shelf
MSIV	Main Steam Isolation Valve
NBCC	National Building Code of Canada
NDE	Non-Destructive Examination
NFCC	National Fire Code of Canada
NFPA	National Fire Protection Association
NGS	Nuclear Generating Station
NPCS	Negative Pressure Containment System
NPP	Nuclear Power Plant
NPSH	Net Positive Suction Head
NSCA	Nuclear Safety and Control Act
NSDR	Nuclear Safety Design Requirement
OPEX	Operating Experience
OPG	Ontario Power Generation
OP&Ps	Operating Policies & Principles
OSR	Operational Safety Requirements
PAM	Post Accident Monitoring
PAR	Passive Autocatalytic Recombiner
PARTS	Pickering A Return to Service
PB	Pressure Boundary
PBRA	Pickering B Risk Assessment
PEL	Program Element
PFU	Predicted Future Unavailability
PHT	Primary Heat Transport
PIE	Postulated Initiating Event
PIP	Periodic Inspection Program
PIV	Post Indicator Valve
PLHIS	Post LOCA Hydrogen Ignition System
PRA	Probabilistic Risk Assessment
PRD	Pressure Relief Device
PROL	Power Reactor Operating Licence

PRV	Pressure Regulating Valve
PSA	Probabilistic Safety Assessment
PSR	Periodic Safety Review
PSR1	Periodic Safety Review 1 (Earlier OPG PSR work and other associated assessments)
PSR2	Periodic Safety Review 2 (Subsequent PSR per CNSC REGDOC-2.3.3)
QA	Quality Assurance
RA	Rapid Access
RAB	Reactor Auxiliary Bay
RCS	Reactor Coolant System
RCU	Reactivity Control Unit
RFI	Request for Interpretation
RFP	Request for Proposal
RFQ	Request for Quotation
RLC	Review Level Condition
RMS	Radionuclide Management System
RLE	Review Level Earthquake
RRS	Reactor Regulating System
RTPC	Real-Time Process Computing
SA	Severe Accident
SAI	Safety Analysis Improvement
SAMG	Severe Accident Management Guidance/Guidelines
SBO	Station Blackout
SCA	Secondary Control Area
SDC	Shutdown Cooling
SDE	Site Design Earthquake
SDM	Safety Design Matrix
SDS2	Shutdown System 2
SDSA	Shutdown System A
SDSE	Shutdown System Enhancement
SDSE IR	Shutdown System Enhancement Instrumentation Room
SES	Site Electrical System
SESA	Scientific, Engineering, and Safety Analysis

SIS	Systems Important to Safety
SMA	Seismic Margin Assessment
SMC	Site Management Center
SMP	Software Maintenance Plan
SOE	Safe Operating Envelope
SPDS	Safety Parameter Display System
SRST	Safety Related System Test
SRV	Steam Reject Valve
SSCs	Structures, Systems and Components
SSS	Special Safety System
STPA	Systems Theoretic Process Analysis
SUI	Start-up Instrumentation
TAB	Turbine Auxiliary Bay
TIMS	Training Information Management System
TSSA	Technical Standards & Safety Authority
UECC	Unit Emergency Control Centre
UHRS	Uniform Hazard Response Spectrum
UHS	Uniform Hazard Spectrum
VESDA	Very Early Smoke Detection Apparatus
VSLORC	Very Slow Loss of Reactivity Control
WPS	Welding Procedure Specifications

Appendix B: L/R/C/S Reviews for Safety Factors 1, 5, 6 and 7

B.1 CSA N285.0-12, "General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants"

B.1.1 Background

The following, paraphrased from CSA N285.0-12 (including Updates No. 1 and No 2) [B.1-1], provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *The CSA N285 series of Standards specifies requirements applicable to nuclear power plants in Canada and references the applicable requirements of the ASME Boiler and Pressure Vessel Code (BPVC).*
>
> *CSA N285.0 provides general requirements for pressure-retaining systems, components, and supports in CANDU nuclear power plants. It specifies the technical requirements for the design, procurement, fabrication, installation, modification, repair, replacement, testing, examination, and inspection of, and other work related to, pressure-retaining systems, components, and supports over the service life of a CANDU nuclear power plant.*

All of N285.0-12 (including Updates No. 1 and No 2) is directly relevant to Safety Factor 1 (Plant Design).

Compliance with CSA N285.0-08 (including Updates No. 1 and No. 2) [B.1-2] is currently a licence requirement for Pickering NGS (per PROL 48.02/2018) as indicated in Section 6.2 and Appendix C.1 of the R04 Pickering Licence Condition Handbook (LCH) [B.1-3].

N285.0-12/N285.6 Series-12 [B.1-1] is the second edition of the CSA N285.0/N285.6 Series. It supersedes the previous edition published in 2008, and the previous editions of CSA N285.0 published in 2006, 1995, 1991 and 1971.

There are two applicable CSA N285.0 Impact Statements: (i) The CSA Impact Statement for the new edition of CSA N285.0-12 [B.1-4] and (ii) CSA Impact Statement for Publication of CSA N285.0-12 (including Updates No. 1 and No. 2) [B.1-5]. They provide a summary of significant changes from the previous editions and are discussed in Section B.1.2 below.

The results of PSR1 CSA N285.0 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.1.2. As identified in Reference [B.1-6], the Pickering PSR2 review of CSA N285.0-12 is an Incremental Review. PSR2 Incremental Review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.

- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.1.2 Compliance Assessment for Pickering PSR2

B.1.2.1 Application of PSR1 Reviews

The versions of N285.0 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

A clause-by-clause review was performed against N285.0-06 [B.1-7] as part of the Pickering B ISR. The review was documented in the Plant Design Safety Factor Report, NK30-REP-03680-00001 R000 [B.1-8]. The report found that Pickering Units 5-8 complied with N285.0-06, except for a discrepancy against Clause A.1.5.

Clause A.1.5 states:

> *Class 3: Systems and sections of systems not classified as Class 1 or 2 and that contain radioactive substances with a tritium concentration exceeding 74 GBq/kg (2 Ci/kg) shall be classified as Class 3.*

whereas CSA N285.0-95 [B.1-9] stipulated a requirement of "tritium concentration exceeding 0.4 TBq/kg (0.01 Ci/g)".

The code review [B.1-8] further states it is improbable that this change would alter the classification of Class 3 & 6 systems.

Subsequent correspondence with the CNSC, contained in NK30-REP-03680-00016 [B.1-10], resulted in this issue not being a gap since the likelihood is low that a system containing radioactive material, initially classified as Class 6, would be impacted as a result of this classification. Therefore this is not a PSR2 gap.

Subsequent to the N285.0-06 code review in [B.1-8], NK30-REP-03680-00016 [B.1-10] documented that a gap exists against Clause A.2.3.1. The issue associated with this clause was initially an Acceptable Deviation, but subsequently identified as a gap per a CNSC request. This clause identifies that for Shutdown Systems, pressure-retaining portions shall be classified as Class 1, except for three listed exceptions. Gap 1-561 identified that some Liquid Injection Shutdown System (LISS) components, which should have been Class 1, were purchased and installed as Class 3.

To resolve the gap, a screening level review of the Shutdown System 2 (SDS2) flow diagram and System Classification List was performed. There is a limited portion of LISS that is Class 3. To meet the intent of the requirement of Clause A.2.3.1(c), the following four actions were proposed [B.1-10]:

- Update Design Documentation
- Prepare a new Design Requirement(s) for the affected Class 3 portion of LISS
- Perform Stress Analysis
- Reconciliation of materials, fabrication and installation of the affected portions with the Technical Standards & Safety Authority (TSSA).

When refurbishment was not being pursued, Section B.3.8 of the Pickering B Integrated Implementation Plan, NK30-PLAN-03680-00002 [B.1-11] indicated that the LISS code classification concession that was previously accepted by the CNSC in NK30-CORR-00531-03047 [B.1-12] was considered acceptable for Continued Operations. Closure of Gap 1-561 was requested in OPG correspondence NK30-CORR-00531-06027 [B.1-13], and accepted in CNSC letter NK30-CORR-00531-06324 [B.1-14]. This code classification concession and the four actions identified in the Pickering B ISR gap resolution need to be reconsidered in the context of operation of Pickering NGS beyond 2020. Therefore, this is identified as a PSR2 gap (**PSR2 CSA N285.0-12 Gap #1**).

Reference [B.1-8] also identified the following twelve Acceptable Deviations (ADs):

- Clause 5.2.8 identifies requirements for radioactivity concentrations used for system classification purposes. The previous code version did not contain this clause. Instead it refers to the Safety Report to define process failures, which would be used for classification. It was deemed improbable that this clause would affect classification. Therefore this not a safety significant issue and is not a PSR2 gap.
- Clauses 6.1.1.3 and 6.1.4 require the designer to register items as systems, vessels, pumps, fittings, or supports. The previous version of the code excludes supports, so they were not subject to registration. The lack of registration does not impact on the ability of the system or component to perform their required function. Therefore, this is not a PSR2 gap.
- Clause 7.6.2.4 on retention of objective evidence that intervening elements are capable of withstanding the loads imposed, is a new requirement. Manufacturers are required to supply information on support qualification to ensure the support function is taken into account in the design of the supported component. This satisfies the clause and therefore this AD is of low safety significance and is not a PSR2 gap.
- Clause 7.7.2.3 identifies considerations for the qualification of the Heat Transport System for overpressure events. In the updated version of the standard, N285.0-12 [B.1-1], the equivalent Clause 7.6.2.3 (c) has been modified to provide relaxation

consistent with Regulatory Document R-77 [B.1-15], so there is no longer a gap associated with this clause.

- Clause 12.1 identifies requirements for pressure-retaining systems documentation. Records are available for most of the important permanent design records, e.g. Class 1 stress report, but not all records are available. The completed configuration management restoration project established the documentation that is required to safely operate and maintain the station. Therefore, this AD is of low safety significance and is also not a PSR2 gap.

- Clauses 12.3.9, 12.4.7.2 and A.1.6.3.3 are very minor issues and their disposition was accepted by the CNSC in Reference [B.1-16]. They are not PSR2 gaps.

- Clause 14.5.3.1 requires documentation to be produced to demonstrate that any differences between registered designs and the as-built modification have been reviewed and reconciled. The previous code version did not include this specific requirement, therefore this process was performed by the designer informally. Reference [B.1-10] stated there was a program to clear the backlog of reconciliation statements and this was committed to the CNSC and has since been closed [B.1-17]. Related ADs against Clauses 14.5.3 and 14.5.3.2, were addressed with the same disposition as for Clause 14.5.3.1. Therefore these three ADs are not PSR2 gaps.

These ADs are not impacted by Pickering operation beyond 2020.

<u>Pickering Units 1,4</u>

For Pickering A Return to Service, the active version of the Standard at the time was CSA-N285.0-95 [B.1-9]. A review of OPG pressure boundary components was performed against N285.0-95 in an OPG letter to the CNSC, NA44-CORR-00531-00464, "Pickering A – Review of Pickering A Design to CSA-N285.0-95" [B.1-18]. A section by section review of the standard was carried out. The review concluded that the design of Pickering A meets the intent of CSA N285.0-95. The following three ADs were identified:

1. Clause 7.0 on Design Rules Intent states:

   The design will be performed in accordance with system requirement:

   - *Class 1 (ASME class 1-NB-3000)*
   - *Class 2 (ASME class 2-NC-3000)*
   - *Class 3 (ASME class 3-ND-3000)*

   For over pressure protection, systems shall comply with the requirements of the applicable rules of ASME Sections NB/NC/ND-7000.

   The disposition of this AD states:

Equipment was designed to Class A which is class 1 today and fatigue was addressed. Piping level A, B, C and D service loading were not identified as required by NCA-2142 but since the original design included 7000 cycles for fatigue, it is considered that the previous design would bound the current service limits.

For Pickering NGS operation beyond 2020, confirmation is required that the allowable cycles for fatigue will continue to bound current service limits for extended operation. This is identified as **PSR2 CSA N285.0-12 Gap #2.**

2. Another AD was identified against Clause 7.0 with the following disposition:

The original design met the requirements of Class A, B, C or B31.1. The major difference between class 1, 2, 3 and B31.1 is that class 1 piping requires fatigue analysis. For class 2, 3 and B31.1 piping fatigue was accounted for by providing a limit on the number of total cycles (7000). Pickering 'A' piping design for class 2 and 3 systems is expected to be compliant. Also, class 1 piping less than or equal to 1" in size is considered to be compliant. For class 1 piping greater than 1", the Pickering 'A' design is considered to be an acceptable deviation with current code. Since B31.1 conservatively considers fatigue by incorporating 7000 cycles for piping fatigue, it is expected that fatigue analysis would be bounded by the original design. In addition, lower allowable stresses are used by the B31.1 code, provide additional margin to the original design.

For Pickering NGS operation beyond 2020, confirmation is required that the allowable cycles for fatigue for Class 1 piping greater than 1" will continue to bound current service limits for extended operation. This specific AD relating to Class 1 greater than 1" piping is being included in **PSR2 CSA N285.0-12 Gap #2** identified above.

3. An AD was identified against Clause 14.0 which specifies the requirements for supports of pressure retaining systems. The disposition of this AD states:

In general supports were not registered as an integral part of the pressure vessel. The Pickering 'A' support design is considered to be acceptable with current code requirement.

Piping support components were selected from the Piping Standards. Engineering review has concluded that there are relatively minor variations between the present and original standards.

This is not safety significant in the context of Pickering PSR2.

Darlington NGS

The Darlington ISR review of N285.0 was initially performed using version N285.0-08 (June 2008) [B.1-19], which was documented in OPG Report NK38-REP-03680-10030 R000 [B.1-20]. The report documents a review of the compliance of eleven Darlington NGS pressure-retaining systems and concludes:

The review did not identify any ISR Gaps and found that Darlington NGS was compliant with CAN/CSA-N285.0-08.

The report did not identify any Acceptable Deviations.

Subsequent to the issue of Reference [B.1-20], an addendum was prepared to document CNSC comments on the code review. This is documented in Reference [B.1-21], which identified a new gap against Clause 7.6.2.2. The gap is a result of the current Darlington NF jurisdictional boundary not meeting the requirements of ASME BPVC, Section III, Division 1, NF requirements.

This gap was specific to the Darlington design. Further, since CSA N285.0-08 (including Updates No. 1 and No. 2) [B.1-2] is a licence requirement for Pickering NGS per Appendix C.1 of the R04 Pickering Licence Conditions Handbook [B.1-3], and compliance with N285.0-08 (including Update No. 1) [B.1-22] is confirmed in the Pickering NGS PROL Renewal Application [B.1-23], Pickering NGS is in compliance with N285.0-08 (including Update No. 1) and therefore this Darlington gap is not applicable to PSR2. The R04 Pickering Licence Conditions Handbook [B.1-3] includes the transitional provisions to N285.0-08 (including Updates No. 1 and No. 2) [B.1-2], which are discussed in B.1.2.2 below.

Following the issuance of CSA N285.0-12 (including Update No. 1) [B.1-24] in September 2013, a code refresh review was conducted and documented in OPG Report NK38-REP-03680-10136 R000 [B.1-25]. This review also assessed the same eleven Darlington systems for compliance. NK38-REP-03680-10136 R000 [B.1-25] identified that:

The eleven Darlington NGS pressure-retaining systems listed in Table 2 are in compliance with the requirements of CSA-N285.0-12 (August 2012) and Update 1 (September 2013) with the exception that the current Pressure Boundary Program Manual, N-MAN-01913.11-10000 R014 and the Pressure Boundary Program, N-PROG-MP-0004 R014 do not conform to Annex N of CSA N285.0-12 and Update No. 1, as required by Section 6.1.2 of the Darlington application for license renewal. This review also identified a need to reconcile the code effective dates of the current Pressure Boundary Program Manual and the supporting governing documents to align with CSA N285.0-12 and Update 1, upon licence renewal.

The gap is against Clause 15, which states [B.1-24]:

The licensee shall have pressure boundary program documents that indicates how the requirements of this Standard are addressed by the licensee's processes and procedures for a nuclear facility. The pressure boundary program document shall be submitted to the regulatory authority. Annex N outlines a suggested format for this document.

The gap was a result of N-MAN-01913.11-10000 [B.1-26], "Pressure Boundary Program Manual" and N-PROG-MP-0004 [B.1-27], "Pressure Boundary" not conforming to Annex N of N285.0-12 (including Update No. 1). The code refresh review [B.1-25] identified this as a requirement in Section 6.1.2 of the Darlington application for license renewal. However, the current Darlington LCH [B.1-28] only requires a "Pressure Boundary Program Document roadmap in compliance with Annex N of CSA N285.0-12 and Update No. 1" to satisfy Clause 15 of N285.0-12 (including Update No. 1). This requirement was addressed with the submission of N-LIST-00531-10003, "Index to OPG Pressure Boundary Program Elements" to the CNSC in April 2015 [B.1-29]:

> *N-LIST-00531-10003…captures the elements of Annex N, "Pressure Boundary Program Document" of CSA N285.0-12 and Update No. 1, as proposed for adoption by OPG and accepted by the CNSC. The N-LIST correlates OPG's processes and procedures to the pressure boundary program elements identified in Annex N, Table N.1.*

N-LIST-00531-10003 [B.1-30] is a programmatic document and is also listed as a relevant governing document in Section 6.2 of the R04 Pickering LCH [B.1-3]. Therefore, the gap identified in the Darlington ISR against Clause 12 of N285.0-12 (including Update No. 1) has been addressed programmatically and is not a gap for PSR2.

No Acceptable Deviations were identified.

The changes made to N285.0 since these code reviews were performed are discussed in the next section.

B.1.2.2 Application of Post PSR1 Reviews

CSA N285.0-08 (including Updates No. 1 and No. 2) [B.1-2] is a licence requirement for Pickering NGS per Appendix C.1 of the R04 Pickering Licence Conditions Handbook [B.1-3], and compliance with N285.0-08 (including Update No. 1) [B.1-22] is confirmed in the Pickering PROL Renewal Application [B.1-23]. The Licence Conditions Handbook [B.1-3] also includes the transitional provisions to N285.0-08 (including Updates No. 1 and No. 2) [B.1-2]. It states:

> *OPG governance (pressure boundary procedures and manual) will be compliant with CSA N285.0-08 with Update #2 before October 30, 2013.* (It also provides conditions for various deliverables, e.g. design modifications and purchase orders).

Gap analyses were performed to assess the changes required to governance documents as a result of the transition to CSA N285.0-08 (including Updates No. 1 and No. 2) [B.1-2]. Reference [B.1-31] documents the changes applicable to N-PROG-MP-0004 [B.1-27] and the associated reference procedures due to the transition. All of the assignments against the Action Request (AR) referenced in this memorandum are now complete (see AR #28154550, "Transition Project – Update Governance to Align with License Requirements").

Given the above, only the changes made to N285.0 since the 2008 (including Updates No. 1 and No. 2) [B.1-2] version need to be reviewed. These changes are reviewed along with an assessment of their impact on PSR2.

The CSA Impact Statement notification for CSA N285.0-12 [B.1-4] identifies the following changes from the previous edition (N285.0-08 (including Updates No. 1 and No. 2) [B.1-2]):

1. *Clarifies the appropriate rules to be used for process systems (such as classification) and the rules to be used for piping systems (such as the ASME rules)*

2. *Revised the term "Contractor" to "Certificate Holder", thereby clarifying that the organizations performing pressure boundary work must hold an appropriate Certificate of Authorization for the work in accordance with Clause 10 (General Requirements for Quality Assurance).*

3. *Clause 5: The rules for classification were revised to clarify the requirements and provide the information in a more logical sequence.*

4. *Clause 7: The rules for Design were revised to address the following:*

    a. *Clarify the requirements and provide the information in a more logical sequence.*

    b. *Migration of the requirement of the CNSC R-77 document into the CSA standard.*

5. *Clause 10: Significant re-write and revision of the clause to address RFI's received by the Technical Committee and to clarify and simplify the specific QA requirements for pressure boundary activities.*

6. *Annex A – Classification: This change was made to address the following:*

    a. *Clarification of the classification requirements for the Emergency Core Cooling System.*

    b. *Provide rules for determining classification of Class 3 vs. Class 6 systems based on a consequence of failure analysis.*

7. *Annex E – Implementation of Quality Assurance Programs: The Annex was deleted.*

The impact of the changes to N285.0-12 [B.1-32], were reviewed in an OPG Memorandum, N-CORR-00590-0455367, "Code Over Code Review of: CSA N285.0-12 over CSA N285.0-08 and Update #2" [B.1-33]. This memorandum concluded that there were no significant changes with respect to pressure boundary integrity. Therefore, these changes have no impact on PSR2.

A CSA Impact Statement could not be found for changes made in the N285.0-12 (including Update No. 1) [B.1-24] version in comparison to N285.0-12 [B.1-32]. However, these changes are documented in N-REP-00590-0520105, "Code over Code Report: CSA N285.0/CSA N285.6 Series For Year 2014" [B.1-34]. The report documents that except for one change, the changes are not significant. Clause 15 introduces a new requirement for the licensee to have a pressure boundary program document per the requirements in Annex N and submit it to the regulator. This was previously identified in the Darlington PSR1 discussion in Section B.1.2.1, which documents that this requirement was satisfied programmatically, and thus is not a PSR2 gap.

Changes made in N285.0-12 (including Update No. 1 and No. 2) [B.1-1] relative to N285.0-12 (including Update No. 1) [B.1-24] are paraphrased from the related Impact Statement for Publication [B.1-5] as follows:

1. *ASME OM code requirements (Clause 13.3) were clarified to reflect the graded approach adopted by the Technical Committee in accordance with the SAME OM Code and only requires those overpressure protection devices within the scope of the ASME OM Code to be addressed.*

2. *Changes related to nuclear supports in order to clarify and align with ASME III requirements for inspection during fabrication of nuclear class supports. ASME support manufacturers can fabricate ASME certified nuclear class supports without inspection by an Authorized Nuclear Inspector (ANI), in accordance with NF-8100.*

The impact of the changes was reviewed in OPG Report N-REP-00590-0526911, "Code Over Code Review Report: CSA N285.0 For Year 2015" [B.1-35]. This report concluded that there were no significant technical changes and therefore there is no impact on PSR2.

B.1.3 Compliance Summary for Pickering PSR2

There are two PSR2 CSA N285.0-12 (including Updates No. 1 and No. 2) [B.1-1] gaps which relate to Safety Factor 1 (Plant Design):

1. Clause A.2.3.1 of CSA N285.0-06 identifies that for Shutdown Systems, pressure-retaining portions shall be classified as Class 1, except for three listed exceptions. It was identified during the Pickering B Integrated Safety Review (ISR) that a limited number of Liquid Injection Shutdown System (LISS) components, which should have been Class 1, were purchased and installed as Class 3. In follow-up, OPG proposed four actions to address the deficiency. When refurbishment was not pursued, a code classification concession was accepted for continued operations. This code classification concession and the four actions identified in the Pickering B ISR gap resolution need to be reconsidered in the context of operation of Pickering NGS beyond 2020. Therefore, this has been identified as a PSR2 gap.

2. The Pickering A Return to Service review against CSA-N285.0-95 [B.1-9] identified two Acceptable Deviations relating to Clause 7.0 requiring confirmation that the allowable cycles for fatigue would not be exceeded. For Pickering Units 1,4 and Units 5-8 operation beyond 2020, further confirmation is required that the allowable cycles for fatigue will continue to bound current service limits for extended operation. Therefore, this has been identified as a PSR2 gap.

B.1.4 References

[B.1-1] CSA Standard, N285.0-12/N285.6 Series-12 including Updates No. 1 and No. 2, *General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants/Material Standards for Reactor Components for CANDU Nuclear Power Plants, 2012*; Update No. 1: September 2013; Update No. 2: November 2014.

[B.1-2] CSA Standard, N285.0-08 including Updates No. 1 and No.2, *General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants/Material Standards for Reactor Components for CANDU Nuclear Power Plants*, June 2008; Update No. 1: June 2009; Update No. 2: August 2010.

[B.1-3] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.1-4] CSA Impact Statement, Notification of CSA N285.0-12 *General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants/Material Standards for Reactor Components for CANDU Nuclear Power Plants*, Date not provided.

[B.1-5] CSA Impact Statement for Publication, *Notification of CSA N285.0-12 Amendment No. 2, General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants/Material Standards for Reactor Components for CANDU Nuclear Power Plants*, Date not provided.

[B.1-6] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.1-7] CSA Standard, N285.0-06, *General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants*, 2006.

[B.1-8] OPG Report, NK30-REP-03680-00001 R000, *Pickering NGS-B Integrated Safety Review – Plant Design Safety Factor*, August 2007.

[B.1-9] CSA Standard, N285.0-95, *General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants*, 1995.

[B.1-10] OPG Report, NK30-REP-03680-00016 R000, *OPG Response to CNSC Comments On Pickering NGS-B Integrated Safety Review – Plant Design, Safety Analysis, Safety Performance, Ageing and Equipment Qualification Safety Factors and Discrepancy Resolutions*, September 2009.

[B.1-11] OPG Plan, NK30-PLAN-03680-00002 R000, *Pickering B – Integrated Implementation Plan,* December 2011.

[B.1-12] CNSC Letter, # 26-1-8-3-0, OPG File No. NK30-CORR-00531-03047, T.E. Schaubel to T.N. Mitchell, *Pickering NGS-B Code Classification Approval – Legacy Modifications to Liquid Injection Shutdown System (USI 34700), Units 5-8,* February 11, 2005.

[B.1-13] OPG Letter, NK30-CORR-00531-06027, G. Jager to M. Santini, *Pickering B – Summary of Systematic Review of Integrated Safety Review (ISR) Gaps for Continued Operation*, September 1, 2011.

[B.1-14] CNSC Letter, e-Docs # 3947907, OPG File No. NK30-CORR-00531-06324, M. Santini to G. Jager, Pickering NGS-B – CNSC Staff Assessment of OPG's 2011 Continued Operations Plan (Action Item 2010-8-05 (2461)) and Path Forward, June 19, 2012.

[B.1-15] CNSC Regulatory Document R-77, *Overpressure Protection Requirements for Primary Heat Transport System in CANDU Power Reactors Fitted with Two Shutdown Systems*, October 1987.

[B.1-16] CNSC Letter, e-Docs # 3256609, OPG File No. NK30-CORR-00531-04876, T.E. Schaubel to D.P. McNeil, *Pickering NGS-B – Integrated Safety Review (ISR) – CNSC Review of Acceptable Deviations and Discrepancies for the Plant Design Safety Factor Report*, June 27, 2008.

[B.1-17] CNSC Letter, e-Docs # 4425359, OPG File No. NK30-CORR-00531-06815, M. Santini to B. Phillips, *Pickering Units 5 to 8: Systems Registration Update Recovery Plan Project, Action Item 2004-8-16 (RIB #2408)*, May 6, 2014.

[B.1-18] OPG Letter, NA44-CORR-00531-00464, R.J. Strickert to J.S.C. Tong, Pickering A – *Review of Pickering A Design to CSA-N285.0-95*, April 9, 2001.

[B.1-19] CSA Standard, N285.0-08, *General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants/Material Standards for Reactor Components for CANDU Nuclear Power Plants*, June 2008.

[B.1-20] OPG Report, NK38-REP-03680-10030 R000, *Review of CAN/CSA-N285.0-08 (June 2008), General Requirement for Pressure Retaining Systems and Components in CANDU Nuclear Power Plants for Darlington Integrated Safety Review*, June 2011.

[B.1-21] OPG Report, NK38-REP-03680-10030-ADD-001 R000, *Addendum to the CSA N285.0-08 Code Review Report for Darlington ISR*, January 2014.

[B.1-22] CSA Standard, N285.0-08 including Update No. 1, *General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants/Material Standards for Reactor Components for CANDU Nuclear Power Plants*, June 2008; Update No. 1: June 2009.

[B.1-23] OPG Letter, P-CORR-00531-03719, G. Jager to M. A. Leblanc, *Application for Renewal of Pickering Nuclear Generating Station Power Reactor Operating Licence*, July 4, 2012.

[B.1-24] CSA Standard, N285.0-12/N285.6 Series-12 including Update No. 1, *General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants/Material Standards for Reactor Components for CANDU Nuclear Power Plants*, 2012; Update No. 1: September 2013.

[B.1-25] OPG Report, NK38-REP-03680-10136 R000, *Code Refresh Review of CSA-N285.0-12 and Update 1 (September 2013), General Requirement for Pressure Retaining Systems and Components in CANDU Nuclear Power Plants*, February 2014.

[B.1-26] OPG Manual, N-MAN-01913.11-10000 R016, *Pressure Boundary Manual*, February 2015.

[B.1-27] OPG Program, N-PROG-MP-0004 R016, *Pressure Boundary*, November 2015.

[B.1-28] CNSC Report, LCH-DNGS-R000, OPG File No. LCH-DNGS-PROL 13.00/2025, *Licence Conditions Handbook: Darlington Nuclear Generating Station Nuclear Power Reactor Operating Licence PROL 13.00/2025*, January 2016.

[B.1-29] OPG Letter, N-CORR-00531-06868 R000, W.M. Elliott to M. Santini, K. Glenn and F. Rinfret, *Code Effective Date for CSA N285.0 for Ontario Power Generation Nuclear Fleet – Annex N*, April 29, 2015.

[B.1-30] OPG List, N-LIST-00531-10003 R000, *Index to OPG Pressure Boundary Program Elements*, March 2015.

[B.1-31] OPG Memorandum, N-CORR-01913.11-0464159-T5, D. Reiter to file, *Governance Gap Analysis Results for N-PROG-MP-0004, Pressure Boundary, Related to Transition 2013 Project*, May 15, 2013

[B.1-32] CSA Standard, N285.0-12/N285.6 Series-12, *General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants/Material Standards for Reactor Components for CANDU Nuclear Power Plants*, 2012.

[B.1-33] OPG Memorandum, N-CORR-00590-0455367, I. Nicolau to File, *Code over Code Review of CSA N285.0-12 (New) over CSA N285.0-08 and Update No. 2 (2010)*, January 31, 2013.

[B.1-34] OPG Report, N-REP-00590-0520105 R001, *Code-Over-Code Review Report: CSA N285.0/CSA N285.6 Series For the Year 2014*, April 2015.

[B.1-35] OPG Report, N-REP-00590-0526911 R000, *Code Over Code Review Report: CSA N285.0 For Year 2015*, July 2015.

B.2 CSA N293-12, "Fire Protection for Nuclear Power Plants"

B.2.1 Background

The following, extracted from the Scope of CSA N293-12 [B.2-1], provides a brief overview of the purpose of the standard and the requirements expressed therein:

> *This Standard provides the minimum fire protection requirements for the design, construction, commissioning, operation, and decommissioning of nuclear power plants, including structures, systems, and components (SSCs) that directly support the plant and the protected area.*

CSA N293 is relevant to Safety Factor 1 (Plant Design) and Safety Factor 7 (Hazard Analysis). CSA N293 is also relevant to Safety Factor 13 (Emergency Planning), since the CNSC Acts and Regulations webpage [B.2-2] identifies the Standard as being relevant to the Safety Control Area for "Emergency Management and Fire Protection".

Compliance with CSA N293-07 [B.2-3] is currently a licence requirement for Pickering NGS (PROL 48.02/2018) as indicated in Section 11.2 and Appendix C.1 of the R04 Pickering Licence Conditions Handbook [B.2-4].

CSA N293-12 is the fourth edition of this standard, and supersedes the previous editions published in 1987, 1995 and 2007. The CSA N293-12 Impact Statement [B.2-5] identifies the significant changes between the 2012 and 2007 editions of the Standard, which are discussed in Section B.2.2.2 below.

The results of PSR1 CSA N293 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.2.2. As identified in Reference [B.2-6], the Pickering PSR2 review of CSA N293-12 is an Incremental Review. PSR2 Incremental Review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- <u>Compliance:</u> Compliance indicates that the change in the safety requirement, per the topical review, is met.
- <u>Gap</u>: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.2.2 Compliance Assessment for Pickering PSR2

B.2.2.1 Application of PSR1 Reviews

The versions of N293 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

As part of the Pickering B ISR, OPG Report NK30-REP-03680-00005 R000, "Pickering NGS B Integrated Safety Review - Safety Analysis Review" [B.2-7] assessed CSA N293-95 [B.2-8] and stated:

> *An overall program N-PROG-RA-0012 has been established to manage Fire Protection at OPG plants …* [which] *… describes the Nuclear governing documents intended to achieve the fire protection goals of CSA N293-95. N293-95 deals with all aspects of fire protection at a programmatic level…. It is concluded that OPG complies with the requirements of CSA standard N293-95 and the compliance is built into existing governance, so there is no expected impact on Pickering B plant life extension in terms of this version of the CSA standard.*

The scope of OPG Program N-PROG-RA-0012 R011, "Fire Protection" [B.2-9] is as follows:

> *This program outlines the scope and objectives of documentation that comprises the fire protection program and interfacing programs for Ontario Power Generation Nuclear… It describes the fire protection organization, interfacing organizations, and their fire protection accountabilities. These processes and organizations are required to minimize the risks and consequences of fire to Nuclear in accordance with: - Nuclear Power Reactor Operating License (PROL) requirements… - Canadian Standards Association (CSA) N293, Fire Protection for Nuclear Power Plants.*

OPG Report NK30-REP-03680-00001 R000, "Pickering NGS-B Integrated Safety Review - Plant Design Safety Factor" [B.2-10] performed a clause-by-clause review of N293-95 [B.2-8] against Safe Operating Envelope (SOE) systems / Systems Important to Safety (SIS) and found ten gaps, of which four were classified as Discrepancies and six as Acceptable Deviations (Note: OPG Report NK30-REP-03680-00001 R000 [B.2-10] did not assess the Pickering B Fire Protection System specifically as part of the review, as it was not classified as an SOE or SIS system). The six Acceptable Deviations were grouped as Issues 1-040/042/043/047/048/049 [B.2-11]. The four Discrepancies were:

- 1-045: Clauses 5.2.5 (d) - *SOE systems containing hydrogen/deuterium not reviewed against NFPA 50A.*
- 1-046: Clause 6.3.5 - *No specific statement to consider deleterious effects of discharging extinguishing agents on equipment in fire area (flooding, additional loads, cooling, etc.).*
- 1-050: Clause 7.5.2 - *No assessment made of SOE/SIS system components that contain flammable liquids/gases and are not seismically qualified.*
- 1-062: Clause 6.5.2 - *Providing holes or doors for discharging extinguishing agents into enclosures/electrical cabinets was not a design requirement.*

OPG Letter, "Pickering NGS-B Integrated Safety Review - Discrepancy Resolution" [B.2-12] stated that the four Discrepancies (1-045, 1-046, 1-050, 1-062) were being resolved by the implementation of a Corrective Action Plan (CAP) for SCR P-2007-17722, and that no further action was required. The CNSC accepted the four Discrepancy classifications, but requested that more information be provided on the six Acceptable Deviations [B.2-11]. The CNSC further responded [B.2-13] by requesting that OPG provide additional justification that no further action is required as part of life extension. OPG Report NK30-REP-03680-00016 R000, "OPG Response to CNSC Comments on Pickering NGS-B Integrated Safety Review - Plant Design, Safety Analysis, Safety Performance, Ageing and Equipment Qualification Safety Factors and Discrepancy Resolutions" [B.2-14] dispositioned the Acceptable Deviations and Discrepancies as low or very low significance with any follow-up work captured under Issue 1-643. OPG Report NK30-REP-03680-00015 R000, "Pickering NGS-B Integrated Safety Review (ISR) - Final ISR Report" [B.2-15] documented the OPG disposition to Issue 1-643 by stating that "OPG will perform a review of CSA N293-07." The commitment to perform a review of CSA N293-07 was later completed as discussed below.

Before discussing the results of the CSA N293-07 review, it is noted that a fire protection Code Compliance Review (CCR) was performed for Pickering B in 2000 to demonstrate that an adequate level of fire protection is provided for the station. The 2000 CCR was documented in OPG Report NK30-REP-71400-10001 R000, "Fire Protection Code Compliance Review Pickering Nuclear Generating Station B" [B.2-16] and assessed compliance of Pickering B with the requirements of the National Building Code of Canada (NBCC), National Fire Code of Canada (NFCC) and applicable National Fire Protection Association (NFPA) standards. The 2000 CCR was later "updated and upgraded" in 2010 to reflect CSA N293-07 requirements and current plant configuration in OPG Report NK30-REP-71400-10001 R001, "Fire Protection Code Compliance Review Pickering Nuclear Generating Station B" [B.2-17].

A gap analysis was performed and documented prior to initiating the 2010 CCR update, per OPG Report NK30-REP-71400-0299410 R000, "Fire Safety Assessment: Definition of Scope of Work for CSA N293-07 Compliance at PNGS-B" [B.2-18]. As part of the gap analysis, changes between the 1995 and 2007 editions of CSA N293 deemed to impact the CCR were identified. The gap analysis concluded that the changes between the 2007 and 1995 editions of N293 and other standards referenced therein "have no impact on the existing physical features of the station, as they are not intended to be applied retroactively. They would as such, be applicable only to new construction including modifications undertaken at the station subsequent to adoption of the new codes and standards under the site operating license." The gaps identified between N293-95 vs N293-07 were related to Inspection, Testing, and Maintenance (ITM) with no impact on the design/construction portion of the CCR. The original 2000 CCR included an assessment of the ITM requirements that applied to the fire alarm life safety systems at Pickering B. The updated ITM evaluation was removed from the CCR and was addressed under OPG Reports NK30-REP-71400-00018 R000, "Fixed Fire Protection Systems Inspection, Testing and Maintenance" [B.2-19] and NK30-REP-71400-00027 R000, "Third Party Review: Fixed Fire Protection Systems Inspection, Testing and Maintenance Report" [B.2-20]. The objective of the Third Party Review [B.2-20] was to assess OPG's evaluation of the ITM program as it applies to fixed fire protection systems at Pickering B for "compliance with the requirements of applicable codes and standards and other relevant documents referenced therein", which included N293-07 [B.2-3]. The Third Party Review [B.2-20] resulted in the identification of 79 deviations from applicable codes and standards. There were three Deviations specifically related to N293-07 as discussed below (text taken verbatim is in italics):

- N293-07 Clause 8.3.3.1 (a) (Deviation No. 73): *Description - Combustible-material-free fire zones must be inspected daily to ensure that they are free of combustible materials. OPG Disposition - Inspection strategy for direct compliance will be implemented prior to operational compliance.* [Third Party Reviewer] *Concurrence - Accepted, Item Closed.*

- N293-07 Clause 8.3.3.1 (c) (Deviation No. 74): *Description - Area with high fire hazards and fire sensitive area must be inspected daily. OPG Disposition - Inspection strategy for direct compliance will be implemented prior to operational compliance.* [Third Party Reviewer] *Concurrence - Accepted, Item Closed.*

- N293-07 Clause 8.3.3.1 (d) (Deviation No. 75): *Description – Doors identified as fire barriers in the FSSA [Fire Safe Shutdown Analysis] must be inspected weekly. OPG Disposition - Currently, PNGS-B complies with CSA N293-95 as per licensing agreement. Inspection strategy is currently being developed as part of the CSA N293-07 implementation project. A list of performance barrier doors has been prepared. Once the list is approved, recommended inspection frequency will be monthly. AHJ [Authority Having Jurisdiction] approval will have to be obtained for frequency change.* [Third Party Reviewer] *Concurrence - Accepted, Item Closed.*

OPG Report NK30-REP-71400-00027 R000 [B.2-20] states that "a satisfactory disposition has been reached to resolve all noted Deviations." Furthermore, as discussed below, Pickering Units 5-8 compliance with the operational requirements of CSA N293-07, including ITM Deviations, has since been demonstrated.

OPG Report NK30-REP-71400-00033 R000, "Pickering NGS 058 Compliance with CSA N293-07, Fire Protection for CANDU Nuclear Power Plants" [B.2-21] was completed to assess compliance against the operational requirements of N293-07, and no gaps were identified. NK30-REP-71400-00033 R000 states:

> *Based on the information provided in this report, Pickering 058 is in compliance with the operational requirements of CSA N293-07. The fire protection design basis documents listed below were all submitted in NK30-CORR-00531-05774, Letter G. Jager to M. Santini, "Pickering NGS 'B' – Request for CNSC Acceptance of the CCR, the Third Party Review of the Inspection, Testing and Maintenance Report for Fixed Fire Protection Systems, the FSSA and the FHA [Fire Hazard Assessment]", December 9th, 2011* [B.2-22].
>
> - *NK30-REP-71400-10001 R001, Fire Protection Code Compliance Review Pickering Nuclear Generating Station B* [B.2-17]
> - *NK30-REP-71400-10002 R002, Fire Hazard Assessment – Pickering B Nuclear Generating Station* [B.2-23]
> - *NK30-REP-71400-00001 R002, Fire Safe Shutdown Analysis – Pickering B Nuclear Generating Station* [B.2-24]
> - *NK30-REP-71400-00027, Third Party Review Fixed Fire Protection Systems Inspection Testing And Maintenance* [B.2-20]

These documents were accepted by the CNSC as satisfying the intent of submission in NK30-CORR-00531-06167, Letter M. Santini to G. Jager, CNSC Acceptance of the Fire Hazard Assessment, Fire Safe Shutdown Analysis, Code Compliance Review and the Inspection Testing and Maintenance Report for Fixed Fire Protection Systems [B.2-25]*. A technical assessment conducted by CNSC Fire Protection staff for all documents submitted is ongoing. Any issues or questions identified following this technical assessment will be addressed as they are received. Tracking of any outstanding deviations identified in the fire protection design basis documents listed above follows the normal Regulatory process.*

In addition to the above, it is noted that OPG List P-LIST-71400-00001 R000, "Application of CSA N293-07 to Structures, Systems and Components for Pickering Nuclear" [B.2-26] was prepared (and referenced in NK30-REP-71400-00033 R000 [B.2-21]) to identify the SSCs within the protected area that are exempt from the application of N293-07, and the structures outside the protected area that need to follow N293-07. Further, OPG Plan P-PLAN-09100-00001 R003, "Pickering Fire Safety Plan" [B.2-27] provides a description of the fire protection initiatives in place at Pickering NGS. Per P-PLAN-09100-00001 R003, "The intent of the Pickering Nuclear Fire Safety Plan is to: - Protect the plant and its staff from hazards and fires. – Minimize the interruption of power generation due to fires. – Minimize economic loss resulting from fire damage. – Minimize the risk of a fire". P-PLAN-09100-00001 R003 states that the Fire Safety Plan meets the requirements of N293-07.

Based on the above, there are no PSR2 gaps associated with the Pickering B ISR review, or subsequent CCR, which assessed compliance against CSA N293-95 [B.2-8] and N293-07 [B.2-3]. OPG Letter NK30-CORR-00531-06731 R000, "Pickering Units 5-8 - Status Update on CSA N293-07 Outstanding Items List" [B.2-28] and NK30-PLAN-00531-00001 R005, "Pickering 5-8 Continued Operations Plan" [B.2-29] subsequently confirmed successful completion of all remaining Deviations identified against N293-07. These past dispositions are not impacted by Pickering NGS operation past 2020.

Nevertheless, NK30-REP-71400-00033 R000 [B.2-21] did not review the design requirements of N293-07, given that Clause 4.3.1 of N293-07 states:

General application: This Standard applies to all plants where its requirements are referenced as a licence condition by the AHJ. For facilities licensed for construction prior to the publication of this Standard, (a) the design and construction requirements of this Standard shall not be retroactively applied to existing structures, systems, and components; and (b) the operational requirements (e.g., general requirements, concepts, programs, operations, analyses, emergency response) of this Standard shall apply.

As a result, NK30-REP-71400-00033 R000 [B.2-21] states that:[5]

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[5] One of the significant differences between the 1995 and 2007 versions of N293 was the acknowledgement that the design and construction requirements of N293 do not apply to those plants that were licensed before January 2007. Therefore, in the absence of modification, existing plants are not required to retro-actively upgrade facilities to demonstrate compliance with this standard.

As per Clause 4.3.1, design requirements do not apply retroactively to existing facilities. For new designs [i.e., plant modifications]*, see information under Clause 4.3.2 regarding the integration of CSA N293-07 in the ECC [Engineering Change Control] process. Fire protection program implemented as per fire protection program N-PROG-RA-0012* [B.2-9] *and its governing procedures.*

The disposition for Clause 4.3.2 states:

The following documentation was either revised or created to align the CSA N293-07 requirements to the existing modification process:

- *N-PROC-MP-0090, Modification Process* [B.2-30]
- *N-GUID-00700-10000, Guide to Modification Process* [B.2-31]
- *N-FORM-10287, Fire Protection Impact Evaluation* [B.2-32]
- *N-STD-RA-0038, Requirements of Fire Safety Assessments* [B.2-33]
- *N-GUID-09076-10002, Guide to Fire Protection Requirements for Design Modifications* [B.2-34]
- *N-INS-09076-10004, Fire Protection Third Party Review* [B.2-35]
- *N-FORM-11180, Fire Codes and Standards Compliance Record* [B.2-36].

OPG Guideline N-GUID-00590-00002 R001, "Code over Code Review – Guideline" [B.2-37] and OPG List N-LIST-00590-00001 R002, "List of Significant Technical Changes from Code-Over-Code Review" [B.2-38] together ensure that any future design changes made to the Pickering NGS Fire Protection System comply with the latest version of N293.

Pickering NGS compliance against the design requirements of N293-07, as well as the changes in N293-12 [B.2-1], is addressed in Section B.2.2.2 below.

<u>Pickering Units 1,4</u>

OPG Letter NA44-CORR-00531-00381 R000, "Pickering A - Updated Basis for Return to Service Document" [B.2-39], Attachment "Pickering A - Basis for Return to Service", noted that the Atomic Energy Control Board requested that OPG review fire protection at Pickering A against CSA N293-95 [B.2-8]. OPG subsequently performed a clause-by-clause review of N293-95 in OPG Report NA44-REP-71400-10004 R000, "PNGS-A Compliance to CSA N293-95, Fire Protection for CANDU Nuclear Power Plants" [B.2-40] as part of the Fire Safety Assessment (FSA) under the OPG Fire Protection Upgrade Project (IIP EN-008). OPG Report NA44-REP-71400-10004 R000 [B.2-40] states:

[The review assessed]*… the design of Pickering 'A' against the requirements laid out in CSA N293-95 on Fire Protection. As the goal is to identify the requirements that were not known at the time these units were designed, compliance discussion for each requirement is based on finding existence of the requirement in plant documents, or current commitment to implement the requirement, and if such evidence is not available, on reviewing the as-built design to determine if the requirement was implemented…*

This review concludes that following completion of the upgrades being implemented during the Pickering 'A' Return to Service, the design of Pickering 'A' will meet the intent of all requirements of CSA N293-95 and that no further changes are necessary.

Pickering A was deemed to be in either Indirect or Direct Compliance, or "Non-Compliant Acceptable", with the various clauses of N293-95 (note: the Non-Compliant Acceptable clauses generally related to the use of High Pressure Service Water (HPSW) for firewater at Pickering A, which has since been replaced by diesel firewater pumps per OPG Letter NA44-CORR-00531-06269 R000, "Pickering "A" - Installation of Diesel Engine Driven Fire Pumps (MEC 91665)" [B.2-41]). These past dispositions are not impacted by Pickering NGS operation past 2020. The review further stated that, as part of the Pickering A Return to Service Fire Protection Project at the time, Pickering A had committed to installing upgraded fire detection and suppressions systems in four locations within each unit, in particular the Main Control Room, Control Equipment Rooms, Cable Spreading Areas and Turbine Generator Areas.

A fire protection CCR was prepared in 2000 in OPG Report NA44-REP-71400-10001 R000, "Pickering Nuclear Generating Station "A" Fire Protection Code Compliance Review" [B.2-42] to demonstrate compliance of Pickering A with the requirements of the NBCC, NFCC and applicable NFPA standards. The original 2000 CCR was updated in 2011 to reflect current station conditions and N293-07 requirements, as outlined in OPG Report NA44-REP-71400-10001 R001, "Pickering NGS A Fire Protection Code Compliance Review (CCR)" [B.2-43].

A gap analysis was also performed and documented prior to initiating the 2010 CCR update, per OPG Report, NA44-REP-71400-0300967 R000, "Fire Safety Assessment: Definition of Scope of Work for CSA N293-07 Compliance at PNGS-A" [B.2-44]. As part of the gap analysis, changes between the 1995 and 2007 editions of CSA N293 deemed to impact the CCR were identified. The gap analysis concluded that the changes between the 1995 and 2007 editions of N293 and other standards referenced therein "have no impact on the existing physical features of the station, as they are not intended to be applied retroactively. They would as such, be applicable only to new construction including modifications undertaken at the station subsequent to adoption of the new codes and standards under the site operating license. Changes to the N293 standard were also identified as having an impact on the operations of the station. These changes and associated impacts will be addressed in the FHA for the station." Similar to the equivalent Pickering B review, the Pickering A gaps identified between N293-95 and N293-07 were related to ITM, with no impact on the design/construction portion of the CCR.

The ITM evaluation was addressed under OPG Reports NA44-REP-71400-00021 R000, "Pickering A Fixed Fire Protection Systems Inspection, Testing, and Maintenance Code Compliance Report" [B.2-45] and NA44-REP-71400-00022 R000, "Third Party Review: Fixed Fire Protection Systems Inspection, Testing and Maintenance Report" [B.2-46]. The objective of the Third Party Review was to assess OPG's evaluation of the ITM program as it applies to fixed fire protection systems at Pickering A for "compliance with the requirements of applicable codes and standards and other relevant documents referenced therein", which included N293-07. The review of OPG's compliance evaluation resulted in the identification of 91 Deviations from applicable codes and standards. There were three Deviations specifically related to N293-07 which were associated with the same clause (Clause 8.3.3.1 (a), (b) and (d)) assessed as Deviations for Pickering B which were discussed earlier. NA44-REP-71400-00022 R000 [B.2-46] states that "a satisfactory disposition has been reached to resolve all noted Deviations."

Furthermore, as discussed below, Pickering Units 1,4 compliance with the operational requirements of CSA N293-07, including ITM Deviations, has since been demonstrated.

OPG Report NA44-REP-71400-00027 R000, "Pickering NGS 014 Compliance with CSA N293-07, Fire Protection for CANDU Nuclear Power Plants" [B.2-47] was completed to assess compliance against the operational requirements of N293-07, and no gaps were identified. NA44-REP-71400-00027 R000 states:

> *Based on the information provided in this report, Pickering 014 is in compliance with the operational requirements of CSA N293-07. The fire protection design basis documents listed below has been submitted to the CNSC as per the referenced correspondence.*
>
> - *NA44-REP-71400-10001 R001, Fire Protection Code Compliance Review Pickering Nuclear Generating Station A* [B.2-43]
> - *NA44-REP-71400-00022, Third Party Review, Fixed Fire Protection Systems Inspection, Testing and Maintenance* [B.2-46]
>
> *Both submitted in NA44-CORR-00531-06690, Letter G. Jager to M. Santini, Pickering A – Request for CNSC Acceptance of [the] Fire Protection Code Compliance Review, and Third Party Review, Fixed Fire Protection Systems Inspection, Testing and Maintenance, June 21st, 2011.*
>
> - *NA44-REP-71400-10003 R001, Fire Hazard Assessment – Pickering A Nuclear Generating Station* [B.2-48]
> - *NA44-REP-71400-00023 R000, Fire Safe Shutdown Analysis – Pickering A Nuclear Generating Station* [B.2-49]
>
> *Both submitted in NA44-CORR-00531-06935, Letter G. Jager to M. Santini, Pickering NGS 'A' – Request for CNSC Acceptance of the "Fire Safe Shutdown Analysis" (FSSA) and "Fire Hazard Assessment" (FHA) Reports and Status Update on CCR/ITM Deviations, June 28th, 2012* [B.2-50]. *The submission of all documents listed above was acknowledged by the CNSC in NA44-CORR-00531-06992, Letter M. Santini to G. Jager, Pickering NGS-A – Fire Safe Shutdown Analysis and Third Party Review, Fixed Fire Protection Systems Inspection, Testing and Maintenance* [B.2-51]. *This letter accepted the proposed completion actions and timeline for deviations identified in the reports and stated that a more detailed technical review of the submission was ongoing. Any further requests for information or clarification identified as a result of this review will be addressed as they are received.*

OPG Letters NA44-CORR-00531-06837 R000, "Pickering NGS A - CNSC Acceptance of Fire Protection Code Compliance Review and Third Party Review, Fixed Fire Protection Systems Inspection Testing and Maintenance" [B.2-52] and NA44-CORR-00531-07475 R000, "Confirmation of Completion of CSA N293-07 Outstanding Items List Pickering NGS 'A'" [B.2-53] subsequently confirmed successful completion of all remaining Deviations identified against N293-07. Based on the above, there are no PSR2 gaps associated with the Pickering A Return to Service review against N293-95 [B.2-8], or the subsequent CCR which addressed compliance against the operational aspects of N293-07 [B.2-3].

OPG Guideline N-GUID-00590-00002 R001 [B.2-37] and OPG List N-LIST-00590-00001 R002 [B.2-38] together ensure that any future design changes made to the Pickering NGS Fire Protection System comply with the latest version of N293.

As noted for the Pickering B review against N293-07, NA44-REP-71400-00027 R000 [B.2-47] did not review the design requirements of N293-07 as they are not applicable to an existing facility. As a result, Pickering NGS compliance against the design requirements of N293-07, as well as the changes in N293-12, is addressed in Section B.2.2.2 below.

Darlington NGS

OPG Report NK38-REP-03680-10014 R000, "Review of CAN/CSA N293-07 (January 2008) Fire Protection for CANDU Nuclear Power Plants for Darlington Integrated Safety Review (ISR)" [B.2-54] documented a clause-by-clause review of the operational requirements of CSA N293-07 [B.2-3]. OPG Report NK38-REP-03680-10128 R003, "Gap Analysis to CSA N293-07 Design Clauses for the Darlington Integrated Safety Review" [B.2-55] documented a clause-by-clause review of design clauses excluded from NK38-REP-03680-10014 R000 (based on not being required as per Clause 4.3.1, which states "the design and construction requirements of this Standard shall not be retroactively applied to existing structures, systems, and components"). OPG Report NK38-REP-03680-10189 R000, "Code Review Refresh of CAN/CSA N293, Fire Protection for CANDU NPP, 2012 Edition" [B.2-56] evaluated compliance against the design requirement clauses (Sections 5, 6 and 7) of CSA N293-12 [B.2-1]. OPG Report NK38-REP-78000-10084 R000, "Darlington NGS Compliance with CSA N293-12, Fire Protection for Nuclear Power Plants" [B.2-57] documented Darlington's compliance with the operational requirements of N293-12 and stated: "The 2012 edition of CSA N293 did not introduce any significant technical changes from the 2007 edition. CSA N293-12 is written as a nuclear reactor technology neutral Standard and it incorporates clarifications and OPEX received from the Canadian licensees. Based on the information provided in this report, it is concluded that Darlington NGS is in compliance with the operational requirements of the CSA N293-12 Standard."

The Darlington ISR review findings are not discussed further due to the Pickering Units 1,4 and 5-8 N293-07 reviews above, and the N293-07 and N293-12 compliance discussions provided under Section B.2.2.2 below. However, the content of the above mentioned Darlington ISR Reports was utilized in Section B.2.2.2 below to assist in ascertaining Pickering NGS compliance against the 2007 and 2012 editions of CSA N293.

B.2.2.2 Application of Post PSR1 Reviews

Per the CSA N293-12 Impact Statement [B.2-5], the following is a summary of the significant changes from the previous edition of the standard (N293-07 [B.2-3]):

1. *Changed… to be "Technology Neutral" such that it can now be applied to any type of nuclear power plant, not just CANDU plants.*

2. *Incorporated lessons learned following the introduction of the standard into the Canadian regulatory framework, to facilitate its application as a licensing requirement to existing and future plants.*

3. *Added the collective term "Fire Protection Assessments (FPA)" to include the Code Compliance Review (CCR), the Fire Hazard Assessment (FHA), and the Fire Safe Shutdown Analysis (FSSA), and clarified the requirements for each.*

4. *Incorporated changes to address issues arising from Requests for Interpretation (RFI) during the implementation of N293-07.*

5. *Continued the move toward an objective based standard by maintaining the separation between the fire protection goals, objectives and criteria (Chapter 5) and the requirements for their implementation (Chapters 6 to 11). This will assist the user of the standard in identifying suitable alternatives to the requirements which meet the stated goals, objectives and criteria.*

6. *Protection of the environment was added as an additional goal for fire protection in Chapter 5. Although no additional implementation requirements have been added, the user of the standard must consider the effects on the environment of all fire protection measures (e.g., releases of radioactive materials, chemicals, hydrocarbons, etc.).*

7. *The use of the term "listed" as it pertains to fire protection equipment has been eliminated from the standard, being replaced by wording indicating that equipment must be suitable for its intended use. Although in most cases, the use of "listed" equipment would be the means of providing suitable equipment, it would not be the only means of satisfying the requirements of the standard.*

OPG Memorandum, N-CORR-00590-0477035, "Code-over-Code Review of CSA N293, Fire Protection for Nuclear Power Plants - 2012 Edition over the 2007 Edition" [B.2-58] provides a generic nuclear review of N293-12 against N293-07. This clause-by-clause review found no significant technical changes between N293-12 and N293-07 and no mitigation required. Of note to existing stations are:

- Clause 5.2 which is "changed to include another goal pertaining to minimizing the impact of radioactive and hazardous materials on the environment as a result of fire". However, per Reference [B.2-58], the "current revisions of DNGS and PNGS FHA reports addressed safety of radioactive materials as part of their objectives". Therefore, there is no impact on current practices and assessments.

- Clause 5.6.2 which is "changed by replacing the term FSSA with Fire Protection Assessments (FPAs)". However, per Reference [B.2-58], the "current requirement to update the DNGS and PNGS FSSA/FHA/CCR remains unchanged". This change is for clarification only.

- Clause 7.3.3.5 which is "revised to include an additional requirement for cable tray protection"*.* However, per Reference [B.2-58], the "current revisions of DNGS and PNGS

FHA and FSSA reports demonstrate the fire protection goals are met. Therefore, this is not considered a significant technical change."

Although these clauses are relevant to Pickering NGS, applicable changes are addressed in the current Pickering Units 5-8 and 1,4 Fire Safe Shutdown Analyses, Fire Hazard Assessments, and/or Code Compliance Review reports. Based on the above, there are no safety significant changes resulting from CSA N293-12.

As discussed previously, compliance with CSA N293-07 [B.2-3] is currently a licence requirement for Pickering NGS (PROL 48.02/2018) as indicated in Appendix C.1 of the R04 Pickering Licence Conditions Handbook [B.2-4]. Completion of all remaining operational and ITM Deviations identified against N293-07 was confirmed by NA44-CORR-00531-07475 R000 [B.2-53] and NK30-CORR-00531-06731 R000 [B.2-28]. However, Pickering NGS compliance against the design requirements of CSA N293 was not formally documented in the past given that Clause 4.3.1 of both CSA N293-07 and N293-12 state: "the design and construction requirements of this Standard shall not be retroactively applied to existing structures, systems, and components". Since compliance review against the design requirements of CSA N293-12 was not undertaken as part of previous Pickering NGS PSR1 reviews, the following high level assessment has been completed. In the review below, the degree of conformance with clauses or groups of clauses in CSA N293-12 is assessed for Pickering NGS by reference to supporting evidence stating whether the intent of the requirements stipulated in the standard is met. The high level intent review below assessed Pickering NGS against Sections 4 (General Requirements), 5 (Fire Protection Concepts), 6 (Design Requirements) and 7 (Design and Installation Requirements) of CSA N293-12.[6]

CSA N293-12 Clause	PSR2 Review	Compliant or Gap
4 General Requirements 4.1 Effective Date 4.2 Responsibility 4.3 Applicability to Existing Facilities 4.4 Alternatives and Performance-Based Approaches	Clause 4 of N293-12 provides general requirements for the application of the standard. Per N-REP-09076-10006 R000, "Review of Design Requirements of CSA N293-07" [B.2-59]: "Those modifications to existing plants must comply with the design requirements of this standard (applies to the modified portions only). In the event that compliance cannot be met then Clause 4 provides the mechanism for resolution of any deviation. Therefore, in the event of a modification to the plant the modified structure, system or component must either meet the deterministic, design requirements detailed in Clause 6 and Clause 7, or demonstrate an equivalent level of fire	Compliant

6 Sections 1 (Scope), 2 (Reference Publications) and 3 (Definitions and Abbreviations) were not reviewed as there are no requirements and the provided information is context-setting or background/definition/reference information only. Section 9 (Fire Protection Requirements for Decommissioning) was not reviewed as Decommissioning is not in PSR2 scope.

Sections 8 (Implementation of Fire Protection Program), 10 (Fire Response Capability) and 11 (Fire Protection Assessments) contain operational content and were previously addressed in past Pickering NGS reviews. Furthermore, there were no safety significant changes to these sections in N293-12 per N-CORR-00590-0477035 [B.2-58].

The Annexes to N293-12 are non-mandatory commentary only and a review is therefore not required.

CSA N293-12 Clause	PSR2 Review	Compliant or Gap
4.5 Documentation Requirements for Alternatives and Performance-Based Approaches 4.6 Fire Protection Assessments	safety through equivalency or performance based approaches detailed in Clause 4." There are no safety significant changes between Clause 4 of N293-07 and N293-12 [B.2-58]. Clause 4 was reviewed for Pickering PSR2 and it was determined that all sub-clauses except the two below were addressed in prior Pickering NGS N293-07 reviews (i.e., NK30-REP-71400-00033 R000, "Pickering NGS 058 Compliance with CSA N293-07, Fire Protection for CANDU Nuclear Power Plants" [B.2-21], NA44-REP-71400-00027 R000, "Pickering NGS 014 Compliance with CSA N293-07, Fire Protection for CANDU Nuclear Power Plants" [B.2-47]).[7] • Sub-clause 4.3.1: Bullet (a) of this sub-clause states: "the design and construction requirements of this Standard shall not be retroactively applied to existing structures, systems, and components". Sub-clause 4.3.1 (a) was cited in the 2015 Pickering A and B N293-07 reviews ([B.2-21], [B.2-47]) to conclude that design-related clauses did not need to be assessed. Compliance against all design related clauses of N293-12 (Sections 4-7) has been assessed as part of PSR2 in this high level intent review table, as discussed below. • Sub-clause 4.6.2: This sub-clause states: "For facilities licensed for operation prior to the publication of this Standard, the assessments referred to in Clause 4.6.1 shall demonstrate achieving (a) the goals, objectives and criteria of Clause 5; (b) the operational requirements of this standard; and (c) the applicable design and construction requirements of the codes of record. Note: It is intended by this Standard that fire protection assessments required by this Clause can be satisfied by previous analysis performed in accordance with the 2007 edition of this Standard (i.e., CCR, FHA, FSSA, etc.) and maintained in compliance with Clause 11.2.3." This sub-clause has changed from NK30-REP-71400-00033 R000 [B.2-21] and NA44-REP-71400-00027 R000 [B.2-47] to provide additional context and clarity. Although Pickering NGS is not a new facility, it does comply with N293-07 per Pickering NGS PROL requirements [B.2-4] and the information provided earlier in this review (e.g., see NA44-CORR-00531-07475 R000 [B.2-53], NK30-CORR-00531-06731 R000 [B.2-28]). CCRs ([B.2-17], [B.2-43]), FHAs ([B.2-23],	

[7] The 2015 Pickering A and B N293-07 reviews, NK30-REP-71400-00033 R000 [B.2-21] and NA44-REP-71400-00027 R000 [B.2-47], did not identify any gaps. Therefore, if a clause was previously assessed in [B.2-21] and [B.2-47], and has not been revised in N293-12, compliance was previously demonstrated and there is no PSR2 gap.

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	[B.2-48]) and FSSAs ([B.2-24], [B.2-49]) have been prepared for Pickering NGS. Furthermore, an intent review is provided against Clause 5 of N293-12 below. Therefore, there is no PSR2 gap.	
5 Fire Protection Concepts 5.1 General 5.2 Goals 5.3 Defence-in-Depth Principle 5.4 Nuclear Safety Objectives and Performance Criteria 5.5 Life Safety 5.6 Fire Protection Assessments (FPA) 5.7 General Fire Protection Measures 5.8 Fire Protection Program 5.9 Fire Protection for Modifications to Operating Plants 5.10 Quality Assurance	Clause 5 of N293-12 provides general fire protection concepts and performance level requirements for design, construction and operation. Clause 5 also provides details of the requirements for assessments to be conducted on modifications which have the potential to impact fire safety. Per N-REP-09076-10006 R000 [B.2-59], this requirement mirrors the process presently implemented by OPG Nuclear in the ECC process and stated in the Pickering PROL, i.e., that: a) the reviews are the responsibility of the plant Design Authority, b) all modifications shall be screened and those having potential for fire impact will receive a detailed review, c) those modifications screened as having potential to impact fire safety shall be submitted to a qualified third party for review and the AHJ, and d) the qualified third party must not be in the same management and financial operation as the design organization. As noted earlier, OPG Guideline N-GUID-00590-00002 R001 [B.2-37] and OPG List N-LIST-00590-00001 R002 [B.2-38] together ensure that any future design changes made to the Pickering NGS Fire Protection System comply with the latest version of N293. With respect to the individual sub-clauses: • <u>Clause 5.1</u> defines the general scope, purpose and objectives of Clause 5. This information is context setting only, i.e., there are no requirements. One minor editorial change was made in N293-12 which is not safety significant. Therefore, the intent of this clause is met and there is no PSR2 gap. • <u>Clause 5.2</u> states high level fire protection goals for the plant. Although assessed in the 2015 Pickering A and B N293-07 reviews ([B.2-21], [B.2-47]), Clause 5.2 has changed in N293-12 to include another goal (d) pertaining to minimizing "the impact of radioactive and hazardous materials on the environment as a result of fire". The Pickering NGS FHAs ([B.2-23], [B.2-48]) identified potential locations where fire may result in the release of radioactive smoke or contaminated water run-off. The FHAs determined adequate controls are in place to satisfy ALARA (As-Low-as-Reasonably-Achievable) principles and identified areas for improvement. OPG submitted the FHAs to the CNSC, noting that any identified deviations are not nuclear safety significant (per NA44-CORR-00531-06935 R000 [B.2-50], NK30-CORR-00531-05774 R000 [B.2-22]) and submitted plans for their correction which were	Compliant

CSA N293-12 Clause	PSR2 Review	Compliant or Gap
	accepted by the CNSC. These plans have now been implemented (per NA44-CORR-00531-07475 R000 [B.2-53], NK30-CORR-00531-06731 R000 [B.2-28]). The FHAs also assess the impact of fires involving hazardous materials. Therefore, the intent of new item (d) is met and there is no PSR2 gap. • Clause 5.3 addresses defence-in-depth principles. All sub-clauses were assessed in the 2015 Pickering A and B N293-07 reviews ([B.2-21], [B.2-47]) and have not been revised in N293-12, except sub-clause 5.3.2 which has changed in N293-12 to replace reference to "materials, oxidizers" with "combustible materials". This change is a clarification only. The station has policies, programs and procedures in place to reduce and control combustible materials and ignition sources per N-PROG-RA-0012 R011, "Fire Protection" [B.2-9]. Hence, the revision has no impact from a compliance perspective and the intent of this clause is met. There is no PSR2 gap. • Clause 5.4 specifies high level nuclear safety objectives and performance criteria, as well as goals for limiting the release of radioactive material and fission products. All sub-clauses were assessed in the 2015 Pickering A and B N293-07 reviews ([B.2-21], [B.2-47]) and have not been revised in N293-12, except for Clause 5.4.2.4 which has changed to replace "primary heat transport" by "reactor coolant" to make it technology-neutral, and to incorporate "moderator system" into "reactor auxiliary systems". These changes do not impact compliance. Therefore, the intent of this clause is met and there is no PSR2 gap. • Clause 5.5 specifies life safety performance objectives to be met during all operational modes and plant configurations. All sub-clauses were assessed in the 2015 Pickering A and B N293-07 reviews ([B.2-21], [B.2-47]) and have not been revised in N293-12. Therefore, the intent of this clause is met and there is no PSR2 gap. • Clause 5.6 title has changed from "Fire Safe Shutdown Analysis" to "Fire Protection Assessments" in N293-12. Sub-clause 5.6.1 has been expanded in N293-12 to require at least a CCR and a FHA along with the FSSA stipulated in N293-07. Pickering NGS has completed the CCRs ([B.2-17], [B.2-43]), FHAs ([B.2-23], [B.2-48]) and FSSAs ([B.2-24], [B.2-49]) and will continue to update them. All other sub-clauses were assessed in the 2015 Pickering A and B N293-07 reviews ([B.2-21], [B.2-47]). Therefore, the intent of this clause is met and there is no PSR2 gap.	

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	• Clause 5.7 provides general fire protection measures, including control of combustible materials, control of ignition sources, fire alarm systems, fire suppression, fire hazard control, smoke management, protection against seismic hazards and control room complex. The following sub-clauses were not assessed in the 2015 Pickering A and B N293-07 reviews ([B.2-21], [B.2-47]) and/or were changed in N293-12: ○ Sub-clause 5.7.1.1 is unchanged in N293-12. This sub-clause requires that "Buildings, both in the protected area or external to the protected area but directly supporting the plant, shall be constructed using non-combustible materials, as defined in the NBCC." The Pickering A and B CCRs ([B.2-17], [B.2-43]) have examined the plant for potential deviations from use of non-combustible construction and dispositioned any findings, e.g., Deviation # 99-003 (fiberglass siding of H2 storage area) for Pickering A (per NA44-REP-71400-10001 R001 [B.2-43]), Deviation #'s 01201 and 01202 (removal of fibreglass wall panels and siding attached to H2 storage enclosures) for Pickering B (per NK30-REP-71400-10001 R001 [B.2-17]). The FSSAs ([B.2-24], [B.2-49]) have shown that a fire in any existing building would not jeopardise attaining a safe shutdown state. Therefore, the intent of this sub-clause is met and there is no PSR2 gap. ○ Sub-clause 5.7.1.4 is unchanged in N293-12. This sub-clause requires that "Plant design shall ensure that combustible materials, dangerous goods, and liquids and gases used for plant operations are stored, located, and protected to minimize fire hazards and the resultant threats to nuclear and life safety". The Pickering NGS CCRs ([B.2-17], [B.2-43]), FHAs ([B.2-23], [B.2-48]) and FSSAs ([B.2-24], [B.2-49]) have demonstrated that design measures in place to locate and store combustible materials, dangerous goods, and liquids and gases used for plant operation adequately minimize fire hazards and the resultant threats to nuclear and life safety. Therefore, the intent of this sub-clause is met and there is no PSR2 gap. ○ Sub-clause 5.7.2.1 is unchanged in N293-12. This sub-clause requires that "Installed devices and process operations that, by design, pose a fire hazard shall be identified and analyzed or addressed in the design stage of the plant and shall be eliminated or controlled in order to minimize the occurrence of fires". The Pickering FSSAs ([B.2-24],	

CSA N293-12 Clause	PSR2 Review	Compliant or Gap
	[B.2-49]) and FHAs ([B.2-23], [B.2-48]) assessed permanent ignition sources as potential fire hazards or contributing to potential fire hazards. The consequences of these fires were assessed from a nuclear and building safety perspective. The fire safe shutdown goals were determined to be satisfied. Modifications to the plant require identification of their potential as a fire hazard and implementation of suitable mitigation as per the following plant procedures: N-PROC-MP-0090 R014, "Modification Process" [B.2-30], N-GUID-00700-10000 R015, "Guide to Modification Process" [B.2-31], N-FORM-10287 R005, "Fire Protection Impact Evaluation" [B.2-32], N-STD-RA-0038 R003, "Requirements of Fire Safety Assessments" [B.2-33], N-GUID-09076-10002 R000, "Guide to Fire Protection Requirements for Design Modifications" [B.2-34], N-INS-09076-10004 R002, "Fire Protection Third Party Review" [B.2-35] and N-FORM-11180 R002, "Fire Codes and Standards Compliance Record" [B.2-36]. Therefore, the intent of this sub-clause is met and there is no PSR2 gap. o Sub-clause 5.7.5.1 had minor editorial changes made to two sub-clauses in N293-12 that are not safety significant. This sub-clause specifies requirements relating to layout of SSCs and separation between floors and areas within buildings. Fire safety assessments conducted over the life of the Station (i.e., Pickering NGS CCRs ([B.2-17], [B.2-43]), FHAs ([B.2-23], [B.2-48]) and FSSAs ([B.2-24], [B.2-49]) have demonstrated that the fire protection goals of CSA N293 are satisfied for the existing plant. Therefore, the intent of this sub-clause is met and there is no PSR2 gap. o Sub-clause 5.7.5.2.2 is unchanged in N293-12. This sub-clause requires that "The turbine generator building (hall) shall be designed and separated from other areas of the plant such that a fire involving the turbine generator area will not (a) spread to other areas; and (b) result in progressive structural collapse." The 2015 Pickering A and B N293-07 reviews ([B.2-21], [B.2-47]) note that the CCRs and FHAs have evaluated separation between the turbine building and adjacent areas, and turbine generator sprinkler protection. All recommendations of the Fire Protection Assessment reports have been implemented or dispositioned (per NA44-CORR-00531-07475 R000 [B.2-53], NK30-CORR-00531-06731 R000 [B.2-28]).	

CSA N293-12 Clause	PSR2 Review	Compliant or Gap
	Therefore, the intent of this sub-clause is met and there is no PSR2 gap. ○ Sub-clause 5.7.8.1 is unchanged in N293-12 (Note that N293-07 sub-clause 5.7.8.1 was deleted as redundant, so 5.7.8.2 to .5 in N293-07 are now 5.7.8.1 to .4 in N293-12). This sub-clause requires that the "control room complex shall be separated from adjoining areas by a fire separation with a fire resistance rating as specified in Clause 6.7.1.1." This clause simply points to sub-clause 6.7.1.1 and does not, itself, require review for compliance. Sub-clause 6.7.1 is addressed specifically below. There is no PSR2 gap. ○ Sub-clause 5.7.8.2 is unchanged in N293-12, except that it has been re-numbered to 5.7.8.3. This sub-clause requires that "Special consideration shall be given to the prevention of fires in the control room complex." The sub-clause was assessed as compliant in the Pickering B N293-07 review [B.2-21] since "Impact of fires on the MCR [Main Control Room] envelope has been assessed in the updated FHA report (NK30-REP-71400-10002 R002 [B.2-23]). As there were no deviations reported, the station may be considered in compliance with this clause." Although this sub-clause was not reviewed in past reviews of Pickering A against the requirements of N293-07, the compliance basis for Pickering B is equally applicable to Pickering A as a FHA has been conducted for Pickering A as well and all its recommendations implemented or dispositioned per NA44-CORR-00531-07475 R000 [B.2-53], similar to that for Pickering B per NK30-CORR-00531-06731 R000 [B.2-28]. Thus, the intent of this sub-clause is met for both Pickering Units 1,4 and Units 5-8 and there is no PSR2 gap. • Clause 5.8 specifies Fire Protection Program requirements. All sub-clauses were assessed in the 2015 Pickering A and B N293-07 reviews ([B.2-21], [B.2-47]). No sub-clauses have been revised in N293-12, except sub-clause 5.8.3 which has changed in N293-12 to replace the acronym "FHA" by "FPA" and include "combustible waste control" in housekeeping activities, and sub-clause 5.8.5 which has changed in N293-12 to replace "structures and equipment" with "SSCs". These are editorial changes and do not impact compliance. Therefore, the intent of this clause is met and there is no PSR2 gap. • Clause 5.9 specifies requirements for modifications to operating plants. All sub-clauses were assessed in the	

CSA N293-12 Clause	PSR2 Review	Compliant or Gap
	2015 Pickering A and B N293-07 reviews ([B.2-21], [B.2-47]). No sub-clauses have been revised in N293-12 except sub-clause 5.9.2.2 which has changed in N293-12 to replace "systems, structures, or components" with "SSCs". This is an editorial change and has no impact on compliance. Therefore, the intent of this clause is met and there is no PSR2 gap. • Clause 5.10 addresses fire protection program Quality Assurance requirements. All sub-clauses were assessed in the 2015 Pickering A and B N293-07 reviews ([B.2-21], [B.2-47]) and have not been revised in N293-12. Therefore, the intent of this clause is met and there is no PSR2 gap.	
6 Design Requirements for the Prevention and Mitigation of Fires 6.1 General 6.2 Objectives 6.3 Separation 6.4 Protection of Fire Safe Shutdown Systems and Equipment 6.5 Reducing the Spread of Fire 6.6 Life Safety 6.7 Maintaining Plant Operation During a Fire 6.8 Fire Prevention by Design	Clause 6 of N293-12 specifies design requirements to prevent fires from occurring (as well as means to limit and mitigate fires, once initiated). With respect to the individual sub-clauses: • Clause 6.1 provides general design requirements for mitigation of fires. Clause 6.1 has not been revised in N293-12, and was not assessed in the 2015 Pickering A and B N293-07 reviews ([B.2-21], [B.2-47]). However, this clause is of a general nature (context-setting) and does not require review for compliance. • Clause 6.2 requires that the plant be provided with redundant fire safe shutdown systems to ensure that nuclear safety objectives are satisfied. Clause 6.2 was not assessed in the 2015 Pickering A and B N293-07 reviews ([B.2-21], [B.2-47]). This Clause has changed in N293-12. It now references sub-clause 5.4.1 where N293-07 referenced Clause 5.4. This change is editorial, and as discussed earlier, the intent of Clause 5.4 is met (in particular, the Pickering FSSAs ([B.2-24], [B.2-49]) modelled sufficient equipment and relevant support services to satisfy the requirements of Clause 5.4.1). Analysis was done for all credible fire scenarios in fire zones where the components or cables associated with the safe shutdown equipment are located. All recommendations of the Fire Protection Assessment reports (CCRs ([B.2-17], [B.2-43]), FHAs ([B.2-23], [B.2-48]) and FSSAs ([B.2-24], [B.2-49]) have been implemented or dispositioned (per NA44-CORR-00531-07475 R000 [B.2-53], NK30-CORR-00531-06731 R000 [B.2-28]). Therefore, the intent of this clause is met and there is no PSR2 gap. • Clause 6.3 addresses separation requirements between redundant fire safe shutdown systems. The 6.3 sub-clauses that were not assessed in the 2015 Pickering A	Compliant

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	and B N293-07 reviews ([B.2-21], [B.2-47]) are discussed below: o Sub-clauses 6.3.1.1 and 6.3.1.2 are unchanged in N293-12, and are general statements regarding methods of achieving fire separation of redundant fire safe shutdown systems (i.e., context-setting only). Therefore, a review is not required. o Sub-clauses 6.3.1.3 and 6.3.1.4, which relate to fire resistance rating of separations, have been changed in N293-12 to replace the acronym "FHA" with "FPA" to clarify that fire protection program goals can be assessed not just in the FHA but also the CCR, FSSA and other assessments. Sub-clause 6.3.1.4 was not assessed in the 2015 Pickering A and B N293-07 reviews ([B.2-21], [B.2-47]). The Pickering NGS Fire Protection Assessments that have been performed include FHAs, FSSAs and CCRs ([B.2-23], [B.2-48]; [B.2-24], [B.2-49]; and [B.2-17], [B.2-43], respectively). The FSSAs have identified the fire areas/zones where redundant safe shutdown functions are located in the same plant area (fire area/zone). The FHAs then evaluated all potential fires in the area to identify situations where a single fire can impact both redundant functions due to insufficient fire resistance rating of fire separation where provided. For redundant functions that can be impacted by a single fire, corrective measures were identified to resolve the issue. The FSSAs modelled sufficient equipment and support services to demonstrate that safe shutdown can be achieved for all credible fires following the identified corrective actions. All recommendations of the Fire Protection Assessment reports have been implemented or dispositioned (per NA44-CORR-00531-07475 R000 [B.2-53], NK30-CORR-00531-06731 R000 [B.2-28]). Further, control of transient materials is governed by N-PROC-RA-0054 R015, "Control of Space Allocation for Transient Material and Extended Storage of Material within the Site" [B.2-60] and the impact of transient material on fire protection is conducted in accordance with N-GUID-09076-10001 R004, "Guideline for Fire Protection Reviews of Space Allocation and Transient Material Permits" [B.2-61]. Thus, existing barriers to fire initiation and propagation are adequate to ensure nuclear safety objectives are met. The intent of this sub-clause is met and there is no PSR2 gap.	

CSA N293-12 Clause	PSR2 Review	Compliant or Gap
	○ Sub-clause 6.3.1.5 has changed in N293-12 to replace "nuclear safety" by "nuclear safety related". This change does not impact compliance. This sub-clause applies standard requirements for life safety as per NBCC sub-clause 3.1.9.1 to protect both personnel and nuclear safety related systems. The Pickering CCRs ([B.2-17], [B.2-43]) have reviewed Pickering NGS compliance with the NBCC. All recommendations of the CCR reports have been implemented or dispositioned (per NA44-CORR-00531-07475 R000 [B.2-53], NK30-CORR-00531-06731 R000 [B.2-28]). Thus, existing barriers to fire initiation and propagation are adequate to ensure nuclear safety objectives are met. The intent of this sub-clause is met and there is no PSR2 gap. ○ Sub-clauses 6.3.2.2 and 6.3.2.3, which relate to the structure housing the Turbine Generator, have not changed in N293-12 but were not assessed in the 2015 Pickering A and B N293-07 reviews ([B.2-21], [B.2-47]). The turbine building was designed based on the relevant standards at the time of construction. The FHAs and FSSAs have demonstrated the effectiveness of the combination of fire separation, fire detection and suppression (both manual and automatic). The design of the upgraded fire suppression system for the turbine generator areas in Pickering NGS has been performed consistent with the requirement for prevention of structural steel collapse, as noted in Section 4.0 of Design Manual NA44-DM-71400-00002 R000, "Fire Protection Systems (Water)" [B.2-66] and Section 3.4.1.2 of Design Manual NK30-DM-71400-00001 R006, "Fire Protection System" [B.2-68]. Turbine building roof fans are rated for smoke removal. Therefore, the intent of these sub-clauses is met and there is no PSR2 gap. ○ Sub-clauses 6.3.3.2 and 6.3.3.3, which relate to spatial separation requirements, were not assessed in the 2015 Pickering A and B N293-07 reviews ([B.2-21], [B.2-47]). Sub-clause 6.3.3.3 has changed in N293-12 to replace the acronym "FHA" with "FPA" and to clarify that fire protection program goals can be assessed not just in the FHA but also the CCR, FSSA and other assessments. Sub-clause 6.3.3.2 prohibits the use of spatial separation in lieu of firewalls for egress except inside containment. The Pickering A CCR [B.2-43] identified Deviation # 2010-505 from this requirement and stated the following: "Exiting from	

CSA N293-12 Clause	PSR2 Review	Compliant or Gap
	the NE corner of the ground floor includes travel through the NE stair which is considered part of the basement floor level which does not comply with the applicable exiting requirements." However, it concluded: "However, based on the limited occupant load, limited combustible loading, the trained and knowledgeable personnel operating in the building and the special nature of the building operations, the existing arrangement does not significantly impact the level of fire and life safety in the building. Therefore, the existing arrangement is deemed acceptable." This argument also applies to Pickering B. With respect to sub-clause 6.3.3.3 (which lists additional compensatory measures where spatial separation is used), associated bullets (a) and (b) are demonstrated to be met by the FHAs and FSSAs. The FSSAs have identified the fire areas/zones where redundant safe shutdown functions are located in the same plant area (fire area/zone). The FHA then evaluated all potential fires in the area to identify situations where a single fire can impact both redundant functions. For redundant functions that can be impacted by a single fire, corrective measures were identified to resolve the issue. The FSSA modelled sufficient equipment and support services to demonstrate that safe shutdown can be achieved for all credible fires following the identified corrective actions. Bullet (c) is met by definition as the FPA only credits provided fire detection, suppression and protection measures. All recommendations of the CCR reports have been implemented or dispositioned (per NA44-CORR-00531-07475 R000 [B.2-53], NK30-CORR-00531-06731 R000 [B.2-28]). Therefore, the intent of these sub-clauses is met and there is no PSR2 gap. • <u>Clause 6.4</u> specifies requirements relating to fire safe shutdown systems and equipment, in particular the potential to damage or disable the system or its components. Clause 6.4 was not assessed in the 2015 Pickering A and B N293-07 reviews ([B.2-21], [B.2-47]). Only sub-clause 6.4.2 has changed in N293-12 to replace the acronym "FSSA" by "FPA", which is editorial. The FSSA modelled sufficient equipment and relevant support services to satisfy these requirements. Analysis was done for all credible fire scenarios in fire zones where the components or cables associated with the safe shutdown equipment are located. The completion of the FSSAs has also demonstrated that, with some	

CSA N293-12 Clause	PSR2 Review	Compliant or Gap
	corrective actions implemented, a single fire will not result in loss of required safety functions as defined in sub-clause 5.4.1. New cable tray routing is designed to follow these separation requirements to the extent practical. All recommendations of the Fire Protection Assessment reports have been implemented or dispositioned (per NA44-CORR-00531-07475 R000 [B.2-53], NK30-CORR-00531-06731 R000 [B.2-28]). Therefore, the intent of this clause is met and there is no PSR2 gap. • Clause 6.5 specifies requirements for reducing the spread of fire. The sub-clauses were not assessed in the 2015 Pickering A and B N293-07 reviews ([B.2-21], [B.2-47]) and are discussed below: o Sub-clause 6.5.1 specifies requirements for storage of combustible materials (including fire resistance rating). Sub-clause 6.5.1.1 has changed in N293-12 to provide clarification on the minimum fire resistance rating required for the fire separations serving combustible material storage rooms. This change resulted in the removal of N293-07 sub-clause 6.5.1.3 which stated this same requirement. Sub-clause 6.5.1.1 sets general requirements for the design of storage rooms, and the Pickering CCRs and FHAs did not identify any non-compliance with these requirements. For sub-clause 6.5.1.2, the adequacy of storage of combustible fluids was evaluated during the CCRs. All recommendations of the CCR reports have been implemented or dispositioned (per NA44-CORR-00531-07475 R000 [B.2-53], NK30-CORR-00531-06731 R000 [B.2-28]). Therefore, the intent of this sub-clause is met and there is no PSR2 gap. o Sub-clause 6.5.2 specifies requirements for fire stopping. All sub-clauses are the same as N293-12, except 6.5.2.1 which has changed in N293-12 to add structural supports to the list of penetration examples. FHAs and CCRs were completed for Pickering NGS which meet the requirements of N293-07. These assessments indicated that, at almost all locations of the plant, there is an appropriate combination of fire prevention, fire detection and suppression, and provisions for limiting or mitigating the effects of fire. Modifications to each unit have been made to correct any exceptions. All recommendations of the Fire Protection Assessment reports have been implemented or dispositioned (per NA44-CORR-00531-07475 R000 [B.2-53], NK30-CORR-00531-	

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	06731 R000 [B.2-28]). Further, information on fire stops is available in various Pickering NGS documentation and is utilized and cited as required in the CCRs ([B.2-17], [B.2-43]), FHAs ([B.2-23], [B.2-48]) and FSSAs ([B.2-24], [B.2-49]). These assessments demonstrate the fire stops adequately limit fire propagation so nuclear safety objectives are met. Therefore, the intent of this sub-clause is met and there is no PSR2 gap. o Sub-clause 6.5.3 specifies requirements for layout of cable trays, and is unchanged in N293-12. The FSSAs ([B.2-24], [B.2-49]) have identified locations where additional separation or protection was required between cable trays. Modifications have been implemented to resolve identified deficiencies as required. All recommendations of the Fire Protection Assessment reports have been implemented or dispositioned (per NA44-CORR-00531-07475 R000 [B.2-53], NK30-CORR-00531-06731 R000 [B.2-28]). Further, the CCRs, FHAs and FSSAs demonstrate that cable trays and risers are located sufficiently away from fire hazards that nuclear safety objectives are met. Therefore, the intent of this sub-clause is met and there is no PSR2 gap. o Sub-clause 6.5.4 specifies requirements for fire protection of structures, and is unchanged in N293-12. Fire protection measures are in place to protect structures, including active protection measures designed and installed in accordance with industry standards measures (e.g., turbine building), passive protection measures and operational protection measures. Similar to the compliance assessment for this clause carried out for Darlington NGS (NK38-REP-03680-10128 R003 [B.2-55]), these measures are commensurate with the reduced requirements for industrial occupancies characterized by the low density of employee population relative to offices or residential buildings. The volume of open space in the Powerhouse is such that, with the exception of the immediate fire area, the remainder of the space would not reach untenable exposure conditions for structural fire safety. Major internal fires have been analyzed for structural impact as documented in the FHAs and indicate low potential for structural damage. Therefore, the intent of this sub-clause is met and there is no PSR2 gap.	

CSA N293-12 Clause	PSR2 Review	Compliant or Gap
	• <u>Clause 6.6</u> specifies requirements for life safety. The associated sub-clauses were not assessed in the 2015 Pickering A and B N293-07 reviews ([B.2-21], [B.2-47]) and are discussed below: ○ Sub-clause 6.6.1 specifies requirements for egress routes, and is unchanged in N293-12. Station airlock controls and lighting are provided power from uninterruptible power supplies with battery and standby generator back up. The Stations' standby generators, in conjunction with station batteries, provide the capability for more than 2 hours of emergency lighting. For the Pickering A CCR [B.2-43], there were no code deviations identified for emergency lighting. For the Pickering B CCR [B.2-17], modifications have been made to ensure emergency lighting levels are as per code. The adequacy of fire exits was evaluated in the CCRs. For the Pickering A CCR, two egress route Deviations requiring action were identified (#'s 2010-501 and -502). For the Pickering B CCR, five egress route Deviations requiring action were identified (#'s 03002, 03301, 05601, 2010-0501 and 2010-0502). All recommendations of the CCRs have been implemented or dispositioned (per NA44-CORR-00531-07475 R000 [B.2-53], NK30-CORR-00531-06731 R000 [B.2-28]). Therefore, the intent of this sub-clause is met and there is no PSR2 gap. ○ Sub-clause 6.6.2 specifies requirements for egress from containment structures, and is unchanged in N293-12. The adequacy of fire exits was evaluated during the Pickering A and B CCRs ([B.2-17], [B.2-43]), and no deviations requiring changes were identified. The airlocks are provided with the capability for manual operation at all times should an abnormal event such as a fire result in loss of power-assisted operation (e.g., per Design Manual NA44-21130 R000c [B.2-62]). Therefore, the intent of this sub-clause is met and there is no PSR2 gap. ○ Sub-clause 6.6.3 specifies requirements for access for firefighting, and is unchanged in N293-12. This sub-clause provides exemption from NBCC, Division B, Clause 3.2.5.1(1): "where compensatory measures acceptable to the AHJ are provided for window and access panel openings". The intent of the NBCC sub-clause in question is to ensure adequacy of response to fire. This is demonstrated by Appendix O of the FHAs ([B.2-23], [B.2-48]) which assess the capability of the Station's emergency response by the Emergency Response	

CSA N293-12 Clause	PSR2 Review	Compliant or Gap
	Team (ERT). Therefore, the intent of this sub-clause is met and there is no PSR2 gap. • Clause 6.7 specifies requirements for maintaining plant operation during a fire. The associated sub-clauses were not assessed in the 2015 Pickering A and B N293-07 reviews ([B.2-21], [B.2-47]) and are discussed below. ○ Sub-clause 6.7.1 specifies requirements for the separation, smoke infiltration and contaminated air content for the control room complex, and is unchanged in N293-12. As per the Pickering A FHA [B.2-48], Section 4.4: "The results of this assessment determined that the existing barriers would provide a 2 h fire resistance rating around the MCR Complex on the 274'-0" elevation." For Pickering B, an equivalent statement is not contained in the FHA [B.2-23]. However, it is demonstrated in Appendix N of the Pickering B FHA that safe shutdown can be achieved or that fire prevention by design (e.g., qualified cables) and procedures (e.g., transient combustible controls) would prevent fire spread. Further, as discussed earlier under sub-Clause 5.7.8.2, the Pickering B N293-07 review [B.2-21] states that the "Impact of fires on the MCR [Main Control Room] envelope has been assessed in the updated FHA report (NK30-REP-71400-10002 R002 [B.2-23]). As there were no deviations reported, the station may be considered in compliance with this clause." As noted in FHA Section 4.5.1 for both Pickering A and B ([B.2-48], [B.2-23]): "Under abnormal conditions such as when smoke is detected in the control equipment rooms, the air conditioning system can be manually switched to provide smoke clearing capability from the rooms. Each control equipment room is equipped with a VESDA [Very Early Smoke Detection Apparatus] air sampling detection system that is interlocked with the air conditioning system to lock out the supply air fans upon detection of smoke, thus stopping the operation of the entire system. Following the appropriate procedures to initiate smoke clearing, the switch on the control panel can be turned to the ON position thereby initiating the return air fan." While the FHAs do not demonstrate that the MCRs will not have more than 1% of contaminated air for a 2 hour period following a fire as required by sub-clause 6.7.1.3, they state the following: "In the event that the Control Centre becomes uninhabitable due to smoke or a toxic gas, MCR	

CSA N293-12 Clause	PSR2 Review	Compliant or Gap
	Uninhabitable, Abnormal Incidents Manual, NA44-AIM-014-09013-09, will be implemented to allow for an orderly planned shutdown" [B.2-48], and "a fire impacting cables and devices in the control equipment rooms would not affect the station capabilities to shut down through the UECCs [Unit Emergency Control Centres]" [B.2-23]. Thus, the Pickering MCRs are sufficiently protected in the event of a fire that nuclear safety objectives are met. The intent of this sub-clause is met and there is no PSR2 gap. o Sub-clause 6.7.2 specifies requirements for travel routes between control rooms, and is unchanged in N293-12. The concept of a secondary control room in the context of CSA N293 is taken to be the location from where fire mitigation activities would be orchestrated in the event that the MCR becomes un-inhabitable. In the case of Pickering this is the Emergency Operating Centre (EOC) located in the station's service wing. There are multiple independent means of travelling to the EOC from the MCR. Furthermore, CCRs have assessed conformance of Pickering NGS with requirements for emergency lighting. All recommendations of the CCR reports have been implemented or dispositioned (per NA44-CORR-00531-07475 R000 [B.2-53], NK30-CORR-00531-06731 R000 [B.2-28]). Therefore, the intent of this sub-clause is met and there is no PSR2 gap. • <u>Clause 6.8</u> specifies design requirements for fire prevention. The 6.8 sub-clauses that were not assessed in the 2015 Pickering A and B N293-07 reviews ([B.2-21], [B.2-47]) are discussed below: o Sub-clause 6.8.1 specifies requirements related to combustible materials in buildings and interior finishes. All sub-clauses are unchanged in N293-12, except for sub-clause 6.8.1.1 on control of combustible materials which has replaced "non-combustible materials" with the more general "non-combustible construction". Extensive evaluations have been conducted of fixed and transient combustibles on a room by room basis for the FHAs and FSSAs ([B.2-23], [B.2-48] and [B.2-24], [B.2-49], respectively). The FHAs and the FSSAs have concluded that under present conditions, the safety objectives of the Station are maintained. Evaluations of combustibles in the FHAs and FSSAs have not identified any issues related to exposed foam plastics that would adversely impact meeting	

CSA N293-12 Clause	PSR2 Review	Compliant or Gap
	safety objectives. With respect to sub-clause 6.8.1.4 which relates to requirements for interior finishes, although it cannot be conclusively stated that the Station meets all the detailed requirements of this sub-clause, the Pickering NGS CCRs ([B.2-17], [B.2-43]), FHAs ([B.2-23], [B.2-48]) and FSSAs ([B.2-24], [B.2-49]) have demonstrated that the plant as built has sufficient protection from fire initiation and spread that nuclear safety objectives are met. Therefore, the intent of this sub-clause is met and there is no PSR2 gap. o Sub-clause 6.8.2 specifies design requirements to facilitate control of transient materials, including requirements for storage and laydown areas, storage facilities and storage rooms. All sub-clauses are unchanged in N293-12, except for sub-clause 6.8.2.5 which has changed in N293-12 to incorporate improved wording for clarity. The Pickering NGS FHAs and FSSAs ([B.2-23], [B.2-48] and [B.2-24], [B.2-49], respectively) analyzed the storage rooms, laydown areas and transient combustible hazards and the potential impact on credited safe shutdown equipment. The FHAs also included a review of the Station's process controls for the use and storage of transient materials. The FHAs and FSSAs did not identify any findings regarding the storage and laydown areas at the Station. As discussed earlier, control of transient materials is governed by N-PROC-RA-0054 R015, "Control of Space Allocation for Transient Material and Extended Storage of Material within the Site" [B.2-60] and the impact of transient material on fire protection is conducted in accordance with N-GUID-09076-10001 R004, "Guideline for Fire Protection Reviews of Space Allocation and Transient Material Permits" [B.2-61]. Further, as part of the FSSAs and FHAs, combustible materials (fixed and transient) were documented on a room by room basis and fire scenarios developed to analyze the impacts and consequences of fires involving combustibles on fire safety objectives. The locations of such materials, for the purposes of the evaluations, were conservatively adjusted to achieve the greatest impact on safety. Based on the results of the evaluations, it was concluded that under the reviewed configuration, the safety objectives of the Station are met. Based on the above, the intent of this sub-clause is met and there is no PSR2 gap.	

CSA N293-12 Clause	PSR2 Review	Compliant or Gap
	○ Sub-clause 6.8.3 specifies requirements for control of combustible materials in Heating, Ventilation and Air Conditioning (HVAC) equipment, including requirements for air handling ducts and combustibility of air filter media. All sub-clauses are unchanged in N293-12. With respect to air handling ducts, the basic materials of the ventilation system ductwork are galvanized sheet metal and steel plate with sheet metal joints and welded joints. These materials are non-combustible. As per the CCRs (e.g., NK30-REP-71400-10001 R001 for Pickering B [B.2-17]), the station complies with NFPA 90A-1964 item 151 (a) which states "Air filters shall be of approved types that will not burn freely or emit large volumes of smoke or other objectionable products of combustion when attacked by flames." Therefore, the intent of this sub-clause is met and there is no PSR2 gap. ○ Sub-clause 6.8.4 specifies requirements for the control of combustible materials in electrical equipment and cables, including minimizing flame spread. All sub-clauses are unchanged in N293-12, except sub-clause 6.8.4.4 which has changed to include a specific reference to Clause 4.11.4 of CSA C22.2. The FHAs ([B.2-23], [B.2-48]) indicate that at all locations of the plant there is an appropriate combination of fire prevention, fire detection and suppression, and provisions for limiting or mitigating the effects of fire. Further, the FSSAs ([B.2-24], [B.2-49]) analyzed failure of all credited electrical cabinets to assess the impact of fire on achievement of safe shutdown goals. In the event the electrical cabinets were located in a room where a hot gas layer would form, the fire impacts were assumed to result in damage of all safe shutdown credited components within the room. The results of these evaluations demonstrated that the safe shutdown goals would be met. With respect to CSA C22.2, although not all electrical cables meet the 1.5 m criterion specified in sub-clause 6.8.4.4 (e.g., 2 m for polyvinyl chloride jacketed cables past the point of flame impingement, as per Design Manual NK30-DM-57000-00001 R000, "Cabling" [B.2-63]), the cables meet Ontario Hydro test standards and are considered acceptable. Therefore, the intent of this sub-clause is met and there is no PSR2 gap. ○ Sub-clause 6.8.5 specifies requirements for control of flammable liquids and combustible liquids. All sub-clauses are unchanged in N293-12 except for	

CSA N293-12 Clause	PSR2 Review	Compliant or Gap
	numbering (which was necessary since N293-07 Clause 6.8.5 was removed in N293-12). The FSSAs and FHAs assessed the major flammable and combustible liquid hazards at Pickering NGS. Equipment containing combustible liquids was assessed for containment and potential fire spread in the FHAs and FSSAs ([B.2-23], [B.2-48] and [B.2-24], [B.2-49], respectively). The effect of an uncontrolled spill fire and a fire involving the turbine generator on FSSA-credited equipment and cables was also assessed. The results indicated that the safety goals would be met. Therefore, the intent of this sub-clause is met and there is no PSR2 gap. o Sub-clause 6.8.6 specifies requirements for control of gases. All sub-clauses are unchanged in N293-12. Severe hydrogen/deuterium burns are prevented by design features such as: a) the Hydrogen Ignition System in containment to enable combustion of flammable mixtures at low concentrations, b) the catalytic recombination unit in the Liquid Zone System to reduce the concentration of radiolytic hydrogen in the helium to an acceptable level, and c) recombination units and flame arrestors in the moderator cover gas system. Further, compressed gas cylinders and piping were analyzed as part of the Pickering NGS CCRs ([B.2-17], [B.2-43]) from a life safety perspective and in the FHAs/FSSAs ([B.2-23], [B.2-48] and [B.2-24], [B.2-49], respectively) from a nuclear safety perspective. Based on the design, locations and safe storage arrangements, no discrepancies were identified from these assessments. Therefore, the intent of this sub-clause is met and there is no PSR2 gap. o Sub-clause 6.8.7 specifies requirements for bulk storage of dangerous goods. This sub-clause is unchanged in N293-12. The Pickering NGS CCRs and FHAs reviewed major hazards at the Station which included the location, storage, quantity, and potential exposure conditions. No deviations pertaining to bulk storage of dangerous goods were identified as a result of these assessments. Therefore, the intent of this sub-clause is met and there is no PSR2 gap. o Sub-clause 6.8.8 specifies requirements for storage of radioactive materials. All sub-clauses are unchanged in N293-12. The Pickering NGS FHAs conducted an evaluation of the radioactive material storage rooms. These evaluations reviewed the construction, ventilation and drainage provided for these rooms. Areas for improvement were	

CSA N293-12 Clause	PSR2 Review	Compliant or Gap
	identified and dispositioned to ensure adequate controls are provided for these rooms to satisfy the intent of this clause. All recommendations of the Fire Protection Assessment reports have been implemented or dispositioned (per NA44-CORR-00531-07475 R000 [B.2-53], NK30-CORR-00531-06731 R000 [B.2-28]). Therefore, the intent of this sub-clause is met and there is no PSR2 gap. o Sub-clause 6.8.9 specifies requirements for the control of ignition sources. All sub-clauses are unchanged in N293-12. Existing ignition sources were identified for the Pickering NGS FHAs ([B.2-23], [B.2-48]) and found to meet fire protection goals. Process controls for operations that present an ignition source hazard were confirmed to be established and used at the Station (see N-PROC-RA-0057 R008, "Control of Ignition Sources and Hot Work Activities" [B.2-64]). Adequate protection against lightning is provided for all structures as per Ontario Hydro Grounding standards and practices. New structures would be assessed against NFPA 780 as called for in CSA N293-07 and -12. External fire hazards (e.g., Standby Generator Fuel Storage, Power Transformer, site vehicle and site vegetation fires) were also assessed and documented in the FSSA and FHA reports. These external hazard assessments determined that the nuclear safety criteria specified in N293 would not be impacted. Therefore, the intent of this sub-clause is met and there is no PSR2 gap.	
7 Design and Installation Requirements for Fire Protection Systems 7.1 General 7.2 Fire Alarm Systems 7.3 Fire Suppression 7.4 Seismic Qualification	Clause 7 of N293-12 specifies requirements for design and installation of fire protection systems. None of the Section 7 sub-clauses were assessed in the 2015 Pickering A and B N293-07 reviews ([B.2-21], [B.2-47]). With respect to the individual sub-clauses: • Clause 7.1 specifies requirements for qualification of fire protection devices and equipment. All sub-clauses are unchanged in N293-12. The Pickering NGS CCRs ([B.2-17], [B.2-43]) and FHAs ([B.2-23], [B.2-48]) reviewed the fire protection systems for appropriateness, effectiveness and reliability. Part of this assessment involved the review of system components from design documentation to confirm they were from accredited organizations. No deviations or findings were noted in these assessments. Therefore, the intent of this clause is met and there is no PSR2 gap.	Gap **PSR2 CSA N293-12 Gap #1** - Clause 7.2.1.10.1 on Fire Alarm System Control **PSR2 CSA N293-12 Gap #2** - Clause 7.2.1.13 on Fire Endurance

CSA N293-12 Clause	PSR2 Review	Compliant or Gap
	• Clause 7.2 specifies requirements for fire alarm systems: o Sub-clause 7.2.1 specifies general fire alarm system concepts, including requirements relating to use of early warning technology, voice communication systems, staged operation, power supplies for fire alarm and voice communication systems, display and control centres, and extinguishing-agent-releasing systems. A number of the sub-clauses have changed in N293-12, but all changes are editorial except for assignment of the very early warning fire detection installation requirements (from N293-07 Clause 7.2.1.1) to a separate N293-12 sub-clause 7.2.1.2. Pickering NGS Fire Protection System Design Manuals NA44-DM-71400.2-00001 R001, "Fire Protection – Smoke Detection" [B.2-65], NA44-DM-71400-00002 R000, "Fire Protection Systems (Water)" [B.2-66], NK30-DM-67140-00001 R011, "Fire Detection - Fire Protection System" [B.2-67] and NK30-DM-71400-00001 R006, "Fire Protection System" [B.2-68] state that the fire alarm systems are designed and commissioned to the requirements of CAN/ULC-S524 and S537 as required by sub-clause 7.2.1.1. As per Safety Report NA44-SR-01320-00001 R015 [B.2-69], Pickering A is equipped with a public address system for general communication of messages and emergency signals to station personnel throughout the plant. Tone alarm signals can be initiated over the public address system by the unit operators in the MCRs. An emergency tone takes priority over all other forms of paging with a fire alarm tone taking second priority. Similar provisions apply to Pickering B. Two-way radios are provided as the main means of communication between emergency responders and the shift supervisor with telephone backup in the event of inadequate radio reception. The fire alarm systems at the Station consist of alarm devices that alarm to a control panel in the MCR, a Public Address system to provide notification, and a dedicated ERT to investigate alarms and respond as required. The automatic fire suppression systems in use at Pickering NGS are equipped with hardware listed for use as an extinguishing-agent releasing system. Data Gathering Panels and the Display Annunciation Stations (DAS) are provided un-interruptible Class II	of Electrical Conductors **PSR2 CSA N293-12 Gap #3** - Clause 7.3.2.2 (d) on Fire Pumps

CSA N293-12 Clause	PSR2 Review	Compliant or Gap
	120Vac regulated electrical power (as per NA44-DM-71400.2-00001 R001 [B.2-65] and NK30-DM-67140-00001 R011 [B.2-67]). Clause 7.2.1.10.1 states: "A display and control centre shall be located in the MCR… capable of providing detailed information on the location and nature of the signal. In addition, the panel operator shall be able to control the fire alarm system without having to leave his or her station." Per Section 3.2.3 of Design Manual NK30-DM-67140-00001 R011 [B.2-67], DAS 058-67140-TVM2 in the EOC (254' Service Wing Extension) provides the ability to control the fire alarm system. DAS 058-67140-TVM1 installed in the Pickering 058 MCR is capable of providing fire alarm system annunciation. While TVM1 does not have control capability, it "can be configured to provide the control capability by changing the software key at the back of TVM1" [B.2-67]. Pickering 014 DAS 014-67140-WS2342 in the EOC is capable of providing annunciation only, and there is no DAS in the Pickering 014 MCR (although there is limited annunciation). Therefore, this is identified as a PSR2 gap (**PSR2 CSA N293-12 Gap #1**). Clause 7.2.1.13 states: "Electrical conductors that are installed in service spaces containing other combustible materials and that are used in connection with fire alarm systems and emergency equipment, including fire alarm cables… shall be capable of performing their intended functions for not less than 1 hour after the start of a fire." Modifications to the Fire Protection System meet the requirements of CAN/ULC-S524 which mandates a 1 hour fire rating as described in Section 2.5 of NA44-DM-71400.2-00001 R001 [B.2-65], Section A.2 of NA44-DM-71400-00002 R000 [B.2-66], and Section 2 of NK30-DM-71400-00001 R006 [B.2-68]. This is achieved by the use of Edwards System Technology (EST) that connects the fire alarm control panels via a data communication link with dual redundant circuit wiring paths. However, existing Pyrotronics fire alarm control panels (not replaced in fire protection upgrade projects) are not similarly connected and, hence, may be susceptible to loss of alarm signal due to spot burning of a cable. While measures such as lack of combustible material in service spaces, combustible transient material control practices, and inherent protection afforded by Pickering NGS cable routing practices used in the	

CSA N293-12 Clause	PSR2 Review	Compliant or Gap
	Fire Protection systems mitigate the lack of such a feature, it could not be confirmed based on existing documentation that all essential fire alarm cables are capable of performing their intended functions for not less than 1 hour after the start of a fire to meet the requirement of N293-12 sub-clause 7.2.1.13. Therefore, this is identified as a PSR2 gap (**PSR2 CSA N293-12 Gap #2**). o Sub-clause 7.2.2 specifies requirements for input devices, including manual pull stations, early warning fire detection technology and alternate fire detection methods. Sub-clause 7.2.2.4 has changed in N293-12 to replace the acronym "FHA" with "FPA", which is editorial. Manual pull stations are not provided in areas other than the Service Wing Extension of the station. Instead, the station emergency telephone system is used. This has been assessed and accepted as identified in Third Party Review reports NA44-CORR-00531-05151 R000 [B.2-70] and NK30-REP-71400-00011 R000 [B.2-71], as described in the CCRs ([B.2-17], [B.2-43]). The Pickering NGS FSSAs and FHAs demonstrate, with technical justification, the acceptability of fire detection methods installed in the Station to achieve safe shutdown goals. Smoke detectors are provided in the MCR that transmit alarm signals to the Very Early Smoke Detection (VESDA) panels as per NA44-DM-71400.2-00001 R001 [B.2-65] and NK30-DM-67140-00001 R011 [B.2-67]. Actions to take if the system is out of service are identified in the operating documentation as per standard content of Operating Manuals. Therefore, the intent of this sub-clause is met and there is no PSR2 gap. o Sub-clause 7.2.3 specifies requirements for output devices, including audible and/or visual fire detection and alarm signal devices and telephone handsets. All sub-clauses are unchanged in N293-12. The Pickering NGS CCRs, FHAs and FSSAs demonstrate acceptability of fire detection methods installed in the Station to achieve safe shutdown goals. Design Manuals on Communications, e.g., NK30-DM-60200-00001 R001, "Communications" [B.2-72], demonstrate compliance with this Clause. Therefore, the intent of this sub-clause is met and there is no PSR2 gap.	

CSA N293-12 Clause	PSR2 Review	Compliant or Gap
	• Clause 7.3 provides requirements for fire suppression. ○ Sub-clause 7.3.1 specifies general fire suppression requirements, including requirements relating to selection of a fire suppression system (via review of the design basis fire, performance levels, reliability, and potential damage resulting from fire suppression agents), automatic sprinkler systems, and design, installation, and registry of fire suppression systems in accordance with pressure-retaining component requirements. Some sub-clauses were changed in N293-12 involving minor editorial revisions (e.g., replacement of "FHA" with "FPA"). Design basis fires have been analyzed as part of the FSSAs and FHAs ([B.2-23], [B.2-48] and [B.2-24], [B.2-49], respectively) with and without suppression credited. The FHAs also reviewed the appropriateness and effectiveness of fire suppression systems. Performance and reliability requirements have been taken into consideration for all fire suppression systems (see NA44-DM-71400-00002 R000 [B.2-66] and NK30-DM-71400-00001 R006 [B.2-68]). The potential damage resulting from the fire suppression agent was analyzed in the FSSAs and documented under the heading "Fire Protection System Operation". As per Design Manuals NA44-DM-71400-00002 [B.2-66] and NK30-DM-71400-00001 [B.2-68], fire suppression systems are provided at Pickering NGS. More generally, the CCRs, FHAs and FSSAs demonstrate acceptability of fire suppression methods installed in the Station to achieve safe shutdown goals. Pressure retaining components of the fire suppression systems have been designed and installed per ASME B31.1 "Power Piping Code" (or ASME B31.3 "Process Piping Code"), CSA B51 "Boiler, Pressure Vessel and Pressure Piping Code" and N285.0 "General Requirements for Pressure Retaining Components in CANDU Nuclear Power Plants" (per NA44-DM-71400-00002 R000 [B.2-66] and NK30-DM-71400-00001 R006 [B.2-68]). Therefore, the intent of this sub-clause is met and there is no PSR2 gap. ○ Sub-clause 7.3.2 specifies requirements for the fire protection water supply, including sources of water, fire protection water supply volume, the ability to draft water from the supply source with fire trucks, and the use of diesel fire pumps in accordance with	

CSA N293-12 Clause	PSR2 Review	Compliant or Gap
	NFPA 20. Some sub-clauses have changed in N293-12 involving minor editorial revisions (e.g., replacement of "FHA" with "FPA"). Sub-clause 7.3.2.1.2 has changed in N293-12 to clarify that 88.9 mm (3.5 inch) and larger hoses need only be included in the hose demand when required by the Station's pre-fire plans. The sources of water used in the fire protection systems at Pickering NGS meet the requirements of this clause. The primary supply of water is Lake Ontario. As per NK30-DM-71400-00001 R006 [B.2-68], at Pickering B "the water supply is sized to accommodate the most demanding calculated design flow". As per NA44-DM-71400-00002 R000 [B.2-66], Pickering A meets the requirements of N293-95 whose equivalent Clause 6.4.4 states: "the water supply system shall be designed to supply the most demanding fire-extinguishing system plus a supply for hose streams". Both Pickering Units 1,4 and Units 5-8 meet OPG Engineering Standard N-STM-78220-10000 R001, "Fire Protection for Turbine Generator Area" [B.2-73] which requires a hose stream allowance of 750 USgpm. Pickering Units 1,4 also meet N-STM-67140-10000 R001, "Fire Protection for Main Control Room, Control Equipment Room and Cable Spreading Areas" [B.2-74]. Even if the hose stream demand were as in N293-95 (500 USgpm), instead of as in N293-12 (750 USgpm), this would not be considered to impact the capability of the fire protection systems to the extent that nuclear safety objectives are jeopardised. The Pickering station has the ability to draft water from the forebay. The capability of fire response actions has been assessed in the FHAs and FSSAs and shown to be adequate to meet nuclear safety objectives. However, Clause 7.3.2.2 (d) of CSA N293-12 states that: "At a minimum, the fire protection water pumping system shall consist of at least one diesel-engine-driven fire pump and one electric-motor-driven fire pump set, with each pump set being capable of providing, the flow rate and pressure specified in Item (a)". This Clause is met at Pickering Units 1,4 with the provision of diesel-driven firewater pumps, backed up by supplies from the High Pressure Service Water (HPSW) system (as noted in the Pickering A Safety Report NA44-SR-01320-00001 R015, Section 11.5.1.1). It is not met	

CSA N293-12 Clause	PSR2 Review	Compliant or Gap
	at Pickering Units 5-8, where the Fire Protection System comprises the HPSW supplies from the four B-side units. As a result, Pickering Units 5-8 does not comply with Clause 7.3.2.2 (d) of CSA N293-12 and this has been identified as a PSR2 gap (**PSR2 CSA N293-12 Gap #3**). ○ Sub-clause 7.3.3 specifies requirements for automatic and manual water-based fire suppression systems, including automatic sprinkler system design and installation (in accordance with NFPA 13 and NFPA 15), water hose stream allowance, measures for diking and/or drainage and transformer fires. No sub-clauses have changed in N293-12 except sub-clause 7.3.3.5, which has added a new requirement for passive heat barriers to protect cables. Modifications to the Fire Protection System have been made to meet the requirements of NFPA 13 and NFPA 15, per Design Manuals NA44-DM-71400-00002 R000 [B.2-66] and NK30-DM-71400-00001 R006 [B.2-68]. Water hose stream allowance has been added to sprinkler system demand as per Design Manuals NA44-DM-71400-00002 R000 [B.2-66] and NK30-DM-71400-00001 R006 [B.2-68]. The new requirement for passive heat barriers to protect cables at Pickering NGS could not be confirmed to be fully met. However, this does not translate into a measurable reduction in the plant's fire suppression capability or nuclear safety margins, since all cables requiring heat protection were identified in the FSSA and provided with heat shields. Measures for diking and/or drainage have been provided as required and assessed in the CCRs, FHAs and FSSAs. Nuclear safety objectives were demonstrated to be met. As discussed earlier, the FSSAs and FHAs analyzed potential transformer fires and determined that fire suppression is sufficient such that there is no impact on fire safe shutdown capabilities. Therefore, the intent of this sub-clause is met and there is no PSR2 gap. ○ Sub-clause 7.3.4 specifies requirements for special extinguishing systems. One sub-clause has changed in N293-12 to use the acronym "FSA" instead of "FHA", which is editorial. Special extinguishing systems are in use at Pickering NGS and have been	

CSA N293-12 Clause	PSR2 Review	Compliant or Gap
	determined in the FHAs ([B.2-23], [B.2-48]) to be suitable for the hazards they are protecting against. Therefore, the intent of this sub-clause is met and there is no PSR2 gap. o Sub-clause 7.3.5 specifies requirements for portable extinguishers. The clause has not changed in N293-12. Portable fire extinguishers are provided at Pickering NGS for use by plant staff prior to ERT response. Adequacy of coverage has been assessed in the CCRs ([B.2-17], [B.2-43]). All recommendations of the FHAs related to portable fire extinguishers have been implemented or dispositioned (per NA44-CORR-00531-07475 R000 [B.2-53], NK30-CORR-00531-06731 R000 [B.2-28]). Therefore, the intent of this clause is met and there is no PSR2 gap. o Sub-clause 7.3.6 specifies requirements for fire hydrants. The clause has not changed in N293-12. Fire hydrants are provided in outdoor areas. Pickering NGS adequacy of coverage has been assessed in the CCRs ([B.2-17], [B.2-43]) and found to be compliant with applicable codes. Fire hydrants at Pickering NGS are clearly enough marked that no deviations have been found in the CCRs, or any adverse impact noted by the FHAs and FSSAs on meeting safe shutdown requirements. Although the specified maximum spacing of 250 feet (per Sub-clause 7.3.6.2) is not met, the provided 300 foot spacing at Pickering A and B (per NA44-DM-71400-00002 R000 [B.2-66] and NK30-DM-71400-00001 R006 [B.2-68]) does not translate into a measurable reduction in the plant's fire suppression capability or nuclear safety margins. Therefore, the intent of this sub-clause is met and there is no PSR2 gap. o Sub-clause 7.3.7 specifies requirements for standpipes. The clause has not changed in N293-12. The Pickering fire protection Design Manuals (NA44-DM-71400.2-00001 R001 [B.2-65], NA44-DM-71400-00002 R000 [B.2-66], NK30-DM-67140-00001 R011 [B.2-67] and NK30-DM-71400-00001 R006 [B.2-68]) state that standpipes in the station meet the requirements of NFPA 14. At Pickering A, a 750 USgpm hose stream allowance is added to the Turbine-Generator sprinkler system demand. At Pickering B, NK30-DM-71400-00001 R006 [B.2-68] states "each standpipe shall be capable of delivering a minimum of 84 USgpm at 45 psi from its most remote hose connection". Standpipe systems inside	

CSA N293-12 Clause	PSR2 Review	Compliant or Gap
	containment are not provided. Although some details of sub-clause 7.3.7 are not met, sufficient fire protection capability has been provided to be able to demonstrate by means of the FHAs and the FSSAs that safe shutdown capability is not impacted and nuclear safety objectives are met. Therefore, the intent of this sub-clause is met and there is no PSR2 gap. o Sub-clause 7.3.8 specifies requirements for manual firefighting. A number of sub-clauses have changed in N293-12 to replace the acronym "FHA" by "FPA", which is editorial. Manual firefighting provisions are in place at the Pickering NGS per N-PROG-RA-0012 R011, "Fire Protection" [B.2-9] and N-STD-RA-0039 R004, "Requirements for Fire Response Capability" [B.2-75]. Therefore, the intent of this sub-clause is met and there is no PSR2 gap. • Clause 7.4 provides requirements for seismic qualification of fire protection systems. This clause has changed in N293-12 to further clarify the seismic design requirements. Sub-clauses 7.4.1, 7.4.2 and 7.4.3 specify general seismic qualification requirements, including the use of seismic categories A and B. The seismic qualification requirements for the Pickering NGS fire protection systems are provided in Design Manuals NA44-DM-71400-00002 R000 [B.2-66] and NK30-DM-71400-00001 R006 [B.2-68]. For Pickering Units 1,4 this includes qualification of the manual valves and piping in the Unit 4 Control Equipment Room (CER), Units 1/2 and 3/4 Auxiliary CERs, as well as the sprinkler piping above the Auxiliary Boiler Feed Pump at floor elevations 254' and 274' in the Turbine Generator Area, to Seismic Category "A". For Pickering Units 5-8 this includes seismic qualification of the Emergency Power and Water Supply Building Air Foam System and the High Pressure Emergency Coolant Pump House Standpipe System. Further, the FHAs ([B.2-23], [B.2-48]) confirm that the consequences of a Design Basis Earthquake are not outside those of the fire scenarios addressed in the FSSAs ([B.2-24], [B.2-49]) and FHAs. Thus, the impact of seismic events on fire protection systems is considered at Pickering NGS and it is confirmed that nuclear safety objectives are met. The safe operation of the Pickering reactors following seismic events has been extensively assessed and mitigating systems and components that are required to function have been identified, along with the degree of their required functionality, i.e., whether Category A or	

CSA N293-12 Clause	PSR2 Review	Compliant or Gap
	Category B. This includes fire protection systems to the extent required. Based on the above, Pickering NGS complies with the intent of Clause 7.4 which is to ensure reactor safety following seismic events, and there is no PSR2 gap.	

Based on the information provided in Section B.2.2 above, there are three PSR2 CSA N293-12 gaps for Pickering NGS compliance with Clauses 7.2.1.10.1, 7.2.1.13 and 7.3.2.2 (d) of CSA N293-12.

B.2.3 Compliance Summary for Pickering PSR2

There are three PSR2 gaps for CSA N293-12 [B.2-1] which relate to Safety Factor 1 (Plant Design):

1. Clause 7.2.1.10.1 of CSA N293-12 states: "A display and control centre shall be located in the MCR [Main Control Room]… capable of providing detailed information on the location and nature of the signal. In addition, the panel operator shall be able to control the fire alarm system without having to leave his or her station." Pickering 014 Display Annunciation Station 014-67140-WS2342 in the Emergency Operating Centre is capable of providing annunciation only, and there is no Display Annunciation Station in the Pickering 014 MCR (although there is limited annunciation). Therefore, this has been identified as a PSR2 gap.

2. Clause 7.2.1.13 of CSA N293-12 states: "Electrical conductors that are installed in service spaces containing other combustible materials and that are used in connection with fire alarm systems and emergency equipment, including fire alarm cables… shall be capable of performing their intended functions for not less than 1 hour after the start of a fire." Modifications to the Fire Protection System meet the requirements of CAN/ULC-S524 which mandates a 1 hour fire rating as described in Section 2.5 of NA44-DM-71400.2-00001 R001, Section A.2 of NA44-DM-71400-00002 R000 and Section 2 of NK30-DM-71400-00001 R006. This is achieved by the use of Edwards System Technology (EST) that connects the fire alarm control panels via a data communication link with dual redundant circuit wiring paths. However, existing Pyrotronics fire alarm control panels are not similarly connected and, hence, may be susceptible to loss of alarm signal due to spot burning of a cable. While measures such as lack of combustible material in service spaces, combustible transient material control practices, and inherent protection afforded by Pickering NGS cable routing practices used in the Fire Protection systems mitigate the lack of such a feature, it could not be confirmed based on existing documentation that all essential fire alarm cables are capable of performing their intended functions for not less than 1 hour after the start of a fire to meet the requirement of N293-12 sub-clause 7.2.1.13. As a result, this has been identified as a PSR2 gap.

3. Clause 7.3.2.2 (d) of CSA N293-12 states: "At a minimum, the fire protection water pumping system shall consist of at least one diesel-engine-driven fire pump and one electric-motor-driven fire pump set, with each pump set being capable of providing, the flow rate and pressure specified in Item (a)". This Clause is met at Pickering Units 1,4 with the provision of diesel-driven firewater pumps, backed up by supplies from the High Pressure Service Water (HPSW) system (as noted in the Pickering A Safety Report NA44-SR-01320-00001 R015, Section 11.5.1.1). It is not met at Pickering Units 5-8, where the Fire Protection System is comprised of the HPSW supplies from the four units only. As a result, Pickering Units 5-8 does not comply with Clause 7.3.2.2 (d) of CSA N293-12 and this has been identified as a PSR2 gap.

B.2.4 References

[B.2-1] CSA Standard, N293-12, *Fire Protection for Nuclear Power Plants*, October 2012.

[B.2-2] CNSC Acts and Regulations web page, *Regulatory Documents,* http://cnsc-ccsn.gc.ca/eng/acts-and-regulations/regulatory-documents/index.cfm#R14http://cnsc-ccsn.gc.ca/eng/acts-and-regulations/regulatory-documents/index.cfm, retrieved: April 2016.

[B.2-3] CSA Standard, N293-07, *Fire Protection for CANDU Nuclear Power Plants*, February November 2008.

[B.2-4] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.2-5] CSA Impact Statement, *Notification of CSA N293-12*, February 2012.

[B.2-6] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.2-7] OPG Report, NK30-REP-03680-00005 R000, *Pickering NGS B - Integrated Safety Review - Safety Analysis Review*, June 2007.

[B.2-8] CSA Standard, N293-95, *Fire Protection for CANDU Nuclear Power Plants*, November 1995.

[B.2-9] OPG Procedure, N-PROG-RA-0012 R011, *Fire Protection*, July 28, 2015.

[B.2-10] OPG Report, NK30-REP-03680-00001-R000, *Pickering NGS-B Integrated Safety Review - Plant Design Safety Factor*, August 2007.

[B.2-11] CNSC Letter, NK30-CORR-00531-04876, *Pickering NGS-B Integrated Safety Review (ISR) - CNSC Review of Acceptable Deviations and Discrepancies for the Plant Design Safety Factor Report*, e-Docs # 3256609, June 27, 2008.

[B.2-12] OPG Letter, NK30-CORR-00531-04739, *Pickering NGS-B Integrated Safety Review – Discrepancy Resolution*, April 17, 2008.

[B.2-13] CNSC Letter, NK30-CORR-00531-05008, *Pickering NGS-B - Discrepancy Resolution for Ageing, Safety Analysis, Emergency Planning, Environment, Plant Design and Management Safety Areas*, e-Docs # 3297754, November 03, 2008.

[B.2-14] OPG Report, NK30-REP-03680-00016 R000, *OPG Response to CNSC Comments on Pickering NGS-B Integrated Safety Review – Plant Design, Safety Analysis, Safety Performance, Ageing and Equipment Qualification Safety Factors and Discrepancy Resolutions*, September 2009.

[B.2-15] OPG Report, NK30-REP-03680-00015 R000, *Pickering NGS-B Integrated Safety Review (ISR) – Final ISR Report*, August 2009.

[B.2-16] OPG Report, NK30-REP-71400-10001 R000, *Fire Protection Code Compliance Review Pickering Nuclear Generating Station B*, May 2000.

[B.2-17] OPG Report, NK30-REP-71400-10001 R001, *Fire Protection Code Compliance Review Pickering Nuclear Generating Station B*, November 23, 2010.

[B.2-18] OPG Report, NK30-REP-71400-0299410 R000, *Fire Safety Assessment: Definition of Scope of Work for CSA N293-07 Compliance at PNGS-B*, August 2009.

[B.2-19] OPG Report, NK30-REP-71400-00018 R000, *Fixed Fire Protection Systems Inspection, Testing and Maintenance Report*, November 2010.

[B.2-20] OPG Report, NK30-REP-71400-00027 R000, *Third Party Review: Fixed Fire Protection Systems Inspection, Testing and Maintenance Report*, February 23, 2011.

[B.2-21] OPG Report, NK30-REP-71400-00033 R000, *Pickering NGS 058 Compliance with CSA N293-07, Fire Protection for CANDU Nuclear Power Plants*, September 23, 2015.

[B.2-22] OPG Letter, NK30-CORR-00531-05774 R000*, Pickering NGS 'B' – Request for CNSC Acceptance of the CCR, the Third Party Review of the Inspection, Testing and Maintenance Report for Fixed Fire Protection Systems, the FSSA and the FHA,* December 9, 2011.

[B.2-23] OPG Report, NK30-REP-71400-10002 R002, *Fire Hazard Assessment - Pickering B Nuclear Generating Station*, November 23, 2011.

[B.2-24] OPG Report, NK30-REP-71400-00001 R002, *Fire Safe Shutdown Analysis - Pickering B Nuclear Generating Station*, October 5, 2011.

[B.2-25] OPG Letter, NK30-CORR-00531-06167 R000*, Pickering NGS 'B' - CNSC Acceptance of the Fire Hazard Assessment, Fire Safe Shutdown Analysis, Code Compliance Review and the Inspection Testing and Maintenance Report for Fixed Fire Protection Systems*, December 28, 2011.

[B.2-26] OPG List, P-LIST-71400-00001 R000, *Application of CSA N293-07 to Structures, Systems and Components for Pickering Nuclear*, July 16, 2009.

[B.2-27] OPG Plan, P-PLAN-09100-00001 R003, *Pickering Fire Safety Plan*, November 2015.

[B.2-28] OPG Letter, NK30-CORR-00531-06731 R000, *Pickering Units 5-8 – Status Update on CSA N293-07 Outstanding Items List*, December 20, 2013.

[B.2-29] OPG Plan, NK30-PLAN-00531-00001 R005, *Pickering 5-8 Continued Operations Plan*, December 2015.

[B.2-30] OPG Procedure, N-PROC-MP-0090 R014*, Modification Process*, October 14, 2016.

[B.2-31] OPG Guide, N-GUID-00700-10000 R015*, Guide to Modification Process*, September 30, 2016.

[B.2-32] OPG Form, N-FORM-10287 R005*, Fire Protection Impact Evaluation*, April 2016.

[B.2-33] OPG Standard, N-STD-RA-0038 R003*, Requirements of Fire Safety Assessments*, November 19, 2014.

[B.2-34] OPG Guide, N-GUID-09076-10002 R000*, Guide to Fire Protection Requirements for Design Modifications*, May 4, 2009.

[B.2-35] OPG Instruction, N-INS-09076-10004 R002*, Fire Protection Third Party Review*, April 30, 2015.

[B.2-36] OPG Form, N-FORM-11180 R002*, Fire Codes and Standards Compliance Record*, September 2016.

[B.2-37] OPG Guideline, N-GUID-00590-00002 R001, *Code over Code Review - Guideline*, August 15, 2016.

[B.2-38] OPG List, N-LIST-00590-00001 R002, *List of Significant Technical Changes from Code-Over-Code Review*, August 2015.

[B.2-39] OPG Letter, NA44-CORR-00531-00381, *Pickering A – Updated Basis for Return to Service Document*, April 20, 2001.

[B.2-40] OPG Report, NA44-REP-71400-10004 R000, *PNGS-A Compliance to CSA N293-95, Fire Protection for CANDU Nuclear Power Plants*, October 2000.

[B.2-41] OPG Letter, NA44-CORR-00531-06269 R000, *Pickering "A" – Installation of Diesel Engine Driven Fire Pumps (MEC 91665)*, February 23, 2010.

[B.2-42] OPG Report, NA44-REP-71400-10001 R000, *Pickering Nuclear Generating Station "A" Fire Protection Code Compliance Review*, April 2000.

[B.2-43] OPG Report, NA44-REP-71400-10001 R001, *Pickering Nuclear Generating Station "A" Fire Protection Code Compliance Review*, March 2011.

[B.2-44] OPG Report, NA44-REP-71400-0300967 R000, *Fire Safety Assessment: Definition of Scope of Work for CSA N293-07 Compliance at PNGS-A*, August 2009.

[B.2-45] OPG Report, NA44-REP-71400-00021 R000, *Pickering A Fixed Fire Protection Systems Inspection, Testing, and Maintenance Code Compliance Report*, January 2011.

[B.2-46] OPG Report, NA44-REP-71400-00022 R000, *Third Party Review: Fixed Fire Protection Systems Inspection, Testing and Maintenance,* March 15, 2011.

[B.2-47] OPG Report, NA44-REP-71400-00027 R000, *Pickering NGS 014 Compliance with CSA N293-07, Fire Protection for CANDU Nuclear Power Plants*, September 23, 2015.

[B.2-48] OPG Report, NA44-REP-71400-10003 R001, *Fire Hazard Assessment - Pickering A Nuclear Generating Station*, April 30, 2012.

[B.2-49] OPG Report, NA44-REP-71400-00023 R000, *Fire Safe Shutdown Analysis - Pickering A Nuclear Generating Station*, April 5, 2012.

[B.2-50] OPG Letter, NA44-CORR-00531-06935 R000, *Pickering NGS 'A' - Request for CNSC Acceptance of the "Fire Safe Shutdown Analysis" (FSSA) and "Fire Hazard Assessment" (FHA) Reports and Status Update on CCR/ITM Deviations*, June 28, 2012.

[B.2-51] OPG Letter, NA44-CORR-00531-06992 R000*, Pickering NGS-A – Fire Safe Shutdown Analysis and Third Party Review, Fixed Fire Protection Systems Inspection, Testing and Maintenance,* July 30, 2012.

[B.2-52] OPG Letter, NA44-CORR-00531-06837 R000, *Pickering NGS A – CNSC Acceptance of Fire Protection Code Compliance Review and Third Party Review, Fixed Fire Protection Systems Inspection Testing and Maintenance,* December 9, 2011.

[B.2-53] OPG Letter, NA44-CORR-00531-07475 R000, *Confirmation of Completion of CSA N293-07 Outstanding Items List Pickering NGS 'A'*, February 27, 2015.

[B.2-54] OPG Report, NK38-REP-03680-10014 R000, *Review of CAN/CSA-N293-07 (Jan 2008) Fire Protection for CANDU Nuclear Power Plants for Darlington Integrated Safety Review (ISR)*, August 2011.

[B.2-55] OPG Report, NK38-REP-03680-10128 R003, *Gap Analysis to CSA N293-07 Design Clauses for the Darlington Integrated Safety Review*, October 2013.

[B.2-56] OPG Report, NK38-REP-03680-10189 R000, *Code Review Refresh of CAN/CSA N293, Fire Protection for CANDU Nuclear Power Plants, 2012 Edition*, December 2013.

[B.2-57] OPG Report, NK38-REP-78000-10084 R000, *Darlington NGS Compliance with CSA N293-12, Fire Protection for Nuclear Power Plants*, April 2015.

[B.2-58] OPG Memorandum, N-CORR-00590-0477035, *Code-over-Code Review of CSA N293, Fire Protection for Nuclear Power Plants – 2012 Edition over the 2007 Edition*, October 1, 2013.

[B.2-59] OPG Report, N-REP-09076-10006 R000, *Review of Design Requirements of CSA N293-07*, April 12, 2007.

[B.2-60] OPG Procedure, N-PROC-RA-0054 R015, *Control of Space Allocation for Transient Material and Extended Storage of Material within the Site*, April 2015.

[B.2-61] OPG Guide, N-GUID-09076-10001 R004, *Guideline for Fire Protection Reviews of Space Allocation and Transient Material Permits*, June 12, 2015.

[B.2-62] Pickering A Design Manual, NA44-21130 R000c, *Reactor Building and Pressure Relief Duct Airlocks*, July 15, 2005.

[B.2-63] OPG Design Manual, NK30-DM-57000-00001 R000, *Cabling*, 2004.

[B.2-64] OPG Procedure, N-PROC-RA-0057 R008, *Control of Ignition Sources and Hot Work Activities*, April 14, 2015.

[B.2-65] Pickering Design Manual, NA44-DM-71400.2-00001 R001, *Fire Protection – Smoke Detection*, September 27, 2013.

[B.2-66] Pickering Design Manual, NA44-DM-71400-00002 R000, *Fire Protection Systems (Water)*, 2014.

[B.2-67] Pickering Design Manual, NK30-DM-67140-00001 R011, *Fire Detection – Fire Protection System*, July 15, 2016.

[B.2-68] Pickering Design Manual, NK30-DM-71400-00001 R006, *Fire Protection System*, January 20, 2016.

[B.2-69] OPG Report, NA44-SR-01320-00001 R015, *Pickering A Safety Report*, July 24, 2012.

[B.2-70] OPG Letter, NA44-CORR-00531-05151 R000, *Pickering "A" Unit 1 – Fire Protection Code Compliance Review - Noise Survey*, February 14, 2006.

[B.2-71] OPG Report, NK30-REP-71400-00011 R000, *Pickering B Fire Protection Code Compliance Review - Additional Review of Alternate Resolutions – Telephones, Public Address System, Stairwell Detection, and Solid Waste Handling Facility,* May 12, 2005.

[B.2-72] Pickering Design Manual, NK30-DM-60200-00001 R001, *Communications,* 21 October, 2013.

[B.2-73] OPG Engineering Standard, N-STM-78220-10000 R001, *Fire Protection for Turbine Generator Areas*, December 2002.

[B.2-74] OPG Engineering Standard, N-STM-67140-10000 R001, *Fire Protection for Main Control Room, Control Equipment Room and Cable Spreading Areas*, July 16, 2009.

[B.2-75] OPG Standard, N-STD-RA-0039 R004, *Requirements for Fire Response Capability*, April 30, 2013.

B.3 CNSC REGDOC-2.4.1 (2014), "Deterministic Safety Analysis"

B.3.1 Background

The following paraphrased from the preface and introduction of CNSC REGDOC-2.4.1 [B.3-1] provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *The purpose of REGDOC-2.4.1 is to help assure that during the construction, operation or decommissioning of a nuclear power plant, adequate safety analyses are completed by, or on behalf of, the applicant or licensee in accordance with the Nuclear Safety and Control Act and regulatory requirements.*
>
> *REGDOC-2.4.1 sets out requirements and guidance for the preparation and presentation of a safety analysis that demonstrates the safety of a nuclear facility. To the extent practicable, the document is technology-neutral and provides information on preparing and presenting deterministic safety analysis reports, including the selection of events to be analyzed, acceptance criteria, safety analysis methods, safety analysis documentation, and the review and update of safety analysis.*

CNSC REGDOC-2.4.1 is relevant to Safety Factors 5 (Deterministic Safety Analysis) and 7 (Hazard Analysis). Compliance with REGDOC-2.4.1 is currently a licensing requirement for Pickering NGS (per PROL 48.02/2018) as indicated in Appendix C.2 of the R04 Pickering Licence Conditions Handbook [B.3-2].

The current version of REGDOC-2.4.1 [B.3-1] is the first edition of the regulatory document and supersedes the following CNSC documents:

- RD-310, "Safety Analysis for Nuclear Power Plants" [B.3-3] (which replaced the draft CNSC document S-310 [B.3-4] and content from the AECB Draft Regulatory Guide C-006, Rev. 1 [B.3-5]);
- RD-308, "Deterministic Safety Analysis for Small Reactor Facilities" [B.3-6]; and
- GD-310, "Guidance on Safety Analysis for Nuclear Power Plants" [B.3-7].

Per the CNSC Document History for REGDOC-2.4.1 [B.3-8]:

> *The CNSC's Fukushima Task Force Report and the CNSC Integrated Action Plan on the lessons learned from the Fukushima Daiichi nuclear accident identified improvements to existing regulatory documents to strengthen the regulatory framework. REGDOC-2.4.1 includes targeted amendments to existing requirements that are specified in regulatory documents RD-310, "Safety Analysis for Nuclear Power Plants", and RD-308, "Deterministic Safety Analysis for Small Reactor Facilities", to address the lessons learned from the Fukushima event.*

REGDOC-2.4.1 also includes content from GD-310, which already included changes driven by the lessons from the Fukushima accident. As a result, the existing content has been integrated into REGDOC-2.4.1. The changes between RD-310 and REGDOC-2.4.1 are discussed further in the Sections below.

The results of PSR1 REGDOC-2.4.1 (and its predecessor) reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.3.2. As identified in Reference [B.3-9], the Pickering PSR2 review of CNSC REGDOC-2.4.1 is an Incremental Review. PSR2 Incremental Review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.
- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.3.2 Compliance Assessment for Pickering PSR2

B.3.2.1 Application of PSR1 Reviews

The versions of REGDOC-2.4.1 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

A review against Draft S-310 was performed as part of the Pickering B ISR, as documented in OPG Report NK30-REP-03680-00005 R000 [B.3-10]. Compliance with S-310 was demonstrated on the basis that it allows use of pre-2004 dose limits, and that "direct compliance with the modern requirements could likely be achieved with more realistic analysis" [B.3-10]. The report concludes:

> *It is judged that there are not any issues relating to S-310 in terms of radiological consequences that would be significant impediments to life extension.*

The same report also documents a review of C-006 Rev. 1 [B.3-5] to confirm that all initiating events are adequately addressed in the Pickering B Safety Report, which concluded the following:

> *In terms of Initiating Events, it is concluded that all of the C-006 R1 events are adequately addressed as being bounded by events explicitly analyzed in the Safety Report or are addressed in supplementary reviews including the Pickering B risk assessment, or in the Pickering A C-006 review Initiating Event dispositions.*

There are therefore no PSR2 gaps arising from the Pickering B ISR review of S-310 [B.3-4] or C-006 [B.3-5].

<u>Pickering Units 1,4</u>

AECB Draft Regulatory Document C-006 Rev. 1 [B.3-5] was reviewed for Pickering A Return to Service, as documented in NA44-REP-03500-00001 R00 [B.3-11] and submitted to the CNSC in NA44-CORR-00531-00560 [B.3-12]. The report concluded the following:

> *No significant deficiencies were identified as a result of the review with respect to plant design and operation of the completeness and correctness of the existing Pickering A Safety analysis in relationship to applicable license requirements. …*

The report concluded that if the Pickering A safety analysis were redone using C006-Rev. 1 as a standard, acceptable results would be obtained [B.3-11].

As such, there are no Pickering PSR2 gaps arising from the Pickering A Return to Service review of C-006 Rev. 1.

Pickering NGS compliance against REGDOC-2.4.1, which was not assessed as part of the Pickering B ISR or Pickering A Return to Service, is discussed below.

Darlington NGS

The Darlington ISR was originally performed against RD-310 [B.3-3], comprising a clause-by-clause review documented in OPG Report NK38-REP-03680-10102 R000 [B.3-13]. The review concluded that: "OPG governance and Darlington Design Guides are compliant with RD-310 for the objectives, responsibility, documentation, review, update, and quality assurance for safety analysis" [B.3-13]. OPG governance applies across OPG's nuclear stations so this is also applicable to PSR2. Five gaps in the Darlington Design Guides were identified for events to be analyzed, acceptance criteria, methods and assumptions. An additional five gaps were identified in response to the CNSC comments on the code review provided in NK38-REP-03680-10102-ADD-001 [B.3-14]. These ten gaps are described in the table below [B.3-15], [B.3-16].

Gap #	Associated RD-310 Clause	Darlington ISR Gap
0844	Clause 5.2.1 requires that a systematic process be used to identify events, event sequences, and event combinations that can potentially challenge the safety of control functions of the Nuclear Power Plant.	There are gaps in the analysis program for identifying initiating events, accounting for all power states and operational experience.
0845	Clause 5.2.3 requires that initiating events be classified into three categories: Abnormal Operational Occurrences (AOO), Design Basis Accidents (DBA), and Beyond Design Basis Accidents (BDBA).	Darlington NGS still uses the Class 1 to Class 5 classification scheme.

Gap #	Associated RD-310 Clause	Darlington ISR Gap
0846	Clause 5.3.2 requires that analysis for AOO demonstrate that radiological doses to members of the public do not exceed the established limits and that the derived acceptance criteria are met.	Documentation issues for some AOO analyses.
0847	Clauses 5.3.3 requires that Analysis for BDBA shall be performed as part of the safety assessment.	Design Guides do not require analysis of BDBA.
0848	Clause 5.4.4 provides requirements for BDBA analysis assumptions.	BDBA analysis has not been systematically performed.
1735	Clause 5.3.4 and 5.4.2 require qualitative acceptance criteria to be established for each AOO and to demonstrate that the criteria are met through identification of quantitative derived acceptance criteria prior to performing the analysis. The analysis method is to identify the applicable acceptance criteria, safety requirements and limits.	This work is in progress via pilot AOO analyses and therefore cannot be classified as compliant.
1736	Clauses 5.3.4 and 5.4.3 require the results of AOO and DBA safety analyses to meet appropriate derived acceptance criteria with margins sufficient to accommodate uncertainties associated with the analysis and significant uncertainties in analysis data, including those associated with nuclear power plant performance, operational measurements, and modelling parameters to be identified.	The claim that 'margins to accommodate uncertainty are included in derived acceptance criteria and in the use of bounding operating parameters' requires demonstration. It is expected that the industry safety analysis improvement (SAI) working group will develop a plan and implement it to demonstrate the Limit of Operating Envelope (LOE) conservatism and adequacy for RD-310 compliance, however, since the work is in progress this represents a gap to these clauses.
1746	Clause 5.4.1 requires the analysis to provide the appropriate level of confidence in demonstrating conformity with the acceptance criteria.	Per the CNSC comment: "Whether the current LOE methodology accommodates modelling uncertainties requires demonstration."
1747	Clause 5.4.2 requires the analysis method to account for uncertainties in the analysis data and models.	CNSC staff conclude that no modelling uncertainty is considered.
1748	Clause 5.4.6 requires the safety analysis to build in a degree of conservatism to off-set any uncertainties associated with both Nuclear Power Plant (NPP) initial and boundary conditions and modeling of nuclear power plant performance in the analyzed event.	CNSC staff expects formal documentation on where conservatisms are built in and where they are used to off-set uncertainties.

These gaps, which are specific to RD-310 implementation, also relate to Governance, Programs, Procedures and Practices that apply across OPG's Nuclear operations and are therefore also applicable to PSR2. These gaps are further discussed below with respect to the ongoing work toward compliance with RD-310 and subsequently REGDOC-2.4.1 post-PSR1.

The gaps outlined above were assigned to 5 Darlington ISR Issues (D027, D028, D030, D399 and D400) in the Final ISR Report and Addenda [B.3-15], [B.3-16]. The resolution of all gaps associated with RD-310 was tracked through CNSC Action Item 2010-OPG-05 "Completion of RD-310 compliance activities." Action Item 2010-OPG-05 included implementation of RD-310 by 2021 Q4, as described in The Second Addendum to the Final ISR Report, NK38-REP-03680-10102-ADD-002 [B.3-16]:

> *A detailed plan for OPG implementation of RD-310 was submitted in December, 2012. It includes a timeline for OPG implementation of RD-310 in safety analysis and its incorporation into the upgrade of the Darlington Safety Report. The timeline made a commitment to have the Safety Report comprehensively comply with RD-310 by Q4 2021, with a staged implementation of updated accident analyses and appendices.*

According to the Final ISR Report, the RD-310 implementation process "is managed under the Reactor Safety Program N-PROG-MP-0014 and is formally communicated to the CNSC via this program on a semi-annual basis" [B.3-17].

With the release of REGDOC-2.4.1, the five issues identified above were regrouped into two issues specific to REGDOC-2.4.1 compliance in Appendix A, Table 4, of the latest revision of the Darlington IIP [B.3-18]:

1. IIP-OI-035 (D027): "A systematic analysis of BDBA and Severe Accidents is required and Severe Accident Management Guidelines must be fully implemented".

2. IIP-OI-043 (D028, D030, D399 and D400): "Comply with the new requirements of CNSC REGDOC-2.4.1".

Under CNSC Action Item 2010-OPG-05, the RD-310 Compliance activities to address the gaps found during the Darlington ISR were to be addressed through OPG's Safety Analysis Improvement (SAI) initiative. With the release of REGDOC-2.4.1, the SAI plans were revised to reflect the graded approach permitted by REGDOC-2.4.1. According to N-CORR-00531-07338 [B.3-19], the CNSC staff agreed in principle with the new direction of OPG's SAI efforts. Action Item (AI) 2010-OPG-05 was closed, while AI 2014-OPG-5461 was opened to track compliance with REGDOC-2.4.1 with a completion date of 2024 [B.3-19].

Closure of AI 2014-OPG-5461 was requested [B.3-20] and granted [B.3-21] with the development and acceptance of the REGDOC-2.4.1 Implementation Plan [B.3-22], which superseded the RD-310 Implementation Plan [B.3-23]. Action Request #28189400 was opened to track the implementation of REGDOC-2.4.1 to meet Pickering PROL Licence Condition 5.1. This work is still in progress and therefore is identified as **PSR2 REGDOC-2.4.1 Gap #1**.

B.3.2.2 Application of Post-PSR1 Reviews

Pickering A and B RD-310 Assessments

With the publication of RD-310, assessments of each of the Pickering Units 1,4 and Units 5-8 Safety Report Appendices against the requirements of RD-310 were summarized in the Pickering A and Pickering B RD-310 Implementation Plan [B.3-23]. The gaps found were consistent with those identified in the Darlington ISR review with the addition of several gaps, summarized as follows:

1. For all Safety Report appendices, gaps were found with RD-310 Clause 5.2.2 which requires the list of events identified for the safety analysis to include operator errors and common-cause internally and externally initiated events. The clause also requires that the selected cut-off frequency is justified and excluded events documented.

2. For all Safety Report appendices, it was found that the safety analysis does not meet Clause 5.4.5 which requires that computer codes used in the safety analysis shall be developed, validated and used in accordance with a quality assurance program that meets the requirements of CSA N286.7-99.

3. For all Safety Report appendices, it was found that the safety analysis does not meet the requirements for documentation as required in Clause 5.5.

4. For all Safety Report appendices, it was found that the quality assurance program does not adequately address the validation of NPP and analytical models as required in Clause 5.7.

The RD-310 Implementation Plan was developed through the prioritization of the gaps requiring resolution along with regulatory considerations, OPG business drivers, and analyses already underway for purposes other than RD-310 compliance. The prioritization of the gaps to be addressed as part of the RD-310 Implementation Plan was based on the expected end of commercial operation date of Pickering A and B of 2020, with the focus of RD-310 compliance on continued demonstration of adequate safety margins in light of ongoing aging mechanisms [B.3-23]. The progress of the RD-310 Implementation Plan was tracked under Action Item 2010-OPG-05. As noted in OPG letter to the CNSC, N-CORR-00531-06076 [B.3-24]:

> *In contrast to the Darlington Implementation Plan, limited upgrades are proposed in the Pickering A and B Plan, which has been developed with consideration for demonstration of continued safe operation while accounting for the limited remaining operating life of the Pickering Units.*

As documented in N-CORR-00531-06256 [B.3-25], the CNSC accepted the RD-310 Implementation Plan, concluding that the RD-310 gaps in the Pickering A and B Safety Report analyses were adequately identified and documented, and that the Implementation Plan was based on a well-documented process and the selected scope for RD-310 compliant analyses was justified [B.3-25]. While the selection of scope was accepted, this was based on a projected end of commercial operation date of 2020. The same prioritization criteria used to select the scope of the RD-310 implementation were carried forward into the REGDOC-2.4.1 Implementation Plan, described further below. As a result, Pickering NGS operation beyond 2020 results in a gap with respect to the basis used for identifying selected analysis updates to comply with REGDOC-2.4.1 for the PSR2 period. This gap is identified as **PSR2 REGDOC-2.4.1 Gap #2**.

REGDOC-2.4.1 Implementation Plan

CNSC REGDOC-2.4.1 was published in 2014. As indicated in the REGDOC-2.4.1 Implementation Plan [B.3-22]:

> *REGDOC-2.4.1 is very similar to RD-310 and GD-310 in technical requirements and guidance.*

According to the OPG REGDOC-2.4.1 Gap Assessment [B.3-26]:

> *The most significant change between RD-310 and REGDOC-2.4.1, for OPG, is that REGDOC-2.4.1 permits a graded approach to updating existing analyses.*

Based on this graded approach, only the most significant gap for Pickering (the absence of a Common Mode Event Appendix) was identified for development to REGDOC-2.4.1 requirements [B.3-22]. Therefore, the gap assessment for REGDOC-2.4.1, which mapped the REGDOC-2.4.1 clauses to their corresponding clauses of RD-310 and assessed the differences, focused on the Darlington RD-310 gaps.

The Darlington REGDOC-2.4.1 gap assessment identified two additional gaps related to cliff-edge effects that are programmatically applicable to Pickering. As discussed above, the REGDOC-2.4.1 Implementation Plan needs to be reconsidered in the context of extended operation beyond 2020. Since REGDOC-2.4.1 requirements will be revisited as part of that review, there is no incremental gap for PSR2.

The Implementation Plan for REGDOC-2.4.1 at Pickering NGS is summarized in the PROL Amendment request [B.3-27]:

> *In alignment with current Pickering licensing requirements, and with the graded approach permitted by REGDOC-2.4.1 requirements, OPG will be upgrading the Pickering safety reports only to the extent that a new appendix will be included to address the development and analysis of common mode events in 2017. The analysis of common mode events represents the single largest gap in the Pickering Safety Reports with respect to REGDOC-2.4.1.*
>
> *Beyond 2017, Pickering NGS will be maintaining a safety analysis program compliant with REGDOC-2.4.1 requirements. OPG will perform to the extent practicable, any new Pickering specific safety analyses in accordance with REGDOC-2.4.1 requirements. OPG will continue to utilize the existing OPG safety report update process to comply with the regulatory requirements or updating safety reports periodically.*

The graded approach to evaluating and addressing the gaps with REGDOC-2.4.1 compliance is recognized in the Licence Conditions Handbook, which identifies the following criteria for implementation of REGDOC-2.4.1 [B.3-2]:

- *Assessment of the current safety analysis practices against REGDOC 2.4.1 to identify gaps;*
- *Prioritization of the identified gaps using formal methods;*
- *Justification of non-conformances (e.g., full compliance with REGDOC 2.4.1 is not practicable or does not provide a demonstrable safety benefit); and*
- *Development and execution of corrective action plans to address the important gaps.*

As identified in OPG letter N-CORR-00531-06865 R000, "Resolution of Large Break LOCA (LBLOCA) Safety Margin Issues" [B.3-28], OPG has initiated a project to update the LBLOCA analyses using the LOE methodology for the OPG Nuclear fleet as part of the REGDOC-2.4.1 compliance initiative. The status of this project is discussed in N-CORR-00531-18022 R000, "Resolution of Large Break LOCA Safety Analysis Margin Issues" [B.3-29].

B.3.3 Compliance Summary for Pickering PSR2

There are two PSR2 CNSC REGDOC-2.4.1 (2014) gaps which relate to Safety Factor 5 (Deterministic Safety Analysis):

1. The REGDOC-2.4.1 Implementation Plan and associated gap assessments capture all gaps related to REGDOC-2.4.1 and incorporate a systematic selection of the scope of work to address the most pertinent gaps in accordance with the graded approach to upgrading existing analyses. REGDOC-2.4.1 compliant analysis activities and progress related to REGDOC-2.4.1 implementation in the Pickering Licence Conditions Handbook are tracked according to the CNSC Compliance Verification Criteria. Since the implementation is in progress, this has been identified as a PSR2 gap for Pickering NGS REGDOC-2.4.1 compliance.

2. As described in the REGDOC-2.4.1 Implementation Plan: "Limited upgrades are proposed in the Pickering A and B Plan, which has been developed with consideration for demonstration of continued safe operation while accounting for the limited remaining operating life of the Pickering Units". The REGDOC-2.4.1 Implementation Plan for Pickering did not consider operation past 2020 and therefore the need for review and update of the Implementation Plan in the context of operation of Pickering NGS beyond 2020 is identified as a PSR2 gap. This will be informed by the timeline of the Darlington REGDOC-2.4.1 Implementation Plan, and the limited additional years of Pickering NGS operation.

B.3.4 References

[B.3-1] CNSC Regulatory Document REGDOC-2.4.1, *Deterministic Safety Analysis*, May 2014.

[B.3-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.3-3] CNSC Regulatory Document RD-310, *Safety Analysis for Nuclear Power Plants*, February 2008.

[B.3-4] CNSC Draft Regulatory Standard S-310, *Safety Analysis for Nuclear Power Plants,* January 2005 (Draft).

[B.3-5] AECB DRAFT Regulatory Guide C-006 Rev. 1, *Safety Analysis for CANDU Nuclear Power Plants*, September 1999 (Draft).

[B.3-6] CNSC Regulatory Document RD-308, *Deterministic Safety Analysis for Small Reactor Facilities*, June 2011.

[B.3-7] CNSC Guidance Document GD-310, *Guidance on Safety Analysis for Nuclear Power Plants*, March 2012.

[B.3-8] CNSC website, *Document History of REGDOC-2.4.1, Deterministic Safety Analysis*, http://nuclearsafety.gc.ca/eng/acts-and-regulations/regulatory-documents/history/regdoc2-4-1.cfm, 21 August, 2014.

[B.3-9] OPG Report P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.3-10] OPG Report, NK30-REP-03680-00005 R000, *Pickering NGS B – Integrated Safety Review - Safety Analysis Review*, June 2007.

[B.3-11] OPG Report, NA44-REP-03500-00001 R000, *Pickering NGS A -Review of Safety Analysis Against CNSC Consultative Document C6 (Revision 1), Safety Analysis of CANDU Nuclear Power Plants*, May 2001.

[B.3-12] OPG Letter, NA44-CORR-00531-00560 R000, R.J. Strickert to Dr. J. S. C. Tong, *Pickering A – Review of Safety Analysis to CNSC Consultative Document C6 (Rev. 1)*, May 31, 2001.

[B.3-13] OPG Report, NK38-REP-03680-10102 R000, *Review of CNSC RD-310 (February 2008) Safety Analysis for Nuclear Power Plants for Darlington Integrated Safety Review*, August 2011.

[B.3-14] OPG Report, NK38-REP-03680-10102-ADD-001 R000, *Addendum to the CNSC RD-310 Code Report for Darlington ISR*, January 2014.

[B.3-15] OPG Report, NK38-REP-03680-10104-ADD-001 R00, *Darlington NGS Integrated Safety Review (ISR) – Final ISR Report Addendum,* June 2013.

[B.3-16] OPG Report, NK38-REP-03680-10104-ADD-002 R00, *Darlington NGS Integrated Safety Review (ISR) – Final ISR Report Addendum 002*, November 2013.

[B.3-17] OPG Report, NK38-REP-03680-10104 R000, *Darlington NGS Integrated Safety Review (ISR) – Final ISR Report*, October 2011.

[B.3-18] OPG Report, NK38-REP-03680-10185 R002, *Darlington NGS Integrated Implementation Plan (IIP)*, April 2015.

[B.3-19] CNSC Letter, e-Doc #4502142 File 2.01 (OPG File No. N-CORR-00531-07338 R000), M. Santini and F. Rinfret to W. M. Elliott, *Darlington & Pickering NGS: Safety Analysis Improvement and REGDOC-2.4.1 Compliance, New Action Item 2014OPG-5461, Closure of Action Item 2010OPG-05*, October 8, 2014.

[B.3-20] OPG Letter, N-CORR-00531-07409 R000, S. Woods to M. Santini and F. Rinfret, *OPG Safety Analysis Improvement and REGDOC-2.4.1 Implementation – Action Item 2014OPG-5461*, October 13, 2015.

[B.3-21] CNSC Letter, E-Doc 4947467 File 4.01.03 (OPG File No. N-CORR-00531-18016 R000), M. Santini and F. Rinfret to S. Woods, *Darlington & Pickering NGS: Safety Analysis Improvement and REGDOC-2.4.1 Implementation – Closure of Action Item 2014-OPG-5461*, March 24, 2016.

[B.3-22] OPG Plan, N-PLAN-03500-0500515 R003, *REGDOC-2.4.1 Implementation Plan*, May 2015.

[B.3-23] OPG Plan, N-PLAN-03500-0439621 R001, *Pickering A & Pickering B RD-310 Implementation Plan*, March 2013.

[B.3-24] OPG Letter, N-CORR-00531-06076 R000, W. M. Elliott to M. Santini and F. Rinfret, *Progress Report on OPG Safety Analysis Improvement and RD-310 Implementation Activities – Action Item 2010OPG-05*, April 23, 2013.

[B.3-25] CNSC Letter, e-Doc #4182716 File 4.01.03 (OPG File No. N-CORR-00531-06256 R000), M. Santini to W. M. Elliott, *Pickering NGS-A and B: RD-310 Gap Assessment and Implementation Plan – Action Item 2010OPG-05 (RIB #2041)*, August 12, 2013.

[B.3-26] OPG Report, N-REP-03500-0524958 R000, *OPG REGDOC-2.4.1 Gap Assessment*, December 2014.

[B.3-27] OPG Letter, P-CORR-00531-04373, B. McGee to M. A. Leblanc, *Pickering NGS – Request for an Amendment to Pickering PROL 48.01/2018 to Implement New Regulatory Documents REGDOC-2.4.1 and REGDOC-2.4.2*, June 10, 2015.

[B.3-28] OPG Letter, N-CORR-00531-06865 R000, W. M. Elliott to M. Santini and F. Rinfret, *Resolution of Large Break LOCA (LCLOCA) Safety Margin Issues*, April 15, 2015.

[B.3-29] OPG Letter, N-CORR-00531-18022 R000, W. S. Woods to M. Santini and F. Rinfret, *Resolution of Large Break LOCA (LBLOCA) Safety Analysis Margin Issues*, April 25, 2016.

B.4 CNSC REGDOC-2.4.2 (2014), "Probabilistic Safety Assessment (PSA) for Nuclear Power Plants"

B.4.1 Background

The following, paraphrased from the preface and introduction of CNSC REGDOC-2.4.2 [B.4-1], provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *The objectives of a probabilistic safety assessment (PSA) include provision of a systematic analysis to give confidence that the design will align with the fundamental safety objective. REGDOC-2.4.2 assures that the licensee conducts a PSA in accordance with defined requirements when incorporated into a licence to construct or operate a nuclear power plant [NPP].*
>
> *REGDOC-2.4.2 sets out the requirements for the PSA for a licence to construct or operate an NPP, when required by the applicable licence.*

CNSC REGDOC-2.4.2 is relevant to Safety Factors 6 (Probabilistic Safety Assessment) and 7 (Hazard Analysis).

Compliance with REGDOC-2.4.2 is currently a licence requirement for Pickering NGS (per PROL 48.02/2018) as indicated in Appendix C.2 of the R04 Pickering Licence Conditions Handbook [B.4-2]. When OPG requested that the CNSC amend the Operating Licence to replace S-294 with REGDOC-2.4.2, a transition plan was also submitted. The transition plan is discussed in paragraph 15 of Reference [B.4-3].

REGDOC-2.4.2 supersedes S-294, "Probabilistic Safety Assessment (PSA) for Nuclear Power Plants" [B.4-4]. As outlined in Reference [B.4-5]:

> *The CNSC's Fukushima Task Force Report and the CNSC Integrated Action Plan on the lessons learned from the Fukushima Daiichi nuclear accident, identified improvements to existing regulatory documents to strengthen the regulatory framework. REGDOC-2.4.2 includes targeted amendments to existing requirements that are specified in regulatory document S-294, to address the lessons learned from the Fukushima event… As a result of additional feedback from stakeholders, guidance has been included to clarify the intent of certain requirements.*

The results of PSR1 REGDOC-2.4.2 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.4.2. As identified in Reference [B.4-6], the Pickering PSR2 review of CNSC REGDOC-2.4.2 is an Incremental Review. PSR2 Incremental Review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.

- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.4.2 Compliance Assessment for Pickering PSR2

B.4.2.1 Application of PSR1 Reviews

The versions of REGDOC-2.4.2 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

The Pickering B ISR included a Safety Analysis Review report [B.4-7] which addressed Probabilistic Risk Assessment (also referred to as PSA) and included a clause-by-clause review of S-294 compliance. A discrepancy was identified as follows:

> *CNSC regulatory document S-294 identifies external events to be included in the PRA (e.g., seismic, flooding, and tornados). The PBRA [8] does not currently include these events, or fire, as an initiating event. However, S-294 allows assessment of events by other methods. OPG has addressed these events via hazard assessments and design (i.e., fire and seismic).*

Subsequently OPG has completed PSAs for Pickering Units 5-8 [B.4-8] and for Pickering Units 1,4 [B.4-9], which were prepared in accordance with CNSC Regulatory Standard S-294 and updated to include the enhancements mandated in the Fukushima Action Plan. The PSAs include detailed Level 1 at-power assessments for seismic events, internal floods, and high winds (including tornados), bounding Level 2 at-power and Level 1/2 outage assessments for those hazards, and screening assessment for external flooding events.

The foregoing is further recognized in Section 5.1 of the R04 Pickering Licence Conditions Handbook [B.4-2] which states:

> *OPG submitted a S-294 compliant Pickering B PSA in 2012, and completed the compliant Pickering A PSA in 2014. In addition, in preparation for the hold point hearing in 2014, OPG incorporated the credits due to the enhancements required under the Fukushima Action Plan in the PSAs for both Pickering A and B.*

There are therefore no PSR1 gaps that are applicable in PSR2. The status of implementation of REGDOC-2.4.2 is discussed further in Section B.4.2.2 below.

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[8] Pickering B Risk Assessment

Pickering Units 1,4

REGDOC-2.4.2 did not exist when the Pickering A RTS assessments were performed. REGDOC-2.4.2 and S-294 were not within the scope of PSR1 as it related to Pickering Units 1,4. However, as discussed above, compliance with S-294 is recognized in Section 5.1 of the R04 Pickering Licence Conditions Handbook [B.4-2]. The status of implementation of REGDOC-2.4.2 is discussed further in Section B.4.2.2 below.

Darlington NGS

The Darlington ISR performed an assessment of the Safety Analysis Safety Factor, which at that time included PSA together with Deterministic Safety Analysis and Hazard Analysis. A detailed clause-by-clause review of S-294 [B.4-10] concluded that:

> *OPG Nuclear governance and the practices at Darlington NGS do not currently meet all of the requirements of S-294. The key issues are summarized as follows:*
>
> 1. *The overall OPG PRA Standard and specific instructions for Level 1 At-Power Internal Events PRA are compliant with the requirements of S-294 and have been accepted by the CNSC. While the intent of specific instructions for other assessments (Level 2; Outage; External Events) should be consistent, a gap in compliance exists until such time that the methodology documents for these other assessments are formally issued.*
>
> 2. *The existing draft DARA [Darlington Risk Assessment] models comply with most but not all of the requirements of the governance documents. This has been recognized, and will be rectified by completion of the updated DARA for submission to the CNSC in December 2010. When the update project is complete, the DARA (Level 1 and 2, internal events, at-power and outage) will comply with the requirements of S-294; until then, the existing draft DARA represents a gap in compliance of station practice with the requirements of S-294.*
>
> 3. *Projects for Darlington fire, seismic, and flooding risk assessments are underway but no formal models or results have been issued to date. Until these projects are complete, this represents a gap in compliance of station practice with the requirements of S-294.*

Based on the S-294 code review [B.4-10], seven gaps were identified in the Safety Analysis Safety Factor report [B.4-11]. The conclusion in terms of PSA was that "Darlington has recently initiated an update of DARA to be compliant with the CNSC regulatory standard S-294". Since OPG has completed PSAs for Pickering Units 5-8 [B.4-8] and for Pickering Units 1,4 [B.4-9] in accordance with CNSC Regulatory Standard S-294, the PSR1 gaps from Darlington do not apply to Pickering and there is no PSR2 gap.

B.4.2.2 Application of Post PSR1 Reviews

A comparison of S-294 to REGDOC-2.4.2 has been performed by CANDU Owner's Group [B.4-12]. The assessment was a factor when a transition plan was developed to include

REGDOC-2.4.2 in the PROL as noted in the following discussion from Section 5.1 of the R04 Pickering Licence Conditions Handbook [B.4-2]:

> *REGDOC-2.4.2 Probabilistic Safety Assessment (PSA) for Nuclear Power Plants Implementation Strategy: OPG will update the Pickering A PSA and Pickering B PSA using a graded approach as permitted by REGDOC-2.4.2. The PSA elements created as part of S-294 will be updated and the updated requirements of REGDOC-2.4.2, such as Irradiated Fuel Bay (IFB) risk assessment, will be addressed.*
>
> *The next Pickering B PSA update will be completed in 2017, including detailed risk re-quantification, in accordance with S-294. Similarly, the next Pickering A PSA update will be completed in 2018, including detailed risk re-quantification, in accordance with S-294.*
>
> *All the Pickering A PSA and Pickering B PSA updates extended to 2020 will be solely focused on the additional updated requirements of REGDOC-2.4.2 going beyond S-294 requirements, including for example, IFB risk assessment, and which are risk contributors of less significance. The updated requirements of REGDOC-2.4.2 may be dealt with through alternative methods to PSA for which guidance is currently being developed by industry.*
>
> *OPG is required to inform the Commission by June 30, 2017 as to whether or not the Pickering units will end commercial operation by December 2020. Should a decision be made to continue to operate the Pickering units beyond 2020, CNSC and OPG staff will engage in further discussions in 2017 on what will be required for Pickering NGS PSA updates beyond 2020 ...*

The Licence Conditions Handbook identifies that following the Pickering 5-8 update in 2017 and the Pickering 1,4 update in 2018, subsequent PSAs will be focused on the additional updated requirements of REGDOC-2.4.2. Also, some of the additional REGDOC-2.4.2 requirements may be dealt with through alternative methods to PSA for which guidance is currently being developed by the industry. The REGDOC-2.4.2 Implementation Plan for Pickering that was agreed to with the CNSC did not consider operation past 2020 and therefore the need for review and update of the Implementation Plan in the context of operation of Pickering NGS beyond 2020 is identified as a gap. (**PSR2 REGDOC-2.4.2 Gap #1**).

B.4.3 Compliance Summary for Pickering PSR2

There is one PSR2 CNSC REGDOC-2.4.2 (2014) gap which relates to Safety Factor 6 (Probabilistic Safety Assessment):

1. The REGDOC-2.4.2 Pickering Implementation Plan agreed to with the CNSC did not consider operation beyond 2020 and therefore, the review and update of the Implementation Plan in the context of operation of Pickering NGS beyond 2020 is required. Therefore, this has been identified as a PSR2 gap.

B.4.4 References

[B.4-1] CNSC Regulatory Document, REGDOC-2.4.2, *Probabilistic Safety Assessment (PSA) for Nuclear Power Plants*, May 2014.

[B.4-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.4-3] CNSC Report, P-CORR-00531-04632, *Application to Amend the Power Reactor Operating Licence for the Pickering Nuclear Generating Station*, December 18, 2015.

[B.4-4] CNSC Regulatory Standard, S-294, *Probabilistic Safety Assessment (PSA) for Nuclear Power Plants*, April 2005.

[B.4-5] CNSC website, Document History of REGDOC-2.4.2, *Probabilistic Safety Assessment (PSA) for Nuclear Power Plants*, http://nuclearsafety.gc.ca/eng/acts-and-regulations/regulatory-documents/history/regdoc2-4-2.cfm, 21 August 2014.

[B.4-6] OPG Report P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.4-7] OPG Report, NK30-REP-03680-00005 R000, *Pickering NGS B - Integrated Safety Review - Safety Analysis Review*, June 2007.

[B.4-8] OPG Letter, NK30-CORR-00531-06764, B. Phillips to M. Santini, *Results of the Pickering NGS B Probabilistic Safety Assessment (PSA)*, March 19, 2014.

[B.4-9] OPG Letter, NA44-CORR-00531-07346, B. Phillips to M. Santini, *Results of the Pickering NGS A Probabilistic Safety Assessment (PSA)*, March 19, 2014.

[B.4-10] OPG Report, NK38-REP-03680-10007 R000, *Review of S-294 (April 2005), Probabilistic Safety Assessment (PSA) for Nuclear Power Plants for Darlington Integrated Safety Review*, August 2011.

[B.4-11] OPG Report, NK38-REP-03680-10081 R001, *Darlington NGS Integrated Safety Review – Safety Analysis Safety Factor Report*, September 2010.

[B.4-12] COG Report, COG-15-9020, *Discussion Papers for Selected PSA Focus Areas,* March 2016.

B.5 CSA N287.1-14, "General Requirements for Concrete Containment Structures for Nuclear Power Plants"

B.5.1 Background

The following, paraphrased from the Preface of CSA N287.1-14 [B.5-1], provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *The purpose of CSA N287.1 is to provide general requirements to ensure that the design, construction, and testing of concrete containment structures will meet a quality and standard commensurate with the safety principles necessary to comply with the Canadian nuclear safety philosophy.*
>
> *CSA N287.1 specifies general requirements for the design, construction, testing, and commissioning of concrete containment structures for nuclear power plants designated as class containment and is directed to the owners, designers, manufacturers, fabricators, and constructors of the concrete components and parts.*

As identified in Section 1 of N287.1-14 [B.5-1]:

> *This Standard is applicable to new nuclear power plants' concrete containment structures to be built in Canada. Application of the Standard to concrete containment structures to be built outside of Canada is subject to approval of the authority having jurisdiction (AHJ). The application of the Standard to existing or operating nuclear power plants is as agreed upon by the owner/operator and the AHJ.*

CSA N287.1 is directly relevant to Safety Factor 1 (Plant Design).

CSA N287.1 is identified in Appendix E.1 of the R04 Pickering Licence Conditions Handbook [B.5-2] as "Guidance or Criteria". N287.1-14 is the fourth edition of this standard, which supersedes the previous editions published in 1993, 1982 and 1977 under the title "General requirements for Concrete Containment Structures for CANDU Nuclear Power Plants". The CSA N287.1-14 Impact Statement [B.5-3] provides a "Summary of significant changes from previous edition" which identifies several changes to the Standard as described in Section B.5.2.2 below.

The results of PSR1 CSA N287.1 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.5.2. As identified in Reference [B.5-4], the Pickering PSR2 review of CSA N287.1-14 is an Incremental Review. PSR2 Incremental Review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- <u>Compliance:</u> Compliance indicates that the change in the safety requirement, per the topical review, is met.

- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.5.2 Compliance Assessment for Pickering PSR2

B.5.2.1 Application of PSR1 Reviews

The versions of N287.1 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

As part of the Pickering B ISR Plant Design Safety Factor, OPG Report NK30-REP-03680-00001 R000, "Pickering NGS-B Integrated Safety Review - Plant Design Safety Factor" [B.5-5] performed a clause by clause review of N287.1-93 (R2004) [B.5-6] against OPG governance. NK30-REP-03680-00001 R000 identified minor deviations and concluded that the design of the existing Concrete Containment Structures (CCSs) at Pickering B are in general compliance with CSA N287.1-93. NK30-REP-03680-00001 R000 established that a number of Acceptable Deviations resulted from the review, as discussed below (text in italics taken verbatim from [B.5-5]):

- Clause 3.1.1 Classification of Class Containment: *The original and subsequent revisions of CSA N287 series of standards were issued after completion of design and construction of Pickering 'B' project. Therefore the description related to containment structure, parts, materials, design, fabrications, construction, inspection examination, testing and commissioning in accordance with the series of standards of CSA N287 were not available. At that time, the CCSs were designed to meet the requirements, the National Building Code (NBC) of Canada and Pickering B design guides and design requirements documents... The fact that "Classification of Class Containment" did not exist at the time of design is not safety significant. The CCSs have served their intended function satisfactorily for 25 years and remain sound and fully functional. In service inspections are being conducted in accordance with the requirements of CSA N287.7 and the resultant inspection reports attest to the quality of the design being maintained and satisfactorily meeting service requirements.*

- Clause 3.3 Jurisdictional Boundaries: *The definitions of the Jurisdictional Boundaries for portions of containment structure systems and components were not specifically identified at the time the CCSs were designed as the N287 standards had not yet been issued. Essentially, the design of all components or portions of the containment structure interfacing with nuclear systems are required to satisfy ASME Section III Class 2 or MC standards. Those interfacing with non-nuclear systems must conform to the standard for the applicable system. Note that Portions B and C of the CCSs, per clause definition, are considered outside the scope of this Standard. Therefore, the intent of this clause is considered to be met.*

- Clause 4.1.2.1 Design Specifications: *This clause specifies the general Design Specifications for Nuclear Power Plants. Pickering "B" CCSs underwent the process of design, fabrication, construction, installation, testing & commissioning as per existing NBCC (National Building Code of Canada) and ASME Section III (for some P/B embedded parts) requirements, Design guides and design requirements documents were prepared at the time and reflect the design of the CCSs as built. All such documents were reviewed and approved by professional engineers. Design Requirements Document NK30-DR-63420-10001 is a comprehensive design specification that covers all requirements listed in clause 5.1 for the NPC [Negative Pressure Containment] system, except those that apply to the vacuum building.*

- Clause 4.1.2.4 Site Seismicity: *Seismic assessment has been done for Pickering 'B'.*

- Clause 4.1.3 Quality Assurance: *OPG maintains a Quality Assurance program which complies with the CSA Standard N286. Although [vendor] has not reviewed this program, it is prepared to rely upon its knowledge that OPG maintains a QA program that meets the requirements of CAN/CSAN286.0 and that the program has been audited by the regulator. The intent of this clause is considered to be met.*

- Clause 4.2.1 Designer's Responsibility: *The designers of the Pickering B CCSs met NBC of Canada 1970 requirements and the requirements of various OPG design manuals and design requirements documents. The CCSs have consistently met their performance requirements throughout some 25 years of station operation and remain in good condition and fit for continuing service. The intent of this clause and the responsibilities listed in clauses 4.2.2 through 4.2.8 can be considered to be met.*

- Clause 5.1 Design Specifications: *This clause and the sub-clauses specify the general Design Specifications for Nuclear Power Plants. Pickering "B" has undergone all the processes listed in this clause during construction, installation, testing & commissioning in accordance with the existing NBCC (National Building Code of Canada) requirements at that time. The intent of this clause is considered to be met.*

- Clause 5.3 Drawings: *The drawings for project were developed manually whereas presently all the drawings are required to be developed electronically based on CAD software.*

- Clause 5.4 Design/Stress Report: *Pickering "B" Design / Stress Reports were prepared for Reactor Building & Vacuum Building and the documents are identified as NK30-PH-21140-01 & 973-NA44/NK30-25100 respectively. In addition, they are referenced in the Safety Report, Refer Part 1, Section 1.2.3.1 of Safety Report NK30-SR-01320-00001, Rev. 02. The document refers to the containment boundary formed by the Reactor Buildings, Pressure Relief Duct, Vacuum Building & Vacuum Ducts of CCSs. (The structures were designed for internal & external pressures which are detailed in this report.) Refer to NK30-DR-63240-10001, Section 6.*

- Clause 7.1 General: *Pickering 'B' Quality Assurance program was initially based on the project requirements of Part 4, 5, 6, 7 & 8 of NBC, Canada and supplemented in Design Manual. Later, the Quality Assurance program of CSA Z299.2 for manufacture and* *fabricated parts were applied as per the requirements. Visual inspection of concrete confirms that the concrete is in good condition and results of in-service leak tests have been satisfactory. This to be considered adequate for civil structures…. The Quality Assurance program applied to the design of the CCSs met the requirements of CSA Standard Z299.2. The generic requirements stipulated in Z299.0 were also considered at the time.*

- Clause 7.2 Quality Verification: *The Quality Verifications of Pickering "B" were prepared as per the project requirements in accordance with the Part 4, 5, 6, 7 & 8 of NBC, Canada. Visual inspection of concrete performed by qualified persons as part of the ongoing negative pressure containment system test program indicates that the CCSs remain in good condition.*

- Clause 7.3 Qualification of Inspection Personnel: *Inspection of the CCSs during construction and the inaugural inspection met the requirements of the NBC of Canada and inspectors carrying out the work were appropriately qualified. Inspectors who perform inspections as part of the on-going in-service and safety are competent by evidence of training and experience in the industry. Therefore the intent of this clause is considered to be met.*

- Clause 7.4 Quality Assurance Records: *The Quality Assurance Records will have been prepared for Pickering "B" CCSs in accordance with the OPG QA Program which complies with the Standard CSA N286.2 & CSA N286.3… the intent of this clause is considered to be met.*

OPG Report NK30-REP-03680-00015 R000, "Pickering NGS-B Integrated Safety Review (ISR) - Final ISR Report" [B.5-7] grouped the above findings into ISR Issues:

- 1-219/222/223/225/227, which identified these findings as Discrepancies with low safety significance (i.e., priority level 3 or 4) with no further action required, and

- 220/221/224/226/228/229/230/231/232, which provided additional justification for earlier classification as Acceptable Deviations.

The rationale for these findings being Acceptable Deviations is not impacted by Pickering NGS operation past 2020. CSA N287.1 applies to the design, construction, testing, and commissioning of CCSs. The CCSs at Pickering B were built and tested to meet 1970 National Building Code of Canada (NBCC) requirements [B.5-8], supplemented by specific loading requirements and the requirements of Design Manuals (e.g., see [B.5-9], [B.5-10], [B.5-11], [B.5-12]) and Design Guides (e.g., see [B.5-13], [B.5-14], [B.5-15]). The original Pickering concrete specifications (L-715-80 [B.5-16] and NK30-LH-20541-01 [B.5-17]) included requirements for quality control and compliance with CSA A23.1, A23.2 and A23.3 (which address concrete materials, methods of concrete construction and test methods and standard practices for concrete). As such, the standards that applied during original construction of Pickering NGS included requirements for tests and quality control procedures to ensure that the concrete used in the as-built structures met the original design requirements. Ongoing confirmation that the Pickering NGS CCSs remain fit for service is demonstrated via the following:

- As required by N-PROG-MA-0017 R008 [B.5-18]:
  - Periodic examinations of in-service inspections and positive leakage rate testing of CCSs that are designated as class containment components are performed in accordance with the requirements of CSA N287.7, "In Service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants". N-PROC-MA-0066 R005, "Administrative Requirements for In-Service Examination and Testing for Concrete Containment Structures" [B.5-19] describes the administrative process and the roles of various organizations across OPG Nuclear involved in the execution of N-PROG-MA-0017 R008 in order to comply with the requirements of CSA N287.7.
  - Periodic inspection of containment system metallic and plastic components (including the containment pressure suppression systems) is performed in accordance with the requirements of CSA N285.5, "Periodic inspection of CANDU Nuclear Power Plant Containment Components". N-PROC-MA-0064 R005, "Administrative Requirements for the Periodic Inspection of Nuclear Power Plant Containment Components" [B.5-20], describes the administrative process and the roles of various organizations across OPG Nuclear involved in the execution of the Program in order to comply with the requirements of CSA N285.5.
- With respect to aging management, N-PROG-MA-0017 R008 interfaces with the Integrated Aging Management Program to evaluate the condition of equipment (including containment structures) on an ongoing basis as defined by component programs, including management of equipment aging activities that: 1) identify aging related degradation mechanisms, 2) define maintenance, inspection, test, and other activities, and 3) initiate preventive action to preclude component or system failure. A condition assessment process is used to evaluate the health of critical components and establish actions necessary to maintain long-term health. Condition assessments are prepared in accordance with N-PROC-MP-0060 R005 B, "Aging Management Process" [B.5-21]. Requirements for reporting (such as summarizing significant aging issues in component health reports) are identified in this procedure.
- Non-Destructive Examination required for the periodic inspection of containment components is performed in accordance with N-STD-MA-0021 R001, "Non-Destructive Examination" [B.5-22] to ensure that these examinations meet code and jurisdictional requirements.

N-PROG-MP-0001 R014, "Engineering Change Control" [B.5-23], N-FORM-10959 R016, "Design Scoping Checklist" [B.5-24] and N-PROC-MP-0090 R014, "Modification Process" [B.5-25] ensure that any future design changes made to the CCSs comply with modern applicable codes, standards, and regulations (including CSA N287.1-14, per OPG Guideline N-GUID-00590-00002 R001, "Code over Code Review - Guideline" [B.5-26]) as applicable.

The above-mentioned Programs, Procedures and Standards are credited with the ability to detect and monitor any safety significant degradation mechanisms and thus to provide assurance of continued fitness for service of the Pickering NGS CCSs.

Pickering Units 1,4

Pickering A Basis for Return to Service identified N287.1-93 [B.5-6] for review based on a "Direct & Immediate Effect on Installed Design Features" [B.5-27]. AECL Assessment Document 44RS-00531-ASD-001 Rev. 04 [B.5-28] documents a clause by clause review against N287.1-93. 44RS-00531-ASD-001 Rev. 04 concluded:

> *Based on the present study and the evaluation of documents, it is concluded that the design of Pickering 'A' containment structure meets the intent of the current CSA standards* [CSA N287.1 and N287.3]*. As the design was done years ago utilizing NBC 1965, it is not possible to meet every aspect of newly developed codes. However after a review of the requirements in old and new codes and project specific documents it can be concluded that the changes and additions that have occurred in new codes do not have impact on the performance of the containment structure, namely the pressure retaining capability and leak tightness… The compliance discussion provided justifies that the containment structure for Pickering A Nuclear Power Plant meets the intent of the current CSA standards. Therefore, no special measures or recommendations are necessary.*

Similar to the Pickering B ISR review of N287.1-93, 44RS-00531-ASD-001 Rev. 04 identified a number of areas Pickering A was considered to be in "Indirect Compliance"; these were primarily related to the inability to recover old records or to demonstrate the standard the in-situ plant was built to. The rationale for these findings being Indirect Compliances was reviewed for PSR2 and confirmed to not be impacted by Pickering NGS operation past 2020. The concrete structures at Pickering A were built and tested to meet the 1965 NBCC requirements [B.5-29] and associated CSA A23 Series Standards, supplemented by specific loading requirements and the requirements of Design Manuals (e.g., see [B.5-30], [B.5-31], [B.5-32], [B.5-33], [B.5-34], [B.5-35]). As discussed earlier, ongoing confirmation that the Pickering NGS CCSs remain fit for service is demonstrated via periodic inspection and in-service testing. Further, the Engineering Change Control (ECC) process ensures that any future design changes made to the CCSs comply with the latest version of N287.1 [B.5-26] as applicable.

Darlington NGS

A review against CSA N287.1-93 (R2004) [B.5-6] was performed as part of the Darlington ISR. This review is documented in OPG Report NK38-REP-03680-10034 R000 [B.5-36] which concluded "The review did not identify any ISR Gaps and found that Darlington NGS was compliant with CAN/CSA-N287.1-93." CNSC staff comments and OPG responses related to the N287.1-93 code review were documented in OPG Report NK38-REP-03680-10034-ADD-001 R000, "Addendum to the CSA N287.1-93 Code Review Report for Darlington ISR" [B.5-37], which identified three new gaps to reflect CNSC/OPG dialogue.

Similar to the Pickering Units 5-8 and 1,4 reviews above, the Darlington gaps against N287.1 were related to the retrievability of old records and the ability to demonstrate the standard the in-situ plant was built to. In addition, similar to the previously mentioned Pickering reviews, it was argued in OPG Report NK38-REP-03680-10104-ADD-01 R000, "Darlington NGS Integrated Safety Review (ISR) - Final ISR Report Addendum" [B.5-38] that the changes in N287.1 will not impact the safe operation of the CCSs given robust design practices at the time, together with ongoing periodic inspection and in-service testing. As a result, the gaps were ranked as low safety significance and resolved to be Acceptable Deviations [B.5-38]. Therefore, there are no PSR2 gaps associated with the Darlington ISR assessment of N287.1-93.

B.5.2.2 Application of Post PSR1 Reviews

Per the CSA N287.1-14 Impact Statement [B.5-3], the following is a summary of the significant changes from the previous edition of the standard (N287.1-93 (R2004) [B.5-6], which was reviewed as part of PSR1):

- The scope has been broadened so that the standard applies to CCSs for nuclear power plants regardless of plant design type (technology neutral). This is reflected in the changed title of the standard.
- Additional figures have been included to clarify the definitions of jurisdictional boundaries.
- Requirements for aging management and repairs have been added; Consideration to aging is given, especially in view of the life extensions.
- In-service examination and testing requirements have been added, to clarify the objectives of in-service examination and defines the responsibilities of the personnel involved.
- Requirements for personnel qualification to perform construction inspections as well as in-service examinations and testing have been added.
- Reference standards and definitions are updated to current standards and to align with other N287 standards.
- Adjustments and verification for consistency with updated CNSC regulatory and guidance documents and international safety standards.

OPG Report N-REP-00590-00004 R000, "Code-Over-Code Review Report: N287.1 for the Year 2014" [B.5-39] provides a clause by clause review of N287.1-14 versus N287.1-93, and states:

> This code over code review has *identified significant technical changes and appropriate mitigation plans have been identified, in Appendix A. For details of mitigation plans see items #3, 13, 14, 22-24, and 28-30 of table in Appendix A.*

The following table provides extracts from the Appendix A table [B.5-39] outlining the changes and associated mitigation plans:

N287.1 Clause	Change	Significance	Mitigation
4.1.3 Concrete containment structures and concrete components affecting the leak-tightness of the containment boundary shall be designed to allow leakage rate testing (LRT) …	The new clause requires penetrations to CCSs be able to individually leak testable, where practical [sic].	Although unlikely, the requirement may be applicable to any newly designed penetrations for containment structure as part of a modification to existing power plants.	This new requirement will be included in the next revision of N-LIST-00590-00001.
6.2 Design specifications The design specifications shall include, but not be limited to …	The existing clause has been modified to include 4 new sub-clauses in design specifications.	These new requirements apply to design specifications for new build or modifications of existing CCSs.	This new requirement will be included in the next revision of N-LIST-00590-00001.
6.4 Drawings g) the position, details, and size of any modular-type structures	New bullet is added to show location of modular type structures in containment structures.	New requirement.	This new requirement will be included in the next revision of N-LIST-00590-00001.
7.2.2.1 Commissioning documents shall …	New bullets are added to include more information in commissioning documents.	New requirement, may apply to modifications carried out to existing power plants.	This new requirement will be included in the next revision of N-LIST-00590-00001.
7.2.3 Test procedures i) personnel qualification requirements;	New bullet is added to include personnel qualification requirements in commissioning test procedures.	New requirement, may apply to modifications carried out to existing power plants.	This new requirement will be included in the next revision of N-LIST-00590-00001.
7.2.4 Report The commissioning report shall include …	New bullets are added to include more information in commissioning reports.	New requirement, may apply to modifications carried out to existing power plants.	This new requirement will be included in the next revision of N-LIST-00590-00001.
9.3.4.2 To qualify as a Level I inspector, a person shall have completed …	The new clause is more restrictive in terms of personnel qualifications, applicable to inspectors performing inspections related to modifications or new construction.	New requirement.	This new requirement will be included in the next revision of N-LIST-00590-00001.

N287.1 Clause	Change	Significance	Mitigation
9.3.4.4 To qualify as a Level II inspector, a person shall have completed …	The new clause is more restrictive in terms of personnel qualifications, applicable to inspectors performing inspections related to modifications or new construction.	New requirement.	This new requirement will be included in the next revision of N-LIST-00590-00001.
9.3.4.6 To qualify as a Level III inspector, a person shall have completed …	The new clause is more restrictive in terms of personnel qualifications, applicable to inspectors performing inspections related to modifications or new construction.	New requirement.	This new requirement will be included in the next revision of N-LIST-00590-00001.

The mitigation instituted for each gap was to include the changes to CSA N287.1-14 in the next revision of N-LIST-00590-00001 R002, "List of Significant Technical Changes from Code-over-Code Review" [B.5-40]. All modifications require review of this document as identified in N-FORM-10959 R016, "Design Scoping Checklist" [B.5-24], as per N-PROC-MP-0090 R014, "Modification Process" [B.5-25] (N-FORM-10959 R016 Section 2.19 requires that a review of N-LIST-00590-00001 R002 "shall be completed to determine if any code change improvement actions apply to the modification"). As a result, significant technical changes will be applied, as appropriate, for modifications to existing Pickering NGS installations going forward in accordance with OPG governance. Based on the above, CSA N287.1-14 has no gaps that need to be addressed for Pickering NGS, unless modifications are made in the future. Therefore, there are no gaps for PSR2 against the changes in N287.1-14.

In summary, the ECC process (which includes review of N-LIST-00590-00001 R002 [B.5-40]), together with a yearly examination of any incremental (Code-over-Code) review requirements due to changes to N287.1 [B.5-26], ensures that any design changes made to the Pickering NGS CCSs comply with the latest version of N287.1 going forward, as applicable. With respect to the retroactive application to Pickering NGS, it is not practicable to make design changes to CCSs without rebuilding them, and ongoing confirmation that the CCSs remain fit for service is demonstrated via periodic inspections and in-service testing. Therefore, there are no PSR2 gaps for Pickering NGS compliance with CSA N287.1-14.

B.5.3 Compliance Summary for Pickering PSR2

There are no PSR2 gaps for CSA N287.1-14 [B.5-1]. Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with CSA N287.1-14.

B.5.4 References

[B.5-1] CSA Standard N287.1-14, *General Requirements for Concrete Containment Structures for Nuclear Power Plants*, February 2014.

[B.5-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.5-3] CSA Impact Statement for Publication, *Product: New Edition; Product Designation: CSA N287.1*, *Date of Release: February, 2014*, Date not provided.

[B.5-4] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.5-5] OPG Report, NK30-REP-03680-00001 R000, *Pickering NGS-B Integrated Safety Review – Plant Design Safety Factor*, August 2007.

[B.5-6] CSA Standard N287.1-93 (R2004), *General Requirements for Concrete Containment Structures for Nuclear Power Plants*, July 1993.

[B.5-7] OPG Report, NK30-REP-03680-00015 R000, *Pickering NGS-B Integrated Safety Review (ISR) – Final ISR Report*, August 2009.

[B.5-8] National Research Council of Canada, *National Building Code of Canada, 1970,* 1971.

[B.5-9] Ontario Hydro Pickering Generating Station B Design Manual, NK30-21000 R002, *Reactor Building - General*, issued February 1980, revised September 1988 and February 1989.

[B.5-10] Ontario Hydro Pickering Generating Station B Design Manual, NK30-21040 R001, *Reactor Building Floor Loadings*, issued August 1979, revised September 1988.

[B.5-11] Ontario Hydro Pickering Generating Station B Design Manual, NK30-21140 R002, *Reactor Building Foundation and Perimeter Wall*, issued November 1979, revised August 1982 and November 1988.

[B.5-12] Ontario Hydro Pickering Generating Station B Design Manual, NK30-21149 R002, *Reactor Building Dome*, issued January 1979, revised August 1982 and December 1988.

[B.5-13] AECL Engineering Design Guide, NK30-REF-68000-0379145 (DG-30-68000-6), *Containment Provisions for Extensions of the Containment Envelope for Pickering NGS B*, May 1977.

[B.5-14] AECL Engineering Design Guide, DG-00-01040-1 Rev. 02, *Earthquake Design Requirements for CANDU Nuclear Power Plants*, November 1974.

[B.5-15] OPG Design Basis Document, NK30-DBD-34200-00001 R000, *Containment System DBD*, February 2000.

[B.5-16] Specification L-715-80, *Specification for Concrete Placing and Workmanship.*

[B.5-17] Tendering and Contract Document, NK30-REF-20541-{47603}, *Supply of Pre-Mix Concrete in Ready Mix Trucks - Units 5-8*, NK30-LH-20541-01, April 1974.

[B.5-18] OPG Program, N-PROG-MA-0017 R008, *Component and Equipment Surveillance*, June 1, 2015.

[B.5-19] OPG Procedure, N-PROC-MA-0066 R005, *Administrative Requirements for In-Service Examination and Testing for Concrete Containment Structures*, April 24, 2014.

[B.5-20] OPG Procedure, N-PROC-MA-0064 R005, *Administrative Requirements for the Periodic Inspection of Nuclear Power Plant Containment Components*, October 24, 2013.

[B.5-21] OPG Procedure, N-PROC-MP-0060 R005 B, *Aging Management Process*, October 1, 2015.

[B.5-22] OPG Standard, N-STD-MA-0021 R001, *Non-Destructive Examination*, July 30, 2015.

[B.5-23] OPG Program, N-PROG-MP-0001 R014, *Engineering Change Control*, January 7, 2015.

[B.5-24] OPG Form, N-FORM-10959 R016, *Design Scoping Checklist*, June 2016.

[B.5-25] OPG Procedure, N-PROC-MP-0090 R014, *Modification Process*, October 14, 2016.

[B.5-26] OPG Guideline, N-GUID-00590-00002 R001, *Code over Code Review - Guideline*, August 15, 2016.

[B.5-27] OPG Letter, NA44-CORR-00531-00381, R.J. Strickert to J.S.C Tong, *Pickering A – Updated Basis for Return to Service Document*, April 20, 2001.

[B.5-28] AECL Assessment Document, 44RS-00531-ASD-001 Rev. 04, *Review of Pickering A Design Against Current Codes and Standards*, November 2000.

[B.5-29] National Research Council of Canada, *National Building Code of Canada, 1965*, 1966.

[B.5-30] Hydro Electric Power Commission of Ontario Pickering Generating Station Design Manual, NA44-21000 R000, *Reactor Building - General*, December 1972.

[B.5-31] Hydro Electric Power Commission of Ontario Pickering Generating Station Design Manual, NA44-21040 R000, *Reactor Building Floor Loadings*, April 1969.

[B.5-32] Hydro Electric Power Commission of Ontario Pickering Generating Station Design Manual, NA44-21140 R000, *Reactor Building Foundation and Perimeter Wall*, April 1970.

[B.5-33] Hydro Electric Power Commission of Ontario Pickering NGS Design Manual, NA44-21149, *Reactor Building Dome*, February 1970.

[B.5-34] Hydro Electric Power Commission of Ontario Pickering Generating Station Design Manual, NA44-25000 R001, *Vacuum Building*, August 1982.

[B.5-35] OPG System Design Requirements, NA44-DR-63420-10001 R000, *Negative Pressure Containment System – Overview*, 2004.

[B.5-36] OPG Report, NK38-REP-03680-10034 R000, Review of CAN/CSA-N287.1-93 (R2004) (July 1993), *General Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants for Darlington Integrated Safety Review*, June 2011.

[B.5-37] OPG Report, NK38-REP-03680-10034-ADD-001 R000, *Addendum to the CSA N287.1-93 Code Review Report for Darlington ISR*, January 2014.

[B.5-38] OPG Report, NK38-REP-03680-10104-ADD-001 R000, *Darlington NGS Integrated Safety Review (ISR) – Final ISR Report Addendum*, June 2013.

[B.5-39] OPG Report, N-REP-00590-00004 R000, *Code-Over-Code Review Report: N287.1 for the Year 2014*, November 2014.

[B.5-40] OPG List, N-LIST-00590-00001 R002, *List of Significant Technical Changes from Code-Over-Code Review*, August 2015.

B.6 CSA N287.3-14, "Design Requirements for Concrete Containment Structures for Nuclear Power Plants"

B.6.1 Background

The following, paraphrased from the Preface of CSA N287.3-14 [B.6-1], provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *The purpose of CSA N287.3 is to ensure concrete containment structures are designed for sufficient strength, structural integrity, and required leak-tightness under operating and test conditions, postulated accident, environmental conditions, or combinations thereof.*
>
> *CSA N287.3 provides requirements for the design of concrete containment structures of a containment system, designated as "class containment" components and parts as defined in CSA N287.1, and addresses their beyond design basis assessment.*

As identified in Section 1 of N287.3-14 [B.6-1]:

> *This Standard is applicable to new nuclear power plants' concrete containment structures to be built in Canada. Application of the Standard to concrete containment structures to be built outside Canada is subject to approval of the authority having jurisdiction (AHJ). The application of the Standard to existing or operating nuclear power plants is as agreed upon by the owner/operator and the AHJ.*

CSA N287.3 is directly relevant to Safety Factor 1 (Plant Design).

CSA N287.3 is identified in Appendix E.1 of the R04 Pickering Licence Conditions Handbook [B.6-2] as "Guidance or Criteria". N287.3-14 is the fourth edition of this standard which supersedes the previous editions published in 1993, 1982 and 1978 under the title "Design Requirements for Concrete Containment Structures for CANDU nuclear power plants". The CSA N287.3-14 Impact Statement [B.6-3] provides a "Summary of significant changes from previous edition" which identifies several changes to the Standard as described in Section B.6.2.2 below.

The results of PSR1 CSA N287.3 reviews (Pickering A Return to Service assessments, and Pickering and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.6.2. As identified in Reference [B.6-4], the Pickering PSR2 review of CSA N287.3-14 is an Incremental Review. PSR2 Incremental Review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.
- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.6.2 Compliance Assessment for Pickering PSR2

B.6.2.1 Application of PSR1 Reviews

The versions of N287.3 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

<u>Pickering Units 5-8</u>

As part of the Pickering B ISR Plant Design Safety Factor, OPG Report NK30-REP-03680-00001 R000 [B.6-5] performed a clause by clause review of N287.3-93 (R2004) [B.6-6] against OPG governance. NK30-REP-03680-00001 R000 concluded:

> *The compliance evaluation of the CSA N287.3-93 Standard has determined that in itself it does not introduce any new requirements that impact the design basis of the CCSs* [Concrete Containment Structures]*. No discrepancies were identified. However, a number of clauses of CSA N287.3 refer to CSA A23.1, CSA A23.3 and other standards. These standards also contain design requirements and they have been revised as recently as 2005. They may, therefore, contain requirements that were not present in the versions of these standards that applied at the time the CCSs were designed and built.* [The vendor] *has not conducted a review of A23.1, A23.3 or any other standard referred to, as it is not part of its current scope of work.*
>
> *… It is noted that the objectives of an independent safety review are to assess the fitness of the existing plant design for life extension and then to establish a plan to ensure continued fitness for service. It is also noted that it is not feasible to make design changes that would extend the life of the CCSs without rebuilding them. Hence, conducting a detailed assessment of the standards called up in CSA N287.3 to determine what new design requirements have been introduced would provide no safety benefit and would not help to achieve the objectives of the ISR.*

A large number of findings were identified in the NK30-REP-03680-00001 R000 review, which generally related to reference in N287.3 to other standards that apply to CCS design (including CSA A23 and the N287 Series of standards), as well as considerations for seismic ground motions. All findings were assessed to be Acceptable Deviations given robust design practices at the time, together with ongoing periodic inspections and in-service testing. OPG Report NK30-REP-03680-00015 R000, "Pickering NGS-B Integrated Safety Review (ISR) - Final ISR Report" [B.6-7] grouped the findings into ISR Issues 1-274/276, I-253 to -273 and 1-275/277/278/279, and confirmed their status as Acceptable Deviations with very low safety significance (i.e., priority level 4) with no further action required.

The rationale for these findings being Acceptable Deviations is not impacted by Pickering NGS operation past 2020. CSA N287.3 ensures CCSs are designed for sufficient strength, structural integrity, and required leak-tightness under operating and test conditions, postulated accident and environmental conditions. The CCSs at Pickering B were built and tested to meet 1970

National Building Code of Canada (NBCC) requirements [B.6-8], supplemented by specific loading requirements and the requirements of Design Manuals (e.g., see [B.6-9], [B.6-10], [B.6-11], [B.6-12]) and Design Guides (e.g., see [B.6-13], [B.6-14], [B.6-15]). The original Pickering concrete specifications (L-715-80 [B.6-16] and NK30-LH-20541-01 [B.6-17]) included requirements for quality control and compliance with CSA A23.1, A23.2 and A23.3 (which address concrete materials, methods of concrete construction and test methods and standard practices for concrete). As such, the standards that applied during original construction of Pickering NGS included requirements for tests and quality control procedures to ensure that the concrete used in the as-built structures met the original design requirements. Ongoing confirmation that the Pickering NGS CCSs remain fit for service is demonstrated via the following:

- As required by N-PROG-MA-0017 R008 [B.6-18]:

    - Periodic examinations of in-service inspections and positive leakage rate testing of CCSs that are designated as class containment components are performed in accordance with the requirements of CSA N287.7, "In Service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants". N-PROC-MA-0066 R005, "Administrative Requirements for In-Service Examination and Testing for Concrete Containment Structures" [B.6-19] describes the administrative process and the roles of various organizations across OPG Nuclear involved in the execution of N-PROG-MA-0017 R008 in order to comply with the requirements of CSA N287.7.

    - Periodic inspection of containment system metallic and plastic components (including the containment pressure suppression systems) is performed in accordance with the requirements of CSA N285.5, "Periodic inspection of CANDU Nuclear Power Plant Containment Components". N-PROC-MA-0064 R005, "Administrative Requirements for the Periodic Inspection of Nuclear Power Plant Containment Components" [B.6-20], describes the administrative process and the roles of various organizations across OPG Nuclear involved in the execution of the Program in order to comply with the requirements of CSA N285.5.

- With respect to aging management, N-PROG-MA-0017 R008 interfaces with the Integrated Aging Management Program to evaluate the condition of equipment (including containment structures) on an ongoing basis as defined by component programs, including management of equipment aging activities that: 1) identify aging related degradation mechanisms, 2) define maintenance, inspection, test, and other activities, and 3) initiate preventive action to preclude component or system failure. A condition assessment process is used to evaluate the health of critical components and establish actions necessary to maintain long-term health. Condition assessments are prepared in accordance with N-PROC-MP-0060 R005 B, "Aging Management Process" [B.6-21]. Requirements for reporting (such as summarizing significant aging issues in component health reports) are identified in this procedure.

- Non-Destructive Examination required for the periodic inspection of containment components is performed in accordance with N-STD-MA-0021 R001, "Non-Destructive

Examination" [B.6-22] to ensure that these examinations meet code and jurisdictional requirements.

N-PROG-MP-0001 R014, "Engineering Change Control" [B.6-23], N-FORM-10959 R016, "Design Scoping Checklist" [B.6-24] and N-PROC-MP-0090 R014, "Modification Process" [B.6-25] ensure that any future design changes made to the CCSs comply with modern applicable codes, standards, and regulations (including CSA N287.3-14, per OPG Guideline N-GUID-00590-00002 R001, "Code over Code Review - Guideline" [B.6-26]) as applicable.

The above-mentioned Programs, Procedures and Standards are credited with the ability to detect and monitor any safety significant degradation mechanisms and thus to provide assurance of continued fitness for service of the Pickering NGS CCSs.

<u>Pickering Units 1,4</u>

Pickering A Basis for Return to Service identified N287.3-93 [B.6-6] for review based on a "Direct & Immediate Effect on Installed Design Features" [B.6-27]. AECL Assessment Document 44RS-00531-ASD-001 Rev. 04 [B.6-28] documents the clause by clause review of N287.3-93. 44RS-00531-ASD-001 Rev. 04 concluded:

> *Based on the present study and the evaluation of documents, it is concluded that the design of Pickering 'A' containment structure meets the intent of the current CSA standards* [CSA N287.1 and N287.3]*. As the design was done years ago utilizing NBC 1965, it is not possible to meet every aspect of newly developed codes. However after a review of the requirements in old and new codes and project specific documents it can be concluded that the changes and additions that have occurred in new codes do not have impact on the performance of the containment structure, namely the pressure retaining capability and leak tightness…*
>
> *The compliance discussion provided justifies that the containment structure for Pickering A Nuclear Power Plant meets the intent of the current CSA standards. Therefore, no special measures or recommendations are necessary.*

Similar to the Pickering B ISR review of N287.3-93, 44RS-00531-ASD-001 Rev. 04 identified a number of areas in which Pickering A was considered to be in "Indirect Compliance". The rationale for these findings being Indirect Compliances was reviewed for PSR2 and confirmed to not be impacted by Pickering NGS operation past 2020. The concrete structures at Pickering A were built and tested to meet the 1965 NBCC requirements [B.6-29] and associated CSA A23 Series Standards, supplemented by specific loading requirements and the requirements of Design Manuals (e.g., see [B.6-30], [B.6-31], [B.6-32], [B.6-33], [B.6-34], [B.6-35]). As discussed earlier, ongoing confirmation that the Pickering NGS CCSs remain fit for service is demonstrated via periodic inspection and in-service testing. Further, the Engineering Change Control (ECC) process ensures that any future design changes made to the CCSs comply with the latest version of N287.3 [B.6-26] as applicable.

Darlington NGS

A review against N287.3-93 (R2004) [B.6-6] was performed as part of the Darlington ISR. This review is documented in OPG Report NK38-REP-03680-10036 R000 [B.6-36] which concluded:

> *Based on the results of the review of Darlington NGS concrete containment structures against CSA Standard N287.3-93 (R2004), it is declared that Darlington NGS concrete containment structures are compliant, except for the gaps identified … by comparing the clauses of CSA Preliminary Standard N287.3-1978 which is the original design standard against the clauses of CSA Standard N287.3-93 (R2004). After reviewing the design related documents of the Darlington NGS concrete containment structures, the gaps were confirmed.*

OPG Report NK38-REP-03680-10036 R000 also includes an appendix with gap revisions made subsequent to the code review report. The Darlington gaps against N287.3 were related to new requirements involving anchorage systems, redundancy of mechanical splices, and maximum concrete tensile stresses for liners of CCSs. OPG Report NK38-REP-03680-10201 R001, "ISR Open Issues and Acceptable Deviations - Adequacy Review" [B.6-37] stated that all three of these gaps were Acceptable Deviations, providing arguments similar to those made for the Pickering Units 5-8 and 1,4 reviews discussed above, i.e.:

- although there were no prescriptive specifications for, and testing of, the materials used in concrete anchors at the time of construction, manufacturers did meet testing requirements at that time; and

- the periodic inspections and routine leakage tests of the containment envelope verify that there is no significant deterioration in the integrity of the CCSs, and any potential degradation from an overall concrete containment perspective will be observed and action taken accordingly if required.

These arguments also apply to Pickering NGS.

CNSC staff comments and OPG responses related to the CSA N287.3-93 code review were documented in OPG Report NK38-REP-03680-10036-ADD-001 R000 [B.6-38]. Two new gaps were opened to reflect CNSC/OPG dialogue which related to documentation for concrete reinforcement covers and anchorage systems. OPG Report NK38-REP-03680-10201 R001 [B.6-37] was follow-up work to demonstrate that the in-situ plant meets requirements. This work was tracked under Action Request (AR) #28153581 and calculations were subsequently completed which demonstrated that: a) anchorage systems meet the intent of N287.3-93, and b) the construction specifications for concrete covers exceeded N287.3-93 requirements. Both gaps were subsequently closed. The Pickering B ISR and Pickering A Return to Service assessments previously assessed against these N287.3-93 clauses were classified either as Acceptable Deviations or Indirect Compliances, citing the rationale discussed earlier. As a result, there are no PSR2 gaps which result from review of the Darlington ISR results.

B.6.2.2 Application of Post PSR1 Reviews

Per the CSA N287.3-14 Impact Statement [B.6-3], the following is a summary of the significant changes from the previous edition of the standard (N287.3-93 (R2004) [B.6-6], which was reviewed as part of PSR1):

- *The scope has been broadened so that the standard applies to concrete containment structures for nuclear power plants regardless of plant design type (technology neutral). This is reflected in the changed title of the standard.*
- *The new edition clarifies several ambiguities in the previous edition. These include several definitions, load and load combinations, and stress and strain limits.*
- *New load and load factors are introduced. Three additional load combinations are defined to represent the construction loads.*
- *New requirements for beyond design basis are introduced by adding a new Annex. A new Annex C provides guidelines for predicting the ultimate pressure capacity of concrete containments. This new Annex provides a measure of the safety margin above the design-basis internal accident pressure.*
- *The new edition considers the requirements in the revised A23.3-04 (R2010).*
- *Design requirements for openings or penetrations, penetration assemblies, and attachments are added.*
- *Significant changes are made to the clause on Anchorage. The Annex D in A23.3-04 (R2010) has been adopted and the sections are revised to reflect this change.*
- *New requirements on liners are added. Strain limits for service load category is defined for metallic liners.*
- *New requirements for serviceability are added. Stress limits for service load category are defined.*
- *Reference standards and definitions are updated.*
- *New definitions and list of symbols are added.*
- *New requirements for seismic design of walls, slabs, shells, and domes are introduced.*

OPG Report N-REP-00590-00005 R000, "Code-Over-Code Review Report: N287.3 for the Year 2014" [B.6-39] provides a clause by clause review of N287.3-14 versus N287.3-93, and states:

> *This code over code review has identified significant technical changes and appropriate mitigation plans have been identified, in Appendix A. Some of the significant changes can only be met during design, construction, or fabrication of new concrete containment structures, such as new build. Such changes have been*

identified to be non-applicable to existing OPGN facilities. Mitigation plans, for requirements applicable to existing OPGN facilities, have been detailed in Appendix A.

The following table provides extracts from the Appendix A table [B.6-39] outlining the changes and mitigation plans:

N287.3-14 Clause	Change	Significance	Mitigation
6.5.6 Penetration assemblies shall be analyzed and designed following the methodologies and procedures specified in …	Further clarifications have been added as a requirement.	New requirement can be met during design of new nuclear power plants with liner. May apply to modifications done in existing facilities.	This new requirement will be added in the revision to the N-LIST-00590-00001.
6.5.7 When attachments in the form of steel structures are mounted onto the metal liner or containment wall, they shall be designed and analyzed…	Requirement applicable to design of attachments to steel liner have been added.	New Requirement may be applicable to any future modification done in existing facilities.	This new requirement will be added in the revision to the N-LIST-00590-00001.
Table 7.1 Construction Dead Load (D) factor = 1.4	A new load combination is introduced in accordance with NBCC 2010.	New Requirement may be applicable to any future modification done in existing facilities	This new requirement will be added in the revision to the N-LIST-00590-00001.
8.2.1.2 For the service load category (factored), the design shall also meet the requirements of CAN/CSA-A23.3	This new Requirement applies for design of concrete containment for new power plants.	Any potential Modifications may have to comply with this requirement.	This new requirement will be added in the revision to the N-LIST-00590-00001.
9.2.3 Companies responsible for welded splices and welded connections shall comply with CSA W186.	Qualifications for companies have been included in.	Any potential modifications may have to comply with this requirement.	This new requirement will be added in the revision to the N-LIST-00590-00001.
11.6.3 Flat slabs with or without beams shall be reinforced and detailed in accordance with …	New clause provides reinforcement detailing provisions during construction/ design of new CCSs. This requirement cannot be applied to existing concrete structures.	This requirement can be met by new build power plants. This new requirement may apply to existing nuclear power plants.	This new requirement will be added in the revision to the N-LIST-00590-00001.

N287.3-14 Clause	Change	Significance	Mitigation
13.3.1.5 Penetration assemblies shall be designed to accommodate all design loads and deformations to ensure structural integrity and leak-tightness.	New requirement for design of penetrations for CCSs.	The new requirement may be applicable to modifications to existing CCSs.	This new requirement will be added in the revision to the N-LIST-00590-00001.
13.3.4.2 Consideration of the following effects that can cause shear loads and displacements shall include …	Three new bullets have been added to address design of embedded parts.	May apply to design of modifications for existing plants.	This new requirement will be added in the revision to the N-LIST-00590-00001.
13.3.5.4 Companies responsible for the welding of metallic parts shall comply with the requirements of CSA W47.1 Division 1 or 2.	Clarification to use W47.1 for welding companies' qualifications.	May apply to modification of existing plants.	This new requirement will be added in the revision to the N-LIST-00590-00001.
14.1 Objective … requirements for the design of cast-in and post-installed anchors and inserts used in a nuclear containment structure …	Anchorage design requirements have been updated to align with CSA A23.3.	It may apply to modifications to concrete structures involving anchorage and a Document Change Request (DCR) has been initiated to update direction on use of anchors in containment to comply with CSA N287.3-14.	This new requirement will be added in the revision to the N-LIST-00590-00001.
14.2.1 Anchor and anchor groups shall be designed for critical effects of factored loads as determined by elastic analysis. …	Anchorage design requirements have been updated to align with CSA A23.3.	It may apply to modifications to concrete Structures involving anchorage and a DCR has been initiated to update direction on use of anchors in containment to comply with CSA N287.3-14.	This new requirement will be added in the revision to the N-LIST-00590-00001.
14.2.2 Post-installed anchors shall be designed assuming cracked concrete unless it can be demonstrated …	Anchorage design requirements have been updated to align with CSA A23.3.	It may apply to modifications to concrete Structures involving anchorage and a DCR has been initiated to update direction on use of	This new requirement will be added in the revision to the N-LIST-00590-00001.

N287.3-14 Clause	Change	Significance	Mitigation
		anchors in containment to comply with CSA N287.3-14.	
14.3.1 The design of concrete anchorage systems shall be in accordance with CAN/CSAA23.3 …	Anchorage design requirements have been updated to align with CSA A23.3.	It may apply to modifications to concrete Structures involving anchorage and a DCR has been initiated to update direction on use of anchors in containment to comply with CSA N287.3-14.	This new requirement will be added in the revision to the N-LIST-00590-00001.
14.3.2.2 When shear is transmitted by the bearing of shear lugs and embedded plates on the concrete, the factored concrete breakout resistance shall …	Anchorage design requirements have been updated to align with CSA A23.3.	It may apply to modifications to concrete Structures involving anchorage and a DCR has been initiated to update direction on use of anchors in containment to comply with CSA N287.3-14.	This new requirement will be added in the revision to the N-LIST-00590-00001.
Table B1 Recommended Dynamic Increase Factors (DIF)	New clause/ Table has added DIF factors for Carbon steel and Stainless steel.	Applies to design of modifications, if applicable.	This new requirement will be added in the revision to the N-LIST-00590-00001.
Table B2 Permissible Ductility Ratios for Beams, Slabs, and Walls	New requirements have increased ductility ratios.	Applies to design of modifications, if applicable.	This new requirement will be added in the revision to the N-LIST-00590-00001.
B.4.2.4 In the design, the required barrier or wall thickness to prevent perforation shall …	New requirement	Applies to design of modifications, if applicable.	This new requirement will be added in the revision to the N-LIST-00590-00001.
B.4.3.2 When the force-time history is not used (e.g., in the simplified method), the conservation of momentum and energy may be applied …	New requirement	Applies to design of modifications, if applicable	This new requirement will be added in the revision to the N-LIST-00590-00001.

The mitigation instituted for these gaps was to include the changes to CSA N287.3-14 in the next revision of N-LIST-00590-00001 R002, "List of Significant Technical Changes from Code-over-Code Reviews" [B.6-40]. All modifications require review of this document as identified in N-FORM-10959 R016, "Design Scoping Checklist" [B.6-24], as per N-PROC-MP-0090 R014, "Modification Process" [B.6-25] (N-FORM-10959 R016 Section 2.19 requires that a review of N-LIST-00590-00001 R002 "shall be completed to determine if any code change improvement actions apply to the modification"). As a result, significant technical changes will be applied, as appropriate, for modifications to existing Pickering NGS installations going forward in accordance with OPG governance.

It is noted that Code-over-Code report N-REP-00590-00005 R000 [B.6-39] also identified a number of changes that can only be met during design, construction, or fabrication of new CCSs (i.e., clauses 9.3.2.3 (mechanical splices), 11.5 (frame joints), 11.6.2 (flat shear walls), 11.6.4/.5/.6 (shear strength of curved walls), 13.3.2.1 (liner requirements) and A.3.2.3 (abnormal/environmental load combinations)); these changes were identified to be non-applicable to existing OPG Nuclear facilities and therefore not placed in N-LIST-00590-00001 R002. Although these changes have not been listed in N-LIST-00590-00001 R002, it is generally not practicable to make design changes to CCSs without rebuilding them. As a result, CSA N287.3-14 has no gaps that need to be addressed for the existing stations, unless modifications are made in the future. Therefore, there are no gaps for PSR2 against the changes in N287.3-14.

In summary, the ECC process (which includes review of N-LIST-00590-00001 R002 [B.6-40]), together with a yearly examination [B.6-26] of any incremental (Code-over-Code) review requirements due to changes to N287.3, ensures that any design changes made to the Pickering NGS CCSs comply with the latest version of N287.3 going forward, as applicable. With respect to the retroactive application to Pickering NGS, it is not practicable to make design changes to CCSs without rebuilding them, and ongoing confirmation that the CCSs remain fit for service is demonstrated via periodic inspections and in-service testing. Therefore, there are no PSR2 gaps for Pickering NGS compliance with CSA N287.3-14.

B.6.3 Compliance Summary for Pickering PSR2

There are no PSR2 gaps for CSA N287.3-14 [B.6-1]. Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with CSA N287.3-14.

B.6.4 References

[B.6-1] CSA Standard, N287.3-14, *Design Requirements for Concrete Containment Structures for Nuclear Power Plants*, February 2014.

[B.6-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.6-3] CSA Impact Statement for Publication, *Product: New Edition; Product Designation: CSA N287.3, Date of Release: February 2014*, Date not provided.

[B.6-4] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.6-5] OPG Report, NK30-REP-03680-00001 R000, *Pickering NGS-B Integrated Safety Review – Plant Design Safety Factor*, August 2007.

[B.6-6] CSA Standard, N287.3-93 (R2004), *Design Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants*, February 1993.

[B.6-7] OPG Report, NK30-REP-03680-00015 R000, *Pickering NGS-B Integrated Safety Review (ISR) – Final ISR Report*, August 2009.

[B.6-8] National Research Council of Canada, *National Building Code of Canada*, *1970*, 1971.

[B.6-9] Ontario Hydro Pickering Generating Station B Design Manual, NK30-21000 R002, *Reactor Building - General*, issued February 1980, revised September 1988 and February 1989.

[B.6-10] Ontario Hydro Pickering Generating Station B Design Manual, NK30-21040 R001, *Reactor Building Floor Loadings*, issued August 1979, revised September 1988.

[B.6-11] Ontario Hydro Pickering Generating Station B Design Manual, NK30-21140 R002, *Reactor Building Foundation and Perimeter Wall*, issued November 1979, revised August 1982 and November 1988.

[B.6-12] Ontario Hydro Pickering Generating Station B Design Manual, NK30-21149 R002, *Reactor Building Dome*, issued January 1979, revised August 1982 and December 1988.

[B.6-13] AECL Engineering Design Guide, NK30-REF-68000-0379145 (DG-30-68000-6), *Containment Provisions for Extensions of the Containment Envelope for Pickering NGS B*, May 1977.

[B.6-14] AECL Engineering Design Guide, DG-00-01040-1 Rev. 02, *Earthquake Design Requirements for CANDU Nuclear Power Plants*, November 1974.

[B.6-15] OPG Design Basis Document, NK30-DBD-34200-00001 R000, *Containment System DBD*, February 2000.

[B.6-16] Specification L-715-80, *Specification for Concrete Placing and Workmanship.*

[B.6-17] Tendering and Contract Document, NK30-REF-20541-{47603}, *Supply of Pre-Mix Concrete in Ready Mix Trucks - Units 5-8*, NK30-LH-20541-01, April 1974.

[B.6-18] OPG Program, N-PROG-MA-0017 R008, *Component and Equipment Surveillance*, June 1, 2015.

[B.6-19] OPG Procedure, N-PROC-MA-0066 R005, *Administrative Requirements for In-Service Examination and Testing for Concrete Containment Structures*, April 24, 2014.

[B.6-20] OPG Procedure, N-PROC-MA-0064 R005, *Administrative Requirements for the Periodic Inspection of Nuclear Power Plant Containment Components*, October 24, 2013.

[B.6-21] OPG Procedure, N-PROC-MP-0060 R005 B, *Aging Management Process*, October 1, 2015.

[B.6-22] OPG Standard, N-STD-MA-0021 R001, *Non-Destructive Examination*, July 30, 2015.

[B.6-23] OPG Program, N-PROG-MP-0001 R014, *Engineering Change Control*, January 7, 2015.

[B.6-24] OPG Form, N-FORM-10959 R016, *Design Scoping Checklist*, June 2016.

[B.6-25] OPG Procedure, N-PROC-MP-0090 R014, *Modification Process*, October 14, 2016.

[B.6-26] OPG Guideline, N-GUID-00590-00002 R001, *Code over Code Review - Guideline*, August 15, 2016.

[B.6-27] OPG Letter, NA44-CORR-00531-00381, R.J. Strickert to J.S.C Tong, *Pickering A – Updated Basis for Return to Service Document*, April 20, 2001.

[B.6-28] AECL Assessment Document, 44RS-00531-ASD-001 Rev. 04, *Review of Pickering A Design Against Current Codes and Standards*, November 2000.

[B.6-29] National Research Council of Canada, *National Building Code of Canada, 1965*, 1966.

[B.6-30] Hydro Electric Power Commission of Ontario Pickering Generating Station Design Manual NA44-21000, *Reactor Building - General*, December 1972.

[B.6-31] Hydro Electric Power Commission of Ontario Pickering Generating Station Design Manual NA44-21040, *Reactor Building Floor Loadings*, April 1969.

[B.6-32] Hydro Electric Power Commission of Ontario Pickering Generating Station Design Manual NA44-21140, *Reactor Building Foundation and Perimeter Wall*, April 1970.

[B.6-33] Hydro Electric Power Commission of Ontario Pickering Generating Station Design Manual NA44-21149, *Reactor Building Dome*, February 1970.

[B.6-34] Hydro Electric Power Commission of Ontario Pickering Generating Station Design Manual NA44-25000-21300.2, *Vacuum Building*, August 1982.

[B.6-35] OPG System Design Requirements, NA44-DR-63420-10001 R000, *Negative Pressure Containment System – Overview*, 2004.

[B.6-36] OPG Report, NK38-REP-03680-10036 R000, *Review of CAN/CSA-N287.3-93 (R2004) (February 1993), Design Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants for Darlington Integrated Safety Review*, August 2011.

[B.6-37] OPG Report, NK38-REP-03680-10201 R001, *ISR Open Issues and Acceptable Deviations – Adequacy Review*, October 2014.

[B.6-38] OPG Report, NK38-REP-03680-10036-ADD-001 R000, *Addendum to the CSA N287.3-93 Code Review Report for Darlington ISR*, January 2014.

[B.6-39] OPG Report, N-REP-00590-00005 R000, *Code-Over-Code Review Report: N287.3 for the Year 2014*, November 2014.

[B.6-40] OPG List, N-LIST-00590-00001 R002, *List of Significant Technical Changes from Code-Over-Code Review*, August 2015.

B.7 CSA N287.5-11, "Examination and Testing Requirements for Concrete Containment Structures for Nuclear Power Plants"

B.7.1 Background

The following, paraphrased from the Preface of CSA N287.5-11 (R2016) [B.7-1], provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *The purpose of CSA N287.5 is to ensure concrete containment structures are built using techniques and work practises that meet the quality and standards commensurate with the principles necessary to comply with the Canadian nuclear safety philosophy through specification of examination and testing requirements.*
>
> *CSA N287.5 specifies examination and testing requirements that apply to the work of any organization participating in the construction, fabrication, or installation of parts or components of concrete containment structures for nuclear power plants that are designated as class containment, as well as the personnel qualification requirements for work pertaining to this standard.*

CSA N287.5-11 is directly relevant to Safety Factor 1 (Plant Design) and Safety Factor 2 (Actual Condition of Structures, Systems and Components).

CSA N287.5 is identified in Appendix E.1 of the R04 Pickering Licence Conditions Handbook [B.7-2] as "Guidance or Criteria". N287.5-11, which was reaffirmed in 2016, is the third edition of this standard which supersedes the previous editions, published in 1993 and 1981, under the title "Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants". The CSA N287.5-11 Impact Statement [B.7-3] provides a "Summary of significant changes from the previous edition" which identifies several changes to the Standard as described in Section B.7.2 below.

The results of PSR1 CSA N287.5 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.7.2. As identified in Reference [B.7-4], the Pickering PSR2 review of CSA N287.5-11 is an Incremental Review. PSR2 Incremental Review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.

- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.7.2 Compliance Assessment for Pickering PSR2

B.7.2.1 Application of PSR1 Reviews

The versions of N287.5 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

The Pickering B ISR did not undertake a compliance assessment against CSA N287.5. Per OPG Report NK30-REP-03680-00002 R000, "Pickering NGS-B Integrated Safety Review - Actual Condition of Systems, Structures and Components Safety Factor Report" [B.7-5]:

> *CSA N287.5 deals with the requirements for examination and testing of concrete during the construction of the containment system. In addition to the examination and testing performed during construction, the commissioning tests, in-service inspections and in-service testing, ensure that the containment system meets its design requirements. Therefore, this standard is not included in the scope of the review documented in this report.*

The date of issue of the Construction License for Pickering B was July 19, 1974, and CSA N287.5 is not listed as "codes and standards applicable to the NPCS [Negative Pressure Containment System] design, in accordance with the Code Effective Date/Construction Licence date/date of original design" [B.7-6]. CSA N287.5 applies to the testing of Concrete Containment Structures (CCSs) during initial construction, installation and fabrication. The CCSs at Pickering B were built and tested to meet 1970 National Building Code of Canada (NBCC) requirements [B.7-7], supplemented by specific loading requirements and the requirements of Design Manuals (e.g., see [B.7-8], [B.7-9], [B.7-10], [B.7-11]) and Design Guides (e.g., see [B.7-12], [B.7-13], [B.7-14]). The original Pickering concrete specifications (L-715-80 [B.7-15] and NK30-LH-20541-01 [B.7-16]) included requirements for quality control and compliance with CSA A23.1, A23.2 and A23.3 (which address concrete materials, methods of concrete construction and test methods and standard practices for concrete)[9].

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[9] Section 2A2 of NK30-LH-20541-01 [B.7-16] identifies all of the applicable laws, standards, codes, authorities and technical organizations referenced in the concrete tendering and contract documents, including CSA A5, A23 Series, A266 Series, American Concrete Institute and American Society for Testing and Materials standards. NK30-LH-20541-01 was reviewed for major differences against the requirements of CSA N287.5-11 (R2016) [B.7-1] as part of PSR2, and is deemed to meet the intent of Clause 4 (General Requirements), Clause 5 (Concrete), Clause 6 (Reinforcement), Clause 7 (Prestressing Systems) and Clause 11 (Documentation & Records). L-715-80 [B.7-15] is deemed to meet the intent of Clause 9 (Metallic Parts). While the specifications used for the Pickering A/B construction meet the general intent of the N287.5-11 clauses mentioned above, the specification does not meet specific sub-clauses of some of these areas. In addition, Clause 8 (Non-metallic Liners) and Clause 10 (Anchorage Systems) do not appear to be addressed in the above references.

The standards that applied during original construction of Pickering NGS included requirements for tests and quality control procedures to ensure that the concrete used in the as-built structures met the original design requirements. As discussed above, the Pickering B ISR did not confirm that the requirements in effect during the construction of Pickering NGS comply with the requirements of CSA N287.5. Nevertheless, the original design requirements for Pickering 5-8 comply with the intent of N287.5-11. Per OPG Report NK30-REP-03680-00015 R000, "Pickering NGS-B Integrated Safety Review (ISR) - Final ISR Report" [B.7-17]:

> *New material was used in the construction of the PNGS B CCSs including the Pressure Relief Duct and the Vacuum Building common CCS. The concrete and steel materials were specified by qualified and experienced designers, some of whom were members of the technical committee that prepared the N287 series of standards, which were published, starting in 1976 (and later revised). These materials complied with the required code and have performed adequately since the time of construction in the 1970's…. As pointed out in OPG's response to CNSC's Comment as per NK30-CORR-00531-04973, "Pickering NGS-B Integrated Safety Review – Plant Design", September 23, 2008, Section B.6.5.1, the reinforced concrete is stronger than the strength required by modern codes… Therefore, the design of the CCSs meets modern performance requirements.*

Furthermore, ongoing confirmation that the Pickering NGS CCSs remain fit for service is demonstrated via the following:

- As required by N-PROG-MA-0017 R008 [B.7-18]:
  - Periodic inspection of containment system metallic and plastic components (including the containment pressure suppression systems) is performed in accordance with the requirements of CSA N285.5, "Periodic inspection of CANDU Nuclear Power Plant Containment Components". N-PROC-MA-0064 R005, "Administrative Requirements for the Periodic Inspection of Nuclear Power Plant Containment Components" [B.7-19], describes the administrative process and the roles of various organizations across OPG Nuclear involved in the execution of the Program in order to comply with the requirements of CSA N285.5.
  - Periodic examinations of in-service inspections and positive leakage rate testing of CCSs that are designated as class containment components are performed in accordance with the requirements of CSA N287.7, "In Service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants". N-PROC-MA-0066 R005, "Administrative Requirements for In-Service Examination and Testing for Concrete Containment Structures" [B.7-20] describes the administrative process and the roles of various organizations across OPG Nuclear involved in the execution of N-PROG-MA-0017 R008 in order to comply with the requirements of CSA N287.7.
- With respect to aging management, N-PROG-MA-0017 R008 interfaces with the Integrated Aging Management Program to evaluate the condition of equipment (including containment structures) on an ongoing basis as defined by component programs, including management of equipment aging activities that: 1) identify aging

related degradation mechanisms, 2) define maintenance, inspection, test, and other activities, and 3) initiate preventive action to preclude component or system failure. A condition assessment process is used to evaluate the health of critical components and establish actions necessary to maintain long-term health. Condition assessments are prepared in accordance with N-PROC-MP-0060 R005 B, "Aging Management Process" [B.7-21] . Requirements for reporting (such as summarizing significant aging issues in component health reports) are identified in this procedure.

- Non-Destructive Examination (NDE) required for the periodic inspection of containment components is performed in accordance with N-STD-MA-0021 R001, "Non-Destructive Examination" [B.7-22] to ensure that these examinations meet code and jurisdictional requirements.

N-PROG-MP-0001 R014, "Engineering Change Control" [B.7-23] ensures that any future design changes made to the CCSs comply with modern applicable codes, standards, and regulations (including CSA N287.5) as applicable.

The above-mentioned Programs, Procedures and Standards are credited with the ability to detect and monitor any safety significant degradation mechanism and thus to provide assurance of continued fitness for service of the Pickering NGS CCSs.

Pickering Units 1,4

Pickering A Return to Service [B.7-24] did not review N287.5, as it "pertains mostly to operations aspects, or other aspects not having a direct or immediate effect on installed design features" [B.7-25]. Per the Pickering A Containment Design Requirements NA44-DR-63420-10001 R001 [B.7-26], CSA N287.5 is not listed as a "codes and standards (including addenda) applicable to the NPCS design."

The concrete structures at Pickering A were built and tested to meet the 1965 NBCC requirements [B.7-27] and associated CSA A23 Series Standards, supplemented by specific loading requirements and the requirements of Design Manuals (e.g., see [B.7-28], [B.7-29], [B.7-30], [B.7-31], [B.7-32]). As discussed above for Pickering Units 5-8, although Pickering A Return to Service did not demonstrate that the requirements in effect during construction comply with the requirements of CSA N287.5, ongoing confirmation that the Pickering NGS CCSs remain fit for service is demonstrated via periodic inspections and in-service testing. Further, the Engineering Change Control (ECC) process ensures that any future design changes made to the CCSs will comply with N287.5 as applicable.

Although N287.5 was not reviewed for either the Pickering B ISR or Pickering A Return to Service, an assessment was performed for Darlington and the programmatic aspects of the review are applicable to Pickering, as discussed below.

Darlington NGS

Appendix A of the Darlington ISR Code Review Report NK38-REP-03680-10060 R000 [B.7-33] notes gaps against ten clauses of CSA N287.5-93 [B.7-34]. Six of the gaps are due to use of alternate test requirements which could not be compared against original tests:

In summary, there is ample evidence that DN [Darlington Nuclear] was in compliance with the 1981 version of N287.5. However, without a detailed comparison of the specific test procedures which have evolved since 1981, or the results of those tests and possibly newer test methods not in existence in 1981, which would now be called for new NPP [Nuclear Power Plant] construction under the current standards, it must be concluded that based only on lack of review of inactive OPG records, this clause is considered to be a gap.

Four of the CSA N287.5-93 gaps were associated with new requirements that were not present in the 1981 version of the code. These gaps relate to new testing methods and frequencies for concrete materials, requirements for the properties and design proportions of concrete mix, concrete and grout testing methods and frequencies, and requirements on testing frequencies for sister splices, test beam specimens and vacuum boxes.

With respect to these ten gaps, NK38-REP-03680-10060 R000 states:

The report in Appendix A documents the review of CAN/CSA N287.5-93, "Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants". ISR Gaps were identified during the review and are documented in the report. During the ISR Gap resolution process, it was determined that none of the gaps identified in the report in Appendix A could be reclassified as compliant and no additional gaps were discovered. Therefore, the ISR Gaps remain as per the report.

OPG Report NK38-REP-03680-10104 R000, "Darlington NGS Integrated Safety Review (ISR) - Final ISR Report" [B.7-35] combined the ten N287.5-93 gaps into a single issue, stating:

Due to the lack of available compliance documentation, some of the testing requirements for testing concrete that were established by CAN/CSA-N287.5-M81 during DNGS construction have not been shown to satisfy CAN/CSA-N287.5-93. Some new requirements related to concrete testing introduced by CAN/CSA-N287.5-93 have not been incorporated in OPG governance.

NK38-REP-03680-10104 R000 classified the issue as an Acceptable Deviation, stating:

This issue applies to testing of concrete and grout during the initial construction, installation and fabrication of the Concrete Containment Structures. The CAN/CSA-N287.5-M81 standard that applied during original construction included requirements for tests and quality control procedures to ensure that the concrete used in the as-built structures met the original design requirements.

The current edition of the standard, CAN/CSA-N287.5-93, includes updated and additional concrete testing requirements, but due to the lack of available documentation it has not been demonstrated that the requirements previously in effect comply with these updated requirements. However, ongoing confirmation that the Concrete Containment Structures remain fit for service is demonstrated via the Periodic Inspection Program and In-service testing results.

The Periodic inspection of containment system components is performed in accordance with the requirements of CAN/CSA-N285.5-08 "Periodic inspection of CANDU Nuclear Power Plant containment components" and the In-service examinations and positive leakage rate testing of containment components are performed in accordance with the requirements of CAN/CSA-N287.7-08 "In Service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants", as required by N-PROG-MA-0017 R006 "Component and Equipment Surveillance".

N-PROG-MP-0001 R009 "Engineering Change Control" ensures that any future design changes made to the Concrete Containment Structures comply with modern applicable codes, standards, and regulations (including CAN/CSA-N287.5-93).

No further action required.

ISR Issue reclassified as an Acceptable Deviation.

OPG Report NK38-REP-03680-10060-ADD-001 R000, "Addendum to the CAN/CSA N287.5-93 Code Review Report for Darlington ISR" [B.7-36] documents CNSC comments on the Darlington Final ISR Report [B.7-35] specific to N287.5, as well as OPG's responses and CNSC acceptance. No further action was required. OPG Report NK38-REP-03680-10201, "ISR Open Issues and Acceptable Deviations - Adequacy Review" [B.7-37] also concluded that the N287.5-93 gap was an Acceptable Deviation.

The Darlington argument above for classification of the N287.5 non-compliance as an Acceptable Deviation, which was agreed upon by the CNSC, is also applicable to Pickering NGS. Although gaps exist between the original design of Pickering and Darlington NGS CCSs and the current requirements of N287.5, in-service inspections are being conducted in accordance with the requirements of CSA N285.5 and N287.7 and the resultant inspection reports attest to the quality of the CCSs. Although Darlington was originally designed to the 1981 version of N287.5 and Pickering NGS was not, the fact remains that it is not practicable to make design changes to the CCSs without rebuilding them. Furthermore, the ECC process ensures that any design changes made to the Pickering CCSs comply with N287.5 going forward, as applicable.

With respect to changes made in the latest version of N287.5, a Code Refresh review was performed for CSA N287.5-11 and documented in NK38-REP-03680-10144 R000, "Code Refresh Review of CSA N287.5-2011 Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants" [B.7-38]. The review concluded:

The changes made in CSA N287.5-2011 relative to CSA N287.5-93 (R2004) are minor and mostly appear to be for clarification purposes. An additional section is added for documentation and records that do not have any technical requirements. The review of the changed clauses in this code refresh report confirms that OPG Nuclear governance continues to be in compliance with the requirements of CSA N287.5-2011…

In summary, the review did not identify any additional gaps relative to the requirements of CSA N287.5-2011. The review confirms OPG Nuclear governance is in *compliance with the changes in the requirements of CSA N287.5-2011 relative to CSA N287.5-93 (R2004).*

NK38-REP-03680-10144 R000 therefore confirms that no substantive (safety significant) changes were made in the 2011 version of N287.5 [B.7-3], and that OPG governance (which is applicable across OPG's nuclear fleet, including Pickering NGS) is in compliance with the changes in the requirements of CSA N287.5-11.

B.7.2.2 Application of Post PSR1 Reviews

Per the CSA N287.5-11 Impact Statement [B.7-3], the following is a summary of the significant changes from the previous edition of the standard:

- *Align the N287.5 standard with recently revised/updated CSA N series standards that are referenced throughout the document. The impact of aligning the format, definitions, and references in N287.5 to the other recently updated N287 series standards is intended to improve the clarity, and consistency of the standards scope and its linkage to other N287 standards. This will benefit future users of the codes, and has developed dialogue between the various subcommittees.*

- *Table 1 related to Concrete Materials has been aligned with Table 1 of N291. Current standards are references, and withdrawn standards are removed and replaced with relevant specifications. The update of N287.5 Table 1 was warranted to ensure that reference is made to current specifications. Its impact is intended to improve the quality of concrete materials examination and testing which meets current industry needs and technologies.*

- *Sections 6, 7 and 9 were modified based on suggestions by experts in examination and testing of reinforcement, pre-stressing, welding, respectively. The impact of incorporating input from expert professionals in the area of reinforcement, welding, and pre-stressing systems has impacted the code to better reflect updated examination and testing industry practices.*

The latest changes incorporated into CSA N287.5-11 were part of the clause-by-clause Code Refresh review performed for the Darlington ISR [B.7-37]. As discussed earlier, that review did not identify any additional gaps for N287.5-11 relative to the previous version of the Standard (CSA N287.5-93 [B.7-34]), and confirmed that OPG Nuclear governance (which is fleet wide and applicable to Pickering NGS) is in compliance with CSA N287.5-11. Nevertheless, CSA N287.5 was not assessed for Pickering NGS in PSR1, and the Darlington ISR conclusions are not fully applicable to Pickering PSR2. As discussed earlier, the original and subsequent revisions of CSA N287.5 were issued after completion of the design of Pickering A and B, and the CCSs were designed to meet the requirements of the National Building Code of Canada and applicable Design Manuals and Design Requirements at the time. The concrete used in the as-built Pickering NGS CCSs met the original design requirements, and the original Pickering construction included requirements for tests and quality control procedures which generally meet the intent of N287.5. However, evidence of full equivalency with N287.5 requirements is not available. As a result, this is a PSR2 gap for Pickering NGS (**PSR2 CSA N287.5 Gap #1**).

It is expected that this gap can be confirmed to be an Acceptable Deviation during the PSR2 Global Assessment process.

B.7.3 Compliance Summary for Pickering PSR2

There is one PSR2 CSA N287.5-11 gap which relates to Safety Factor 1 (Plant Design):

1. The Concrete Containment Structures (CCSs) at Pickering A and B were built and tested to meet the 1965 and 1970 National Building Code of Canada requirements, respectively, prior to the initial issuance of CSA N287.5. No assessments exist which demonstrate that the requirements in effect during construction of Pickering NGS CCSs comply with the requirements of CSA N287.5. Ongoing confirmation that the Pickering NGS CCSs remain fit for service is demonstrated via periodic and in-service inspections conducted in accordance with the requirements of CSA N285.5 and N287.7, and the resultant inspection reports attest to the quality of the design. In addition, the Engineering Change Control (ECC) process ensures that any design changes made to the Pickering CCSs will comply with N287.5 going forward, as applicable.

   The original Pickering construction included requirements for tests and quality control procedures which generally meet the intent of N287.5. Furthermore, retroactive application of N287.5 to the as-built design of CCSs cannot be practically achieved without rebuilding them. Nevertheless, there is a PSR2 gap for Pickering NGS given that compliance with the specific requirements of N287.5 has not been demonstrated.

B.7.4 References

[B.7-1] CSA Standard, N287.5-11 (R2016), *Examination and Testing Requirements for Concrete Containment Structures for Nuclear Power Plants*, 2011.

[B.7-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.7-3] CSA Impact Statement and Publication Notice, *Product: New Edition; Product Designation: CSA N287.5*; Date of Release: May 2011, Date not provided.

[B.7-4] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.7-5] OPG Report, NK30-REP-03680-00002 R000, *Pickering NGS-B Integrated Safety Review - Actual Condition of Systems, Structures and Components Safety Factor Report*, May 2008.

[B.7-6] OPG System Design Requirements, NK30-DR-63420-10001 R001, *Negative Pressure Containment System – Reactor Building Overview*, 2009.

[B.7-7] National Research Council of Canada, *National Building Code of Canada, 1970*, 1971.

[B.7-8] Ontario Hydro Pickering Generating Station B Design Manual, NK30-21000 R002, *Reactor Building - General*, issued February 1980, revised September 1988 and February 1989.

[B.7-9] Ontario Hydro Pickering Generating Station B Design Manual, NK30-21040 R001, *Reactor Building Floor Loadings*, issued August 1979, revised September 1988.

[B.7-10] Ontario Hydro Pickering Generating Station B Design Manual, NK30-21140 R002, *Reactor Building Foundation and Perimeter Wall*, issued November 1979, revised August 1982 and November 1988.

[B.7-11] Ontario Hydro Pickering Generating Station B Design Manual, NK30-21149 R002, *Reactor Building Dome*, issued January 1979, revised August 1982 and December 1988.

[B.7-12] AECL Engineering Design Guide, NK30-REF-68000-0379145 (DG-30-68000-6), *Containment Provisions for Extensions of the Containment Envelope for Pickering NGS B*, May 1977.

[B.7-13] AECL Engineering Design Guide, DG-00-01040-1 Rev. 02, *Earthquake Design Requirements for CANDU Nuclear Power Plants*, November 1974.

[B.7-14] OPG Design Basis Document, NK30-DBD-34200-00001 R000, *Containment System DBD*, February 2000.

[B.7-15] Specification L-715-80, *Specification for Concrete Placing and Workmanship*.

[B.7-16] Tendering and Contract Document, NK30-REF-20541-{47603}, *Supply of Pre-Mix Concrete in Ready Mix Trucks - Units 5-8*, NK30-LH-20541-01, April 1974.

[B.7-17] OPG Report, NK30-REP-03680-00015 R000, *Pickering NGS-B Integrated Safety Review (ISR) – Final ISR Report*, August 2009.

[B.7-18] OPG Program, N-PROG-MA-0017 R008, *Component and Equipment Surveillance*, June 1, 2015.

[B.7-19] OPG Procedure, N-PROC-MA-0064 R005, *Administrative Requirements for the Periodic Inspection of Nuclear Power Plant Containment Components*, October 24, 2013.

[B.7-20] OPG Procedure, N-PROC-MA-0066 R005, Administrative Requirements for In-Service Examination and Testing for Concrete Containment Structures, April 24, 2014.

[B.7-21] OPG Procedure, N-PROC-MP-0060 R005 B, *Aging Management Process*, October 1, 2015.

[B.7-22] OPG Standard, N-STD-MA-0021 R001, *Non-Destructive Examination*, July 30, 2015.

[B.7-23] OPG Program, N-PROG-MP-0001 R014, *Engineering Change Control*, January 7, 2015.

[B.7-24] AECL Assessment Document, 44RS-00531-ASD-001 Rev. 04, *Review of Pickering A Design Against Current Codes and Standards*, November 2000.

[B.7-25] OPG Letter, NA44-CORR-00531-00381 R000, R.J. Strickert to J.S.C. Tong, *Pickering A - Updated Basis for Return to Service Document*, April 20, 2001.

[B.7-26] OPG System Design Requirements, NA44-DR-63420-10001 R000, *Negative Pressure Containment System – Overview*, 2004.

[B.7-27] National Research Council of Canada, *National Building Code of Canada, 1965*, 1966.

[B.7-28] Hydro Electric Power Commission of Ontario Pickering Generating Station Design Manual NA44-21000, *Reactor Building - General*, December 1972.

[B.7-29] Hydro Electric Power Commission of Ontario Pickering Generating Station Design Manual NA44-21040, *Reactor Building Floor Loadings*, April 1969.

[B.7-30] Hydro Electric Power Commission of Ontario Pickering Generating Station Design Manual NA44-21140, *Reactor Building Foundation and Perimeter Wall*, April 1970.

[B.7-31] Hydro Electric Power Commission of Ontario Pickering Generating Station Design Manual NA44-21149, *Reactor Building Dome*, February 1970.

[B.7-32] Hydro Electric Power Commission of Ontario Pickering Generating Station Design Manual NA44-25000-21300.2, *Vacuum Building*, August 1982.

[B.7-33] OPG Report, NK38-REP-03680-10060 R000, Review of CSA N287.5-93 *Examination and Testing Requirements For Concrete Containment Structures For CANDU Nuclear Power Plants for Darlington Integrated Safety Review*, August 2011.

[B.7-34] CSA Standard, N287.5-93 (R2009), *Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants*, July 1993, Reaffirmed 2009.

[B.7-35] OPG Report, NK38-REP-03680-10104 R000, *Darlington NGS Integrated Safety Review (ISR) – Final ISR Report*, October 2011.

[B.7-36] OPG Report, NK38-REP-03680-10060-ADD-001 R000, *Addendum to the CAN/CSA N287.5-93 Code Review Report for Darlington ISR*, January 2014.

[B.7-37] OPG Report, NK38-REP-03680-10201 R001, *ISR Open Issues and Acceptable Deviations – Adequacy Review*, October 2014.

[B.7-38] OPG Report, NK38-REP-03680-10144 R000, *Code Refresh Review of CSA N287.5-2011 Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants*, January 2014.

B.8 CSA N290.0-11, "General Requirements for Safety Systems of Nuclear Power Plants"

B.8.1 Background

The following, paraphrased from the preface and introduction of CSA N290.0-11 [B.8-1], provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *The purpose of CSA N290.0 is to cover the design, qualification, installation, operation, maintenance, inspection, and documentation of the safety systems for a water-cooled nuclear power plant.*
>
> *N290.0 provides the general requirements for the safety systems and is a companion document of CSA N290.2 and N290.3, which outline specific requirements.*

The CSA N290.0 title refers to *Safety Systems*; however, it is only applicable to Special Safety Systems. In addition to being a companion document to N290.2 (Emergency Coolant Injection) and N290.3 (Containment), it is also a companion document to N290.1 (Shutdown Systems). Whereas CSA N290.1 was first issued in 1980, CSA N290.2 and N290.3 are far more recent and were initially issued in 2011, with the common requirements only in N290.0. These standards in conjunction with N290.0 have incorporated requirements from CNSC Regulatory Documents R-7 [B.8-2] and R-9 [B.8-3]. In addition, N290.1 has been revised to incorporate requirements from CNSC Regulatory Documents R-8 [B.8-4] and R-10 [B.8-6].

The relationship between and the PSR2 review approach for standards N290.0, N290.1, N290.2 and N290.3 are outlined here. The common generic requirements for the Special Safety Systems (previously addressed in earlier versions of N290.1, R-9 and R-7) were amalgamated into a companion document, CSA N290.0. The common requirements will be reviewed within the PSR2 N290.0 assessment. The system-specific requirements identified in N290.1, N290.2 and N290.3 will be reviewed within the PSR2 assessments for these standards. Any gaps identified as generic to the Special Safety Systems will be recorded as N290.0 gaps. If a gap is specific to only one of the Special Safety Systems, or system specific standard, it will be recorded as a gap against that standard (N290.1, N290.2, and N290.3) only.

All of N290.0-11 is directly relevant to Safety Factor 1 (Plant Design).

CSA N290.0 is identified in Appendix E.1 of the R04 Pickering Licence Conditions Handbook [B.8-7] as "Guidance or Criteria". Section 6.1, "Design Program" of the Licence Conditions Handbook [B.8-7] states that "Recommendations and guidance are found in… N290.0, which covers general requirements for safety systems". CSA N290.0-11 is the first edition of this standard.

The results of PSR1 CSA N290.0 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.8.2. As identified in Reference [B.8-8], the Pickering PSR2 review of CSA N290.0 is an Incremental Review. PSR2

Incremental Review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.
- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.8.2 Compliance Assessment for Pickering PSR2

B.8.2.1 Application of PSR1 Reviews

The versions of N290.0 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

CSA N290.0-11 is the first edition of this Standard, published in October 2011. Therefore, there were no previous reviews of this standard conducted for Pickering. Many of the requirements contained in the standard are consistent with the content of the following CNSC Regulatory Documents:

- R-7, "Requirements for Containment Systems for CANDU Nuclear Power Plants" [B.8-2];
- R-8, "Requirements for Shutdown Systems for Nuclear Power Plants" [B.8-4];
- R-9, "Requirements for Emergency Core Cooling Systems for CANDU Nuclear Power Plants" [B.8-3]; and,
- R-10, "The Use of Two Shutdown Systems in Reactors" [B.8-5].

Code reviews of the CNSC regulatory documents R-7 [B.8-2], R-8 [B.8-4], and R-9 [B.8-3] were performed as part of the Pickering B ISR. The results of these reviews are included in the respective PSR2 CSA code reviews for CSA N290.1, CSA N290.2 and CSA N290.3, as well as CSA N290.0 for common requirements.

Darlington NGS

Following the issuance of CSA N290.0-11 [B.8-1] in October 2011, a clause-by-clause code review was conducted and documented in OPG Report NK38-REP-03680-10211 R000 [B.8-9].

The code review concluded that compliance was demonstrated for the majority of the requirements, except for twelve gaps against eight clauses. Ten were associated with approved design guide exceptions for Darlington special safety systems where separation and independence requirements could not be completely met. These are all related to Darlington-specific design features, and are not specifically applicable to Pickering. Therefore, these are not PSR2 gaps. The other two gaps are addressed below.

The first of these was a gap against clause 4.12.5, which states:

> *SSCs postulated to perform mitigating functions during and after exposure to harsh environmental conditions resulting from BDBAs [Beyond Design Basis Accidents] shall be assessed for their potential to perform their intended function under the expected harsh environmental conditions. Assessment of capability may be based on design specifications, environmental qualification testing, or other considerations.*

This was a gap for Darlington due to the Instrument Survivability Assessment for BDBAs not being complete at that time. This assessment has now been completed for all OPG stations and is documented in References [B.8-10] and [B.8-11].

The last gap was on clause 4.14.10 on human-machine interface requirements. The gap was a result of the lack of design standards related to Human Factors Engineering (HFE) or HFE activities being formally documented when the Darlington control centre was originally designed and constructed. This gap also applies to the Pickering Units 1,4 and 5-8 control rooms, due to the lack of formally documented HFE design standards at the time of the design and therefore is identified as **PSR2 CSA N290.0-11 Gap #1**. This gap is generic to each of the Special Safety Systems at both Pickering 5-8 and Pickering 1,4 and is, therefore, identified under N290.0.

B.8.2.2 Application of Post PSR1 Reviews

Section B.8.1 states that the PSR2 Basis Document [B.8-8] specifies the review of CSA N290.0-11 to be Incremental. However, more detail has been provided than is typical for an Incremental review in the assessment below, as the Darlington ISR contains minimal information specific to Pickering Special Safety Systems (SSSs). In the review below, the degree of conformance with clauses or groups of clauses in the Standard is demonstrated by supporting evidence stating whether the intent of the requirements stipulated in the requirement document is met.

CSA N290.0-11 Clause	PSR2 Review	Compliant or Gap
1. to 3.	Do not contain requirements.	N/A
4.1 General	The compliance of the Pickering SSSs to N286 standards is addressed in a separate PSR2 code review.	N/A
4.2 Plant States	This clause requires plant states be grouped into categories including Anticipated Operational Occurrences (AOOs), Design Basis Accidents (DBAs) and BDBAs. BDBAs are now considered in design; however, AOOs have not been identified or analyzed in the current Pickering Safety Reports. Hence, the requirements and	Gap

CSA N290.0-11 Clause	PSR2 Review	Compliant or Gap
	credits attributed to the Special Safety Systems for AOOs, if any, cannot be readily ascertained. This is identified as **PSR2 CSA N290.0-11 Gap #2.**	**PSR2 CSA N290.0-11 Gap #2**
4.3 System Operating States	Poised, unpoised, test and post-accident states are all included in SSS operating states defined in SSS Operating Manuals.	Compliant
4.4 General Design Requirements	N-STD-MP-0016, “Safe Operating Envelope” [B.8-12] identifies the requirement to establish safety limits for all SSSs, which are contained in the Operational Safety Requirements documents [B.8-13], [B.8-14], [B.8-15],[B.8-16], [B.8-17], [B.8-18]. All SSSs meet the requirements in this section and there are no PSR2 gaps.	Compliant
4.5.1 General Reliability Requirements	N-STD-RA-0033, "Reliability Monitoring and Reporting of Systems Important To Safety" [B.8-19], section 1.2.3, identifies the requirements for reliability targets for SSSs as follows: "For the Special Safety Systems the target shall be set equal to the licensing target. If site specific licensing targets are already established and accepted by the CNSC, then the licensee shall continue referring to these targets". Also, support systems are taken into account in the reliability analyses and probabilistic safety evaluations. All SSSs meet the requirements in this section and there are no PSR2 gaps.	Compliant
4.5.2 Reliability Analysis	Section 1.2 of N-STD-RA-0033, "Reliability Monitoring and Reporting of Systems Important To Safety" [B.8-19], identifies the requirements for the production of unavailability models for SSSs. These models use the test frequencies as defined in the Safety-Related Test Program. Models are updated to take into account current component failure rates and any other new information, e.g., new failure modes. All SSSs meet the requirements in this section and there are no PSR2 gaps.	Compliant
4.6 Separation and Independence	Requirements for Independence and Separation are contained in system Design Manuals and these requirements have been addressed in the SSS design. Engineering Change Control governance addresses these requirements for design changes. Also, the requirements for this set of clauses are contained in CNSC Regulatory Documents for each SSS: R-7, R-8 and R-9. The reviews	Compliant

CSA N290.0-11 Clause	PSR2 Review	Compliant or Gap
	of these documents for Pickering Units 1,4 [B.8-19] and Units 5-8 [B.8-21] demonstrated compliance with the requirements for separation and independence, with the following exceptions: • For the Units 1,4 Shutdown System, there are approved exceptions for Shutdown System A (SDSA) and Shutdown System Enhancement (SDSE) not being independent of each other. This is assessed in the PSR2 N290.1 assessment. • The Units 1,4 Emergency Coolant Injection (ECI) system is not independent from the moderator system since they share components in the design of ECIS recovery mode. This is assessed in the PSR2 N290.2 assessment. With respect to clause 4.6.2 (b), independence between SSS sub-systems is addressed by having separate sensitivity cases in the Safety Report analyzing impairments of sub-systems of the SSS.	
4.7 Single Failure Criteria	The requirements for this set of clauses are partially contained in CNSC Regulatory Documents clauses on Availability Requirements for each SSS: R-7, R-8 and R-9. The reviews of these documents for Pickering Units 1,4 [B.8-20] and Units 5-8 [B.8-21] demonstrated compliance with the requirement that SSS do not contain any active single components that can result in system unavailability. The only exception is the Unit 1,4 ECI system, which contains singleton components required to operate for the system to operate successfully. The PSR2 assessment of CSA N290.2-11, "Requirements for Emergency Core Cooling Systems of Nuclear Power Plants", reviews this PSR1 gap, which was assessed as an acceptable deviation, since all practical modifications were made to the Pickering Units 1,4 ECI system during Pickering A Return to Service and the system meets its unavailability target. Therefore this is not a PSR2 gap. In addition, to meet the other clauses in this section, all Units 1,4 and 5-8 SSSs consist of redundant components and channels (other than for the approved exceptions for Units 1,4 ECIS). This design permits testing, maintenance and other necessary activities to be completed without rendering the system unavailable. All SSSs meet the requirements in this section and there are no PSR2 gaps.	Compliant

CSA N290.0-11 Clause	PSR2 Review	Compliant or Gap
4.8 Fail Safe	The Pickering SSSs are generally designed to be fail safe as far as practical (as this clause permits), e.g., Instrument channels fail safe. There are exceptions to this, e.g., the Pickering containment Instrumented Pressure Relief Valves (IPRVs). The valves which operate the IPRVs are designed to fail closed on loss of support services, to prevent spuriously connecting the Vacuum Building to containment. Design exceptions of this type are acceptable by designing in sufficient redundancy and identifying component failures via Abnormal Condition annunciations. Failures of a single support system component do not result in unavailability of SSSs. This is addressed by supplying odd and even electrical power to redundant supplies, and by having redundant pumps and compressors supplying water and air respectively. All SSSs meet the requirements in this section and there are no PSR2 gaps.	Compliant
4.9 Safety Support Systems	Requirements for Safety Support Systems (electrical, water and air supplies) credited in the Safety Report to ensure the proper functioning of a SSS are contained in the Operational Safety Requirements (OSRs) (see the compliance discussion for clause 4.4). The PSR2 CSA N290.5 review of Electrical and Instrument Air systems did not contain any specific PSR2 Gaps related to support systems for SSSs. The SSSs are designed to not rely on operation of the turbine-generator or off-site electrical grid (per the compliance with R-7, R-8 and R-9), except for the High Pressure ECI (HPECI). The HPECI pumps are supplied by Class IV power supplied from the bulk electrical system and backed up by the Site Electrical System (SES), which supplies Class IV power from non-accident running units. In addition, the Pickering 1,4 HPECI injection valves are supplied power from the SES. The reliability of SES is such that the reliability of the HPECIS pumps meets requirements. However, in addition the SES supplied HPECI pumps and valves can be powered from diverse Class III Standby Generators supplies. Therefore, this is not a PSR2 gap. Lastly, any required ventilation provisions are provided for SSS design or its support systems, e.g., Class II power ventilation system.	Compliant

CSA N290.0-11 Clause	PSR2 Review	Compliant or Gap
	All SSSs meet the requirements in this section and there are no PSR2 gaps.	
4.10 Pressure-Retaining SSCs	The design requirements of the Pickering SSSs include meeting the requirements of CSA N285.0 and N287.7. Separate reviews of these Standards are being performed for PSR2.	Compliant
4.11.1 Selection of Instrumentation	The design of the Pickering SSSs includes all the Environmentally Qualified (EQ) instrumentation required for correct operation of the system as credited in the Safety Report. The requirements for this instrumentation are contained in the OSRs (see the compliance discussion for clause 4.4). Required surveillance and testing are documented in the OSRs. The Operating Manuals prescribe which instrumentation is to be used to confirm the system has operated as required after an accident.	Compliant
4.11.2 Instrumentation and Control	All of the Pickering SSSs are designed to initially operate automatically, such that operator action is not required for at least 15 minutes after an accident. The Main Control Room (MCR) contains the controls necessary for manual action/intervention. SSS instrumentation is included in the unavailability models and accuracy requirements are defined in Instrument Uncertainty Calculations (IUCs) referred to in the OSRs, and take into account applicable errors and uncertainties. The response times of critical instrument loops is also defined in the SSS OSRs and compliance is demonstrated during device calibrations or on-line testing. The requirements for testing in clause 4.11.2.10 are met for all SSS by preparing procedures that test and monitor SSS instrumentation as required by the OSR Compliance tables, referred to in the OSRs. All SSSs meet the requirements in this section and there are no PSR2 gaps.	Compliant
4.11.3 System Performance	The required process and SSS parameters are directly monitored in the MCR via redundant and reliable instrumentation, in station operating procedures to confirm the operability of the SSS, e.g., Heat Transport System pressure and temperature.	Compliant

CSA N290.0-11 Clause	PSR2 Review	Compliant or Gap
	Annunciations are provided in the MCR, meeting the requirements of clause 4.11.3.6, e.g. loss of ECI system pressure, loss of support systems. All SSSs meet the requirements in this section and there are no PSR2 gaps.	
4.12 Environmental Qualification	All Pickering SSSs meet the requirements of N-PROG-RA-0006, "Environmental Qualification" [B.8-22] to ensure credited equipment will operate in the conditions present following a design basis accident. Pickering Units 5-8 SSSs are designed to meet Seismic Design Basis Earthquake (DBE) or Site Design Earthquake (SDE) categories following a seismic event. Required Pickering Units 1,4 SSS equipment has been demonstrated to meet requirements of the Seismic Margin Assessment. N-PROG-MP-0008, "Integrated Aging Management" [B.8-23] defines the requirements for aging management of SSS components. Sub-clause 4.12.4 identifies that the impact of accident generated debris be addressed. This is specifically applicable to ECI recovery. The Units 1,4 and 5-8 ECI systems are equipped with ECIS recovery screens designed to accommodate debris and other contaminants generated after a DBA. All SSSs meet the requirements in this section and there are no PSR2 gaps.	Compliant
4.13 Dynamic Piping Effects	Requirements for dynamic piping effects, e.g., pipe whip, are contained in R-7, R-8 and R-9. The code reviews for these regulatory documents for Pickering Units 1,4 [B.8-20] and Units 5-8 [B.8-21] demonstrated compliance with this requirement except for an Acceptable Deviation raised for each of the Units 1,4 SSSs. There is an outstanding CANDU Safety Issue relevant to this clause [B.8-24]. There is a lack of a systematic assessment of high energy line break effects. Closure of this issue requires confirmation that the impacts of high energy line breaks have been captured in the design basis documents and the Safety Report. OPG has completed the High Energy Line Break Assessment (HELBA) for Pickering 5-8 and the Pickering 1,4 assessment is expected to be completed in 2017. Results to date, demonstrate that there would be no consequential damage caused by the rupture of high	Gap **PSR2 CSA N290.0-11 Gap #3**

CSA N290.0-11 Clause	PSR2 Review	Compliant or Gap
	energy pipes inside containment to safety related equipment, beyond that already accounted for in the Safety Reports. Since the Pickering 1,4 work has not been completed, this is identified as **<u>PSR2 CSA N290.0-11 Gap #3</u>**.	
4.14 Human Factors	Currently Human Factors Engineering (HFE) is considered for all design changes. The changes are managed through the Engineering Change Control Program and plant modifications follow the Modification Process. The Design Scoping Checklist [B.8-25] is a component of the modification process that identifies the level of HFE required. If the modification is judged to have an HFE impact, then the Human Factors Level of Activity Form [B.8-26] is completed to determine the scope of the HFE. A Human Factors Engineering Specialist must concur with the Human Factors Level of Activity. The modification may require the preparation of the Human Factors Engineering Plan or the Human Factors Worksheet [B.8-27]. These instructions and processes ensure that the human-system interface elements for the modification are addressed. The technical, design and operator reviews during and following the design process and via the Availability for Service (AFS) process, ensure the usability requirements will be achieved. The compliance discussion for clause 4.14.10 for Darlington states the following: *…While it is noted that the control centre for the plant was designed to the best standards of the day to provide "operator friendly" facilities, due to the lack of any design standards related to HFE in effect or HFE activities formally documented when the plant was originally designed and constructed, Darlington NGS is not considered to be in compliance with clause 7.21 (in CNSC Reg Doc RD-337).* The Pickering units have many years of safe operation and operating experience with special safety system monitoring, operation (including testing), maintenance and training (including simulator). Any potentially significant human interaction deficiencies due to the design would be identified during these activities and be addressed. Hence, the extent of application of human factors engineering in the original design is not expected to have an impact on nuclear safety relating to the SSSs. Ontario Hydro design practice in the original design phase recognized the need for focus on the operator interfaces in the control centres, and on recognition of the integrated whole of the control centre and the related	Gap **<u>PSR2 CSA N290.0-11 Gap #1</u>**

CSA N290.0-11 Clause	PSR2 Review	Compliant or Gap
	human-systems interfaces. However, as already identified in Section B.8.2.1 above, the Darlington gap is applicable to the design of the Pickering Units 1,4 and 5-8 control rooms and this is previously identified **PSR2 CSA N290.0-11 Gap #1**. Other clauses are addressed by the compliance discussion for clause 4.11.	
4.15 Fire Protection	A separate code review is being prepared for CSA N293.0 as part of PSR2. Also, Fire Hazard Assessments have been performed to ensure SSSs meet requirements for safe shutdown of the reactor after a fire. All SSSs meet the requirements in this section and there are no PSR2 gaps.	Compliant
4.16 System Health Monitoring	N-PROC-MA-0024,” System Performance Monitoring” defines the requirements for monitoring the SSSs and meets the requirements of clause 4.16.	Compliant
4.17 Operability	Station Operating Policies and Principles (OP&Ps) and other operating procedures define when a SSS can be removed from service. Abnormal Incidents Manuals (AIMs) define the required actions to take if a system is found to be inoperable. The Pickering SSSs are designed such that no manual action is required in the MCR for 15 minutes after a DBA and 30 minutes for required field actions. Shielding requirements to gain access to equipment after a DBA are contained in the Design Requirements documents for ECI and Containment systems. The compliance assessment for clause 4.11.2 applies to other clauses in this section (4.17.7 & 4.17.8). All SSSs meet the requirements in this section and there are no PSR2 gaps.	Compliant
4.18 Maintainability	The compliance assessment for clause 4.7 applies to the majority of clauses in this section addressing design features to optimize maintenance practices, e.g., ensuring maintenance can be performed without making the system unavailable. Isolation devices are in the design of SSSs to permit maintenance.	Compliant

CSA N290.0-11 Clause	PSR2 Review	Compliant or Gap
	All SSSs meet the requirements in this section and there are no PSR2 gaps.	
4.19 Maintenance Program	N-PROG-MA-0004, "Conduct of Maintenance" [B.8-28] defines the maintenance requirements for plant SSCs, including SSS components and includes the Preventative Maintenance program [B.8-29]. Surveillance of equipment condition is performed via System Monitoring (per clause 4.16) and Components and Equipment surveillance per N-PROG-MA-0017 [B.8-30]. To perform required maintenance, operating procedures direct that the component be placed in a safe state, where possible. All SSSs meet the requirements in this section and there are no PSR2 gaps.	Compliant
4.20.1 Testing - General	N-STD-OP-0018, "Operability Testing of Safety Related Systems" [B.8-31] defines the methods/processes for SSS testing. Section 1.5.1 states: *Operability Tests shall be conducted at frequencies required for meeting station's Predicted Future Unavailability (PFU) targets and frequencies shall be determined by Reactor Safety Support in accordance with approved procedures.* This requirement ensures that the SSSs will meet the required reliability targets. The other clauses of 4.20.1 are met by having N-STD-0018 and Pickering Units 1,4 and 5-8 Safety Related System Tests (SRSTs) in place.	Compliant
4.20.2 Functional Testing	N-STD-OP-0018, "Operability Testing of Safety Related Systems" [B.8-31] defines the methods/processes for SSS testing. SSS tests are in place with acceptance criteria consistent with the requirements of the OSRs.	Compliant
4.20.3 Post-Maintenance Testing	Station OP&Ps require testing be performed following SSS component maintenance. Requirements for post-maintenance testing are specified in maintenance instructions and procedures. Operational testing to confirm availability upon completion of maintenance is done in accordance with operating procedures and manuals.	Compliant

CSA N290.0-11 Clause	PSR2 Review	Compliant or Gap
4.20.4 Commissioning	Commissioning Tests were performed for all Pickering SSSs prior to unit start-up per the requirements of this section.	Compliant
4.21 Sharing Within a Unit	The Pickering SSSs generally do not share components (including instrumentation) with a process system. There are exceptions, all of which satisfy the design considerations of this clause. For example: Units 1,4 ECIS shares moderator system components, e.g., the main moderator pumps, which is identified and assessed as part of the CSA N290.2 PSR2 review for Pickering Units 1,4. Containment System Air Cooling Units (ACUs) perform both a process and safety function, but the requirements of this clause are met, e.g., the essential safety functions are unaffected by the initiation of the Loss of Coolant Accident (LOCA) event. All SSSs meet the requirements in this section and there are no PSR2 gaps.	Compliant
4.22 Sharing Between Units	The SSS OSRs are based on the limiting DBA and credited components are environmentally qualified per the requirements in clause 4.12. Common cause events were considered in the design of the Pickering Units 5-8 SSSs by adopting the two Group philosophy. Group 2 systems have been qualified to ensure they can operate after a common cause event, e.g., a seismic event or main steam line failure. For Units 1,4, capabilities have been assessed to ensure safe shutdown following a seismic event (per the Seismic Margin Assessment). For main steam line failures in the powerhouse, an emergency heat sink, Emergency Boiler Water Supply (EBWS) and alternate electrical supply to critical loads (Inter-Station Transfer Bus (ISTB)) were added to Pickering Units 1,4 to provide comparable capabilities to Units 5-8. The sharing of systems between units has been assessed for internal and external common cause events and the nuclear safety consequences demonstrated to be acceptable. All SSSs meet the requirements in this section and there are no PSR2 gaps.	Compliant

CSA N290.0-11 Clause	PSR2 Review	Compliant or Gap
4.23 Documentation	The following SSS documents fulfill the requirements of this section: Design Manuals Design Descriptions Design Requirements Operational Safety Requirements (OSRs) Operating Manuals	Compliant

B.8.3 Compliance Summary for Pickering PSR2

There are three PSR2 N290.0-11 gaps which relate to Safety Factor 1 (Plant Design):

1. The Darlington Integrated Safety Review (ISR) identified a gap against Clause 4.14.10 of N290.0-11 as a result of the lack of design standards related to Human Factors Engineering (HFE) or HFE activities being formally documented when the control rooms were originally designed and constructed. Pickering NGS has many years of successful Special Safety System (SSS) operation and the absence of formal HFE in the original design is not expected to have any nuclear safety significance relating to SSSs. However, the Darlington gap is also applicable to Pickering NGS and is therefore identified as a PSR2 gap.

2. Clause 4.2 of N290.0-11 requires that Plant States be grouped into several categories, including Anticipated Operational Occurrences (AOOs). This is consistent with clauses of REGDOC-2.4.1 and REGDOC-2.5.2 related to identification and classification of initiating events. Since AOOs have not been identified and analyzed in the current Pickering Safety Reports, the requirements and credits attributed to the Special Safety Systems for AOOs, if any, cannot be readily ascertained. This issue has therefore been identified as a PSR2 gap. It is being addressed as part of REGDOC-2.4.1 implementation.

3. OPG is currently in the process of completing the High Energy Line Break Assessment (HELBA) for Pickering NGS. Preliminary results show that there would be no consequential damage caused by the rupture of high energy pipes inside containment to safety related equipment, beyond that already accounted for in the Safety Reports. The final HELBA reports for Pickering Units 5-8 have been completed, while Pickering Units 1,4 are expected to be completed in 2017. Since this work has not been completed for Pickering 1,4, this is identified as a PSR2 gap.

B.8.4 References

[B.8-1] CSA Standard, N290.0-11, *General Requirements for Safety Systems of Nuclear Power Plants*, October 2011.

[B.8-2] AECB Regulatory Document, R-7, *Requirements for Containment Systems for CANDU Nuclear Power Plants*, February 1991.

[B.8-3] AECB Regulatory Document, R-9, *Requirements for Emergency Core Cooling Systems for CANDU Nuclear Power Plants*, February 1991.

[B.8-4] AECB Regulatory Document, R-8, *Requirements for Shutdown Systems for CANDU Nuclear Power Plants*, February 1991.

[B.8-5] AECB Regulatory Document R-10, *The Use of Two Shutdown Systems in Reactors*, January 1977.

[B.8-6] AECB Regulatory Document, R-10, *The Use of two Shutdown Systems in Reactors*, January 1977.

[B.8-7] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.8-8] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.8-9] OPG Report, NK38-REP-03680-10211 R000, *Code Refresh Review Of CSA-N290.0-11 General Requirements For Safety Systems of Nuclear Power Plants*, January 2014.

[B.8-10] OPG Report, N-REP-09013-10007 R000, *Ontario Power Generation Severe Accident Management Guidance Instrumentation and Equipment Survivability - Summary Report*, December 2013.

[B.8-11] OPG Report, N-REP-09013-10008 R001, *Ontario Power Generation Severe Accident Management Guidance Instrumentation and Equipment Survivability - Appendices in Support of Summary Report*, May 2015.

[B.8-12] OPG Standard, N-STD-MP-0016 R002, *Safe Operating Envelope*, 2012.

[B.8-13] OPG Operational Safety Requirements, NA44-OSR-08131.02-00001 R002, *Pickering NGS A Operational Safety Requirements: Shutdown System*, June 2010.

[B.8-14] OPG Operational Safety Requirements, NK30-OSR-08131.02-00004 R003, *Pickering NGS B Operational Safety Requirements: Shutdown Systems*, February 2016.

[B.8-15] OPG Operational Safety Requirements, NA44-OSR-08131.02-00004 R002, *Pickering NGS A Operational Safety Requirements: Emergency Coolant Injection System*, October 2010

[B.8-16] OPG Operational Safety Requirements, NK30-OSR-08131.02-00001 R003, *Pickering NGS-B Operational Safety Requirements: Emergency Coolant Injection System*, June 2012.

[B.8-17] OPG Operational Safety Requirements, NA44-OSR-08131.02-00002 R003, *Pickering 1-4 Operational Safety Requirements: Negative Pressure Containment*, March 2015.

[B.8-18] OPG Operational Safety Requirements, NK30-OSR-08131.02-00003 R004, *Pickering 5-8 Operational Safety Requirements: Negative Pressure Containment*, March 2015.

[B.8-19] OPG Standard, N-STD-RA-0033 R002, *Reliability Monitoring and Reporting of Systems Important to Safety*, October 2015.

[B.8-20] AECL Assessment Document, 44RS-00531-ASD-001 Rev. 04, *Review of Pickering A Design Against Current Codes and Standards*, November 2000.

[B.8-21] OPG Report, NK30-REP-03680-00001 R000, *Pickering NGS B Integrated Safety Review - Plant Design Safety Factor*, August 2007.

[B.8-22] OPG Program, N-PROG-RA-0006 R008, *Environmental Qualification*, May 2015.

[B.8-23] OPG Program, N-PROG-MP-0008 R006A, *Integrated Aging Management*, October 2015.

[B.8-24] OPG Letter, N-CORR-00531-06904, W.S. Woods to B. Howden, *Progress Update on Category 3 CANDU Safety Issues – Implementation of Risk Control Measures*, June 24, 2015.

[B.8-25] OPG Form, N-FORM-10959 R015, *Design Scoping Checklist*, April 2015.

[B.8-26] OPG Form, N-FORM-10580 R006, *Identifying Human Factors Level of Activity*, December 2015.

[B.8-27] OPG Form, N-FORM-10221 R008, *Human Factors Worksheet*, December 2015.

[B.8-28] OPG Program, N-PROG-MA-0004 R011, *Conduct of Maintenance*, April 2015.

[B.8-29] OPG Procedure, N-PROC-MA-0020 R027, *Predefined Process*, June 2016.

[B.8-30] OPG Program, N-PROG-MA-0017 R008, *Components and Equipment Surveillance*, June 2015.

[B.8-31] OPG Standard, N-STD-OP-0018 R004, *Operability Testing of Safety Related Systems*, June 2014.

B.9 CSA N290.1-13, "Requirements for the Shutdown Systems of Nuclear Power Plants"

B.9.1 Background

The following, paraphrased from the preface and introduction of CSA N290.1-13 [B.9-1], provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *The purpose of CSA N290.1 is to act as a general standard for reactor shutdown systems of nuclear power plants. It establishes the design, procurement, installation, commissioning, operation, testing, and maintenance requirements of the shutdown system(s) to terminate the fission chain reaction in the event of an accident.*
>
> *CSA N290.1 applies to existing and new water-cooled nuclear power plants, both CANDU and non-CANDU. CSA N290.1 incorporates the relevant requirements of the CNSC regulatory documents R-8, "Requirements for Shutdown Systems for CANDU Nuclear Power Plants, and R-10, "The Use of Two Shutdown Systems in Reactors".*

All of N290.1-13 is directly relevant to Safety Factor 1 (Plant Design).

Compliance with N290.1 is not currently a licence requirement for Pickering NGS (PROL 48.02/2018) and is not referenced in the Pickering Licence Condition Handbook [B.9-2].

CSA N290.1-13 is the second edition of CSA N290.1. It supersedes the previous edition: N290.1-80 published in December 1980 under the title "Requirements for the Shutdown Systems of CANDU Nuclear Power Plants" [B.9-3]. The CSA Impact Statement notification for CSA N290.1-13 [B.9-4] provides a "Summary of Significant Changes from the Previous Edition", which identifies changes to the Standard that are discussed in Section B.9.2 below.

The relationship between and the PSR2 review approach for standards N290.0, N290.1, N290.2 and N290.3 are outlined here. The common generic requirements for the Special Safety Systems (previously addressed in earlier versions of N290.1, R-9 and R-7) were amalgamated into a companion document, CSA N290.0. The common requirements will be reviewed within the PSR2 N290.0 assessment. The system-specific requirements identified in N290.1, N290.2 and N290.3 will be reviewed within the PSR2 assessments for these standards. Any gaps identified as generic to the Special Safety Systems will be recorded as N290.0 gaps. If a gap is specific to only one of the Special Safety Systems, or system specific standard, it will be recorded as a gap against that standard (N290.1, N290.2, and N290.3) only.

The results of PSR1 N290.1 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.9.2. As identified in Reference [B.9-5], the Pickering PSR2 review of CSA N290.1-13 is an Incremental Review. PSR2 Incremental Review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.

- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.9.2 Compliance Assessment for Pickering PSR2

B.9.2.1 Application of PSR1 Reviews

The versions of N290.1 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

A clause-by clause review of the Pickering Units 5-8 Shutdown Systems (SDSs) against CSA N290.1-80 [B.9-3] was documented in NK30-REP-03680-00001 R000 [B.9-6].

This review concluded that with the exception of the 14 documentation deficiencies that posed gaps, the Shutdown Systems were found to be in direct compliance with the design requirements stated in CSA-N290.1-80. Actions were completed to address these document deficiencies. The following revised documents were issued: NK30-DR-63730-10002 [B.9-7], NK30-DR-63720-10001 [B.9-8], and NK30-DR-31730-10001 [B.9-9].

In addition Reference [B.9-6] identified eight Acceptable Deviations. Their relevance to extended Pickering operation is reviewed below. The eight Acceptable Deviations are:

- Clause 4.4.1.2 identifies a requirement that during the design phase, a target unavailability of $1x10^{-4}$ yrs/yr be used. This is not a requirement in N290.1-13, so is not relevant to PSR2.

- Clause 4.4.1.3 identifies a requirement that the design target for spurious operation of a shutdown system be 10^{-1} yrs/yr or less. This is related to clause 4.2.2 in N290.1-13, which requires a spurious trip frequency target be established. The justification provided for the acceptable deviation for PSR1 remains valid for PSR2 and is not impacted by operation beyond 2020.

- Clause 4.4.1.4 identifies a requirement that in the design phase, the system unavailability be calculated on the basis of the most demanding accidents and shall be based on main trip parameters only. This is related to clause 4.2.1.2 in N290.1-13 and to reliability requirements contained in CSA Standard N290.0-11, "General Requirements for Safety Systems for Nuclear Power Plants" [B.9-10]. These requirements are met by performing unavailability calculations for the Pickering Units 1,4 and 5-8 SDSs per the regulatory requirements in RD-98. Therefore this is not a PSR2 issue.

- Clause 4.4.1.5 identifies requirements for single failures, e.g., failure of any single instrumentation device shall not render the system unavailable or should be fail-safe. This requirement is now contained in CSA N290.0, Clause 4.7 and there are no gaps relating to the Shutdown Systems.

- Clause 4.4.2.5 requires that the grouping of components of a channel be adequate to allow commissioning, operations and maintenance without affecting the operation of other channels. The Pickering Units 1,4 and 5-8 SDSs designs meet this requirement by having redundant trip channels and procedures are in place for channelized maintenance and operational testing. Therefore this is not a PSR2 issue.

- Clause 4.4.14.1 on Quality of Equipment identifies basic requirements, e.g., of proven design, have predictable failure modes and last the life of the plant. It is related to requirement 4.7.1 in N290.1-13. These requirements have been demonstrated to be satisfied given the reliable operation of the Shutdown Systems and, therefore, is not an issue for PSR2.

- Clauses 6.1.1 and 6.2.2 on design documentation, require the requirements of the standard be contained in the design documentation and identify specific information to be included. These are not requirements in N290.1-13, so these clauses are not relevant to PSR2.

In addition to the review of N290.1, a review of the Shutdown Systems was performed against regulatory document R-8 "Requirements for Shutdown Systems for Nuclear Power Plants" [B.9-11]. This review was documented in NK30-REP-03680-00001 R000 [B.9-6]. This review will be used in the compliance discussion of the latest N290.1 version. This report concluded that other than one gap, the Pickering Units 5-8 Shutdown Systems comply with the requirements of R-8. The one gap was raised since the design documentation did not contain the requirements of clause 3.6.2, which states the following:

> *Manual actuation may be considered acceptable in place of one of the automatic parameters provided it is shown to the satisfaction of the AECB[10] that all of the following requirements are met:*
>
> > *(a) There is instrumentation designed to give the operator clear and unambiguous indication of the need to actuate the shutdown system.*
> >
> > *(b) The reliability of such instrumentation is commensurate with the requirements for availability of the shutdown system as specified in Section 3.4. If indication of only a single parameter is required, the instrumentation shall be part of the shutdown system…*

Requirement (a) is a Safety Analysis requirement which is met for all special safety systems. Requirement (b) is satisfied by including the required instrumentation in SDS unavailability

[10] Now the CNSC.

models. Given this, there is no gap associated with this clause and, therefore, the Pickering Units 5-8 Shutdown Systems comply with the requirements of R-8.

<u>Pickering Units 1,4</u>

As part of Pickering A Return to Service (PARTS), code reviews were conducted. The main submission for PARTS [B.9-12] committed to perform a review of N290.1-80 [B.9-3], which was the active version at that time. This review was completed in AECL Report, "Review of Pickering A Design Against Current Codes and Standards" [B.9-13]. This report reviewed the level of compliance of the Shutdown System (including both Shutdown System A (SDSA) and Shutdown System Enhancement (SDSE) logic).

This report provides a high level review of the degree that the Shutdown System complies with N290.1-80 through the addition of the new SDSE.

The SDSE for Units 1,4 was not intended to adopt all the CSA requirements since: (1) SDSE is an enhancement to an existing system, and (2) there are practical design constraints inherent in back-fitting changes to an existing shutdown system and reactor structure [B.9-13].

The design principles in the N290.1-80 requirements that could not be fully complied with were requirements for independent shutdown systems with conceptually different reactivity components, diverse trip parameters, and trip parameter displays in a Secondary Control Area.

Practical limitations on the SDSE design arose due to the layout of existing equipment, space availability, and the presence of high radiation fields in certain areas. Prior to developing the Nuclear Safety Design Requirements (NSDRs) for SDSE, these issues were discussed extensively with the CNSC. As a result of these discussions, the conceptual design and NSDRs of the Pickering Units 1,4 SDSE were finalized and accepted by the CNSC. The NSDRs also contain a requirement that for any deviations from requirements, the SDSE design must be shown to meet the intent of the requirement. This information is documented in the NSDR as an exception. This formal process ensured that the SDSE design met the intent of N290.1-80.

Reference [B.9-13] concludes that:

> *The intended overall performance of the Pickering 'A' Shutdown Systems will not be significantly affected by the identified deviations from full compliance with CSA N290.1-80. The enhanced shutdown system at Pickering 'A' provides further defence-in-depth through increased shutdown system reliability, trip coverage, reactivity depth, and protection against common-mode cross-link effects, such that the overall shutdown system effectiveness approaches that normally attributed to reactors having two completely separate and independent shutdown systems.*

There were six clauses that were not satisfied by the SDSE design and based on the above were classified as Acceptable Deviations. The reasons for not fully complying with the CSA requirements and the implications on shutdown system performance were assessed as being acceptable in [B.9-13] and were accepted by the CNSC. The basis for the acceptability of the exceptions continues to be applicable and is not impacted by operation beyond 2020. Note, the requirements for shutdown system separation and independence have been moved from CSA

N290.1 to N290.0 Clause 4.6 when the 2013 version of N290.1 was issued. However, the rationale and acceptance for classifying this issue as an Acceptable Deviation in PSR1 remains valid for PSR2.

In addition to the above noted deviations related to SDSE, Reference [B.9-13] identified seven other Acceptable Deviations. Their relevance to extended Pickering operation is reviewed below. The seven Acceptable Deviations are:

- Clause 4.4.3.3 requires that where instrumentation does not fail-safe on a loss of power, the loss of power condition shall be annunciated. The review concluded that there were sufficient alternate indications, including panel checks, such that these failures do not detract from the SDS availability. Note, this requirement has been moved to CSA N290.0-11, "General Requirements for the Safety Systems of Nuclear Power Plants" [B.9-10], Clause 4.8. The rationale for this acceptable deviation classification is applicable for PSR2 and is not impacted by operation beyond 2020.

- Clause 4.4.8.1 prescribes requirements for display of trip variables and their trip setpoints in the Main Control Room (MCR)/secondary control areas. The code review states:

  > *Continuous displays of the trip variables are provided in the MCR for both SDSA and SDSE. However, displays of trip setpoints are provided only for SDSE and some SDSA trips that use indicating alarm units on the control room panels. Given that the setpoints are generally fixed values, their display is not a significant benefit to safety.*

  This acceptable deviation is not impacted by operation beyond 2020.

- Clause 4.4.9 related to testing states:

  Dedicated on-line test facilities shall be provided to confirm the correct operation of the Shutdown System, including:

  a) Confirmation of the actual trip set point values;

  b) The response times of the critical elements of the system;

  c) The independence of redundant portions of the system;

  d) The necessary system availability requirements.

  On-line testing of critical parameter response times is not performed for all SDS components at Pickering. In lieu of this, bench testing is performed. The Safe Operating Envelope Compliance Tables for Pickering Units 1,4 [B.9-14] and Pickering Units 5-8 [B.9-15] demonstrate this. The updated version of N290.1 does not contain the requirement that response times be measured on-line, so both Units 1,4 and 5-8 SDSs comply with requirements and, therefore, this is not a PSR2 gap.

- Clause 4.4.9.6 contains specific requirements for testing of SDS sensors and how it should be performed, e.g., by manipulating the input variable until it reaches the setpoint. This specific requirement is no longer in N290.1-13 Section 4.5 on in-service testing and therefore is not relevant to PSR2.

- Clause 4.4.14.2 on Quality of Equipment requires manufacture and fabrication of equipment meet the requirements of the CSA Z299 series of standards. SDSA did not meet this standard as it was not available at the time of the original design. Subsequent modifications complied with this requirement. Pickering Units 5-8 SDSs meet this requirement. This remains not safety significant and is not a PSR2 issue.

- Clause 4.4.16.4 states that where a number of channels of one system are in close proximity, colour coding to differentiate between channels is the preferred identification method. This is not a mandatory requirement and not a PSR2 issue.

- Clause 5.1 requires the design, procurement and installation of SDSs meet the N286.0 series of standards. SDSA does not reference these standards as they were not available at the time of the original design. Subsequent modifications complied with this requirement. Pickering Units 5-8 SDSs meet this requirement. This remains not safety significant and is not a PSR2 issue.

In addition to the review of N290.1, a review of the Shutdown System was performed against regulatory document R-8 "Requirements for Shutdown Systems for Nuclear Power Plants" [B.9-11]. This review will be used in the compliance discussion of the latest N290.1 version. This review documented in [B.9-13] demonstrated that other than for the approved exceptions for the SDSE design described above, one gap and one acceptable deviation, the Pickering Units 1,4 Shutdown System complied with R-8.

The gap relates to Safety Report trip coverage assessments not directly addressing a loss of service water/loss of moderator cooling. However, since that time these events have been addressed in Appendix 9 of the Units 1,4 and 5-8 Safety Reports [B.9-16] and [B.9-17].

The one Acceptable Deviation was on the requirements for the design of SDSA in consideration of dynamic effects or jet forces. The applicable clause is no longer contained in N290.1 and an equivalent clause is contained in N290.0. The PSR2 review of CSA N290.0-11, "General Requirements for Safety Systems of Nuclear Power Plants", Clause 4.13 assesses the impact on all Special Safety Systems and identifies a Gap. However, because it is a generic gap, it is assigned as a PSR2 CSA N290.0-11 Gap.

Darlington NGS

The original code review of the Darlington ISR, as documented in OPG Report NK38-REP-03680-10011 R000 [B.9-18], was performed against CSA N290.1-80 [B.9-3]. In general it was found that the Darlington design of the Shutdown Systems met the requirements of N290.1-80. NK38-REP-03680-10011 does identify the following three gaps:

1. Common manufacturers are used for the SDS1 and SDS2 in-core flux detectors, in-core flux detector amplifiers, ion chamber amplifiers and computer termination modules.

Therefore, this resulted in a gap against clause 4.1.3.2. However, appropriate design and production separation was maintained during the manufacturing process.

Pickering Units 1,4 used the same manufacturer for SDSA and SDSE in-core flux detectors. However, compensating measures were taken such as: they were procured from different manufacturing plants and there is a slight difference in diameter between the Shutdown Systems. Pickering Units 5-8 complies with this clause [B.9-6]. In addition, this clause is no longer contained in N290.1-13. Therefore, this issue is not a PSR2 gap.

2. Response times of all critical components are not confirmed during on-line tests. Shutoff Rods and Liquid Injection Shutdown System valve timing are confirmed on-line and loop response testing of parameters is available on-line (Clause 4.4.9.2). However, component response time is confirmed as part of commissioning and post-maintenance testing. This above approach to response time testing was accepted by CNSC.

   This issue was later re-classified in Appendix B of NK38-REP-03680-10011 [B.9-18] as not a gap. Appendix B cites a procedure that had previously been prepared for Darlington, NK38-PROC-68000-002, "Requirements for Darlington NGS SDS1 & 2 – Safety Related Systems Tests" [B.9-19], which outlines how Darlington complies with the requirements for SDS critical parameter response times. Part of the basis for this is that Darlington utilizes digital components in its SDS trip logic, which do not degrade over time.

   This issue has already been addressed above for Pickering Units 1,4 and Units 5-8 and is not a PSR2 gap.

3. A minor documentation gap existed only for SDS1 Design Requirement documentation under Clause 4.4.15.4 dealing with qualification program testing. The clause is explicitly stated in the Design Requirements for SDS2 but not in the Design Requirements for SDS1 even though both systems are environmentally qualified. Both Pickering Units 1,4 and 5-8 comply with the requirements of this clause per the above code reviews and, therefore, this is not relevant to PSR2.

Following the issuance of CSA N290.1-13 [B.9-1] in December 2013, a code refresh review was conducted and documented in OPG Report NK38-REP-03680-10155 R000 [B.9-20]. A clause-by-clause review was done given the nature of the revisions made to the standard. NK38-REP-03680-10155 identifies that Darlington is in compliance with the requirements of CSA-N290.1-13 and no new gaps were identified. The changes made between version N290.1-80 and the new version N290.1-13 are discussed in the next section.

B.9.2.2 Application of Post PSR1 Reviews

As discussed above, CSA N290.1 has been updated since the last code reviews were performed for Pickering. It was updated to N290.1-13 [B.9-1] in 2013 from the original version N290.1-80 [B.9-3]. There were a number of changes made in the update of the code. The code was re-organized, the requirements made more precise and modified to be technology neutral. Of the changes listed in the CSA Impact Statement for the issue of N290.1-13 [B.9-4], only three potentially have some nuclear safety impact. These are:

a) It incorporates regulatory documents R-8 and R-10 (preserves the important requirements).

b) It has been aligned with regulatory document RD-337 (incorporation of design extension conditions and new build requirements).

c) Requirements have been made more precise (specific changes are not listed in the Impact Statement).

Since the changes referred to in items (a) and (c) are not specific, the new version was reviewed to identify any significant issues that are relevant to PSR2.

From an overall perspective, since the design of the Darlington Shutdown Systems complies with this new version of N290.1, it is expected that the same is the case for Pickering Units 5-8 SDS1 and SDS2. The designs are very similar and have the same design requirements with respect to: their function, two group philosophy, independence and separation, channelization, diversity, trip logic design, and having a fail-safe design, etc.

This is also predominantly the case for Pickering Units 1,4 Shutdown System, including SDSA and SDSE logic (with the exceptions discussed in section B.9.2, which have been assessed as being acceptable).

In addition, the following information and analyses demonstrate that the Units 1,4 and 5-8 Shutdown Systems meet nuclear safety objectives and licensing requirements:

- Performance requirements to ensure credits in the Safety Report are satisfied and Safety Report objectives are met, are contained in the Operational Safety Requirements (OSR) for Pickering Units 1,4 and 5-8 [B.9-21] and [B.9-22].

- The Probabilistic Safety Assessments have demonstrated that public safety goals have been met with the design performance of the Shutdown Systems.

- Per Regulatory Document RD/GD-98 [B.9-23], unavailability analyses are completed that demonstrate that reliability requirements are met for the Shutdown Systems.

- Environmental and seismic qualification requirements are in place to ensure the required performance following applicable design basis accidents.

- Compliance with regulatory document R-8 [B.9-11] has been demonstrated for both Pickering Units 5-8 and Pickering Units 1,4 with any exceptions being acceptable, as described above.

The presence of all the above information, provides assurance that the N290.1-13 requirements are met for Pickering Units 1,4 and Units 5-8 at a high level.

For additional assurance, the requirements of N290.1-13 and N290.1-80 were compared to determine if there were any safety significant changes where Pickering SDSs may not meet the intent of the revision (not including those described above for Pickering Units 1,4 Shutdown System that have approved exceptions). The majority of clauses in N290.1-80 are covered in N290.1-13. The following clauses fall into the category described above and are the clauses with potential safety significant intent changes:

- Clause 4.3.3.4 states:

  Trip set points shall be selected to provide sufficient allowance between the set points and the corresponding safety limits to account for uncertainties. The uncertainties include but are not limited to:…

  This clause is new in N290.1-13. It is satisfied by completion of Instrument Uncertainties Calculations for Pickering Units 1,4 and 5-8. These are referenced in the Operational Safety Requirements documents [B.9-21] and [B.9-22].

- Clause 4.4.2 on MCR displays states:

  For certain trip parameters, such as those that originate from in-core flux detectors, display of margins to trip should be provided in the main control room.

  The MCR design provides for a display of margin to trip for in-core flux detectors for Pickering Units 1,4 SDSA and SDSE. The same applies to Pickering Units 5-8 SDS1 and SDS2. Therefore this clause is met.

- Clause 4.6 on Cyber Security

  This clause is new and identifies considerations for cyber security. OPG has a program in place to ensure cyber security requirements are in place as per OPG Procedure, N-PROC-MP-0103, "Security for Real-Time Process Computing Systems" [B.9-24]. Therefore this clause is met.

In addition to the above there is one new requirement relating to new plant designs. This is,

- Clause 4.1.8.2 requires remote tripping and monitoring capability to be available for both Shutdown Systems in a secondary control room. Pickering 5-8 has tripping capability for both Shutdown Systems in the main control room. Pickering 1,4 has SDSA and SDSE tripping capability from the main control room.

  Pickering 5-8 has SDS2 tripping capability in the Unit Emergency Control Centres (UECCs). Pickering 1,4 has remote SDSE tripping capability in the SDSE instrument room. The purpose of this requirement is to ensure that probability of failure to shut down is sufficiently low for events that result in the main control room becoming unavailable (typically for common mode events). Failure to shut down the reactor using manual trips for common mode events is unlikely to be a significant contributor to plant risk; however, this has not been confirmed. Therefore, the absence of remote SDS tripping and monitoring capability for Pickering 5-8 SDS1 and Pickering 1,4 SDSA is **PSR2 CSA N290.1-13 Gap #1**.

Given the previous PSR1 reviews and additional reviews of specific clauses, Pickering is largely compliant with the significant safety clauses in the standard, with the exception of the gap identified.

B.9.3 Compliance Summary for Pickering PSR2

There is one PSR2 CSA N290.1-13 gap which relates to Safety Factor 1 (Plant Design):

1. Clause 4.1.8.2 of CSA N290.1-13 is for a new plant and requires remote tripping and monitoring capability for both Shutdown Systems. Pickering Units 1,4 only have one Shutdown System with tripping capability from separate logic (SDSA and SDSE). Remote tripping capability is available for Pickering 5-8 SDS2 and Pickering 1,4 SDSE. However, Pickering Units 5-8 and 1,4 do not have remote tripping and monitoring capability for SDS1 or SDSA respectively. Therefore, this has been identified as a PSR2 gap.

B.9.4 References

[B.9-1] CSA Standard, N290.1-13, *Requirements for the Shutdown Systems of Nuclear Power Plants*, December 2013.

[B.9-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.9-3] CSA Standard, N290.1-80, *Requirements for the Shutdown Systems of CANDU Nuclear Power Plants*, December 1980.

[B.9-4] CSA Impact Statement for Public Review, *Notification of CSA N290.1 Publication; Product Designation: CSA N290.1-13*, Date not provided.

[B.9-5] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.9-6] OPG Report, NK30-REP-03680-00001 R000, *Pickering NGS B Integrated Safety Review - Plant Design Safety Factor*, August 2007.

[B.9-7] OPG System Design Requirements, NK30-DR-63730-100022 R001, *Shutdown System Number Two (SDS2)*, October 2012.

[B.9-8] OPG System Design Requirements, NK30-DR-63720-10001 R002, *Shutdown System Number One (SDS1)*, October 2012.

[B.9-9] OPG System Design Requirements, NK30-DR-31730-10001 R001, *Shutoff Units*, October 2012.

[B.9-10] CSA Standard, N290.0-11, *General Requirements for Safety Systems of Nuclear Power Plants*, October 2011.

[B.9-11] AECB Regulatory Document, R-8: *Requirements for Shutdown Systems for Nuclear Power Plants,* February 1991.

[B.9-12] OPG Letter, NA44-CORR-00531-00381, R.J. Strickert to J.S.C. Tong, *Pickering A Updated Basis for Return to Service Document*, April 20, 2001.

[B.9-13] AECL Assessment Document, 44RS-00531-ASD-00 Rev. 04, *Review of Pickering A Design Against Current Codes and Standards*, November 2000.

[B.9-14] OPG Operational Safety Requirements, NA44-OSR-08131.02-00001-TABLE-01 R001, *Pickering NGS A Operational Safety Requirements: Shutdown Systems Compliance Table*, June 2014.

[B.9-15] OPG Operational Safety Requirements, NK30-OSR-08131.02-00004-TABLE-01 R001, *Pickering NGS B Operational Safety Requirements: Shutdown Systems Compliance Table*, April 2014.

[B.9-16] OPG Safety Report, NA44-SR-01320-00002 R004, *Pickering Nuclear 1-4 Safety Report: Part 3 - Accident Analysis*, September 2013.

[B.9-17] OPG Safety Report, NK30-SR-01320-00003, *Pickering Nuclear 5-8 Safety Report: Part 3 - Accident Analysis*, October 2014.

[B.9-18] OPG Report, NK38-REP-03680-10011 R000, *Review of CAN/CSA-N290.1-80 (R2001) (Jan 1980) Requirements for Shutdown Systems for CANDU Nuclear Power Plants for Darlington Integrated Safety Review*, August 2011.

[B.9-19] OPG Procedure, NK38-PROC-68000-002, *Requirements for Darlington Shutdown Systems One and Two – Safety Related Systems Tests*, December 1996.

[B.9-20] OPG Report, NK38-REP-03680-10155 R000, *Code Refresh Review of CSA-N290.1-13 Requirements for Reactor Shutdown Systems for CANDU Nuclear Power Plants*, February 2014.

[B.9-21] OPG Operational Safety Requirements, NA44-OSR-08131.02-00004 R002, *Pickering NGS A Operational Safety Requirements: Shutdown Systems*, June 2010.

[B.9-22] OPG Operational Safety Requirements, NK30-OSR-08131.02-00004 R003, *Pickering NGS B Operational Safety Requirements: Shutdown Systems*, February 2016.

[B.9-23] CNSC Regulatory Document, RD/GD-98, *Reliability Programs for Nuclear Power Plants*, June 2012.

[B.9-24] OPG Procedure, N-PROC-MP-0103 R003, *Security for Real-Time Process Computing Systems*, March 2015.

B.10 CSA N290.2-11, "Requirements for Emergency Core Cooling Systems of Nuclear Power Plants"

B.10.1 Background

The following, paraphrased from the preface and scope of CSA N290.2 [B.10-1], provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *The purpose of CSA N290.2 is to ensure the emergency core cooling system meets its primary purpose, which is to transfer heat from the reactor core following a loss of reactor coolant that exceeds the makeup capability of process systems.*
>
> *CSA N290.2 sets the requirements for the design, qualification, installation, operation, maintenance, inspection, and documentation of the emergency core cooling system for a water-cooled nuclear power plant. CSA N290.2 also applies to all support systems required to ensure that the ECC [Emergency Core Cooling] system is able to maintain adequate heat transfer for as long as necessary to maintain the release of radioactive material within reference dose limits by limiting fuel failure.*

All of N290.2-11 is directly relevant to Safety Factor 1 (Plant Design).

CSA N290.2 is identified in Appendix E.1 of the R04 Pickering Licence Conditions Handbook [B.10-2] as "Guidance or Criteria". Section 6.1, "Design Program" of [B.10-2] states that "Recommendations and guidance are found in… N290.2, which covers emergency core cooling."

CSA N290.2-11 is the first edition of the standard. It replaces Regulatory Document, R-9: "Requirements for Emergency Core Cooling Systems for CANDU Nuclear Power Plants" [B.10-3] and is largely based on it. Previous code reviews for Pickering were performed against R-9.

The relationship between and the PSR2 review approach for standards N290.0, N290.1, N290.2 and N290.3 are outlined here. The common generic requirements for the Special Safety Systems (previously addressed in earlier versions of N290.1, R-9 and R-7) were amalgamated into a companion document, CSA N290.0. The common requirements will be reviewed within the PSR2 N290.0 assessment. The system-specific requirements identified in N290.1, N290.2 and N290.3 will be reviewed within the PSR2 assessments for these standards. Any gaps identified as generic to the Special Safety Systems will be recorded as N290.0 gaps. If a gap is specific to only one of the Special Safety Systems, or system specific standard, it will be recorded as a gap against that standard (N290.1, N290.2, and N290.3) only.

The results of PSR1 CSA N290.2 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.10.2. As identified in Reference [B.10-4], the Pickering PSR2 review of CSA N290.2-11 is an Incremental Review. PSR2 Incremental Review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.

- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.10.2 Compliance Assessment for Pickering PSR2

B.10.2.1 Application of PSR1 Reviews

CSA N290.2-11 is the first edition of the standard. Therefore it was not subject to previous reviews conducted for Pickering. A code review was conducted for Darlington against CSA N290.2 and is addressed below. CSA N290.2 replaces Regulatory Document, R-9: "Requirements for Emergency Core Cooling Systems for CANDU Nuclear Power Plants" [B.10-3] and is largely based on it. Previous code reviews for Pickering were performed against R-9.

The versions of N290.0 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

CSA N290.2 was not in effect at the time of the Pickering B ISR code reviews. A clause-by clause review of the compliance of the Pickering Units 5-8 Emergency Coolant Injection System (ECIS) against R-9 (CSA N290.2 predecessor) was documented in NK30-REP-03680-00001 R000 [B.10-5].

With the exception of three gaps associated with document discrepancies, the ECIS design was found to comply with the requirements of R-9. The three gaps were:

1. ECIS design requirements shall be based on the assumption that the least effective of the shutdown systems has operated successfully (clause 2.3). This requirement is contained in clause 5.2.1.2 of N290.2-11 and, therefore, this is still applicable.

2. ECIS cooling requirements must be capable of addressing a Design Basis Accident coincident with an impairment of containment (clause 3.2). This requirement is not contained in N290.2-11 and therefore this gap is not applicable to PSR2.

3. Discrepancies between ECI Heat Transport low pressure signal setpoint between the Design Requirements and the OSR (Clause 3.4.9).

These three gaps have been addressed by revising the Pickering Units 5-8 ECIS Design Requirements documentation [B.10-6] to include the two requirements and to address the discrepancy. However, the requirement in clause 5.2.1.2 of N290.2-11 is not achievable for Pickering Units 1,4 ECIS because there is only one Shutdown System [B.10-7]. Therefore, this is identified as **PSR2 CSA N290.2-11 Gap #1**.

One Acceptable Deviation was raised in the assessment, regarding clause 3.4.9 on the use of manual intervention. There are certain conditions when High Pressure (HP) injection may no longer be essential after a loss of coolant accident and thus can be terminated by the Operator. For example, following successful initial HP injection, once the HTS has been filled and temperatures lowered, HP injection can be terminated and Low Pressure (LP) pumps will automatically be started. The required actions to make this operating mode transition are controlled and documented in system Operating Manuals. Therefore, this is not a safety significant issue for PSR2.

Pickering Units 1,4

As part of Pickering A Return to Service (PARTS), code reviews were conducted. CSA N290.2 did not exist at that time. The main submission for PARTS [B.10-8] committed to perform a review of Regulatory Document R-9 (CSA N290.2 predecessor). This review was completed in AECL Report, "Review of Pickering A Design Against Current Codes and Standards" [B.10-9].

The review identified the following seven gaps:

1. Clause 2.2 (b) on the requirement that support systems shall be considered as safety support equipment and meet all the requirements of R-9. There was only one requirement in clause 3.4.5 where compliance could not be shown. A loss or high temperature of Low Pressure Service Water (LPSW) will affect the moderator heat exchangers and moderator room Air Cooling Units (ACUs), incapacitating ECIS recovery. To address this issue a design change was made prior to units restarting, to provide a back-up source of cooling water to the moderator HXs. Therefore there is no PSR2 gap.

2. Clause 3.4.1 requires the ECI system to satisfy availability requirements. ECIS could not be shown to meet the legacy R-9 [B.10-3] unavailability target of $1x10^{-3}$ yrs/yr for all scenarios. Design changes recommended in the "Core Damage Frequency Reduction Study – Pickering NGS A Risk Assessment (PARA) Review" [B.10-10] were made to improve ECI recovery availability. CSA N290.2-11 does not specify an ECI system reliability target. Instead, there is now a generic special safety system reliability requirement in CSA N290.0-11, Clause 4.5.1. Limits are established as part of CNSC RD/GD-98 compliance and the licencing target for ECI is $2x10^{-3}$ yrs/yr [B.10-11], which has been demonstrated to be met per [B.10-11]. Therefore, this is not a PSR2 gap.

3. Clause 3.4.3 requires the design to have redundancy such that a single failure cannot result in a loss of system capability. Thirteen instances of single failures existed in the design, of which five were being addressed by design changes. Reference [B.10-9] provided justification for addressing those five as they resulted in the greatest reduction in core damage frequency. The single failures not addressed were either of low probability or there would be high radiological exposure (person-rem) to make changes. This rationale remains valid and does not introduce a PSR2 gap.

4. Clause 3.4.5 requires that as far as practicable, all ECIS equipment shall be designed such that its most probable failure modes will not result in a reduction in safety. Changes were made to address this issue, as identified in Reference [B.10-10]. Therefore there is no PSR2 gap.

5. Clause 3.5.2 requires that as far as practicable, the ECIS shall be independent from all process systems. The Pickering 1,4 ECI system utilizes the moderator system in its low pressure recovery mode. However, based on the improvements made to the design and changes proposed in Reference [B.10-10], the degree of commonality between the moderator system and ECI recovery has been reduced. Therefore this requirement was met to the degree practicable and was an Acceptable Deviation for PARTS PSR1. This rationale remains valid and there is no PSR2 gap.

6. Clause 3.5.4 requires that if ECIS sub-systems are considered to be independent for the purpose of safety analysis, principles for separation and independence of each sub-system shall be prepared and approved by the regulator. It was recommended to review the Safety Report to identify credits for independence between sub-systems and document the basis for the assumed independence. This R-9 clause is included in N290.0-11, Clause 4.6.2 (b) [B.10-12] rather than in N290.2-11. Assessment of that clause is contained in the PSR2 assessment of CSA N290.0 and it did not identify any gaps relating to sub-system independence, with the exception of the dependence on the moderator system, which has already been addressed in Item 5 above.

7. Clause 3.11 on seismic requirements specifies requirements for equipment to function following a site design earthquake. To achieve full compliance with this requirement, the moderator pump motors need to be seismically qualified. To comply with this requirement all components in the ECI recovery flowpath have been seismically assessed [B.10-13]. Therefore there is no PSR2 gap associated with this issue.

The review also identified one Acceptable Deviation on the requirements for the design of ECIS in consideration of dynamic effects or jet forces. This requirement is included in N290.0-11, Clause 4.13 rather than in N290.2. Further, this issue is a PSR2 Gap common to all Special Safety Systems and is, therefore, addressed in the PSR2 assessment of CSA N290.0.

Darlington NGS

The code review of the Darlington ISR, as documented in OPG Report NK38-REP-03680-10004 R000 [B.10-14], was performed against R-9. It was found that the Darlington design of the ECI system meets the requirements of Regulatory Document R-9[11].

Following the issuance of CSA N290.2-11 [B.10-1] in October 2011, a Code Refresh review was conducted and documented in OPG Report NK38-REP-03680-10212 R000 [B.10-15]. A clause-by-clause review was done given this was the first edition of the standard and due to the significant differences between R-9 and N290.2. For example, all of the requirements common to all special safety systems that were contained in R-9 for ECIS, are now contained in CSA N290.0-11, "General Requirements for the Safety Systems of Nuclear Power Plants" [B.10-12].

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[11] A gap on clause 3.5.4 was initially identified in the report, but was later modified to be compliant in Appendix B.

In addition, Reference [B.10-15] states that: "N290.2 is devoted to modern day requirements specific to ECC performance and provides a greater level of detail than R-9." This will be considered in Section B.10.2.2 on the application of Post-PSR1 reviews.

NK38-REP-03680-10212 [B.10-15] identifies that Darlington is in compliance with the requirements of CSA-N290.2-11 and no gaps were identified.

B.10.2.2 Application of Post PSR1 Reviews

The CSA N290.2-11 Impact Statement [B.10-16] provides a summary of significant features of the new standard. Of the features described, the following are considered safety significant for the purposes of PSR2:

1. *The standard reflects current Canadian regulatory requirements.*
2. *The standard includes requirements for debris interceptors and controls to minimize fouling of interceptors.*

In addition, the Darlington Code Refresh [B.10-15] summarized that N290.2 is devoted to modern day requirements specific to Emergency Core Cooling performance and provides a greater level of detail than R-9.

With respect to the standard reflecting current regulatory requirements, this is not a PSR2 issue. It was confirmed that both Pickering Units 1,4 and 5-8 comply with the requirements of R-9, or that any exceptions for Units 1,4 are acceptable [B.10-9], [B.10-5].

<u>Requirements for Debris Interceptors or Strainers</u>

Clause 5.14 contains requirements for debris interceptors. Clause 5.14.1 states:

> *Existing plants shall demonstrate that their existing debris interceptors meet minimum allowable performance standards. Clauses 5.14.2 through 5.14.16 shall apply to new builds.*

Clause 5.14.1 is satisfied for Units 1,4 by the following:

- The ECIS design contains strainers in the flowpath to the suction of the moderator pumps, which assure that minimum pump Net Positive Suction Head (NPSH) requirements are satisfied during ECIS recovery operation.
- Significant work has been performed to formulate a conservative methodology used to determine the debris quantities to be used for assessing strainer design adequacy [B.10-17].
- Sections 5.2, A.1.3, A.4.1, C.4.3.7, D.4.3.3 of the Pickering Units 1,4 Operational Safety Requirements (OSR) [B.10-18] discuss and provide the allowable performance

standards for the ECIS strainers and identify the required actions if standards (limits) are not met.

Clause 5.14.1 is satisfied for Units 5-8 by the following:

- The ECIS design contains strainers in the flowpath to the suction of the ECI recovery pumps, which assure that minimum pump NPSH requirements are satisfied during ECIS recovery operation.

- Significant work has been performed to formulate a conservative methodology used to determine the debris quantities to be used for assessing strainer design adequacy [B.10-17].

- Sections 5.2, A.1.3, A.4.7, A.4.8, Table A.4 and D.4.3.3 of the Pickering Units 5-8 OSR [B.10-19] discuss and provide the allowable performance standards for the ECIS strainers and identify the required actions if standards (limits) are not met.

The other aspect of this first edition of N290.2 identified in Reference [B.10-15] is that requirements have been made more precise. These changes were not specified in the CSA Impact Statement. The approach taken was to review the new version to identify any significant issues that could impact Pickering's PSR2 compliance given the following context.

From an overall perspective, since the Darlington ECI System design complies with N290.2, it is expected that the same is the case for Pickering Units 5-8. The designs are similar and have the same design requirements with respect to: their function, two group philosophy, independence and separation, channelization, diversity and being fail-safe, etc. The Units 1,4 ECI design contains a significant difference in that portions of the moderator system are used for ECIS recovery mode. This difference was previously assessed in the Pickering A Return to Service R-9 review [B.10-9] and demonstrated to not pose a significant safety impact.

In addition the following information and analyses demonstrate that the Units 1,4 and 5-8 ECI systems meet nuclear safety objectives and licensing requirements:

- Performance requirements to ensure credits in the Safety Report are satisfied, are contained in the OSR for Pickering Units 1,4 and 5-8 [B.10-18] and [B.10-19].

- The Probabilistic Safety Assessment has demonstrated that public safety goals have been met with the design performance of the Pickering ECI systems.

- Per Regulatory Document RD/GD-98 [B.10-20], unavailability analyses are completed that demonstrate that reliability requirements are met for the ECI systems.

- Also, environmental and seismic qualification requirements are in place to ensure the required performance following applicable design basis accidents.

The presence of all the above information provides assurance that the N290.2-11 requirements are met for Pickering at a high level. For additional assurance, the requirements of N290.2-11 were reviewed to determine if there were any new requirements, or clauses containing more precise wording than contained in R-9, or differences in the Darlington design compared to Pickering, that could challenge Pickering's PSR2 compliance with the new standard.

As a result of this review, the following clause was identified, which requires review:

- Clause 5.7 on Chemistry, Water Quality and Inventory Management is new and contains requirements that are potentially safety significant. The following sub-clause requires review:

  Clause 5.7.1 (c) states:

  > *Water chemistry, specifically pH and oxygen levels, shall be controlled to prevent: interactions of chemicals that might lead to fouling (e.g., plugging of instrument tubing, loading of debris interceptors, poison precipitation).*

  For Pickering Units 1,4 safety limits are in place for the pH level and concentration of hydrazine as specified in Table A.1 of the OSR [B.10-18]. For Pickering Unit 5-8, similar safety limits are contained in the Table A.1 of the Pickering Units 5-8 OSR [B.10-19]. Chemistry Control procedures ensure these safety limits are met during operation, per Chemical Control Procedure, "Emergency Coolant Injection Unit 058", NK30-CCP-33350-00001 [B.10-21]. This satisfies this clause.

  The following clauses related to 5.7.1 (c) are also satisfied with the same compliance basis: 5.7.2.1, 5.7.2.3, 5.7.3.1, 5.7.4.1 and 5.7.4.3.

Clauses 5.12 and 5.13 on Venting and Draining and Leakage Collection respectively, are new clauses not contained in R-9. However, they do not contain safety significant requirements and both Pickering Units 1,4 and 5-8 have design features similar to Darlington. Other clauses are either not safety significant, or Pickering Units 5-8 and 1,4 have design features similar to Darlington.

As noted above, CSA N290.2-11 has requirements for debris interceptors (strainers) that are specific to new plants in Clauses 5.14.2 through 5.14.16. Several of these clauses (e.g., 5.14.9, 5.4.13, 5.14.15, 5.14.16) relate to controls of materials and compounds used inside containment in order to minimize the potential for consequential strainer plugging. OPG initiated and completed additional assurance work for both Pickering 5-8 and Pickering 1,4 to address these issues under CNSC Generic Action Item (GAI) 06G01 *Emergency Core Cooling Strainer Deposits.*

A detailed assessment of the potential sources of strainer debris and contaminants is documented in [B.10-22] for Pickering 5-8. This assessment demonstrated that there was sufficient margin to ensure post-accident ECI recovery strainer effectiveness. A similar assessment was performed for Pickering 1,4 in [B.10-23]; however, this review identified vulnerabilities to strainer failure due to the incremental debris loading. In order to restore margin, a project was initiated to modify the Pickering 1,4 ECI recovery strainer design and to install new strainer modules in Units 1 and 4. This installation is now complete. Therefore, the above demonstrates that Pickering is compliant with the requirements for new plant debris interceptors.

However, there is one ancillary requirement relating to debris interceptors for new plants with which Pickering does not comply. This relates to Clause 5.14.11 which requires instrumentation to be available to monitor post-accident effectiveness and to determine the extent of plugging of the interceptor. While relative health of a strainer can be inferred by a combination of ECI recovery pump performance and reactor building water level, there is no direct correlation between these conditions and debris loading available. This is, therefore, **PSR2 CSA N290.2-11 Gap #2** and is applicable to both Pickering 5-8 and 1,4.

Given the above review of specific clauses, Pickering 5-8 complies with the significant safety clauses in the standard. For Pickering 1,4 there was one gap identified. Additionally, there is one gap relating to specific requirements for a new plant design which is common to all Pickering Units.

B.10.3 Compliance Summary for Pickering PSR2

There are two PSR2 CSA N290.2-11 gaps which relate to Safety Factor 1 (Plant Design):

1. Clause 5.2.1.2 of CSA N290.2-11 requires that Emergency Coolant Injection System (ECIS) design requirements be based on the assumption that the least effective of the Shutdown Systems has operated successfully. The Pickering Units 5-8 Safety Report analysis does address this requirement and the requirement is also contained in the Pickering Units 5-8 Design Requirements. However, this requirement cannot be met for Pickering Units 1,4 since there is only one Shutdown System (albeit with tripping capability from separate SDSA and SDSE logic). Therefore, this has been identified as a PSR2 gap.

2. Clause 5.14.11 of CSA N290.2-11 requires instrumentation to be available to monitor post-accident effectiveness and to determine the extent of plugging of Emergency Coolant Injection System (ECIS) debris interceptors (strainers). While relative health of a strainer can be inferred by a combination of ECIS recovery pump performance and reactor building water level, there is no direct correlation between these conditions and debris loading available. Therefore, this has been identified as a PSR2 gap (which is applicable to both Pickering Units 5-8 and 1,4).

B.10.4 References

[B.10-1] CSA Standard, N290.2-11, *Requirements for Emergency Core Cooling Systems of Nuclear Power Plants*, October 2011.

[B.10-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.10-3] AECB Regulatory Document, R-9, *Requirements for Emergency Core Cooling Systems for CANDU Nuclear Power Plants*, February 1991.

[B.10-4] OPG Report P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.10-5] OPG Report, NK30-REP-03680-00001 R000, *Pickering NGS B Integrated Safety Review - Plant Design Safety Factor*, August 2007.

[B.10-6] OPG System Design Requirements, NK30-DR-33350-10004 R001, *Emergency Coolant Injection System*, September 2012.

[B.10-7] OPG System Design Requirements, NA44-DR-33350-10001 R000, *Emergency Coolant Injection System*, January 2004.

[B.10-8] OPG Letter, NA44-CORR-00531-00381, R.J. Strickert to J.S.C. Tong, *Pickering A Updated Basis for Return to Service Document*, April 20, 2001.

[B.10-9] AECL Assessment Document, 44RS-00531-ASD-001 Rev. 04, *Review of Pickering A Design Against Current Codes and Standards*, November 2000.

[B.10-10] AECL Report, 44RS-03611-ASD-001 Rev. 03, *Reduction of Severe Core Damage Frequency - Picketing NGS 'A' Risk Assessment (PARA) Review*, September 2000.

[B.10-11] OPG Report, NA44-REP-09051.1-00014 R000, *2014 Annual Reliability Report - Pickering Units 1 & 4*, March 2015.

[B.10-12] CSA Standard, N290.0-11, *General Requirements for Safety Systems of Nuclear Power Plants*, October 2011.

[B.10-13] OPG Report, NA44-REP-02004-00002 R001, *Pickering NGS A Seismic Success Path Addendum Including the Safe Shutdown Equipment List*, August 2013.

[B.10-14] OPG Report, NK38-REP-03680-10004 R000, *Review Of CNSC R-9 (February 1991) Requirements for Emergency Core Cooling Systems for CANDU Nuclear Power Plants for Darlington Integrated Safety Review*, September 2011.

[B.10-15] OPG Report, NK38-REP-03680-10212 R000, *Code Refresh Review Of CSA-N290.2-11 Requirements For Emergency Core Cooling Systems Of CANDU Nuclear Power Plants*, January 2014.

[B.10-16] CSA Impact Statement, *Notification of CSA N290.2-11 Publication*, Date not provided.

[B.10-17] OPG Report, N-REP-34320-985099 R001, *Assessment Methodology of LOCA-Generated Debris Impact on ECIS Recovery Operation*, June 1999.

[B.10-18] OPG Operational Safety Requirements, NA44-OSR-08131.02-00004 R002, *Pickering NGS A Operational Safety Requirements: Emergency Coolant Injection System*, October 2010.

[B.10-19] OPG Operational Safety Requirements, NK30-OSR-08131.02-00001 R003, *Pickering NGS-B Operational Safety Requirements: Emergency Coolant Injection System*, June 2012.

[B.10-20] CNSC Regulatory Document, RD/GD-98, *Reliability Programs for Nuclear Power Plants*, June 2012.

[B.10-21] OPG Chemical Control Procedure, NK30-CCP-33350-00001 R005, *Emergency Coolant Injection Unit 058*, January 2013.

[B.10-22] OPG Letter, P. Pasquet to T. Schaubel, NK30-CORR-00531-05194 R001, *Pickering B – Generic Action Item 06G01 Emergency Core Cooling System Strainer Deposits – Status Update and Request for Closure*, June 30, 2009.

[B.10-23] OPG Letter, W.M. Elliot to T. Schaubel, NA44-CORR-00531-06062 R000, *GAI 06G01: Emergency Core Cooling System Strainer Deposits – Status Update,* June 30, 2009.

B.11 CSA N290.3-11, "Requirements for the Containment System of Nuclear Power Plants"

B.11.1 Background

The following, paraphrased from the preface and scope of CSA N290.3-11 [B.11-1], provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *The purpose of CSA N290.3 is to ensure the containment system fulfills the following safety functions:*
>
> - *Controls and minimizes radioactive releases for normal operation, AOOs [Anticipated Operational Occurrences], and DBAs [Design Basis Accidents];*
> - *Considers BDBAs [Beyond Design Basis Accidents], including severe accident conditions; and*
> - *Minimizes the impact of radiological exposure to plant personnel from within the containment boundary.*
>
> *CSA N290.3 sets the requirements for the design, instrumentation, shielding, support systems, operations, and maintenance of the containment system. In addition, it sets the design requirements for containment subsystems.*

All of N290.3-11 is directly relevant to Safety Factor 1 (Plant Design).

CSA N290.3 is identified in Appendix E.1 of the R04 Pickering Licence Conditions Handbook [B.11-2] as "Guidance or Criteria". Section 6.1, "Design Program" of [B.11-2] states that "Recommendations and guidance are found in… N290.3, which covers containment systems."

CSA N290.3-11 is the first edition of the standard. It replaces Regulatory Document, R-7: "Requirements for Containment Systems for CANDU Nuclear Power Plants" [B.11-3] and is largely based on it. Previous code reviews for Pickering were performed against R-7.

The relationship between and the PSR2 review approach for standards N290.0, N290.1, N290.2 and N290.3 are outlined here. The common generic requirements for the Special Safety Systems (previously addressed in earlier versions of N290.1, R-9 and R-7) were amalgamated into a companion document, CSA N290.0. The common requirements will be reviewed within the PSR2 N290.0 assessment. The system-specific requirements identified in N290.1, N290.2 and N290.3 will be reviewed within the PSR2 assessments for these standards. Any gaps identified as generic to the Special Safety Systems will be recorded as N290.0 gaps. If a gap is specific to only one of the Special Safety Systems, or system specific standard, it will be recorded as a gap against that standard (N290.1, N290.2, and N290.3) only.

The results of PSR1 CSA N290.3 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.11.2. As identified in Reference [B.11-4], the Pickering PSR2 review of CSA N290.3-11 is an Incremental Review. PSR2 Incremental Review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.
- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.11.2 Compliance Assessment for Pickering PSR2

B.11.2.1 Application of PSR1 Reviews

CSA N290.3-11 is the first edition of the standard. Therefore, it was not subject to previous reviews conducted for Pickering. A code review was conducted for Darlington on CSA N290.3-11 and is addressed below. CSA N290.3 replaces Regulatory Document, R-7: "Requirements for Containment Systems for CANDU Nuclear Power Plants" [B.11-3] and is largely based on it. Previous code reviews for Pickering were performed against R-7.

The versions of CSA N290.3 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

CSA N290.3 was not in effect at the time of the Pickering Units 5-8 code reviews. A clause-by clause review of the compliance of the Pickering Units 5-8 Containment System against R-7 (predecessor to N290.3) was documented in NK30-REP-03680-00001 [B.11-5].

With the exception of five gaps, the Containment design was found to comply with the requirements of R-7. The five gaps were:

1. Analysis could not be found to demonstrate that containment can adequately withstand overpressures resulting from complete failure of dousing in combination with other initiating events. This addressed three of the discrepancies against clauses 3.4.1, 3.4.3 and 3.4.4. This requirement does not exist in the revised N290.3-11, therefore this is not a PSR2 gap.

2. Regarding clause 3.7.4, the design documentation does not include the requirement that the containment system is not dependent on power supplies from the turbine generators associated with any reactor unit within that containment system. This is a documentation issue only. The containment systems at Pickering Units 1,4 and Units 5-8 do not rely on Class IV power to fulfill any of their safety functions. This requirement is addressed in N290.0-11, Clause 4.9 rather than in N290.3-11. This issue has no safety significance and is not a PSR2 gap against CSA N290.0.

3. There is a gap on clause 3.9 which refers to Appendix A clause A.2.2 (c). It pertains to containment penetrations consisting of one closed isolation valve for lines of 50 mm or less, which are normally closed to the containment atmosphere and connected to an easily defined closed system outside containment. The requirement is that they be part of the containment envelope and be constructed to the requirements of the ASME Code (Section III, Class 2). The code review identifies this gap as a documentation issue since the requirement is not contained in the Design Requirements document. This requirement is still contained in N290.3-11. OPG committed to assess the level of compliance with Appendix 2.2 and revise the required design documentation [B.11-6]. The gap was later closed on the basis that "a review of our design requirements and applications has concluded that PNGS-B Containment appurtenance isolation valves are in compliance with R-7 Clause A2.2(c)".

In addition the following six Acceptable Deviations (ADs) were raised in the assessment:

1. Clause 3.4.2 identifies that the negative design pressure of containment must not be greater than that predicted in the Safety Report for a set of postulated events with failure of dousing. It is not necessary to evaluate event combinations involving failure of dousing for negative containment design pressure impacts, because these event combinations result in higher rather than lower containment pressure. Therefore, this is not a PSR2 gap.

2. Clause 3.8.3 requires design principles for separation of redundant instrument channels be prepared and be approved by the regulator. These requirements relating to separation are not included in N290.3-11, rather they have been incorporated into clauses under 4.6 of CSA N290.0-11, "General Requirements for the Safety Systems of Nuclear Power Plants" [B.11-7]. However, there is no specific requirement for regulatory approval of the separation principles, therefore, there is no gap relating to this issue for PSR2.

3. Clause 3.8.4 requires that if containment sub-systems are considered to be independent for the purpose of safety analysis, principles for separation and independence of each sub-system shall be prepared and approved by the regulator. This requirement is not contained in N290.3-11, but the principles of separation and independence for safety analysis have been incorporated into Clause 4.6.2 of CSA N290.0-11. However, regulatory approval is not required in CSA N290.0, therefore, there is no gap relating to this issue for PSR2.

4. Clause 3.11.2 requires that a report be prepared demonstrating the adequacy of shielding provisions to ensure required operator access after an accident. The reports have been prepared for both Units 1,4 and 5-8 as required by this clause but they have not been referenced in the containment design documentation. This issue is a documentation issue only and not safety significant. Clause 11.1 in updated N290.3-11 requires adequate shielding provisions be present, but does not contain a requirement for the information to be referenced in the design documentation, therefore this is not considered a PSR2 gap.

5. Clause 3.13.1 requires that the application for a construction approval identify any aspects of the design that fail to comply with applicable codes and standards in: CSA N285 and N287. Any applicable codes and standards relevant to containment design are assessed in PSR2.

6. Clause 3.13.2 requires a list of codes and standards to be applied to the containment system be prepared and have approval by the CNSC prior to construction approval. This clause is not included in N290.3 but has been incorporated in the clauses under 4.23 of N290.0. Codes and standards applicable to Containment are referenced in the design documentation; however, there is no longer an explicit requirement for regulatory approval in the standard. Therefore, this is not considered a PSR2 gap.

Pickering Units 1,4

As part of Pickering A Return to Service (PARTS), code reviews were conducted. N290.3 did not exist at that time. The main submission for PARTS [B.11-8] committed to perform a review of Regulatory Document R-7. This review was completed in AECL Report, "Review of Pickering A Design Against Current Codes and Standards" [B.11-9].

The review identified the following gap:

Clause 3.8.4 requires that if containment sub-systems are considered to be independent for the purpose of safety analysis, principles for separation and independence of each sub-system shall be prepared and approved by the regulator. This requirement does not exist in N290.3-11 but it has been included in N290.0, Clause 4.6.2 [B.11-7]. However, regulatory approval is not required in CSA N290.0. Therefore, there is no gap relating to this issue for PSR2.

The review also identified the following five ADs:

1. Clause 3.6.2 requires that dynamic effects or jet forces caused by the event cannot result in impairment of the containment system.

   This clause is no longer contained in N290.3 and an equivalent clause is contained in N290.0, which has been reviewed separately for PSR2.

2. Clause 3.7.1 requires that containment system unavailability be less than $1x10^{-3}$ yrs/yr and that systems supporting containment operation be commensurate with this requirement.

   At the time of the code review the predicted unavailability was greater than this target. The latest available Pickering Units 1,4 Annual Reliability Report [B.11-10] documents that the predicted future unavailability of the containment system is less than $0.74 x10^{-3}$ yrs/yr. In addition many support systems are included in the list of systems important to safety, e.g., Class III power, and are shown to meet their unavailability targets as well. Therefore, this AD is no longer applicable and compliance with this clause can now be demonstrated. Therefore this is not a PSR2 gap.

3. Clause 3.10.3 requires that the design of the plant be such that non-essential sources of compressed air be isolated.

   Installation of the rupture panel system on Units 1,4 has resulted in a reduction of non-accident unit compressed air in-leakage. With this modification the Safety Report demonstrates that the post-accident containment pressurization times exceed 48 hrs, which meets requirements set out by the CNSC.

   For Pickering Units 5-8, the Reactor Building (RB) bulkheads would be isolated post-accident to minimize compressed air in-leakage into containment after an accident. As a result the Safety Report demonstrates that the post-accident containment pressurization times exceed 48 hours for DBAs in Units 5-8 as well.

   Therefore this is not a PSR2 gap.

4. Clause 5.1.1 requires that negative pressure proof pressure tests be performed.

   This requirement is not met because the containment envelope was not tested to the lowest predicted negative pressure following an accident. However, the negative design pressure is limited due to the multi-unit design of Pickering Containment and the code review demonstrates that leakage is not expected to be significant at these pressures. This rationale remains valid for PSR2 and the issue is not a gap.

5. Clause 2 in Appendix A requires containment extensions meet ASME code Class 2 requirements.

   R-7 was not in place during the design of Pickering and the design of containment penetrations/extensions did not meet Class 2 requirements. The code review provides the justification for this AD being acceptable, as meeting this requirement was not practicable, and the Pickering Units 1,4 design has been demonstrated to be reliable over 40 years of testing and operation. This rationale remains valid for PSR2 and the issue is not a PSR2 gap.

Darlington NGS

The original code review of the Darlington ISR, as documented in OPG Report NK38-REP-03680-10002 R000 [B.11-11], was performed against R-7 [B.11-3]. It was found that the Darlington design of the containment system meets the requirements of regulatory document R-7, with one exception relating to locking closed single isolation valves. CSA N290.3 only contains requirements relating to locking closed containment boundary valves in Annex B, which is applicable to new build applications only. This is addressed under the requirements for new plants below.

Following the issuance of CSA N290.3-11 [B.11-1] in October 2011, a Code Refresh review was conducted and documented in OPG Report NK38-REP-03680-10213 R000 [B.11-12]. A clause-by-clause review was done given this was the first edition of the standard and due to the differences between R-7 and N290.3. For example, all of the requirements common to all special safety systems that were contained in R-7, are now contained in CSA N290.0-11, "General Requirements for the Safety Systems of Nuclear Power Plants" [B.11-7].

The Darlington Code Refresh review of N290.3-11 documented in NK38-REP-03680-10213 [B.11-12] resulted in 27 gaps. Of these gaps, eight are related to mitigating the impacts of Beyond Design Basis Accidents (BDBAs) and Severe Accidents, which are new requirements not contained in R-7, eight of the gaps apply to new plant design while the remaining eleven gaps cover various topics.

The Darlington BDBA gaps relate to the assurance of containment integrity, radionuclide management system (RMS) and energy management system (EMS). Because of differences between the Darlington and Pickering unit designs, containment designs and BDBA progression, Pickering is utilizing a different approach to address BDBA response (systems and mitigation strategies). The Pickering Filtered Air Discharge System (FADS) is used as the BDBA RMS while FADS in conjunction with Reactor Building Air Cooling Units (ACUs) provides the EMS function [B.11-13].

Completion of Phase 2 Emergency Mitigating Equipment (EME) implementation is required to provide further enhancement for BDBA response providing AC power for FADS and Reactor Building ACUs and their services. Since Phase 2 EME implementation is not complete, this is **PSR2 CSA N290.3-11 Gap #1** relating to enhancements to the containment RMS and EMS for BDBAs.

The CSA N290.3-11 new build clauses and gaps are addressed in a subsection further below, while the specific Darlington gaps that need to be considered for applicability to Pickering PSR2 are discussed immediately below:

1. Clause 6.4, Note (2) requires the selection of materials used inside containment shall consider coverings, coatings and cladding. The code review refers to a gap against clause 8.6.11 against previous regulatory document RD-337 [B.11-14]. This was classified as an AD for Darlington. Although there were no specifications for coatings in the original Darlington design basis, the impact of this issue has been addressed in safety analysis and in the design of the Emergency Coolant Injection System (ECIS) recovery strainer. No other safety issues were raised. This disposition also applies to Pickering. For example, the modified ECIS recovery strainer analysis considers the liberation of chemicals and other materials that could result in a pressure drop across the strainers that could affect pump operation (also refer to PSR2 review of N290.2-11). Therefore, this is not a PSR2 gap.

2. Clause 7.4 requires that leakage limits be defined for both gas and liquid phases. The gap exists since the Safety Report and Darlington Operational Safety Requirements (OSRs) have no separate leakage limits for liquids through containment penetrations. The Pickering Units 1,4 and Units 5-8 OSRs do not contain these limits either. However, per Darlington Issue D613 [B.11-15], this had a low safety significance and was determined to be an acceptable deviation. This was because Derived Release Limits have been established and are complied with for liquid releases during normal operation; bounding safety analysis is done for failures outside of containment and failures that result in some containment by-pass (e.g., Steam Generator consequential

leaks); and means are available to recover and mitigate waterborne leakage. Because of similarities in the safety analysis assumptions and methodology between Pickering and Darlington, the issue resolution is also applicable to Pickering. In addition, Pickering reactor building airlock solid seals on the 254' elevation provide an additional barrier to liquid release post-accident. Therefore, this is not a PSR2 gap.

3. Clause 7.6 requires that for BDBAs, the containment boundary leakage rate shall be maintained below the maximum allowable long enough to allow for implementation of off-site emergency procedures. For Pickering, FADS procedures for BDBA have been developed such that operation with containment above atmospheric pressure is possible. Allowing more containment pressurization time provides more time to implement off-site response. The assessments of accident scenarios as part of the Environmental Assessment for Pickering B Refurbishment Life Extension [B.11-16] confirmed there to be more than 24 hours available for off-site actions for the representative BDBA accident category (Ex-Plant Release Category 5 (EPRC5)). Therefore, this is not a PSR2 gap.

4. Clause 9.2.1.1 requires that all containment penetrations be designed to be leak tight. The design of Darlington penetrations meets the intent of this requirement as per Nuclear Safety Design Guide NK38-DG-03650.7 [B.11-18] Section 3.5 which identifies penetration seal requirements. The intent of the clause was met, but the explicit requirement to design a leak tight penetration was considered a gap.

   For Pickering, this is an implicit requirement, since the entire containment boundary, including penetrations is designed to be leak tight to the degree practicable. This is demonstrated in successful periodic reactor building and vacuum building leakage rate tests. The results of these tests demonstrate there is significant margin between the measured leakage rate and Safety Report assumptions. The trend of measured leakage rates demonstrates that margin will be maintained for operation beyond 2020. Therefore this is not a PSR2 gap.

5. Clause 9.5 on combustible gas management requires recombiners. Passive Autocatalytic Recombiners (PARs) were installed at Darlington meeting this clause. Since the issue of the code review, installation of PARs has been completed on all OPG units, therefore this gap no longer exists.

6. Clause 9.5.1 requires management of combustible gases be provided to control concentrations to preclude destructive combustion modes. Management of combustible gases is addressed in the Safety Report, but the code review identifies a gap related to RD-337 [B.11-14] clause 8.6.10. Since the issue of the code review, installation of PARs has been completed on all OPG units, therefore this gap no longer exists.

7. Clause 9.5.6 requires that in the case of a severe accident, emergency venting of containment shall consider the build-up of combustible gases to minimize injury to plant personnel and damage to SSCs from deflagration. Darlington addressed this in Issue D607 [B.11-17] citing implementation of Severe Accident Management Guidelines (SAMG) and plant modifications with minimizing the risk of hydrogen deflagration and addressing this gap.

For Pickering, design differences from Darlington require a different approach. Emergency venting with FADS and hydrogen mitigation are addressed in the SAMG. The applicable SAMG documents require that negative implications of any SAMG actions be assessed including unacceptable impacts on credited SSCs and personnel safety.

8. Clause 10.2.2 requires a list of containment conditions that need to be monitored for BDBAs be developed. Containment parameters to be monitored for a severe accident are developed in SAMG documentation. This was a gap for Darlington as an equivalent list had not been developed for BDBAs.

   This is not a gap for Pickering as the EME guides used for BDBAs contain a list of channelized indications powered by EME generators for Units 5-8. This list is in Appendix I of [B.11-19]. For Pickering Units 1,4 there is a list of EME indications in Table 2 of [B.11-20].

9. Clause 11.1.1 requires that the containment system include provisions for radiation shielding for all plant states defined in N290.0. For Pickering, this has been addressed in a BDBA habitability assessment [B.11-21].

10. Clause 12.1.2 requires the containment system design provide for isolation of all sources of compressed air and other non-condensable gases that are not needed for operation of the plant following an accident. This is a gap at Darlington since there is a design guide exception for the Powerhouse Service Air system.

    There was a similar gap for Pickering Units 1,4 against R-7 clause 3.10.3 [B.11-9]. Installation of the rupture panel system on Units 1,4 has resulted in a reduction of non-accident unit compressed air in-leakage. With this modification the Safety Report demonstrates that the post-accident containment pressurization times exceed 48 hours, which meets requirements set out by the CNSC.

    For Pickering Units 5-8, the RB bulkheads would be isolated post-accident to minimize compressed air in-leakage into containment after an accident. As a result the Safety Report demonstrates that the post-accident containment pressurization times exceed 48 hours for DBAs in Units 5-8. Therefore this gap is not applicable to PSR2.

11. Clause A.2 contains containment piping barrier requirements for systems connected to the containment atmosphere and provides acceptable configurations. A gap exists since there is an open design guide exception for the D_2O Leakage Collection System.

    This gap is specific to the design of a Darlington system and is not applicable to Pickering 1,4 or 5-8, and is therefore, not a PSR2 gap.

12. Clause A.5 (a) contains containment piping barrier requirements for crimping. The gap exists as there is an open design guide exception for two systems that do not have procedures in place to perform crimping of the required lines.

    This gap is specific to the design of Darlington systems and is not applicable to Pickering 1,4 or 5-8 and is therefore, not a PSR2 gap.

New Plant Requirements

In addition to the above, there were eight Darlington gaps identified against eight clauses relating to requirements for new plants. These are:

13. Clause 9.1.2 requires that the selection of the containment concrete material for new builds shall take into account the effects of severe accidents involving interaction of a melted reactor core and concrete. This was not addressed in the Darlington design and has not been addressed in the Pickering design. For Darlington, this was Gap 2286 and combined under Issue D607 [B.11-17] relating to BDBA and SAMG mitigation. The implementation of modifications in conjunction with improvements to SAMG were deemed to adequately compensate for the concrete used in the Darlington design. The same concept applies to Pickering, with a similar justification. Hydrogen is mitigated by the installation of PARs and improvements to SAMG for hydrogen management and mitigation. BDBA and SAMG strategies to stop accident progression in the In-Vessel Retention (IVR) state significantly reduce the risk of Core-Concrete Interaction (CCI). This clause has been acceptably addressed for Pickering and is not a PSR2 gap.

14. Clause 9.2.2.3 requires piping penetrations in new builds to be designed to be tested for leak tightness. In addition, Clause 9.2.3.2 requires each electrical and fibre optic cable penetration to be designed to be tested for leak tightness. Darlington had gaps relating to both clauses. Pickering did not have any gaps relating to R-7 Clause 5.2.4 which has the same requirement as Clause 9.2.2.3 of CSA N290.3. For Clause 9.2.3.2, Pickering does not use testable penetrations and instead relies on operational leak rate testing and Periodic Inspection Program (PIP) testing to confirm the leak tightness of containment. No fibre optic penetrations are used at Pickering. Darlington Gap 2306 was assigned to Clause 9.2.3.2 and was included in Issue D613 [B.11-15]. This gap was resolved as an acceptable deviation with low safety significance. The resolution notes that although the Darlington penetrations are equipped with testing capability, the benefit of the feature proved to be minimal. Similar to Pickering, Darlington relies on containment envelope leak rate testing. Given the above, although Pickering does not have individually testable electrical penetrations, this is acceptable based on the suitable alternate surveillance/testing being performed to assure the safety function is maintained. Therefore, this issue is not a PSR2 gap.

15. Clause 9.4.2.4 (b) requires that for BDBAs, if emergency venting is necessary to protect the structural integrity of containment, new plants shall be designed to minimize and to monitor the release of radionuclides. This was Gap 2290 noting that Darlington was not designed for BDBA venting and monitoring capability. A PSR2 Gap relating to enhancement of the power supply for the BDBA containment EMS and RMS has already been identified above. As a result of providing Phase 2 EME power to the FADS, the existing provisions for monitoring capability will be available. Hence, the ability to monitor releases is also addressed under the **PSR2 CSA N290.3-11 Gap #1** above.

16. Clauses 9.5.5 and 10.2.3 relate to provisions for sampling the containment atmosphere and monitoring the concentration of hydrogen. For containments that use inert atmosphere for combustible gas control, provisions for monitoring oxygen concentration shall be provided. These clauses were gaps for Darlington which identified that SAMG

would address measurement of hydrogen in Gaps 2291 and 2294. These have been combined in Issue D607 [B.11-17] relating to BDBA and SAMG mitigation, which notes that modifications are being made to reduce the potential for hydrogen production. Given that SAMG has been enhanced to provide improvements to the determination of hydrogen concentration and its mitigation, the requirement for hydrogen measurement instrumentation was classified as an Acceptable Deviation for Darlington. Pickering has installed PARs and implemented SAMG improvements to enhance hydrogen monitoring. Therefore, the absence of hydrogen monitoring instrumentation is acceptable, given there are alternate means of estimating hydrogen concentration through SAMG.

17. Clause B.3.1 requires pipes that connect to the reactor coolant system and penetrate the containment structure be provided with two isolation barriers. This is a similar requirement to Clause A.3.1 but also applies to pipes less than 25 mm for new plants. This was Gap 2307 and included in Issue D613 [B.11-15] for Darlington. This gap was classified as an Acceptable Deviation and the issue had low safety significance. There are some circumstances where it is possible to crimp small lines as a secondary containment barrier for Pickering Units 1,4 [B.11-22]. However, the rationale and classification of this issue as an Acceptable Deviation is also fully applicable to Pickering. Therefore, this is not a PSR2 gap.

18. Clause B.3.3 requires that for a pipe connected to the reactor coolant system with barriers open for 1 hour per year or more, both barriers shall be automatic and fail closed. The requirement is a change from the existing plant requirements in A.3.2 and A.3.3 where only the barrier inside containment was required to fail closed or to have a power operator. A similar requirement to A.3.2 and A.3.3 exists in the Design Guides for Darlington. However, Pickering 5-8 and Pickering 1,4 do not have reactor coolant systems (i.e., Heat Transport System, Shutdown Cooling, Moderator) that penetrate containment, therefore, this Darlington gap is not a PSR2 Gap for Pickering.

The above review for containment requirements for new plants has not identified any incremental safety significant gaps that are applicable to PSR2.

B.11.2.2 Application of Post PSR1 Reviews

The CSA N290.3-11 Impact Statement [B.11-23] provides a summary of significant features of the new standard. Containment boundary requirements are based on the requirements of CNSC R-7, therefore the compliance discussion for R-7 is still applicable. Of the features described, the only one considered to be safety significant for the purposes of PSR2 is that the standard includes requirements for the containment system for BDBAs and severe accidents.

A review of Darlington against N290.3-11 was discussed in Section B.11.2.1. For the clauses that contain requirements for BDBAs and severe accidents and are gaps for Darlington, the gap was assessed for applicability to Pickering. In addition, a review was completed to review the new clauses for BDBAs and severe accidents where Darlington was in compliance, to ensure the same is the case for Pickering. The following clauses fall into this category and the disposition for Pickering is provided:

- Clause 4.1(b) states [B.11-1]:

  The containment system shall consider BDBAs, including severe accident conditions.

  This is a general requirement which is addressed in more detail in subsequent clauses which have been addressed above for Pickering.

- Clause 5.5 states [B.11-1]:

  For each plant (existing and new build), the scope of DBAs and BDBAs shall be as agreed upon by the authority having jurisdiction (AHJ) and the licensee.

  This has been addressed by completing Fukushima Action Items and completing analyses for BDBAs and severe accidents.

- Clause 10.2.1 states [B.11-1]:

  The monitoring of containment conditions under BDBAs shall consider the guidelines in Annex A of CSA N290.6.

  Annex A is not a mandatory part of CSA N290.6 and therefore, outside the scope of PSR2.

Therefore, none of these clauses result in gaps for PSR2.

In addition, the N290.3 Impact Statement identifies that it captures existing Canadian industry and international standards and practices, including CNSC document R-7, which it replaces. Therefore, it is expected that Pickering will largely comply with N290.3-11. However, since this is the first version of the standard, this was confirmed by reviewing N290.3-11 to identify any significant issues that could impact Pickering's general compliance.

At a high level, the following demonstrates that the Pickering Units 1,4 and 5-8 Containment systems meet applicable nuclear safety objectives and licensing requirements:

- Performance requirements to ensure credits in the Safety Report are satisfied are contained in the OSRs for Pickering Units 1,4 and 5-8 [B.11-24] and [B.11-25].

- The Probabilistic Safety Assessment has demonstrated that public safety goals have been met with the design of the Pickering Containment system.

- Per Regulatory Document RD/GD-98 [B.11-26], unavailability analyses are completed that demonstrate that reliability requirements are met for the Containment systems.

- Environmental and seismic qualification requirements are in place to ensure the required performance following applicable design basis accidents.

For additional assurance, the requirements of N290.3-11 were reviewed to determine if there were any new requirements, or specific differences in the Darlington design compared to Pickering that could challenge Pickering's compliance with the new standard. Based on the review performed, there were no safety significant clauses that could impact on PSR2 other than those identified in B.11.2.1 above.

B.11.3 Compliance Summary for Pickering PSR2

There is one PSR2 CSA N290.3-11 gap which relates to Safety Factor 1 (Plant Design):

1. Per CSA N290.3-11, a Containment Energy Management System (EMS) and Radionuclide Management System (RMS) are required to protect containment and minimize radiological releases for Beyond Design Basis Accidents (BDBAs). The Pickering EMS and RMS use the Filtered Air Discharge System (FADS) and Reactor Building Air Cooling Units (ACUs). Enhancements to the AC power supplies to these systems and related loads are being provided by Phase 2 Emergency Mitigating Equipment (EME), which is not yet fully implemented. This PSR2 gap has been identified to track the implementation of Phase 2 EME such that it can be used to support the EMS and RMS.

B.11.4 References

[B.11-1] CSA Standard, N290.3-11, *Requirements for the Containment System of Nuclear Power Plants*, October 2011.

[B.11-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.11-3] AECB Regulatory Document, R-7, *Requirements for Containment Systems for CANDU Nuclear Power Plants,* February 1991.

[B.11-4] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.11-5] OPG Report, NK30-REP-03680-00001 R000, *Pickering NGS B –Integrated Safety Review - Plant Design Safety Factor*, August 2007.

[B.11-6] CNSC Letter, e-Docs # 3297754, OPG File No. NK30-CORR-00531-05008, T.E. Schaubel to D.P. McNeil, *Pickering NGS-B – Discrepancy Resolution for Ageing, Safety Analysis, Emergency Planning, Environment, Plant Design and Management Safety Areas*, November 3, 2008.

[B.11-7] CSA Standard, N290.0-11, *General Requirements for the Safety Systems of Nuclear Power Plants*, October 2011.

[B.11-8] OPG Letter, NA44-CORR-00531-00381, R.J. Strickert to J.S.C. Tong, *Pickering A Updated Basis for Return to Service Document*, April 20, 2001.

[B.11-9] AECL Assessment Document, 44RS-00531-ASD-001 Rev. 04, *Review of Pickering A Design Against Current Codes and Standards*, November 2000.

[B.11-10] OPG Report, *2014 Annual Reliability Report - Pickering Units 1 & 4*, NA44-REP-09051.1-00014, March 10, 2015.

[B.11-11] OPG Report, NK38-REP-03680-10002 R000, *Review of CNSC R-7 (February 1991) Requirements for Containment Systems for CANDU Nuclear Power Plants for Darlington Integrated Safety Review*, August 2011.

[B.11-12] OPG Report, NK38-REP-03680-10213 R000, *Code Refresh Review of CSA-N290.3-11 Requirements for the Containment Systems of Nuclear Power Plants*, February 2014.

[B.11-13] OPG Report, P-REP-09013-00002 R001, *Pickering NGS – Beyond Design Basis Containment Integrity*, January 27, 2014.

[B.11-14] CNSC Regulatory Document, RD-337, *Design of New Nuclear Power Plants*, November 2008.

[B.11-15] OPG Report, NK38-REP-00770-0489592 R000, Issue D613, *Containment Boundary Leakage,* February 11, 2014.

[B.11-16] OPG Report, NK38-REP-07701-00014, *Credible Malfunctions and Accident Scenarios, Technical Support Document, Refurbishment and Continued Operation of Pickering B Nuclear Generating Station Environmental Assessment,* December 2007.

[B.11-17] OPG Report, NK38-REP-00770-0488762 R000, Issue D607, *Severe Accident and Beyond Design Basis Accident (BDBA) Design/SAMG*, February 11, 2014.

[B.11-18] OPG Guide, NK38-DG-03650.7 R005, *Nuclear Safety Design Guide: Extensions of the Containment Envelope*, September 2012.

[B.11-19] OPG Guide, NK30-EME-09013-OPS-00003 R007, *Sustained Total Loss of Class IV, III and EPS Power*, December 2015

[B.11-20] OPG Guide, NA44-EME-09013-OPS-00003 R011, *Sustained Total Loss of Class IV, III Power*, May 2016.

[B.11-21] OPG Memorandum, N-CORR-03611-0454183 T10, *Pickering Habitability Study – Summary of Results,* January 24, 2014.

[B.11-22] OPG Report, NA44-REP-34280-00001 R000, *Pickering NGS A Containment Penetration List*, August 30, 2002.

[B.11-23] CSA Impact Statement, *Notification of CSA N290.3 Publication; Product Designation: CSA N290.3-11*, Date not provided.

[B.11-24] OPG Operational Safety Requirements, NA44-OSR-08131.02-00002 R003, *Pickering 1-4 Operational Safety Requirements: Negative Pressure Containment*, March 2015.

[B.11-25] OPG Operational Safety Requirements, NK30-OSR-08131.02-00003 R004, *Pickering 5-8 Operational Safety Requirements: Negative Pressure Containment*, March 2015.

[B.11-26] CNSC Regulatory Document, RD/GD-98, *Reliability Programs for Nuclear Power Plants*, June 2012.

B.12 CSA N290.4-11, "Requirements for Reactor Control Systems of Nuclear Power Plants"

B.12.1 Background

The following, paraphrased from the preface and scope of CSA N290.4 [B.12-1], provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *The purpose of CSA N290.4 is to specify provisions for safe and effective control of reactor power and pertains to all components of the system, including mechanical, process, software, electrical, and instrumentation and control design used for the control of the neutron flux and the thermal output of the reactor.*
>
> *CSA N290.4 establishes the minimum requirements for the design, manufacture and fabrication, qualification, and installation of reactor control systems in nuclear power plants, in order to ensure that they will operate as intended.*

All of N290.4-11 is directly relevant to Safety Factor 1 (Plant Design).

CSA N290.4-11 is the second edition of CSA N290.4. It supersedes the previous edition: N290.4-M82 published in January 1982 under the title "Requirements for the Reactor Regulating Systems of CANDU Nuclear Power Plants" [B.12-2].

Compliance with N290.4 is not currently a licence requirement for Pickering NGS (in accordance with PROL 48.02/2018) per the R04 Pickering Licence Condition Handbook [B.12-3].

The CSA Impact Statement notification for CSA N290.4-11 [B.12-4] provides a "Summary of Significant Changes from the Previous Edition" which identifies six changes to the Standard which are discussed in Section B.12.2 below.

The results of PSR1 N290.4 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.12.2. As identified in Reference [B.12-5], the Pickering PSR2 review of CSA N290.4-11 is an Incremental Review. PSR2 Incremental Review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.
- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.12.2 Compliance Assessment for Pickering PSR2

B.12.2.1 Application of PSR1 Reviews

The versions of CSA N290.4 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

A clause-by clause review of the Pickering Units 5-8 Reactor Regulating System (RRS) against the CSA N290.4-M82 [B.12-2] version of the Standard was documented in NK30-REP-03680-00001 R000 [B.12-6].

This review concluded that with the exception of two Acceptable Deviations (ADs), the Reactor Regulating System was in direct compliance with the design requirements stated in CSA-N290.4-82. The two ADs were as follows:

1. Clause 4.3.11 pertains to Reactor Power Start-up. This clause requires that if other special equipment is used for reactor start-up after long periods of shutdown, then that equipment may also have requirements imposed on it by N290.1. Start-up Instrumentation (SUI) is used in these circumstances at both Pickering Units 1,4 and 5-8 for this purpose. SUI may not fully conform with N290.1 such as the required areas of separation. However, the code review states that SUI is used under strict procedural controls and is used very infrequently and for limited periods of time. Therefore, the areas in which SUI may not comply with the requirements of N290.1 are not safety significant. This applies to both Pickering Units 1,4 and 5-8 SUI. In addition, related clauses 5.7 and 5.10.1 in N290.4-11 do not contain the requirement for SUI to meet N290.1 requirements. Therefore, this is not safety significant and not relevant for PSR2.

2. Clause 6.2.1 requires that a specific set of functional and performance requirements (contained in sub-clauses (a) though (j)) be documented in the RRS design manuals. Reference [B.12-6] states that not all of these aspects have been fully documented, but none of the omissions relate directly to the system design. Also, there is no corresponding requirement in the updated N290.4-11 version. This is not safety significant and not a PSR2 gap.

Pickering Units 1,4

As part of Pickering A Return to Service (PARTS), code reviews were conducted. The main submission for PARTS [B.12-7] committed to perform a review of N290.4-82 [B.12-2], which was the active version at that time. This review was completed in AECL Report, "Review of Pickering A Design Against Current Codes and Standards" [B.12-8]. This report concluded that, overall the design of Pickering Units 1,4 Reactor Regulating System complies with the design principles and requirements of CSA Standard N290.4-M82. The review identified four Acceptable Deviations in the following areas:

1. Clause 4.1 requires the reactor regulating system be physically and functionally separated from the special safety systems. (This is related to clause 5.2 in N290.4-11). Special safety systems and the regulating system are mostly independent of the major control loops, but there are three exceptions identified in the code review:

   - Sharing of taps and impulse lines from flow orifices in reactor flow measurement loops between RRS and Shutdown System A (SDSA). The code review states that irrational inputs to RRS control programs are rejected and SDSA channel flows are continually monitored in the control room to ensure they remain in the normal operating range. Therefore the impact of this exception is acceptable and not a PSR2 gap.

   - There are a number of RRS hardware and program interlocks related to the SDS system, e.g., gradually filling all zones when the reactor is tripped. These interlocks do not affect SDS functionality and are standard practice in CANDU design. Also, these interlocks drive RRS reactivity devices in the safe direction. Therefore the impact of this exception is acceptable and not a PSR2 gap.

   - In the control of moderator level, RRS and Emergency Coolant Injection (ECI) share some components, i.e., the common helium bubbler supply in the calandria and dump tank level measurements. The loss of the bubblers does not result in a loss of regulation, therefore there will be no postulated demand on either the in-core logic or ECI injection from this event and, therefore, sharing the bubblers is acceptable and not a PSR2 gap.

2. An AD is identified with clause 4.3.1.2 which requires that all components of the system (including power sources) shall be included in the reliability calculations. This clause is superseded by N290.4-11 clauses 5.4.2 and 5.4.3, which require the design target reliability of RRS be established in the Probabilistic Safety Assessment. The Pickering Units 1,4 Probabilistic Safety Assessment demonstrates that the RRS reliability including any supporting systems is acceptable. Therefore, this is not a PSR2 gap.

3. Clause 4.3.6.1 requires RRS to control neutron flux to obtain an acceptable spatial distribution and be capable of counteracting flux distortions that may otherwise lead to violation of fuel bundle or channel power limits. (This is related to N290.4-11 clause 4.1.2 (c)). Pickering Units 1,4 are not equipped with automatic reactor power setback or an alarm based on high flux tilt. Flux tilts are managed by operator monitoring and by the initiation of a power reduction as specified in operating manuals. This deviation has been accepted by the CNSC [B.12-9]. Therefore, this is not a PSR2 gap.

4. Clause 4.3.25.4 requires that where a number of channels of one system are in proximity, colour coding is the preferred identification method. This was not implemented into the Pickering Units 1,4 RRS. This requirement is referred to in clause 5.11.4 in N290.4-11 as one of several methods of facilitating human-system interface in the design and, therefore, is still relevant. However, the design of the RRS control room panels are clearly organized and labeled, such that colour coding is not required. Therefore, this is not a PSR2 gap.

Darlington NGS

The original code review for the Darlington ISR, as documented in OPG Report NK38-REP-03680-10012 R000 [B.12-10], was performed against CSA N290.4-82 [B.12-2]. In general it was found that the Darlington design of the Regulating Systems meets the requirements of N290.4-M82. NK38-REP-03680-10012 does identify the following gap[12]:

(i) Start-up Instrumentation (SUI) - SUI is a part of both RRS and Shutdown Systems and SUI meets all requirements for RRS with respect to permanently installed equipment. However, for SUI in terms of SDS function, it may not meet all requirements of CSA N290.1 (e.g., separation environmental and seismic), which N290.4-M82 explicitly references (clause 4.3.11).

The code reviews for both Pickering Units 5-8 and Units 1,4 reviewed above addressed this issue. For Pickering Units 5-8 there was an acceptable deviation against this clause. For Pickering Units 1,4 it was stated that there was direct compliance. Given the similarity of the SUI used in Units 5-8 and 1,4, it is judged that there is an acceptable deviation for both stations against this clause. As per the compliance assessment for Pickering Units 5-8 on this clause, the new version of N290.4 does not contain the same clause, therefore there is no PSR2 gap associated with this issue.

Following the issuance of CSA N290.4-11 [B.12-1], a Code Refresh review was conducted and documented in OPG Report NK38-REP-03680-10156 R000 [B.12-11].

The clause-by-clause review in the report does not discuss the differences between CSA N290.4-11 [B.12-1] relative to CSA N290.4-M82 [B.12-2] because the new standard is a major, "technology neutral" revision that is applicable also for light water reactors.

The review confirmed that OPG Nuclear and DNGS governance are in compliance with the requirements in CSA N290.4-11, except for the following requirement. Clauses relating to Anticipated Operational Occurrences (AOOs) (e.g., Clause 4.2) require the capability of the Reactor Regulating System be assessed to deal with AOOs, by preventing them from escalating into Design Basis Accidents (DBAs) that would require Shutdown System action.

In general the Setback function (and Stepback in Pickering Units 5-8) addresses these requirements; however, AOOs have not been identified and analyzed in the current Pickering Safety Reports. This is therefore identified as **PSR2 CSA N290.4 Gap #1.** There are also additional clauses which refer to requirements of RRS during AOOs, specifically: Clause 5.19 RRS and core stability; Clause 5.6.2 reactivity depth and rate of change; and, Clause 5.16.1 environmental qualification. These clauses are also captured under Gap #1.

Clause 5.19 of CSA N290.4-11 is specific to new plant design and relates to RRS control stability. For Darlington, Gap 1702 was identified against this clause, specifically relating to RRS requirements for AOOs. This gap was included in Issue D332 [B.12-12] and the resolution in the Integrated Implementation Plan (IIP) [B.12-13] is to address this issue as part of IIP-OI-

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12 Another gap was originally identified for clause 6.2.2 of CSA N290.4- M82; however, it was re-classified as compliant in Appendix B of [B.12-10].

043 (or AI 2014-OPG-5461) relating to CNSC REGDOC 2.4.1 compliance. This issue is also a gap for PSR2; however, it is included in **PSR2 CSA N290.4 Gap #1** above. Issues related to AOOs are being addressed as part of REGDOC-2.4.1 implementation and the current REGDOC-2.4.1 implementation plan is documented in N-PLAN-03500-0500515 R03 [B.12-14].

In terms of the more general issue of core stability, given the many years of operation for Pickering 5-8 and 1,4, there is operating experience to demonstrate that reactor control is stable and robust for normal operation. Reference [B.12-15] documents a core stability assessment for Pickering 1,4 concluding that spatial control is adequate. Therefore, the issue of core stability for normal operation is not a PSR2 gap.

B.12.2.2 Application of Post PSR1 Reviews

CSA N290.4 has been updated since the last code reviews were performed for Pickering. It was updated to N290.4-11 [B.12-1] in 2011 from the original version N290.4-M82 [B.12-2]. A review of changes to N290.4 and Pickering's compliance with them are discussed in this section.

Given the extent of the changes in N290.4-11 [B.12-1] and lack of specificity in the Impact Statement [B.12-4], a review of changed clauses was performed. There were a number of changes made in the update of the code, with some resulting in intent changes. However, in the majority of cases, though wording was revised compared to N290.4-M82, there were no additional significant intent changes. The code was re-organized, the requirements made less specific and modified to be technology neutral. Detailed requirements were removed in cases where reference to another standard can be made, e.g., N290.13 on Environmental Qualification.

Of the six changes listed in the CSA Impact Statement for the issue of N290.4-11 [B.12-4], only two have the potential to have an impact on nuclear safety.

These are:

a) It has been aligned with regulatory document RD-337 [B.12-16]; and

b) Additional requirements have been added in the following areas: Human-System Interface, Cyber Security, Chemistry, Software Design, Aging Management, Reactivity Management.

Alignment with RD-337

Regulatory Document RD-337, "Design of New Nuclear Power Plants" [B.12-16][13] introduces new requirements for the assessment of AOOs. N290.4-11 contains various clauses referring to RRS requirements in the event of AOOs. This has already been addressed in **PSR2 CSA N290.4 Gap #1**.

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[13] Now superseded by CNSC REGDOC 2.5.2 "Design of Reactor Facilities: Nuclear Power Plants" [B.12-17].

Human-System Interface

Clause 5.11 introduces requirements for the human factors engineering design process, including the development of a human factors plan and usability requirements.

Currently Human Factors Engineering (HFE) is considered for all design changes. The changes are managed through the Engineering Change Control Program [B.12-18] and plant modifications follow the Modification Process [B.12-19]. The Design Scoping Checklist [B.12-20], which is a component of the modification process, identifies the level of HFE required. If the modification is judged to have an HFE impact, then the Human Factors Level of Activity [B.12-21] is completed to determine the scope of the HFE. A Human Factors Engineering Specialist must concur with the Human Factors Level of Activity. The modification may require the preparation of a Human Factors Engineering Program Plan or the Human Factor Worksheet [B.12-22]. These instructions and processes ensure that the human-system interface elements for the modification are addressed. The technical, design and operator reviews during and following the design process and via the Operations Turnover process ensure the usability requirements will be achieved.

The Human Factors Engineering Program Plan prepared by OPG meets the requirements of CNSC G-276 [B.12-23] and CNSC G-278 [B.12-24] and applicable elements from NUREG-0711 [B.12-25]. The results are independently verified. Therefore, the intent of these clauses is met.

Cyber Security

Clause 5.18 identifies the consideration for cyber security. OPG has a program in place to ensure cyber security requirements are in place as per OPG Procedure, N-PROC-MP-0103, "Security for Real-Time Process Computing Systems" [B.12-26]. This clause is met.

Chemistry

Clause 5.20 contains the following requirement:

> *The design of the reactor control system shall facilitate the monitoring and control of isotopic values and soluble neutron absorbers used for reactor power control such that reactivity worth is known.*

The concentrations of the relevant chemicals in the moderator to control reactivity are measured routinely and controlled [B.12-27], [B.12-28]. Therefore, there is no PSR2 gap associated with this new clause.

Software Design

Clause 5.17 identifies requirements for software design. OPG has a comprehensive program for managing software development, including real-time RRS software [B.12-29].

Aging Management

Four new clauses have been added in Section 5.16.4 dealing with equipment Aging Management. The current regulatory requirements for an Aging Management program are contained in CNSC REGDOC-2.6.3, "Aging Management" [B.12-30]. A code review was performed of this regulatory document in Report P-REP-03680-00004 [B.12-31]. The review concludes OPG is in compliance with REGDOC-2.6.3 except in two areas. Neither of these areas affect compliance with the N290.4 clause 5.16.4. Pickering is compliant with this clause.

<u>Reactivity Management</u>

The standard identifies the following relating to reactivity management:

> **Note:** *For the purposes of this Standard, "reactivity management" refers to*
> *(a) the safe, controlled, and conservative performance of plant operation and maintenance activities that affect reactivity;*
> *(b) the active and vigilant monitoring of reactivity changes;*
> *(c) the quick and reliable detection of deviations from expected results; and*
> *(d) limiting rate changes and maximum power changes in response to failures.*

Clause 5.1 relates to incorporation of operating experience from reactivity management events into control system design. Clause 5.13.1 relates to avoidance of reactivity control events due to maintenance activities. There is an OPG nuclear standard specific to reactivity management [B.12-32] that deals with Operations and Maintenance requirements for reactivity management. In addition reference [B.12-32] addresses reactivity management performance monitoring and reporting. The Pickering reactivity control system designs are mature; however, any changes relating to reactivity control mechanisms require operating experience to be considered and dissemination of industry reactivity management operating experience to stakeholders (e.g., control room operators). This is addressed in reference [B.12-32].

B.12.3 Compliance Summary for Pickering PSR2

There is one PSR2 CSA N290.4-11 gap which relates to Safety Factor 1 (Plant Design):

1. Clause 4.2 and Clause 5.19 of CSA N290.4-11 require the capability of the Reactor Regulating System (RRS) to be assessed to deal with Anticipated Operational Occurrences (AOOs), by preventing them from escalating into Design Basis Accidents (DBAs) that would require Shutdown System action. In general, the setback function (and stepback in Pickering Units 5-8) addresses this requirement; however, AOOs have not been identified and analyzed in the current Pickering Safety Reports. Therefore, this has been identified as a PSR2 gap. It is being addressed as part of REGDOC-2.4.1 implementation. Note: There are also additional clauses which refer to requirements of RRS during AOOs (Clauses 5.6.2, 5.19, 5.16.1); however, for convenience, all issues related to AOO requirements for RRS in N290.4-11 are captured under this one PSR2 gap.

B.12.4 References

[B.12-1] CSA Standard, N290.4-11, *Requirements for Reactor Control Systems of Nuclear Power Plants*, October 2011.

[B.12-2] CSA Standard, N290.4-M82, *Requirements for the Reactor Regulating Systems of CANDU Nuclear Power Plants*, 1982.

[B.12-3] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.12-4] CSA Impact Statement, *Notification of CSA N290.4 Publication; Product Designation: CSA N290.4-11*, Date not provided.

[B.12-5] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.12-6] OPG Report, NK30-REP-03680-00001 R000, *Pickering NGS B –Integrated Safety Review - Plant Design Safety Factor*, August 2007.

[B.12-7] OPG Letter, NA44-CORR-00531-00381, R.J. Strickert to J.S.C. Tong, *Pickering A Updated Basis for Return to Service Document*, April 20, 2001.

[B.12-8] AECL Assessment Document, 44RS-00531-ASD-001 Rev. 04, *Review of Pickering A Design Against Current Codes and Standards*, November 2000.

[B.12-9] OPG Letter, NA44-CORR-00531-{131444}, R.O. Schuelke to G.R. Schwarz, *Pickering NGS-A: 100% Full Power Operation with Adjustor Rods AA5 and AA14 Out of Core, Action #90-4-13,* December 1991.

[B.12-10] OPG Report, NK38-REP-03680-10012 R000, *Review Of CAN/CSA-N290.4-M82 (R2001) (January 1982), Requirements For Reactor Regulating Systems of CANDU Nuclear Power Plants for Darlington Integrated Safety Review*, August 2011.

[B.12-11] OPG Report, NK38-REP-03680-10156 R000, *Code Refresh Review of CSA-N290.4-11, Requirements for Reactor Regulating Systems of CANDU Nuclear Power Plants, for DNGS ISR*, July 2013.

[B.12-12] OPG Report, NK38-REP-00770-0463391 R000, Issue D332, *Reactor Control System Requirements for AOOs,* April 5, 2016.

[B.12-13] OPG Report, NK38-REP-03680-10185 R002, *Darlington NGS Integrated Implementation Plan*, April 30, 2015.

[B.12-14] OPG Plan, N-PLAN-03500-0500515 R003, *REGDOC-2.4.1 Implementation Plan*, May 2015.

[B.12-15] OPG Letter, NA44-CORR-00531-05262 R000, *Pickering NGS-A: Core Stability, Action Item 2004-4-01,* June 30, 2006.

[B.12-16] CNSC Regulatory Document, RD-337, *Design of New Nuclear Power Plants*, November 2008.

[B.12-17] CNSC Regulatory Document, REGDOC 2.5.2, *Design of Reactor Facilities: Nuclear Power Plants*, May 2014.

[B.12-18] OPG Program, N-PROG-MP-0001 R014, *Engineering Change Control*, January 2015.

[B.12-19] OPG Procedure, N-PROC-MP-0090 R012, *Modification Process*, April 2015.

[B.12-20] OPG Form, N-FORM-10959 R016, *Design Scoping Checklist*, June 2016.

[B.12-21] OPG Form, N-FORM-10580 R006, *Identifying Human Factors Level of Activity*, December 2015.

[B.12-22] OPG Form, N-FORM-10221 R008, *Human Factors Worksheet*, December 2015.

[B.12-23] CNSC Regulatory Guide, CNSC G-276, *Human Factors Engineering Program Plans*, June 2003.

[B.12-24] CNSC Regulatory Guide, CNSC G-278, *Human Factors Verification and Validation Plans*, June 2003.

[B.12-25] U.S. NRC Document, NUREG-0711 Revision 3, *Human Factors Engineering Program Review Model*, November 2012.

[B.12-26] OPG Procedure, N-PROC-MP-0103 R003, *Security for Real-Time Process Computing Systems*, March 2015.

[B.12-27] OPG Chemistry Control Procedure, NK30-CCP-32000-0001 R010, *Moderator and Auxiliaries Units 014*, June 2014.

[B.12-28] OPG Chemistry Control Procedure, NA44-CCP-32000-0001 R015, *Moderator and Auxiliaries Units 058*, April 2015.

[B.12-29] OPG Program, N-PROG-MP-0006 R009, *Software*, April 2015.

[B.12-30] CNSC REGDOC-2.6.3, *Aging Management*, March 2014.

[B.12-31] OPG Report, P-REP-03680-00004 R000, *Pickering NGS Periodic Safety Review 2: Code and Standard Reviews for Safety Factors 2 (Actual Condition of SSCs), 3 (Equipment Qualification) and 4 (Aging)*, July 2016.

[B.12-32] OPG Standard, N-STD-OP-0009 R010, *Reactivity Management*, February 2016.

B.13 CSA N290.5-06, "Requirements for Electrical Power and Instrument Air Systems of CANDU Nuclear Power Plants"

B.13.1 Background

The following, paraphrased from the Preface and Scope of CSA N290.5-06 (R2011) including Update No. 1 [B.13-1] provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *CSA N290.5 specifies design, procurement, qualification, construction, installation, inspection, and documentation requirements to ensure that CANDU electrical power and instrument air systems meet nuclear safety requirements.*

All of N290.5-06 (R2011) including Update No. 1 is directly relevant to Safety Factor 1 (Plant Design).

CSA N290.5 is identified in Appendix E.1 of the R04 Pickering Licence Conditions Handbook [B.13-2] as "Guidance or Criteria". Section 6.1, "Design Program" of [B.13-2] states that "Recommendations and guidance are found in… N290.5, which covers electrical power and instrument air systems."

CSA N290.5-06 (R2011) including Update No. 1 [B.13-1] was issued in November 2011 and is the second edition of CSA N290.5. This edition including Update No. 1 supersedes the following versions of the standard:

(i) N290.5-M90, published in 1990 under the title "Requirements for the Support Power Systems of CANDU Nuclear Power Plants" [B.13-3], and

(ii) N290.5-06 published in December 2006 under the title "Requirements for Electrical Power and Instrument Air Systems of CANDU Nuclear Power Plants" [B.13-4].

The CSA Impact Statement notification for Update No. 1 to CSA N290.5-06 [B.13-5] provides a "Summary of Significant Changes from the Previous Edition" which identifies five primary changes to the Standard which are discussed in Section B.13.2 below.

The results of PSR1 N290.5 reviews (Pickering A Return to Service (PARTS) assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.13.2. As identified in Reference [B.13-6], the Pickering PSR2 review of CSA N290.5-06 including Update No. 1 is an Incremental Review. PSR2 Incremental Review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard (L/R/C/S) on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.

- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.13.2 Compliance Assessment for Pickering PSR2

B.13.2.1 Application of PSR1 Reviews

The versions of N290.5 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

For Pickering B ISR Safety Factor 1, a clause-by clause review of Pickering Units 5-8 against the CSA N290.5-M90 version of the Standard was documented in NK30-REP-03680-00001 R000 [B.13-7]. CSA N290.5-M90 was titled, "Requirements for the Support Power Systems of CANDU Nuclear Power Plants", which is different from the latest version of the Standard: "Requirements for Electrical Power and Instrument Air Systems of CANDU Nuclear Power Plants". Although the two titles are different, the scope of the two documents is restricted to the same systems, i.e. Electrical and Instrument Air (IA).

NK30-REP-03680-00001 R000 [B.13-7] concluded that the designs of the Pickering Units 5-8 Class I, II, III, IV, Site Electrical System, and Emergency Power Supply electrical systems were found to comply with the design requirements of the Standard. There was one Acceptable Deviation on Reference Publications referred to in the Standard, however, this has no impact on nuclear safety. This report did not address the Instrument Air system in the scope of the review since Instrument Air is not a Safe Operating Envelope (SOE) system or System Important to Safety (SIS) and the local air reservoirs were considered adequate to perform the safety function, per Appendix B of Reference [B.13-7].

Because backup IA has credited safety functions and the PSR2 Basis Document [B.13-6] contains a requirement to address specific components in non-SOE systems when they play a role in the plant safety basis, an assessment of Pickering Units 5-8 against the IA portions of CSA N290.5-06 (R2011), including Update No. 1 [B.13-1], has been provided in Section B.13.2.2 below. As discussed in Section B.13.2.2, the review demonstrated that the Pickering Units U5-8 IA system complies with the updated version of this code.

Pickering Units 1,4

As part of PARTS, code reviews relevant to N290.5 were conducted. The main submission for PARTS [B.13-8] committed to perform a review of N290.5-M90 [B.13-3]. This review was completed in AECL Report, "Review of Pickering A Design Against Current Codes and Standards" [B.13-9]. This report concluded for Electrical Systems that, overall the design of Pickering 'A' Electrical Distribution systems complies with the design principles and requirements of CAN/CSA N 290.5-M90, for Class I, II, III and IV power systems.

One Acceptable Deviation was identified against clause 5.4.4.2 regarding the use of a single rectifier for each Class I battery. This was considered acceptable since rectifier failures are annunciated and the associated battery has adequate capacity to allow time for operator action. This clause has not changed in the latest code version and states:

> *The number of rectifiers for each battery shall be determined by the reliability requirements for dc power.*

Per Regulatory Document RD-98, reliability targets are set for Systems Important to Safety (SIS). The Class I power system is in the list of SIS and therefore it is a monitored system required to meet its reliability target. Since the reliability of the system has been demonstrated to be acceptable with its current design, the Acceptable Derivation remains applicable and there is no PSR2 gap against this clause.

Two instances were identified where non-compliances with the code existed. These dealt with (i) DC contactor ratings for the Motor Generator (MG) Sets and (ii) On-line testing of distribution systems. In both cases, replacement of the MG sets prior to return to service addressed these non-compliances.

A compliance review of the Instrument Air System against N290.5-M90 was not completed as part of this PARTS code review. As discussed under Section B.13.2.2 below, significant work was performed on the IA system during the Pickering A Return to Service and therefore a compliance review was judged as not required at that time. Subsequent to this, the CNSC requested that Pickering Units 1,4 submit a compliance review of N290.5 for instrument air. This was completed in 2011 and documented in Reference [B.13-10]. Since the PARTS review, a new version of the code CSA N290.5-06 including Update No. 1 [B.13-1] was issued and was used for the compliance review. This review demonstrated that the Pickering Units 1,4 IA system complies with the updated version of this code. All requirements of the Standard were reviewed clause-by clause.

Darlington NGS

The original code review of the Darlington ISR, as documented in OPG Report NK38-REP-03680-10013 R000 [B.13-11], was performed against CSA N290.5-06 (2006). In general it was found that the Darlington designs of Electrical Power and Instrument Air systems meet or exceed the requirements of N290.5-06. NK38-REP-03680-10013 does identify gaps against clauses for both systems. The majority of the gaps were related to updating of Electrical Transient Analysis Program (ETAP) calculations to demonstrate system capabilities for all the Classes of power systems as well as for the two Groups of power[14]. This was not identified as a gap in the Pickering Units 5-8 code review of N290.5-06 [B.13-1] or in the Pickering Units 1,4 code review of N290.5-M90 [B.13-3]. Also, there is a requirement in the Engineering Change Control governance to maintain ETAP models current. Therefore this is not considered a PSR2 gap.

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[14] Two other gaps were identified; however they were re-classified as compliant in Appendix B of [B.13-11].

Following the issuance of CSA N290.5-06 (R2011) including Update No. 1 [B.13-1] in November 2011, a Code Refresh review was conducted and documented in OPG Report NK38-REP-03680-10157 R000 [B.13-12].

The significant changes made in CSA-N290.5-06 (R2011) including Update No. 1 relative to CSA-N290.5-06 were the inclusion of new requirements with respect to Beyond Design Basis Accidents (BDBAs), degradation of the off-site electrical grid, and aging management. This aligns with the information provided in the CSA Impact Statement notification for Update No. 1 to CSA N290.5-06 [B.13-5].

NK38-REP-03680-10157 R000 [B.13-12] identifies that Darlington is in compliance with the requirements of CSA-N290.5-06 (R2011) including Update No. 1 [B.13-1] with the exception of three clauses that have ISR Gaps related to BDBAs. These gaps are programmatic in nature and it was stated that OPG governance is being updated to include the Management of BDBAs for all stations. These gaps apply to Pickering and will be discussed in the next section.

B.13.2.2 Application of Post PSR1 Reviews

CSA N290.5 has been updated twice since the last code reviews were performed for Pickering. It was first updated to N290.5-06 in 2006. In 2011 Update No. 1 to N290.5-06 was issued [B.13-1]. A review of changes to N290.5 and Pickering's compliance with them are discussed in this section separately for the Electrical and Instrument Air systems.

Electrical Systems

As described above, a code review for Electrical systems was last completed for Pickering Units 1,4 against N290.5-M90 in Reference [B.13-9]. For Pickering Units 5-8, the latest code version reviewed was also N290.5-M90 in Reference [B.13-7]. Two changes to the code have been made since these reviews, i.e. N290.5-06 and N290.5-06 (2011) including Update No. 1. The significant intent changes to the code that could have an impact on nuclear safety are described below, including Pickering's compliance to them at a high level.

<u>Changes from Code Versions N290.5-M90 to N290.5-06</u>

These changes were not contained in a CSA Impact Statement, so they were determined based on a review of the two code versions to identify significant changes.

- Clause 4.2 – Use of Software (this is a common requirement applicable to IA as well)

  This new clause identifies requirements for software categorization, design, coding, etc.

  The OPG Program N-PROG-MP-0006 R009, "Software" [B.13-13], which includes Procedure, N-PROC-MP-0049 R009, "Procurement of Software and Products Containing Software" [B.13-14], is in place and demonstrates compliance with this clause.

- Clause 5.5.3.3 – Conversion Equipment Characteristics

  This new clause provides requirements for conversion equipment providing AC power (e.g., inverters, MG Sets) with respect to limits on frequency deviation and drift that are consistent with the requirements of the control equipment it supports. Additionally, output AC frequency is not to be dependent on nominal grid frequency. Pickering NGS demonstrates compliance with this clause in that limits on frequency characteristics are prescribed for conversion equipment by Class II Power Design Requirements that meet the requirements of supplied loads with sufficient margin. These Design Requirements are outlined in Appendix B of [B.13-15] and Section 3.4 of [B.13-16] for Pickering Units 1,4 and Units 5-8, respectively.

- Clause 5.5.3.4 – Conversion Equipment Level of Quality

  This new clause provides requirements for the level of quality of the power delivered by inverters or MG Sets to Class II loads, e.g., frequency variation, voltage variation, and voltage distortion. Pickering NGS demonstrates compliance with this clause in that limits on power quality characteristics are prescribed for conversion equipment by Class II Power Design Requirements that meet the requirements of supplied loads with sufficient margin. These Design Requirements are outlined in Appendix B of [B.13-15] and Section 3.4 of [B.13-16] for Pickering Units 1,4 and Units 5-8, respectively.

Changes from Code Versions N290.5-06 to N290.5-06 (R2011) including Update No. 1

These changes are extracted from the CSA Impact Statement [B.13-5] for the amended issue of N290.5-06 to N290.5-06 (R2011) including Update No. 1.

- Clauses 4.1 (m), 5.1.3.2 and 5.1.3.3 - Beyond Design Basis Accidents (BDBAs) (applicable to instrument air as well)

  These are new clauses dealing with requirements for BDBAs. Clause 4.1 (m) states that design features or strategies to manage BDBAs shall be considered in the design of electrical power and instrument air systems. Clause 5.1 provides general design requirements for electrical power systems: Clause 5.1.3.2 requires identification of BDBAs resulting from common cause events; Clause 5.1.3.3 requires consideration of design features or strategies for the identified common cause BDBAs. Clause 6 provides design requirements for instrument air systems; there are no requirements related to BDBAs. Both Pickering Units 1,4 and Units 5-8 have addressed BDBAs in their design, including preparation of operating procedures and provisions for connecting temporary generating equipment. No additional equipment is required to supply instrument air, as back-up air provisions to safety related equipment can fulfil their function after a postulated BDBA. Guides have been issued for both Pickering Units 1,4 and Units 5-8 describing the Functional Safety Requirements for BDBAs [B.13-17], [B.13-18], which include requirements for electrical equipment. This demonstrates compliance with these clauses.

- Clauses 5.1.7.1 and 5.1.7.2 - Degradation of the Off-Site Electrical Grid

  These are new clauses dealing with the requirements for:

  (i) The response of nuclear safety related functions during sustained grid frequency and voltage excursions

(ii) The plant continuing to produce electrical power during sustained grid frequency and voltage excursions

The purpose of these two new requirements is to ensure grid survivability so that safety related systems, e.g. Standby Generators and Emergency Power Generators, are not unnecessarily challenged. With regard to continued power generation during grid excursions, Pickering NGS must be compliant with the Ontario Independent Electricity System Operator (IESO) Market Rules, including MDP_RUL_0002_04A, "Chapter 4 – Grid Connection Requirements – Appendices" [B.13-19]. These rules ensure compliance with Clauses 5.1.7.1 and 5.1.7.2; therefore, no PSR2 gap exists with respect to these Clauses.

- Clause 7.1 on Equipment Qualification under normal operation and Anticipated Operational Occurrences (AOOs) (applicable to instrument air as well).

  This clause introduces the general requirement for components to be qualified to perform their required functions during normal operation and AOOs. Only the portion of this clause on AOOs is pertinent to nuclear safety. AOOs have not been identified and analyzed in the current Pickering Safety Reports. Hence, the requirements and credits attributed to the Electrical Power and Instrument Air systems for AOOs cannot be readily ascertained. This is therefore identified as **PSR2 CSA N290.5-06 Gap #1.**

- Clause 7.4 - Aging Management (applicable to instrument air as well).

  Four new clauses have been added dealing with equipment Aging Management. The current regulatory requirements for an Aging Management program are contained in CNSC REGDOC-2.6.3, "Aging Management" [B.13-20]. A code review was performed of this regulatory document in Report P-REP-03680-00004 R000 [B.13-21]. The review concludes OPG is in compliance with REGDOC-2.6.3 except in two areas. None of these areas affect compliance with the N290.5 clause 7.4.

  However, since clause 7.4.2 raises the AOO plant state in its requirement, as per clause 7.1, there is a gap against clause 7.4.2. This is already addressed by **PSR CSA N290.5-06 Gap #1** above.

Instrument Air

As described above, a code review for the Instrument Air systems was last completed for Pickering Units 1,4 against N290.5-06 in Reference [B.13-10]. For Pickering Units 5-8, there has not been a review performed against any versions. Given this, Pickering Units 1,4 and Pickering Units 5-8 will be discussed separately.

Pickering Units 1,4

As stated above, Reference [B.13-10] demonstrated that the Pickering Units 1,4 Instrument Air System is in compliance with N290.5-06. The changes made in code version N290.5-06 (R2011) including Update No. 1 were described earlier. There were no changes made to the IA system sections of the code. Only the changes to the common sections of the code, i.e. in clauses 4.1(m), 7.1 and 7.4, apply to IA as well. The compliance discussion for these clauses was already addressed above, so there are no additional PSR2 gaps for the IA sections of the current code version.

Pickering Units 5-8

As described above, a compliance review against the IA system portions of CSA N290.5-06 (R2011) including Update No. 1 [B.13-1] was not undertaken as part of previous Pickering Units 5-8 PSR1 reviews and the following assessment has therefore been completed. In the review below, the degree of conformance with clauses or groups of clauses in the L/R/C/S is assessed for Pickering NGS by reference to supporting evidence stating whether the intent of the requirements stipulated in the L/R/C/S is met. (Note that in instances where multiple sub-clauses are addressed they have been summarized and rolled-up into a more general paraphrased summary statement. In cases where only one clause is assessed it has been transcribed here and included in italics.). The review demonstrated that the Pickering Units 5-8 IA system complies with the updated version of this code, and therefore there are no PSR2 gaps.

CSA N290.5-06 Clause	PSR2 Review	Compliant or Gap
6 Design requirements — Instrument air system	See subsections below.	N/A
6.1 General 6.1.1 System design – the instrument air system shall be arranged to ensure a reliable supply of dry, oil-free compressed air supplied from Class III electrical and include permanently connected air storage tanks or compressed gas bottles to ensure an uninterrupted supply as per plant documentation and in accordance with CSA N285. 6.1.2 Failure of non-safety-related loads – shall not impair the instrument air system as a result of Design Basis Accidents (DBAs).	See subsections below. As detailed in Design Manual NK30-DM-75120-00001 R000 [B.13-22], the unitized Pickering U5-8 instrument air supply equipment includes oil free air compressors, dryers, filters and air receivers, and is capable of supplying air at a nominal gauge pressure of 860 kPa (125 psi) suitably conditioned by drying to a dew point below -40°C (-40°F) and filtered to a particle size less than 1 micron. The instrument air compressors are supplied from either Class IV or Class III power. In the event of a loss of Class IV power, Class III power will be able to supply the compressors prior to depleting the instrument air and air receivers [B.13-22]. Per the Design Requirements, NK30-DR-75120-10003 R001 [B.13-23], local emergency bottles provide an alternate, non-interruptible, air supply for nuclear safety related loads. The Instrument Air system is designed and registered in accordance with CSA Standard B51 [B.13-23]. All instrument air stations installed in U5-8 are provided with either single or double manifolds, each having eight outlets complete with isolation valves and a shut-off root valve and drain valve [B.13-22]. The instrument air distribution piping through the plant, including the Turbine Hall, Turbine Auxiliary Bay (TAB), Reactor Auxiliary Bay (RAB) and Reactor Building, includes components such as air receivers, air stations and other major equipment which have isolation and/or non-return valves in order to be able to isolate specific equipment, portions of the system, or specific branches of the piping, should it be required as a result of a DBA.	Compliant

CSA N290.5-06 Clause	PSR2 Review	Compliant or Gap
6.1.3 Instrument air for other purposes – design shall ensure that the instrument air system is not impaired by the operation or failure of non-safety-related loads.	The distribution system consists of a ring header, providing two alternate routes to any point in the Turbine Building, TAB, RAB and the Reactor Building. Manual isolating valves on the ring header have been provided for emergency isolation so that the instrument air supply to the remaining headers will not be jeopardized [B.13-22]. Hence the design of the Instrument Air system provides isolation such that the Instrument Air system supplying safety related loads is not impaired by the failure of non-safety related systems or loads. Furthermore, dedicated backup instrument air supplies, i.e., local air bottles, supply air to safety related loads in the event of the loss of the normal supply. Based on the above, the Pickering U5-8 Instrument Air system meets the intent of Section 6.1 of CSA N290.5-06.	
6.2 Safety-related considerations 6.2.1 Supply to safety-related loads – in accordance with the nuclear safety requirements and plant design documentation. 6.2.2 Isolation of loads – via an instrument air manifold take-off valve to facilitate maintenance and protect upstream supplies.	See subsections below. The Backup Instrument Air Design Requirements, NK30-DR-75120-10003 R001 [B.13-23], state that during normal operating conditions instrument air to each unit's loads are drawn from the normal instrument air supplied by its unitized compressors. Additional redundancy is provided by backup instrument air supplies, consisting of local, seismically qualified, instrument air bottles and associated distribution components that connect to the Instrument Air system. As per Section 11.2.1 of the Safety Report [B.13-24] backup instrument air equipment provides readily available capacity for the nuclear safety related equipment (including Emergency Coolant Injection System, Airlocks and Pressure Relief Panel Bypass Valves) requiring a non-interruptible air supply in the event of a loss of the normal instrument air supplies. Per [B.13-23] backup instrument air equipment is credited following design basis events such as: a) Loss of Normal Instrument Air b) Main Steam Line Breaks c) Design Basis Earthquakes d) Loss of Coolant Accident (LOCA) followed by Site Design Earthquake 24 hours later As described in 6.1.2 all instrument air stations installed in U5-8 are provided with either single or double manifolds, each having eight outlets complete with isolation valves, and a shut-off root valve and drain valve [B.13-22]. Isolation of specific air stations, and individual loads supplied by those stations, is accomplished through these isolation valves. These isolation points both facilitate maintenance and protect upstream supplies.	Compliant

CSA N290.5-06 Clause	PSR2 Review	Compliant or Gap
6.2.3 Supply to redundant loads – as required for the operation of a nuclear-safety-related load via either sufficient air supply separation or an alternate air supply such as local air storage tanks.	As per 6.1.1 a redundant air supply is provided by backup instrument air equipment consisting of local, seismically qualified, instrument air bottles and associated distribution components that connect to the Instrument Air system. Non-return valves are incorporated into the design to provide separation from the normal air supply, if it were to become unavailable. As per Section 11.2.1 of the Safety Report [B.13-24] backup instrument air equipment provides readily available capacity for nuclear safety related equipment.	
6.2.4 Reliability – requirements for air compressors with associated air receivers, filters, dryer equipment and cooling water system.	The Instrument Air system is not a System Important to Safety requiring an unavailability target per RD/GD-98 and does not have reliability requirements specified. However, backup instrument air equipment is credited for safety and has specified reliability targets [B.13-23]. Nevertheless, the system is designed with redundancy that supports reliable operation. The configuration of air supply compressors and associated air receivers, filters, dryer equipment and cooling water systems include redundancy considerations which support system reliability. Instrument air to each unit is supplied by four 33%, 0.307 m^3/s (650 SCFM), 860 kPa gauge (125 psig), two stage, water cooled oil free rotary screw compressors each driven by a 150 kW (200 hp) motor. One set of dual prefilters and afterfilters is provided for the four air dryers to remove both solid and liquid particles. Each filter, of a set of two, is capable of a 0.66 m^3/s (1400 SCFM) capacity [B.13-22]. The four air compressors discharge into four 7.0 m^3 (250 cu ft) air receivers that are arranged in parallel and are connected downstream of the four air dryer units by a common header [B.13-22].	
6.2.5 Common headers - may be used between the redundant air compressor/filter equipment and the air storage tanks.	Interconnections are not used between redundant air compressor equipment and air receivers belonging to another compressor set; however, the air receivers are joined by the common header and can be charged from each operating compressor as shown on the compressor flow diagram for U5 [B.13-25]. Based on the above, the Pickering U5-8 Instrument Air system meets the intent of Section 6.2 of CSA N290.5-06.	

CSA N290.5-06 Clause	PSR2 Review	Compliant or Gap
6.3 Protection against localized and common-cause events 6.3.1 Loss of instrument air – safety related equipment and the required air supplies shall either be isolated from the effects of DBAs or hardened to be immune to them. 6.3.2 Group 1 and Group 2 capability – shall meet nuclear safety requirements for local and common cause events.	See subsections below. As outlined in the Instrument Air Design Manual, NK30-DM-75120-00001 R000 [B.13-22], the air receivers and downstream air stations are distributed spatially in the plant. The air receivers are equipped with non-return valves to prevent them from being depleted, as described in Clause 6.5. The Instrument Air system is not protected for common mode DBAs. Backup instrument air equipment provides a seismically and harsh environment qualified alternate air supply which is credited for a variety of DBAs as described in Clause 6.2.1. This is accomplished by means of compressed air bottles which supply selected safety related loads for given mission times to ensure critical control, cooling and containment functions are maintained, or to provide sufficient time to allow for subsequent field actions. In accordance with the nuclear safety requirements and plant design documentation, backup instrument air equipment provides a seismically qualified alternate air supply to Group 2 pneumatic loads required to operate following a DBA. See the Pickering Safety Report [B.13-24] and Backup Instrument Air Design Requirements, NK30-DR-75120-10003 R001 [B.13-23], for details. Based on the above, the Pickering U5-8 Instrument Air system meets the intent of Section 6.3 of CSA N290.5-06.	Compliant
6.4 Reliability *The reliability of the instrument air supply shall be commensurate with the requirements of the connected safety-related systems.* *Note: Reliability requirements can be achieved by providing air to redundant devices from different instrument air take-off valves and providing an alternate air supply for those devices that need air following events that disable the normal air supply system.*	The reliability requirements for backup instrument air to loads is satisfied by bottled air supplies. The reliability of these supplies is commensurate with the requirements of the load(s) [B.13-23]. Reliability requirements are further enhanced by design provisions, such as the presence of local air receivers in various parts of the system, diverse supplies to redundant loads and alternate flow paths to many parts of the system. See Clause 6.2.4 which outlines additional redundancy provisions that contribute to system reliability. Based on the above, the Pickering U5-8 Instrument Air system meets the intent of Section 6.4 of CSA N290.5-06.	Compliant

CSA N290.5-06 Clause	PSR2 Review	Compliant or Gap
6.5 Isolating devices *Non-return valves shall be provided upstream of each air storage tank to maintain the short-term air supply to safety-related loads after loss of the upstream air supply. Isolation valves shall be provided on the distribution system to facilitate maintenance and testing and to mitigate the impact of header failures.*	The four main compressors' air receivers are fitted with non-return valves upstream of their inlets. As an example refer to the U5 flow diagram, NK30-FEH-75120-0001-U5 R006 [B.13-25]. Furthermore, as per the Instrument Air Design Manual, NK30-DM-75120-00001 R000 [B.13-22], each of the eleven local air receivers in the unitized instrument air distribution systems are equipped with a check valve to prevent back flow into their supply line. The check valves prevent air in the receiver from escaping should a line break occur in the upstream piping. These design provisions ensure there is sufficient air supply to loads in the short term. The distribution system consists of a ring header, providing two alternate routes to any point in the Turbine Building, Turbine Auxiliary Bay, RAB and the Reactor Building. Manual isolating valves on the ring header are provided for emergency isolation so that the instrument air supply to the remaining headers will not be jeopardized. All instrument air stations installed in the units are provided with either single or double manifolds, each having eight outlets complete with isolation valves and a shut-off root valve [B.13-22]. These valves both facilitate maintenance and testing and mitigate the impact of header failures. Based on the above, the Pickering U5-8 Instrument Air system meets the intent of Section 6.5 of CSA N290.5-06.	Compliant
6.6 Monitoring and testing *The instrument air system shall have appropriate monitoring and testing facilities so that the operator can confirm that the system is capable of performing its intended functions. Where a common alternate air system uses high-pressure gas cylinders, system surveillance or an alarm for operator response shall be provided. The action limits for the surveillance or the alarm shall be such that the alternate instrument air system remains available.*	Monitoring and testing of instrument air facilities is facilitated by air sampling throughout the plant in accordance with ANSI standard ISA-S7.3 as per OPG Chemistry Laboratory Procedure, P-CLP-75100-0001 R004 [B.13-26]. As detailed in the Instrument Air Design Manual, NK30-DM-75120-00001 R000 [B.13-22], monitoring of the Instrument Air system is accomplished through a variety of indications and alarms including indications for air pressure and compressor operation, low supply pressure alarms, etc. The backup instrument air surveillance (monitoring and testing) requirements are included with the credited load(s) requirements in the Operational Safety Requirements (OSRs) for the associated system. Compliance with the OSR requirements allows the operator to confirm that the backup instrument air supply to a specific load(s) is available; see the Containment OSR, NK30-OSR-08131.02-00003 [B.13-27], as an example. Furthermore Safety Related System Tests are in place to confirm the availability of backup air supplies. As an example, see NK30-SRS-C-036 R003 [B.13-28], which is an air holding test of backup instrument air tanks 21103-TK1/2. Based on the above, the Pickering U5-8 Instrument Air system meets the intent of Section 6.6 of CSA N290.5-06.	Compliant

CSA N290.5-06 Clause	PSR2 Review	Compliant or Gap
6.7 Limiting air flow into containment *A means to limit instrument air flow into containment following DBAs shall be provided.*	Instrument air lines feeding into containment have isolation valves installed on the RAB side (upstream of the containment penetration). In the event it is required (i.e., for a LOCA), these valves can be closed to stop the air flow into containment. As examples see the instrument air lines into containment shown on the U5 flow diagrams [B.13-29] and [B.13-30]. Based on the above, the Pickering U5-8 Instrument Air system meets the intent of Section 6.7 of CSA N290.5-06.	Compliant
6.8 Normal air supply system including: 6.8.1 Compressors - including filters, compressors, motors, instrumentation & control systems, and other protective and auxiliary systems shall be rated to meet both peak air demand and reliability requirements of the system. 6.8.2 Filters and dryers - shall be capable of maintaining the instrument air quality in accordance with ANSI/ISA 7.0.01 and meet the reliability requirements of the system. 6.8.3 Air receivers – shall meet the air supply and reliability requirements of the system including air supply requirements following a loss of Class IV power until Class III power is restored.	See subsections below. The overall capacity of the compressors, dryers, filters and related equipment exceed the required firm capacity of 1970 SCFM, as stated in the Safety Report [B.13-24], as follows. Four 33% unitized instrument air compressors and associated equipment, including water cooling, rotary screw compressors and 150 kW (200 hp) motors, are rated to supply 0.307 m^3/s (650 SCFM), 860 kPa gauge (125 psig) to a common header. The header feeds four 33% twin tower air dryer units, each of 0.307 m^3/s (650 SCFM) outlet capacity, which are collectively bound by one set of dual prefilters and afterfilters, where each one, of a set of two, is capable of 0.66 m^3/s (1400 SCFM) capacity and able to remove both solid and liquid particles [B.13-22], [B.13-24]. See Clause 6.4 for further discussion on the reliability requirements of the system. As per OPG Chemistry Laboratory Procedure for Station Air Systems, P-CLP-75100-0001 R004 [B.13-26], the instrument air quality is maintained in accordance with ANSI standard ISA-S7.3 (R1981). ANSI standard ISA-S7.3 (R1981) was subsequently incorporated into ANSI/ISA–7.0.01–1996 [B.13-31]. Four main and eleven local air receivers are installed as a source of instrument air, which the Design Requirements, NK30-DR-75120-10003 R001 [B.13-23], state are capable of maintaining loads for 3 to 8 minutes upon loss of power. The Design Manual, NK30-DM-75120-00001 R000 [B.13-22], further states that, in the event of a loss of Class IV power, Class III power will be able to supply the compressors prior to depleting the instrument air and air receivers. Based on the above, the Pickering U5-8 Instrument Air system meets the intent of Section 6.8 of CSA N290.5-06.	Compliant

CSA N290.5-06 Clause	PSR2 Review	Compliant or Gap
6.9 Distribution system	See subsections below.	Compliant
6.9.1 Capability - The system shall be capable of delivering sufficient instrument air to all loads required to maintain pressure under the most severe design basis conditions and include provisions for draining moisture.	As detailed in Section 4.0 of the Design Manual NK30-DM-75120-00001 R000 [B.13-22], the instrument air distribution system is adequately designed to provide the required peak capacity to loads in the station within a pressure control range from 760 kPa (110 psi) to 860 kPa (125 psi). Backup air supply provisions are also provided to ensure operation of safety-related equipment after the most severe design basis conditions (see Clause 6.2.1). Provisions for draining include automatic drain traps to remove condensate from the main air receivers, drain valves included on every instrument air manifold and moisture provisions on the instrument air dryer units, which normally maintain the system at a dew point below minus 40°C (-40°F) [B.13-22].	
6.9.2 Header air storage tanks - shall be located and sized to allow for normal operation of the plant and safe shutdown in the event of loss of instrument air supply and shall meet air supply requirements following a loss of Class IV power until Class III power restored.	As discussed in Design Manual NK30-DM-75120-00001 R000 [B.13-22], four main and eleven local air receivers are installed as a source of instrument air, which the Design Requirements, NK30-DR-75120-10003 R001 [B.13-23], state are capable of maintaining loads for 3 to 8 minutes upon loss of power. The Design Manual, NK30-DM-75120-00001 R000 [B.13-22], further states that, in the event of a loss of Class IV power, Class III power will be able to supply the compressors prior to depleting the instrument air and air receivers. Based on the above, the Pickering U5-8 Instrument Air system meets the intent of Section 6.9 of CSA N290.5-06.	
6.10 Air supply connections to loads	See subsections below.	Compliant
6.10.1 Instrument air manifolds – shall be provided with an inlet manual isolation valve to facilitate maintenance and protect upstream supplies.	As detailed in Clause 6.1.2 all instrument air stations installed in Pickering U5-8 are provided with a shut-off root valve. These isolation points both facilitate maintenance and protect upstream supplies.	
6.10.2 [Manual] Instrument air take-off valves - shall be provided on instrument air manifolds to enable as required isolation.	As detailed in Clause 6.1.2 all instrument air stations installed in Pickering U5-8 are provided with either single or double manifolds, each having eight outlets complete with isolation valves. These isolation points both facilitate maintenance and protect upstream supplies.	

CSA N290.5-06 Clause	PSR2 Review	Compliant or Gap
6.10.3 Self-vented pressure regulating valves shall be used to regulate the air pressure of loads that do not continuously consume air and that have design pressures lower than the air system design pressure. 6.10.4 Direct connections to the distribution headers - may be permissible, via a manual isolation valve, for loads that require a large instrument air supply capacity.	The Pressure Regulating Valves (PRVs) used in the plant are typically of the self-vented type. Examples are shown in the manufacturer information included in NK30-MMM-21130-00004 R001 [B.13-32] and NICR-65640 [B.13-33]. This type of PRV is used for normal instrument air supply loads and for air supplied by compressed air bottles. The hose connections to the resin transfer system, as shown on the U5 Flow Diagram, NK30-FEH-75120-0004-U5 R018 [B.13-29], are connected to the distribution header via manual isolating valves. Based on the above, the Pickering U5-8 Instrument Air system meets the intent of Section 6.10 of CSA N290.5-06.	
6.11 Alternate air system 6.11.1 Local air storage tanks – where provided, shall be sufficient to supply air to their loads for the mission time established by the plant design documentation following loss of normal air supply. 6.11.2 Alternate air supply system – where provided, shall have sufficient redundancy to meet reliability requirements.	See subsections below. As discussed in Design Manual NK30-DM-75120-00001 R000 [B.13-22], four main and eleven local air receivers are installed as a source of instrument air which the Design Requirements, NK30-DR-75120-10003 R001 [B.13-23], state are capable of maintaining loads for 3 to 8 minutes upon loss of power. Class III power will be able to supply the compressors prior to depleting the instrument air and air receivers [B.13-22]. Backup instrument air equipment provides a seismically and environmentally qualified alternate air supply to Group 2 equipment required to operate following a DBA [B.13-23]. As described in Clause 6.2 this is accomplished by means of compressed air bottles which supply selected safety related loads for given mission times to ensure critical shutdown, cooling and containment functions are maintained, or to provide sufficient time to allow for subsequent field actions. Redundancy is further enhanced by providing independent bottles for redundant safety related components. Based on the above, the Pickering U5-8 Instrument Air system meets the intent of Section 6.11 of CSA N290.5-06.	Compliant

B.13.3 Compliance Summary for Pickering PSR2

There is one PSR2 CSA N290.5-06 (R2011) including Update No. 1 [B.13-1] gap which relates to Safety Factor 1 (Plant Design):

1. A gap exists for the Pickering Units 1,4 and 5-8 Instrument Air and Electrical Systems on Clauses 7.1 and 7.4.2 of N290.5-06 (R2011) including Update No. 1 dealing with requirements for Anticipated Operational Occurrences (AOOs). These clauses introduce the requirement for components to be qualified to perform their required functions during normal operation and AOOs. Only the portion of this clause on AOOs is pertinent to nuclear safety. It is likely that AOOs, due to their nature, do not result in a challenge to the qualification of systems, including Instrument Air and Electrical systems. However, AOOs have not been identified and analyzed in the current Pickering Safety Reports. This issue has therefore been identified as a PSR2 gap. It is being addressed as part of REGDOC-2.4.1 implementation.

B.13.4 References

[B.13-1] CSA Standard, N290.5-06 (R2011) including Update No. 1, *Requirements for Electrical Power and Instrument Air Systems of CANDU Nuclear Power Plants*, December 2006; Update No. 1: November 2011.

[B.13-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.13-3] CSA Standard, N290.5-M90, *Requirements for the Support Power Systems of CANDU Nuclear Power Plants*, 1990.

[B.13-4] CSA Standard, N290.5-06, *Requirements for Electrical Power and Instrument Air Systems of CANDU Nuclear Power Plants*, December 2006.

[B.13-5] CSA Impact Statement and Publication Notice, *Product: CSA N290.5-06 Product Designation: CSA 290.5-06*, Date not provided.

[B.13-6] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.13-7] OPG Report, NK30-REP-03680-00001 R000, *Pickering NGS B –Integrated Safety Review - Plant Design Safety Factor*, August 2007.

[B.13-8] OPG Letter, NA44-CORR-00531-00381, R.J. Strickert to J.S.C. Tong, *Pickering A Updated Basis for Return to Service Document*, April 20, 2001.

[B.13-9] AECL Assessment Document, 44RS-00531-ASD-001 Rev. 04, *Review of Pickering A Design Against Current Codes and Standards*, November 2000.

[B.13-10] OPG Letter, NA44-CORR-00531-06686, G. Jager to T.E. Schaubel, *Pickering A - Review Modified Instrument Air System Against CSA N290.5-06, Requirements for*

Electrical Power and Instrument Systems of CANDU Nuclear Power Plants Action Item 2001-4-08, March 28, 2011.

[B.13-11] OPG Report, NK38-REP-03680-10013 R000, *Review Of CAN/CSA-N290.5-06 (December 2006) Requirements for Electrical Power and Instrument Air Systems of CANDU Nuclear Power Plants for Darlington Integrated Safety Review*, August 2011.

[B.13-12] OPG Report, NK38-REP-03680-10157 R000, *Code Refresh Review of CSA N290.5-06 (2011) Requirements For Electrical Power and Instrument Air Systems of CANDU Nuclear Power Plants for DNGS ISR*, July 2013.

[B.13-13] OPG Program, N-PROG-MP-0006 R009, *Software*, April 2015.

[B.13-14] OPG Procedure, N-PROC-MP-0049 R009, *Procurement of Software and Products Containing Software*, June 2016.

[B.13-15] OPG Design Manual Addendum, 44RS-54200-DMA-001 R001, *Pickering 'A' Return to Service – Class II Inverter and Class I Rectifier (C1-038-00-01-008)*, May 2002.

[B.13-16] OPG Design Requirements, NK30-DR-54200-00001 R001, *Pickering GS B– Class II Power Supply*, November 2002.

[B.13-17] OPG Guideline, NA44-GUID-03600-00001 R000, *Pickering 1-4 Beyond Design Basis Functional Safety Requirements*, October 2014.

[B.13-18] OPG Guideline, NK30-GUID-03600-00001 R000, *Pickering 5-8 Beyond Design Basis Functional Safety Requirements*, October 2014.

[B.13-19] IESO Market Rules, MDP_RUL_0002_04A, *Market Rules - Chapter 4 – Grid Connection Requirements – Appendices*, September 2015.

[B.13-20] CNSC REGDOC-2.6.3, *Aging Management*, March 2014.

[B.13-21] OPG Report, P-REP-03680-00004 R000, *Pickering NGS Periodic Safety Review 2: Code and Standard Reviews for Safety Factors 2 (Actual Condition of SSCs), 3 (Equipment Qualification) and 4 (Aging)*, July 2016.

[B.13-22] Pickering Design Manual, NK30-DM-75120-00001 R000, Design Manual for SCI 75120 *Instrument Air System*, November 2010.

[B.13-23] OPG Design Requirements, NK30-DR-75120-10003 R001, *Nuclear Safety Design Requirements for Backup Instrument Air*, April 2005.

[B.13-24] OPG Report, NK30-SR-01320-00002 R004, *Pickering B Safety Report – Part 2*, October 2012.

[B.13-25] OPG Design Flow Diagram, NK30-FEH-75120-0001-U5 R006, *Turbine Hall Instrument Air Supply Flow Diagram*, February 2016.

[B.13-26] OPG Chemistry Laboratory Procedure, P-CLP-75100-0001 R004, *Sampling Station Air Systems*, August 2014.

[B.13-27] OPG Operational Safety Requirements, NK30-OSR-08131.02-00003 R004, *Pickering 5-8 Operational Safety Requirements: Negative Pressure Containment*, March 2015.

[B.13-28] OPG Safety Related System Test, NK30-SRS-C-036 R003, *Units 5 – 8 21103-TK1 and TK2 Air Holding Test*, May 2009.

[B.13-29] OPG Design Flow Diagram, NK30-FEH-75120-0004-U5 R018, *Reactor Building Elev 254-0" Instrument Air Flow Diagram*, October 2013.

[B.13-30] OPG Design Flow Diagram, NK30-FEH-75120-0005-U5 R019, *Reactor Building Instrument Air Flow Diagram*, October 2013.

[B.13-31] ANSI Quality Standard, ISA–7.0.01–1996, *Quality Standard for Instrument Air*, November 1996.

[B.13-32] OPG Manufacturer's Manual, NK30-MMM-21130-00004 Sht: Vol 2 R001*, EQ Equipment Modifications for Equipment and Personnel Airlocks P.O. No 15-32021-24 P.O. No 12-29237-24*, June 2011.

[B.13-33] OPG Engineering Change, NICR-65640, *Replace Instrument Air Pressure Regulator CV164*, August 2008.

B.14 CSA N290.6-09, "Requirements for Monitoring and Display Of Nuclear Power Plant Safety Functions In The Event Of an Accident"

B.14.1 Background

The following paraphrased from the preface and scope of CSA N290.6-09 [B.14-1] provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *The monitoring and display of nuclear power plant variables following an accident are important safety functions; appropriate operator actions following such an event rely on the availability of adequate information.*
>
> *CSA N290.6 provides the requirements and recommendations for the design, manufacture, installation, and qualification of those components specifically involved in the monitoring and display of post-accident information for a nuclear reactor.*

CSA N290.6 is relevant to Safety Factor 1 (Plant Design). CSA N290.6 is not discussed in the R04 Pickering Licence Conditions Handbook [B.14-2].

CSA N290.6-09, which was reaffirmed in 2014, is the second edition of this standard, and supersedes the previous edition published in 1982 under the title "Requirements for Monitoring and Display of CANDU Nuclear Power Plant Status in the Event of an Accident" [B.14-3]. The Preface of CSA N290.6-09 [B.14-1] provides a summary of significant changes from the previous 1982 edition [B.14-3], which are discussed in Section B.14.2.2 below.

The results of PSR1 CSA N290.6 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.14.2. As identified in Reference [B.14-4], the Pickering PSR2 review of CSA N290.6-09 is an Incremental Review. PSR2 Incremental Review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- <u>Compliance:</u> Compliance indicates that the change in the safety requirement, per the topical review, is met.
- <u>Gap</u>: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.14.2 Compliance Assessment for Pickering PSR2

B.14.2.1 Application of PSR1 Review

The versions of N290.6 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

The Post Accident Monitoring (PAM) technical basis for Pickering Units 5-8 is documented in References [B.14-5], [B.14-6] and [B.14-7]. A clause-by-clause review of the Pickering Units 5-8 Critical Safety Parameter Monitoring (CSPM) and display systems design against the CSA N290.6-M82 [B.14-3] version of the Standard was documented in OPG Report NK30-REP-03680-00001 R000, "Pickering NGS-B Integrated Safety Review – Plant Design Safety Factor" [B.14-8]. NK30-REP-03680-00001 R000 [B.14-8] identified one discrepancy for Clause 5.3.1 relating to availability of instruments. This was identified as Gap 1-103, and the disposition was provided in Reference [B.14-9]. The disposition concluded that implementation of the Safe Operating Envelope (SOE) for CSPM would address the discrepancy. The implementation of SOE (including gap analysis and issue resolution) has subsequently been completed, with the compliance framework established in the CSPM Operational Safety Requirements (OSR) [B.14-6] and the CSPM OSR Compliance Table [B.14-10]. Therefore, this issue is no longer a gap for Pickering Units 5-8.

OPG Report NK30-REP-03680-00001 R000 [B.14-8] also identified three Acceptable Deviations associated with the following clauses:

- Clause 5.7.1 and Clause 5.7.3, relating to maintenance accessibility of instruments post-accident, including for calibration. These clauses were combined under Gap 1-104 and an assessment of the impact was provided in Reference [B.14-9]. It was concluded that there were sufficient instrument loops that would be accessible post-accident, there was sufficient redundancy and that calibrations for accuracy during post-accident mission were not necessary. This disposition is not impacted by Pickering NGS operation beyond 2020. Therefore, this is not a gap for Pickering PSR2.

- Clause 5.12.1, relating to separation between channelized instrument tubing. The disposition and acceptance of this gap by the CNSC is documented in Reference [B.14-11]. The disposition stated that a separate review performed for CSA N285.0, "General Requirements for Pressure-retaining Systems and Components in CANDU Nuclear Power Plants", concluded there were no gaps with respect to instrument tube routing. A separate review was also performed for CSA N285.0 as part of PSR2, and the Pickering B ISR disposition is not impacted by Pickering NGS operation beyond 2020. Therefore, this is not a gap for Pickering PSR2.

The above-mentioned Pickering Unit 5-8 Gaps and Acceptable Deviations were deemed to have either low or no safety significance as part of the Pickering B ISR, and are not impacted by Pickering NGS operation beyond 2020. Therefore, none of the above Pickering Units 5-8 ISR findings represent a Pickering PSR2 gap against CSA N290.6-M82 [B.14-3]. Pickering NGS compliance against the most recent version of the Standard, CSA N290.6-09 [B.14-1], is addressed under Section B.14.2.2 below.

Pickering Units 1, 4

The PAM technical basis for Pickering Units 1,4 is documented in References [B.14-12], [B.14-13], [B.14-14], [B.14-15] and [B.14-16]. A review of the Pickering Units 1,4 CSPM and display systems design against CSA N290.6-M82 [B.14-3] was completed in AECL Report 44RS-00531-ASD-001 R004, "Pickering A Return to Service: Review of Pickering A Design Against Current Codes and Standards (CI-007-03-01-007)" [B.14-17]. 44RS-00531-ASD-001 R004 [B.14-17] found that Pickering Units 1,4 were either fully compliant or indirectly compliant with all clauses of the standard. This review was performed prior to the implementation of the SOE or Environmental Qualification (EQ) programs at Pickering A. Subsequently, the CSPM OSR [B.14-12] and CSPM OSR Compliance Table [B.14-16] have been produced and fully implemented. Additionally, an EQ Technical Basis Document for PAM has been produced [B.14-13]. Therefore, the Pickering B ISR Gap 1-103 discussed above is also addressed for Pickering Units 1,4 and is not a PSR2 gap.

The Pickering B ISR Acceptable Deviations discussed above (related to Clauses 5.7.1, 5.7.3 and 5.12.1) were also considered for applicability to Pickering Units 1,4. CSA N290.6-M82 [B.14-3] Clause 5.7.1 was not reviewed for Pickering Units 1,4 in 44RS-00531-ASD-001 R004 [B.14-17]. However, given the similarities between plant design and instrument loop locations, the Clause 5.7.1 dispositions for Pickering Units 5-8 [B.14-9] are also applicable to Pickering Units 1,4. 44RS-00531-ASD-001 R004 [B.14-17] concluded that Clause 5.7.3 had 'Indirect Compliance', noting that "Calibration capability exists through redundant measurements." Clause 5.12.1 relates to requirements for pressure-retaining components used in the information chain and no issues were identified in 44RS-00531-ASD-001 R004 [B.14-17] relating to this clause for Pickering Units 1,4.

Based on the above, there are no PSR2 gaps associated with Pickering Units 1,4 compliance with CSA N290.6-M82 [B.14-3]. Pickering NGS compliance against the most recent version of the Standard, CSA N290.6-09 [B.14-1], is addressed under Section B.14.2.2 below.

Darlington NGS

A clause-by-clause review of Darlington NGS against CSA N290.6-M82 [B.14-3] was conducted in OPG Report NK38-REP-03680-10042-R000, "Review of CAN/CSA-N290.6-M82 (R2001) (January 1983), Requirements for Monitoring and Display of CANDU Nuclear Power Plant Status in the Event of an Accident for Darlington Integrated Safety Review" [B.14-18]. Since PSR1 reviews against CSA N290.6-M82 [B.14-3] were completed for Pickering Units 1,4 and 5-8 as discussed above, and CSA N290.6 is largely design-specific, the Darlington PSR1 results for CSA N290.6-M82 have not been assessed for applicability to PSR2.

It is noted that Darlington also conducted an ISR Code Refresh review of CSA N290.6-09 [B.14-1] in OPG Report NK38-REP-03680-10158 R000, "Code Refresh Review of CSA N290.6-2009, Requirements for Monitoring and Display of Nuclear Power Plant Safety Functions in the Event of an Accident for DNGS ISR" [B.14-19]. Although the associated findings and their resolutions were mostly specific to Darlington design documentation, NK38-REP-03680-10158 R000 has been utilized in Section B.14.2.2 below to assess Pickering NGS compliance against N290.6-09 where there is programmatically applicable content.

B.14.2.2 Application of Post-PSR1 Reviews

As identified above in Section B.14.1, the Preface of CSA N290.6-09 [B.14-1] provides a summary of significant changes from the previous edition of the Standard, N290.6-M82 [B.14-3]. These are:

a) *Harmonization with CSA N285.0 and N286.*

Compliance reviews for CSA N285.0, "General Requirements for Pressure-retaining Systems and Components in CANDU Nuclear Power Plants" and N286 "Management System Requirements for Nuclear Facilities" were completed separately as part of Pickering PSR2. Harmonization of these CSA Standards with N290.6 is largely editorial in nature and addressed via the overlapping PSR2 reviews. Therefore, there is no PSR2 gap.

b) *The addition of an informative annex on severe accidents*.

Informative Annex A, which relates to information and measurement needs for severe accidents, is non-mandatory and provided only for guidance. However, clauses identifying new build related requirements are assessed for applicability to Pickering NGS as part of PSR2. Clause A.1.1 from informative Annex A [B.14-1] states:

> *For existing plants, SAMGs* [Severe Accident Management Guidance] *should be developed with the aim of making maximum use of the existing design. For new plants, SAMGs should be considered in the design.*

SAMG parameters and associated instrumentation were not considered in the original design of Pickering NGS for Beyond Design Basis Accident (BDBA) purposes. However, as part of Fukushima follow-up activities, OPG has conducted extensive assessments for instrumentation and equipment survivability for BDBAs, including severe accidents, per Reference [B.14-20]. The review demonstrated that sufficient instrumentation has a reasonable confidence of surviving post-BDBA such that all Units could be safely stabilized in a shutdown state. Given that Annex A is non-mandatory, and the above-mentioned assessments have been conducted, there is no PSR2 gap.

c) *The addition of information to aid in the selection of parameters to be monitored.*

Clause 4 of CSA N290.6-09 [B.14-1] prescribes guidance on the selection of parameters required to assess whether the plant will continue to operate within Safety Limits following an accident. An intent review against the associated three sub-clauses (sub-clauses 4.1, 4.2 and 4.2) is discussed below for Pickering NGS:

- Sub-clause 4.1 states: "To ensure public safety following an accident, the nuclear power plant shall be monitored for verification of: (a) reactor shutdown; (b) reactor heat removal; (c) a barrier(s) to the release of radioactivity to the environment; and (d) radioactive releases."

As identified in NK38-REP-03680-10158 R000 [B.14-19], Clause 4.1 (a) (b) (c) of N290.6-09 [B.14-1] is the same as Clause 4.1.1 (a) (b) (c) of N290.6-M82 [B.14-3]. In addition, Clause 4.1 (d) in N290.6-09 is the same as Clause 4.1.2 (a) of N290.6-M82. As discussed earlier, N290.6-M82 was reviewed for Pickering Units 1,4 and Units 5-8 and the intent of theses clauses was determined to be met. Therefore, there is no PSR2 gap.

- Sub-clause 4.2 states: "Monitoring of the nuclear power plant shall include the parameters required to assess whether the plant will continue to operate within the safety limits following an accident."

  The Pickering Units 1,4 and Units 5-8 PAM technical basis and operational documentation references discussed earlier in Section B.14.2.1 describe the CSPM indications of important parameters at designated operating locations, as well as qualified instrumentation loops. Per the Pickering B CSPM OSR [B.14-6]:

  > *CSP Monitoring forms the minimum set of post accident monitoring activities that must be carried out in order to meet the safety objective of protecting the public. In this context, it is important to note that Critical Safety Parameters are the parameters that must be maintained within safety limits in order to ensure that the Control, Cool, and Contain safety functions are being continually satisfied post-accident. Critical Safety Support Parameters (CSSPs) are parameters required to determine the appropriate restoration procedure to be followed given one or more CSPs are in an unacceptable range. Support Parameters give advance warning that a CSP may reach its action limit but they might not be available following a Design Basis Earthquake (DBE). CSP monitoring is no longer required when it has been determined that the reactor is adequately controlled (shutdown), cooled (Heat Transport System (HTS) is subcooled), and contained (radioactivity remains within containment). If a subsequent event occurs that could challenge any of these critical parameters, CSPM will be reintroduced…*
  >
  > *Critical Safety Parameter Monitoring instrumentation is required to provide necessary indications following all design basis accidents, including situations where the cause and course of the accident may be indeterminate. The indications are provided in the Main Control room (MCR); and should the MCR become uninhabitable, necessary CSPM indications are provided in the Unit Emergency Control Centres (UECCs).*

  Similar statements are made in the Pickering A CSPM OSR [B.14-12]. CSPM indications are provided in the MCR and the Shutdown System Enhancement Instrumentation Rooms. Based on the above, Pickering NGS meets the intent of sub-clause 4.2.

- Sub-clause 4.3 states: "Monitoring shall be performed for evaluating radiological conditions (a) to determine when conditions are developing that warrant initiating off-site emergency procedures; and (b) that warrant the initiation of off-site emergency procedures."

As identified in NK38-REP-03680-10158 R000 [B.14-19], Clause 4.3 (a) of N290.6-09 [B.14-1] is the same as Clause 4.1.2 (a) of N290.6-M82 [B.14-3]. N290.6-M82 was reviewed for Pickering Units 1,4 and Units 5-8, as discussed earlier, and the intent of this clause was determined to be met. The same conclusion is applicable to PSR2.

The radiation monitoring that is required in the event of a radiation emergency is used to support decisions regarding off-site actions as part of the emergency response framework. Monitoring within the plant is provided by the Automated Source Term Gamma Monitoring System, portable monitors and stack monitoring. Offsite radiation monitoring is provided by the Automated Near Boundary Gamma Monitoring System and dosimeters. Therefore, Pickering is compliant with the requirements of this clause.

Based on the above, Pickering NGS meets the intent of CSA N290.6-09 [B.14-1] and there are no PSR2 gaps.

B.14.3 Compliance Summary for Pickering PSR2

There are no PSR2 gaps for CSA N290.6-09 [B.14-1]. Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with CSA N290.6-09.

B.14.4 References

[B.14-1] CSA Standard N290.6-09, *Requirements for Monitoring and Display of Nuclear Power Plant Safety Functions in the Event of an Accident*, March 2009.

[B.14-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.14-3] CSA Standard, CAN3-N290.6-M82, *Requirements for Monitoring and Display of Nuclear Power Plant Safety Functions in the Event of an Accident*, 1982.

[B.14-4] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.14-5] OPG Manual, NK30-AIM-058-09013-07 R024, *Pickering Nuclear Generating Station – Critical Safety Parameter Monitoring and Restoration,* May 2016.

[B.14-6] OPG Manual, NK30-OSR-08131.02-00021 R001, *Pickering NGS-B Operational Safety Requirements: Critical Safety Parameters Monitoring Instrumentation*, November 10, 2010.

[B.14-7] OPG Report, Pickering NGS B, NK30-REP-03651-0393 R002, *Environmental Qualification Requirements for Harsh Environment DBA Post Accident Monitoring*, April 5, 2000.

[B.14-8] OPG Report, NK30-REP-03680-00001 R000, *Pickering NGS-B Integrated Safety Review – Plant Design Safety Factor,* August 2007.

[B.14-9] OPG Report, NK30-REP-03680-00016 R000, *OPG Response to CNSC Comments on Pickering NGS-B Integrated Safety Review – Plant Design, Safety Analysis, Safety Performance, Ageing and Equipment Qualification Safety Factors and Discrepancy Resolutions,* September 2009.

[B.14-10] OPG Manual, NK30-OSR-08131.02-00021-Table-01 R001, *Pickering NGS-B Operational Safety Requirements: Critical Safety Parameters Monitoring,* November 2013.

[B.14-11] CNSC Letter, NK30-CORR-00531-05312 R000, T. Schaubel to W. Robinson, *Pickering NGS-B Integrated Safety Review (ISR) - CNSC Staff Response to OPG's Comments Disposition Report*, March 4, 2010.

[B.14-12] OPG Manual, NA44-OSR-08131.02-00018 R001, *Pickering NGS-A Operational Safety Requirements: Critical Safety Parameters Monitoring Instrumentation*, December 2011.

[B.14-13] OPG Report, NA44-REP-03651-00005 R004, *Technical Basis Document for Environmental Qualification of Post Accident Monitoring*, November 2013.

[B.14-14] OPG Manual, NA44-AIM-014-09013-07 R011, *Abnormal Incidents Manual - Critical Safety Parameter (CSP) Monitoring and Restoration Procedure*, April 2016.

[B.14-15] OPG Manual, NA44-AIM-014-09013-09.01 R007, *Abnormal Incidents Manual - Critical Safety Parameter (CSP) Monitoring and Restoration Procedure with MCR Uninhabitable*, April 2016.

[B.14-16] OPG Manual, NA44-OSR-08131.02-00018-Table 01 R002, *Pickering NGS-A Operational Safety Requirements: Critical Safety Parameters Monitoring Instrumentation SOE Compliance Table*, February 2014.

[B.14-17] AECL Report, 44RS-00531-ASD-001 Revision 4, *Pickering A Return to Service: Review of Pickering A Design Against Current Codes and Standards (CI-007-03-01-007)*, November 2000.

[B.14-18] OPG Report, NK38-REP-03680-10042 R000, *Review of CAN/CSA-N290.6-M82 (R2001) (January 1983), Requirements for Monitoring and Display of CANDU Nuclear Power Plant Status in the Event of an Accident for Darlington Integrated Safety Review*, August 2011.

[B.14-19] OPG Report, NK38-REP-03680-10158 R000, *Code Refresh Review of CSA N290.6-2009, Requirements for Monitoring and Display of Nuclear Power Plant Safety Functions in the Event of an Accident for DNGS ISR*, July 25, 2013.

[B.14-20] OPG Report, N-REP-09013-10007 R000, *Ontario Power Generation Severe Accident Management Guidance Instrumentation and Equipment Survivability - Summary Report*, December 13, 2013.

B.15 CSA N290.11-13, "Requirements for Reactor Heat Removal Capability During Outage of Nuclear Power Plants"

B.15.1 Background

The following, paraphrased from the preface and scope of CSA N290.11 [B.15-1], provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *The purpose of CSA N290.11 is to ensure that the designs of all water-cooled nuclear power plants include systems that transfer heat from the reactor to the ultimate heat sink during outages.*
>
> *CSA N290.11 sets the requirements for the design, qualification, installation, commissioning, operation, maintenance, testing, inspection, and documentation requirements for systems providing heat removal from the reactor core to the ultimate heat sink(s) for water-cooled nuclear power plants during outages. CSA N290.11 covers only fuel cooling within the reactor core and does not cover spent fuel pool cooling, off-reactor fuelling operations, or the completely defueled core state.*

All of N290.11-13 is directly relevant to Safety Factor 1 (Plant Design).

CSA N290.11-13 is the first edition of this standard. Compliance with N290.11 is not currently a licence requirement for Pickering NGS (PROL 48.02/2018) and is not referenced in the Pickering Licence Conditions Handbook [B.15-2].

As identified in Reference [B.15-3], the Pickering PSR2 review of CSA N290.11-13 is a High Level Review. For a PSR2 High Level Review, the degree of conformance with clauses or groups of clauses in the Law, Regulation, Code or Standard (L/R/C/S) is demonstrated by supporting evidence stating whether the intent of the requirements stipulated in the requirement document is met. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the intent of the safety requirement is met.
- Gap: A Gap indicates that the intent of the safety requirement is not met.

B.15.2 Compliance Assessment for Pickering PSR2

B.15.2.1 Application of PSR1 Reviews

There have not been any programmatic or system/application PSR1 reviews against CSA N290.11 for Pickering or Darlington.

B.15.2.2 Application of Post PSR1 Reviews

The CSA Publication Notice for N290.11-13 [B.15-4], identifies the following Summary of Impacts of the significant features of this standard:

1. Defines and uses the term "process heat sink" instead of "primary/back-up heat sinks".

2. Provides high level reliability requirements for the design of outage heat sinks.

3. Provides testing requirements, including the requirements to test standby heat sink systems during the outage.

4. Clarifies that work can be done on a primary heat sink.

As discussed above, a compliance review of CSA N290.11-13 was not undertaken as part of previous PSR1 reviews and the following High Level assessment has been completed. In the review below, the degree of conformance with clauses or groups of clauses in the L/R/C/S is assessed for Pickering NGS by reference to supporting evidence stating whether the intent of the requirements stipulated in the L/R/C/S is met (Note: Introductory Sections 1 through 3 do not contain requirements and are not included).

CSA N290.11-13 Clause	PSR2 Review	Compliant or Gap
4.1 Functional Requirements - General	The general requirements for heat sink management are contained in N-STD-OP-0025 [B.15-5], Sections 1.1 and 1.2. These contain the requirements stated in Section 4.1. Monitoring of heat sink effectiveness is covered in section 1.4 of [B.15-5] and in the Pickering Units 5-8 and Units 1,4 Shutdown Heat Sinks Operating Manuals [B.15-6] and [B.15-7].	Compliant
4.2 Functional Requirements - Success Criteria	Section 1.2 of [B.15-5] provides the capability requirements and acceptance criteria for both shutdown and emergency heat sinks in compliance with this section.	Compliant
5.1.1 to 5.1.5 Heat Sink Requirements - General	The heat sink standard [B.15-5], Section 1.5.3 prescribes that no "normally anticipated single failure" can fail both the primary and back-up heat sink. N-STD-OP-0025 [B.15-5] also requires: 1. that process and emergency heat sinks be specified in advance; 2. that heat sink strategies consider defence in depth and reliability in the specification of back-up heat sinks; 3. Special considerations for conditions which could affect Heat Transport System (HTS) boundary integrity, e.g. ice plugs. The requirement for back-up heat sinks to mitigate the conditions following an Anticipated Operational Occurrence (AOO) is not specified in governance or procedures. Loss of a division of power, a single component failure, etc., which are likely to be considered in the set of AOOs, are accounted for in the specification of heat sinks. However, AOOs have not been identified	Gap **<u>PSR2 CSA N290.11-13 Gap #1</u>**

CSA N290.11-13 Clause	PSR2 Review	Compliant or Gap
	and analyzed in the current Pickering Safety Reports, and demonstration of heat sink capability following an AOO has not been formally addressed for Pickering. Therefore, this issue is identified as **PSR2 CSA N290.11-13 Gap #1.**	
5.1.6 Heat Sink Requirements - General	The Pickering Units 5-8 and Units 1,4 Shutdown Heat Sinks Operating Manuals [B.15-6] and [B.15-7] specify all of the requirements listed in this section.	Compliant
5.1.7-5.1.8 Application of Single Failure Criterion	For a new nuclear plant, this clause requires that the emergency heat sinks shall meet the single failure criterion. For existing plants (Clause 5.1.8), this is not mandatory. The designated outage emergency heat sinks, as per Reference [B.15-8] are: • Pickering 1,4: Emergency Coolant Injection System (ECIS) and Emergency Boiler Water Supply (EBWS), supported by the Interstation Transfer Bus (ISTB) and Class III standby generators; ECI recovery for HTS open or HTS leaks/Loss of Coolant Accident (LOCA). Additionally, for a seismic event, Shutdown Cooling (SDC) is credited. • Pickering 5-8: ECIS and Emergency Water Supply (EWS), supported by Emergency Power Supply (EPS). A gap relating to the single failure criterion was identified during the Pickering B Integrated Safety Review (ISR) [B.15-9]. This gap was categorized as an acceptable deviation that would be confirmed when the updated Pickering 5-8 Probabilistic Safety Assessment (PSA) was complete. The PSA update was subsequently completed and this issue has since been closed in the Continued Operations Plan [B.15-10]. Both Pickering 5-8 and 1,4 now have outage PSAs which model the above systems, identify dominant contributors to plant risk and significant singletons. The outage plant risk for both Pickering 5-8 and 1,4 has been demonstrated to be acceptable. Hence, the rationale for closure of the Pickering B PSR1 gap is also applicable to outage heat sinks and to Pickering 1,4.	Compliant
5.1.9 to 5.1.12 Heat Sink Requirements - General	The primary and back-up heat sinks for Units 1,4 and 5-8 are identified in [B.15-6], [B.15-7]. They do not rely on safety system initiation and do not impact on the reliability of engineered safety features in the plant.	Compliant

CSA N290.11-13 Clause	PSR2 Review	Compliant or Gap
	The requirements for demonstrating heat sink capability are identified in Section 1.2.2 of [B.15-5]. The emergency heat sinks used at Pickering, the boilers supplied by water from either the EWS, i.e. Group 2 for Units 5-8 and the EBWS for Units 1-4, have redundancy by design.	
5.2.1 Heat Sink Operation - General	N-STD-OP-0025 [B.15-5] defines general requirements and requires that heat sink management be under control of qualified, licensed staff.	Compliant
5.2.2.1 to 5.2.2.8 Heat Sink Operation - Recall	Sections 1.2.2 of [B.15-5] defines the basic requirements for back-up and emergency heat sink recall times. Further detailed requirements and practical considerations are contained in Section 1.3.1, including the requirement to ensure recall times can be achieved by careful work planning, considering conditions when implementing the back-up/emergency heat sink and ensuring there is margin in the required recall times. It also requires a test of the recall strategy where a significant recall margin does not exist (Section 1.3.1 (c) (2) (iii)). For Units 1,4 Section 4.2 of [B.15-6] and 4.2.4 of [B.15-7] for Units 5-8, document the process for calculating and documenting heat sink recall times.	Compliant
5.2.2.9 to 5.2.2.13 Heat Sink Operation - Recall	There are approved procedures for implementing back-up and emergency heat sinks. They have been devised to make use of available engineered heat sinks and methods of circulation which minimize the use of manual actions, most of which are completed from the Main Control Room (MCR). The consequences of failure of not placing a heat sink in service prior to the recall time have been assessed in the Safety Report. Emergency Operating Procedures are written to address a loss of heat sink. Recall times are calculated in Section 4.2 of [B.15-6] for Units 1,4 and 4.2.4 of [B.15-7] for Units 5-8 and take into account all relevant factors, i.e. monitoring, operator actions and other activities. The estimated time to change between various heat sinks is contained in the operating procedures.	Compliant
5.2.3 Heat Sink Operation - Actions on Failure of a Heat Sink	Section 1.6 of [B.15-5] prescribes actions that must be taken in the event of failure of process heat sinks. It details actions to take such as:	Compliant

CSA N290.11-13 Clause	PSR2 Review	Compliant or Gap
	• If failure of any heat sink occurs, work that could adversely affect the other two heat sinks shall be suspended immediately, e.g. the back-up and emergency heat sinks. • Highest priority is given to establishing a primary and back-up heat sink configuration meeting the requirements of [B.15-5]. • If both process heat sinks fail, all work affecting the emergency heat sink shall be suspended. Highest priority shall be given to re-establishing process heat sinks and preparation for implementing the emergency heat sink is also high priority. Section 5.0 of References [B.15-6] and [B.15-7] contain the detailed procedures prescribing the actions to take following process heat sink failures required by this section.	
5.2.4 Heat Sink Operation - Heat Sinks Affecting Other Work	Section 1.3.3 of P-INS-03600-00002, "Reference Guide for Reactor Safety Support of Outages" [B.15-8] outlines the process used for conducting reactor safety assessment during outages. Section 1.3.6 of this instruction describes the process followed to: • *Identify of any vulnerabilities resulting from combinations of reduced coolant inventory operation, electrical power alignments, safety system unavailability, heat sink transitions, and other abnormal system alignments.* • *Identify the need for contingency plans to mitigate the effects of an unexpected loss of function of systems maintaining shutdown safety, or heat sink function.* • *Identify the need for training and incident prevention actions to reduce risk during periods of vulnerability, when needed.* This work is performed prior to and during outages to assess heat sink and other reactor safety issues that could be affected by planned and emergent work. In addition, Section 1.2.2 (a) prescribes that special consideration be given to conditions that affect HTS integrity or relief capability, including fuel channel work, which may affect heat sink capability and recall time.	Compliant

CSA N290.11-13 Clause	PSR2 Review	Compliant or Gap
5.2.5 Heat Sink Operation - Monitoring	Section 1.4 of [B.15-5] contains the requirements for heat sink monitoring, prescribing that sufficient monitoring of reactor heat sinks shall be provided to ensure there is timely indication or warning of degradation in heat sink capability. In addition, a heat sink checksheet is used at the stations, e.g. P-FORM-10308 [B.15-11] for Units 1,4 and P-FORM-10216 [B.15-12] for Units 5-8, to routinely monitor the status of outage heat sinks and their effectiveness. HTS temperature, pressure, flow, level (if in a drained state) and D_2O storage level and cover gas pressure are monitored to ensure in-service heat sink effectiveness. Status of the back-up and emergency heat sinks are also reviewed in the form.	Compliant
5.3 Instrumentation and Control	A number of parameters using directly connected instrumentation, measure heat sink effectiveness. Trends are set-up for monitoring twice per shift, e.g. HTS temperatures, see section 4.1.4 of [B.15-7]. To provide the Authorized Nuclear Operator with enhanced monitoring when Shutdown Cooling is valved in as part of a heat sink, temperature alarm setpoints on all Shutdown Cooling Heat Exchanger inlet temperature elements are lowered to 5 degC above the current HTS temperature. This allows for early warning of HTS temperature increases (see Section 4.1.4 of [B.15-7]). For other heat sinks, alarms occur on failure of the equipment resulting in the loss of heat sink, e.g. HTS pump failure, failure of a service water pump, or other supporting component failure. Neutronic power trips are in place for heat sinks while not in the Guaranteed Shutdown State, as well as for some heat sink combinations while in a Guaranteed Shutdown State.	Compliant
5.4 Containment Boundary	This requirement is addressed in the PSR2 CSA N290.3 code review.	Compliant
5.5 Loop Isolation	The clauses identify that if loops are isolated, then they should be treated independently, in terms of heat sink requirements. Pickering HTS loops are usually isolated for outage heat sinks and the same type of heat sink is used for both HTS loops. This is done to simplify management of the heat sink strategy. The credited heat sinks in each loop ensure adequate cooling.	Compliant

CSA N290.11-13 Clause	PSR2 Review	Compliant or Gap
5.6.1 Reliability - General	There is a requirement that a design reliability be established for each outage heat sink. The process heat sinks used during outages have been designed to remove heat while the reactor is at higher thermal power conditions. The main process heat sink designed for shutdown states is the Shutdown Cooling System, though additional systems (e.g., air conditioning units, boilers/Steam Reject Valves (SRVs), boiler blowdown, bleed cooler) are credited as outage heat sinks, and meet the quality requirements for their application. The emergency heat sinks used at Units 1,4 and Units 5-8, i.e., EBWS and EWS respectively, are the same as those credited for higher power operations. The reliability target exists for Systems Important to Safety per Reference [B.15-13] and the on-demand (i.e., standby for HTS cooldown) unavailability targets for SDC at both stations is included in the Annual Reliability Report [B.15-14]. For Pickering 5-8, the SDC (in normal heat sink mode) unavailability requirements were established in the Safety Design Matrix (SDM) Study [B.15-15]. Although, no formal SDM study was completed for Pickering A, the SDC systems have a very similar configuration between the two stations. Assessment of outage heat sink unavailability is completed as part of outage risk management planning per [B.15-16] from an instantaneous, integrated plant perspective. Although reliability targets are established for some outage heat sinks (e.g., SDC), other credited outage heat sinks do not have a design unavailability or reliability. This issue relating to heat sink design reliability for Pickering Units 1,4 and 5-8 is identified as **PSR2 CSA N290.11-13 Gap #2.** Reliability of all outage heat sinks (including those without explicit targets) is managed through the Risk and Reliability Program (both through unavailability models as well as through PSA), hence the reactor safety impact is assessed and monitored. This clause also has a requirement that the designed reliability for process heat sinks be consistent with the AOO frequency limits (i.e., the emergency heat sink should not be relied on for an AOO). This cannot be verified because the AOO events relating to outage heat sinks have not been identified or analyzed in the current Pickering Safety Reports. However, as described above, heat sink configuration reliability (fuel and core damage frequency) is assessed as part of integrated outage risk monitoring [B.15-16], the results of which are input to	Gaps **PSR2 CSA N290.11-13 Gap #2** **PSR2 CSA N290.11-13 Gap #3**

CSA N290.11-13 Clause	PSR2 Review	Compliant or Gap
	ongoing outage decision-making based on established criteria. This issue relating to heat sink design reliability being required to be consistent with AOO frequency is identified as **PSR2 CSA N290.11-13 Gap #3.**	
5.6.2 Reliability Analysis	Risk models have been developed to enable the tracking of outage risk [B.15-16], [B.15-17].	Compliant
5.6.3 Redundancy	Redundancy for single failures is provided by having primary and back-up heatsinks rather than having redundancy for single failures within a heat sink. Redundancy for the Ultimate Heat Sink to the lake or atmosphere is always provided either via multiplicity of heat exchangers rejecting heat to service water or boilers and SRVs to the atmosphere.	Compliant
5.6.4 Reliability - Diversity	Section 1.5.1 of [B.15-5] requires diversity between primary and back-up heat sinks. Where diversity is not possible, the probability of consequences of losing both heat sinks shall be assessed as being acceptable. The outage reactor safety assessment performed as part of Reference [B.15-8] addresses this. Both Pickering Units 1,4 and 5-8 have a number of heat sinks that are diverse and can be used under different HTS conditions.	Compliant
5.7 Independence and Separation	Section 1.5.1 of [B.15-5] requires independence between the primary and back-up heat sinks. It specifies among other requirements that there shall be no normally anticipated single failures that could fail both the primary and back-up heat sinks. Section 1.5.2 of [B.15-5] specifies the requirement that the emergency heat sink be independent from the process heat sinks. For both Units 1,4 and 5-8 the Operating Manuals contain requirements for primary and back-up heat sink independence, e.g. section 4.3.3 in Reference [B.15-6]. Heat sink tables are provided in the Operating Manuals (e.g. Section 4.2.1 in [B.15-7]), which specify permissible combinations of heat sinks, based on independence and other relevant factors. Separation requirements are met in part by ensuring independence between heat sinks. In addition, primary heat sink equipment and components are designated and	Compliant

CSA N290.11-13 Clause	PSR2 Review	Compliant or Gap
	protected with signs and barriers as required per P-INS-09000-00003, "Protected Equipment Zones" [B.15-18]. Separation between process heat sinks and emergency heat sinks at Pickering is assured by providing an emergency heat sink that is Group 2 in Units 5-8, and EBWS for Units 1,4 (supplied by Units 5-8 service water pumps and power supplies).	
5.8 Pressure-retaining SSCs	The requirement of this section is addressed separately in the PSR2 code review for CSA N285.0.	Compliant
5.9 Environmental Qualification	Equipment required to function after a DBA in the outage unit is environmentally qualified per the requirements of N-PROG-RA-0006, "Environmental Qualification" [B.15-19]. A separate PSR2 code review has been prepared for CSA N290.13. The requirement on seismic qualification is met by having emergency heat sinks capable of withstanding the impact of a Design Basis Earthquake. The seismic heat sink for Units 5-8 is specified in Section 4.3.4 in [B.15-7] and in Section 4.4.2.2 in [B.15-6] for Units 1,4. Aging Management requirements are addressed in N-PROG-MP-0008, "Integrated Aging Management" [B.15-20].	Compliant
5.10 Dynamic Piping Effects	The requirements of this section are met by having emergency heat sinks at Pickering that are separate from the process heat sinks, i.e. Group 2 in Units 5-8 (EWS), and EBWS for Units 1,4 (supplied by Units 5-8 service water pumps and power supplies).	Compliant
5.11 Maintenance	Section 1.3.1 (b) and (c) of Reference [B.15-5] provide the requirements for allowable work and processes to follow on primary and back-up heat sinks. For example, Section 1.3.1 (b) only permits cursory work be performed on the primary heat sink. Section 1.3.1 (c) provides a number of requirements for allowable work on back-up heat sinks, e.g. work on back-up heat sink components requires the component to be placed in a safe state. Section 1.3.1 (d) of [B.15-5] contains requirements for maintenance work on the emergency heat sink. Pickering Units 1,4 Operating Manual [B.15-6] Section 4.5 and Units 5-8 Operating Manual [B.15-7] Section 3.4.1.1 contain additional detailed requirements associated with	Compliant

CSA N290.11-13 Clause	PSR2 Review	Compliant or Gap
	maintenance work on heat sinks. These references address all of the requirements of Section 5.11 of CSA N290.11. In addition, Reference [B.15-8] on Reactor Safety support of outages assesses the risk posed by work performed on outage heat sinks and other outage scope to ensure public safety risk is acceptable.	
5.12 Functional Testing	The heat removal capabilities of the Pickering heat sinks have been assessed and the applicable limits are contained in the Pickering Operating Manuals, for example, Section 4.1.1 in [B.15-7]. Generally, the process heat sinks are designed for much greater heat removal during higher power operation, e.g. boilers/Steam Reject Valve, shutdown cooling. Testing and verification requirements for process heat sinks are outlined in Operating Manuals for Units 1,4 in Section 3.4.12 [B.15-6] and Section 3.4.15 for Units 5-8 [B.15-7]. Testing for transfers from one heat sink to another is outlined in Section 7.0 of the Operating Manuals; supplementary Operating Procedures are referenced in Section 10 of the OMs and in various parts of Section 4 and 5 of the OMs. Confirmation of heat sink function and performance is by completion of the heat sink checklists ([B.15-11], [B.15-12]) and confirming conditions are acceptable. Section 1.3.1 of [B.15-5] requires post-maintenance testing of any work done on a credited heat sink prior to it being placed in-service. A separate PSR2 code review of CSA N290.0 has been performed.	Compliant
5.13 Documentation	Documentation of requirements for heat sinks and operating procedures for their implementation are covered in References [B.15-5], [B.15-6] and [B.15-7].	Compliant
5.14 Heat Sink Support System Requirements	Requirements for systems used to support heat sink operation are generally the same as for the process heat sink, e.g., electrical system independence. A separate PSR2 code review has been prepared for CSA N290.5 on support systems.	Compliant

B.15.3 Compliance Summary for Pickering PSR2

There are three PSR2 CSA N290.11-13 gaps which relate to Safety Factor 1 (Plant Design):

1. The CSA N290.11-13 Clause 5.1.1 to 5.1.5 requirement for back-up heat sinks to mitigate the conditions following an Anticipated Operational Occurrence (AOO) is not specified in governance/procedures. Loss of a division of power, a single component failure, etc., which are likely to be in the set of AOOs, are accounted for in the specification of heat sinks. However, AOOs have not been identified and analyzed in the current Pickering Safety Reports. This issue is therefore a PSR2 gap. It is being addressed as part of REGDOC-2.4.1 implementation.

2. Clause 5.6.1 of CSA N290.11-13 requires design reliability to be established for outage heat sinks. Although some emergency heat sinks (e.g., Emergency Boiler Water Supply and Emergency Water Supply) have design reliability requirements, design reliability requirements have not been established for all normal and back-up heat sinks used at Pickering. Reliability of all outage heat sinks (including those without explicit targets) is managed under the Risk & Reliability Program (both through unavailability models as well as through Probabilistic Safety Assessment), hence reactor safety impact is assessed and monitored. However, there is a PSR2 gap with respect to establishment of design reliability requirements for Pickering Units 1,4 and 5-8 outage heat sinks.

3. Clause 5.6.1 of CSA N290.11-13 requires that the designed reliability for process heat sinks be consistent with AOO frequency limits, such that an emergency heat sink does not need to be used for an AOO. AOOs have not been identified and analyzed in the current Pickering Safety Reports. This issue is therefore a PSR2 gap and is being addressed as part of REGDOC-2.4.1 implementation.

B.15.4 References

[B.15-1] CSA Standard, N290.11-13, *Requirements for Reactor Heat Removal Capability During Outage of Nuclear Power Plants*, December 2013.

[B.15-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook,* December 2015.

[B.15-3] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.15-4] CSA Impact Statement for Public Review (Draft), *Notification of CSA N290.11 Publication; Product: New Standard*, Date not provided.

[B.15-5] OPG Standard, N-STD-OP-0025 R003, *Heat Sink Management*, October 2014.

[B.15-6] OPG Manual, NA44-OM-14-04300-01 R007, *Shutdown Heatsinks*, April 2016.

[B.15-7] OPG Manual, NK30-OM-5-04300-01 R030, *Shutdown Heat Sinks*, February 2016.

[B.15-8] OPG Instruction, P-INS-03600-00002 R000, *Reference Guide for Reactor Safety Support of Outages*, September 2014.

[B.15-9] OPG Report, NK30-REP-03680-00015 R000, *Pickering NGS-B Integrated Safety Review (ISR) Final ISR Report*, September 2009.

[B.15-10] OPG Plan, NK30-PLAN-00531-00001 R000, *Pickering B Continued Operations Plan*, September 2010.

[B.15-11] OPG Form, P-FORM-10308 R018, *Shutdown Heat Sinks Checksheet (Pickering A)*, June 2016.

[B.15-12] OPG Form, P-FORM-10216 R017, *Shutdown Heat Sink Check Sheet PNGS-B*, June 2015.

[B.15-13] OPG Standard, N-STD-RA-0033 R003, *Reliability Monitoring and Reporting of Systems Important To Safety*, April 2016.

[B.15-14] OPG Report, P-REP-09051.1-00015 R000, *Pickering NGS – 2015 Annual Risk and Reliability Report*, March 2016.

[B.15-15] OPG Correspondence, NK30-CORR-N0360008, *Pickering Generating Station 'B' Safety Design Matrix Study – Loss of Shutdown Cooling (Normal Mode)*, August 1981.

[B.15-16] OPG Standard, N-STD-RA-0030 R004, *Risk Management for Outage Planning and On-line Maintenance*, August 2016.

[B.15-17] OPG Report, P-REP-03611-00006 R000, *Pickering NGS PSA Update to Include Enhancements from the Fukushima Integrated Action Plan*, April 2014.

[B.15-18] OPG Instruction, P-INS-09000-00003 R000, *Protected Equipment Zones*, April 2014.

[B.15-19] OPG Program, N-PROG-RA-0006 R008, *Environmental Qualification,* May 2015.

[B.15-20] OPG Program, N-PROG-MP-0008 R006A, *Integrated Aging Management*, October 2015.

B.16 CSA N290.14-15, "Qualification of Digital Hardware and Software for Use in Instrumentation and Control Applications for Nuclear Power Plants"

B.16.1 Background

The following, paraphrased from the preface and scope of CSA N290.14-15 [B.16-1], provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *The purpose of CSA N290.14 is to establish a qualification process for digital instrumentation and control systems and components for use in nuclear power plants [NPPs] and provides guidance for maintaining qualification once it has been established. It addresses application-specific qualification by outlining a set of qualification concerns and failure modes that allow candidate products to be assessed within the context of their application.*
>
> *This standard does not apply to business systems (e.g., business applications, desktop computers, email, business networks), analysis software (e.g., scientific, engineering, and safety analysis software), or passive devices (e.g., wires), unless they are part of a safety-related computing system.*

CSA N290.14-15 is applicable to Safety Factor 1 (Plant Design).

Compliance with CSA-N290.14 is not currently a licence requirement for Pickering NGS (in accordance with PROL 48.02/2018) per the R04 Pickering Licence Conditions Handbook [B.16-2]. CSA N290.14-15 is the second edition of this standard, and supersedes the previous edition published in 2007 (and reaffirmed in 2012) [B.16-3] under the title "Qualification of Pre-Developed Software for Use in Safety-Related Instrumentation and Control Applications in Nuclear Power Plants". The publication notice for the new edition, [B.16-4], identifies the following significant changes from the previous edition:

1. *Expanded scope to include all aspects of digital hardware and software qualification. The user has new guidance for qualification of all aspects of the digital product. This includes custom software, pre-developed software, digital hardware, and software engineering tools.*

2. *The addition of the term digital items (custom software, pre-developed software, digital hardware, and software engineering tools). An informative annex is also added in this standard to give examples of how to identify digital items in a candidate product.*

3. *Added integration concerns of digital items in Annex. This informative annex reflects industry OPEX and best practices when dealing with digital upgrades in the nuclear power plants.*

4. *Added proof through testing as a qualification option for pre-developed software. The proof through testing method may be applied to pre-developed software digital items that are unmodified, and are to be used in Category 3 applications. This adds flexibility for the qualification process when no other method can be used to qualify the*

candidate product. This method is meant for low complexity products that run with discrete executions or used continuously while the system is online and that the manufacturer is not willing to provide information on how the product was developed (e.g. boot firmware, disk controllers, etc.).

5. *Added an informative annex on how to qualify software engineering tools. Software engineering tools used for the configuration, maintenance of digital items, or the development of custom software will now require qualification.*

6. *Guidance regarding emerging I&C [Instrumentation & Control] technology applications [Field Programmable Gate Arrays (FPGA) and Wireless]. Users are required to develop a wireless co-existence management plan and application policy for the use of wireless safety-related applications. Users should proceed with thoughtfulness to ensure the use of wireless systems uphold defense in depth, reliability and security requirements. Capital investment and maintainability costs for the usage of wireless technology may be considered high (>1M) based on lack of industry for safety related applications and OPEX [Operating Experience]. Users are given guidance on the qualification expectations for FPGA applications. FPGAs are required to be qualified with respect to both hardware and software.*

7. *The recognized program list has been updated. Some utility processes which reference withdrawn programs will have to be revised.*

The publication notice further notes that the revised standard reflects current practice at Bruce Power, OPG, and Candu Energy, and does not add new demands on these users [B.16-4].

As identified in Reference [B.16-5], the Pickering PSR2 review of CSA N290.14-15 is a High Level review. For a PSR2 High Level review, the degree of conformance with clauses or groups of clauses in the Law, Regulation, Code or Standard (L/R/C/S) is demonstrated by supporting evidence stating whether the intent of the requirements stipulated in the requirement document is met. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the intent of the safety requirement is met.
- Gap: A Gap indicates that the intent of the safety requirement is not met.

B.16.2 Compliance Assessment for Pickering PSR2

B.16.2.1 Application of PSR1 Reviews

There have not been any programmatic or system/application PSR1 reviews against CSA N290.14 for Pickering or Darlington.

B.16.2.2 Application of Post PSR1 Reviews

The 2007 version of N290.14-07 (R2012) [B.16-3] was introduced into OPG governance in 2009 (i.e., Revision 4 of N-PROC-MP-0049, "Procurement of Software and Products Containing Software" [B.16-6]). Hence, it was not possible for software procured prior to this time to be directly compliant with N290.14. Software applications introduced into safety related systems prior to 2009 were qualified in accordance with OPG governance based on industry standards such as those developed by CANDU Computer Systems Engineering Centre of Excellence, CANDU Owners Group (COG) and International Electrotechnical Commission (IEC).

Qualification of Legacy Systems

This code review needs to consider the level of compliance for systems and applications that are within the scope of the PSR. This is defined in PSR2 Basis Document [B.16-5], Tables B1 and B2. Pickering Units 1, 4 and Pickering Units 5-8, have several legacy installations with Real-Time Process Computing (RTPC) on Safe Operating Envelope (SOE) systems and Systems Important to Safety (SIS) that are included in the PSR2 Assessment Basis [B.16-5], Tables B1 and B2, that pre-date the adoption of N290.14.

Examples of applications that have received formal software categorization, but pre-date the adoption of CSA N290.14-07 in 2009 include:

- Critical Safety Parameter displays
- Data Extraction Systems
- Various microprocessor controllers used in Fuel Handling, Boiler Level Control, Class III/II/I power distribution, Standby Generator control and test loops.

It is difficult to retroactively qualify these applications to N290.14. The qualification of these systems was based on the best-industry practice of the day and is supported by extensive operating experience and successful proven in-service application over many years.

Further, recent correspondence with the CNSC [B.16-7] identifies all of the software application qualifications for software Categories 1, 2 & 3 from January 1, 2007 to the time of the correspondence (June 2016), noting that software qualification in accordance with N290.14-07 was not integrated into OPG governance before May of 2009. Therefore, for any systems containing pre-developed software that were modified after 2007, there is evidence of software qualification per Reference [B.16-7]. In addition, any modifications of RTPC custom software for pre-existing systems are performed per N-PROG-MP-0006 [B.16-8], which outlines qualification requirements in accordance with N290.14-07.

An evaluation of legacy Real-Time Process Computing applications with respect to the requirements of N290.14-15 for Categories 1, 2 & 3 software has not been performed. This is identified as a PSR2 gap (**PSR2 N290.14-15 Gap #1**).

High Level Code Review

As discussed above a compliance review of CSA N290.14-15 was not undertaken as part of previous PSR1 reviews and, therefore, the following High Level assessment has been completed. In the review below, the degree of conformance with clauses or groups of clauses in the L/R/C/S is assessed for Pickering NGS by reference to supporting evidence stating whether the intent of the requirements stipulated in the L/R/C/S is met. The High Level review below assesses the degree of conformance of OPG's current governance and procedures with respect to N290.14, independent of specific applications.

N290.14-15 Clause	PSR2 Review	Compliant or Gap
0. Introduction	No requirements specified	N/A
Figure 1 Process flow diagram	No requirements specified	N/A
Scope	No requirements specified	N/A
1.1 This Standard defines requirements for the process of qualification of digital hardware and software for use in instrumentation and control applications for NPPs. **Notes:** **1)** *This Standard applies to individual safety-related programmable digital devices containing software or programmable logic (e.g., devices such as application-specific integrated circuit (ASICs), complex programmable logic device (CPLD), and field-programmable gate array (FPGAs)).* **2)** *This Standard may provide guidance for nuclear facilities other than NPPs, using a graded approach.*	The OPG governance relating to software N-PROG-MP-0006, "Software" [B.16-8], does not explicitly identify that it is applicable to digital hardware. However, digital hardware is addressed via a combination of N-PROG-MP-0009, "Design Management" [B.16-9], N-PROG-MP-0001, "Engineering Change Control" (ECC) [B.16-10], and Software [B.16-8] programs. Hardware is explicitly addressed in OPG design standards required by the implementing procedures for these OPG programs. These standards include N-STI-69000-10013, "Computer System Requirements and Design" [B.16-11] and N-STI-69000-10005, "Category II/III Software Testing" [B.16-12]. OPG governance does not reference the 2015 version and only refers to the 2007 version of the Standard. However, since N290.14-15 is not currently a licensing requirement, this is not a PSR2 gap.	Compliant
1.2 This Standard refers directly to other industry standards for topics related to the categorization of functions, hardware qualification aspects, and software qualification aspects.	No requirements specified	N/A
1.3 This Standard does not apply to business systems (e.g., business applications, desktop computers, email, business networks), analysis software (e.g., scientific, engineering, and safety analysis software), or passive devices (e.g., wires), unless they are part of a safety-related computing system. **Note:** *For requirements related to analysis software refer to CSA N286.7.*	No requirements specified	N/A

N290.14-15 Clause	PSR2 Review	Compliant or Gap
1.4 Annex E, provides guidance that focuses on activities unique to the integration of a digital item, based on recent best practice and operating experience. **Note:** *Refer to CSA N290.12 for integration concerns with respect to human factors.*	No requirements specified	N/A
1.5 This Standard assumes that the candidate product has been previously assessed as functionally suitable for the proposed application. **Note:** *Functional suitability is a determination of the degree to which a product can meet the specified requirements including confirmation that the use of the digital item does not conflict with the requirements of the application. This Standard is* ***used to qualify the product*** *after this determination is made.*	The Software [B.16-8], Design Management [B.16-9], Configuration Management [B.16-13], and ECC [B.16-10] programs ensure a suitable product is procured. N-PROC-MP-0049, "Procurement of Software and Products Containing Software" [B.16-14], addresses this for new installations and component replacements.	Compliant
1.6 In this Standard, "shall" is used to express a requirement, i.e., a provision that the user is obliged to satisfy in order to comply with the standard; "should" is used to express a recommendation or that which is advised but not required; and "may" is used to express an option or that which is permissible within the limits of the Standard…	N/A - Clarification	N/A
2. Referenced publications 3. Definition and Abbreviations 4. Referenced requirements	N/A – no requirements	N/A
5. General requirements	Heading only	N/A
5.1 Definition of candidate product scope The scope of the candidate product shall be clearly identified, including the following: a) clear definition of which components are included and which are not; b) description of applicable functions, including safety functions, implemented by the candidate product; c) description of applicable system interfaces; and	As part of N-PROC-MP-0090, "Modification Process" [B.16-15], N-FORM-10959, "Design Scoping Checklist" [B.16-16] is completed. Section 2.7 of N-FORM-10959 directs the preparer to identify RTPC implications for the design change. If there is RTPC software involved in the change, N-PROG-MP-0006 will require either procedure N-PROC-MP-0099, "Development of Real-time Process Computing Systems" [B.16-17] or N-PROC-	Compliant

N290.14-15 Clause	PSR2 Review	Compliant or Gap
d) identification of any restrictions or conditions related to the use of the candidate product in the proposed NPP application. Note: *Typically the candidate product is part of a larger system. It is important therefore to define the scope that is to be considered as part of the qualification activities.*	MP-0049, "Procurement of Software and Products Containing Software" [B.16-14], to be followed. The bulleted requirements are addressed in the Computer System Requirements (CSR) and Computer System Design (CSD) in accordance with N-STI-69000-10013, "Computer System Requirements and Design" [B.16-11].	
5.2 Categorization of the function 5.2.1 Functions that are intended to be implemented by the candidate product shall be identified and assigned a safety category in accordance with a standard selected from the reference standards given in Table 1. The safety category of the proposed application shall be determined using the most safety-significant function.	The details provided in Table 1 of CSA N290.14-15 are the same as those detailed in OPG Standard, N-STI-69000-10000, "Software Categorization" [B.16-18]. Note, the categorization in this standard does not apply to hardware requirements.	Compliant
5.3 Personnel qualification The personnel performing a qualification shall collectively demonstrate a) experience in computer engineering (which includes software and hardware),and quality assurance (QA) as they pertain to safety-related instrumentation and control (I&C) applications in NPPs; b) familiarity with relevant standards; and c) an understanding of the target application's impact on safety actions and safety improvements.	a) This qualification and expertise resides under the "Computers and Control Design Department" under the Nuclear Engineering organization. All training requirements and qualifications are maintained in the Training Information Management System (TIMS) II. The core and supplementary training for Engineering support is documented in, N-TQD-403-00001, "Nuclear Engineering Support Personnel Training and qualification Description" [B.16-19]. Additional qualification requirements are documented in N-QA-403-00008 R009, "Nuclear Design Engineering Qualification Guide" [B.16-20]. In addition to the core training, Appendix B of N-TQD-403-00001 identifies the Program Element (PEL) ID 65690, "Real-Time Computer Based Systems and Software Governance – CAL" [B.16-19]. Appendix A of N-QG-403-00008 specifies specific duty area training for staff in the Computers and Control Design Department, including PEL ID 28963, "DE Computer Evaluation – Real Time Software (General)" [B.16-20].	Compliant

N290.14-15 Clause	PSR2 Review	Compliant or Gap
	As part of work assignment, the supervision ensures the personnel are qualified to perform the assigned task. Specific applicable qualifications are [B.16-20], [B.16-21]: • Qual ID 16934, Software Categorization Specialist • Qual ID 7771, DE- Safety Critical Software Engineering • Qual ID 7772, DE-Real Time Computer Systems (Hardware) • Qual ID 6286, DE-Real Time Software In addition to these qualification IDs, required core qualifications include [B.16-19], [B.16-20]: • Qual ID 6168 Engineering Support Personnel - Core Training • Qual ID 33685 Design Engineer N-INS-69000-10002, "Computer Spare Parts and Other Electronic Components Acquisition by Computers and Control Design Department" [B.16-22], Section 4.1 identifies the specific qualifications required for computer related acquisitions (included in the above). b) The applicable standards are identified in the relevant STIs and N-PROC-MP-0099, "Development of Real-Time Process Computing Systems" [B.16-17]. c) This is outlined in Section 3, 4, & 5 of the STI, N-STI-69000-10000-R004, "Software Categorization" [B.16-18], where the standard requires the user to understand the purpose and significance of the application being evaluated. Sections 4.3.1 and 4.3.2 of N-PROC-MP-0099 [B.16-17], provide a detailed list of Performance and Developmental references.	
6 Qualification activities	Heading only	N/A

N290.14-15 Clause	PSR2 Review	Compliant or Gap
6.1 Identifying and classifying digital items	Clarification only. Section 6.1 is a new addition to the 2015 version of the standard that did not explicitly exist in the R2012 version of the standard.	N/A
6.1.1 Comprehensive list A comprehensive list of all digital items within the candidate product shall be compiled. **Note:** *Annex C provides some examples of identification of digital items found in a candidate product.* Annex C is not mandatory. It identifies six general categories: 1) General 2) Smart Transmitter 3) PLC Controller 4) PC-based solution 5) Distributed control System 6) Custom FPGA-based device And, provides advice with respect to the listing.	Depending on the type of application and its Categorization, the level of detail required differs, per N-PROC-MP-0099, "Development of Real-Time Process Computing Systems" [B.16-17], Sections 1.3 and 1.4, (also including custom applications) which specify the standards and requirements that are applicable. For example, N-STI-69000-10003, "Category II / III Software Design" [B.16-24], Section 4.2.3.3 details the types of modules that should be identified. For the software, N-FORM-10409, "Software Quality Assurance Requirements" [B.16-23] identifies the quality assurance requirements for Category I/II/III software, which is in accordance with N290.14, consistent with the 2015 version of the standard. Hence, the 2015 code requirements are met.	Compliant
6.1.2 Identifying item types Each digital item listed in accordance with Clause 6.1.1 shall be identified as one of the following: a) pre-developed software; b) custom software; c) digital hardware; or d) software engineering tool.	The R2012 version of the standard was only applicable to pre-developed software. The 2015 version has expanded the scope to the four identified applications. Item d) *software engineering tool*, has been introduced. This refers to tools used in software testing and development and differs from Scientific, Engineering, and Safety Analysis (SESA) software. The procedures and guidance detailed in clause 6.1.1 address all these aspects.	Compliant
6.1.3 Minor customizations	Heading only	N/A
6.1.3.1 Software with minor customizations may be considered to be pre-developed software digital items.	This is recognized in OPG procedures N-PROC-MP-0049, "Procurement of Software and Products Containing Software" [B.16-14], as Modifiable Off-The Shelf (MOTS).	Compliant

N290.14-15 Clause	PSR2 Review	Compliant or Gap
6.1.3.2 Where a pre-developed software digital item is modified with minor customizations, complementary testing shall be performed on the customizations in order to verify that they do not invalidate any qualification-related conclusions. Failures uncovered during the testing shall be subject to impact analysis, corrective action, and retesting. **Note:** *The scope of testing should be justified based on the impact that the modifications have on the candidate product's ability to meet its functional and safety requirements.*	This is identified in N-FORM-10408, "Software Procurement Planning" [B.16-25], where the Section F selection directs to the mandatory software qualification requirements as listed in Appendix B of N-PROC-MP-0049, "Procurement of Software and Products Containing Software" [B.16-14]. The MOTS software are directed to the Custom (CUST-I/II/III/IV) software quality assurance requirements.	Compliant
6.2 Qualification concerns The candidate product shall be assessed against the concerns identified in Annex A. Annex A is mandatory and includes addressing the following: A.1 General A.2 Silent or undetected failure A.3 Flooding, determinism, and performance A.4 Common mode A.5 Security A.6 Power interruption or restart A.7 Time Dependant behaviour A.8 Modal behaviour A.9 Shared resources A.10 Upgrades A.11 Maintainability issues A.12 Extra functionality A.13 Communications A.14 Coexistence A.15 User interface A.16 Other postulated failure modes	N-PROC-MP-0049, "Procurement of Software and Products Containing Software" [B.16-14], Appendix B identifies that the supplier must qualify software in accordance with N290.14 for Commercial off the Shelf (COTS-I/II/III) and Custom (CUST-I/II/III) software. This list of qualification concerns is essentially the same in the 2012 and 2015 versions of N290.14. However, the requirement to qualify for A.14 'Coexistence' relating to wireless technology has been introduced in the 2015 version. N-PROC-MP-0103, "Security for Real-Time Process Computing Systems" [B.16-26], Section 1.3 applies restrictions on the use of wireless technology and requires a Wireless Technology Risk Assessment to document how potential hazards are mitigated when wireless technology is used. In addition, N-STI-69000-10013, "Computer System Requirements and Design" [B.16-11], requires environmental requirements to be specified (e.g., electromagnetic interference or radio frequency interference). Guidance on transient immunity testing is provided in N-DG-60407-10000, "Guidelines for Electromagnetic Compatibility Test" [B.16-27]. Additionally, A.13 related to 'communications' has been expanded to identify that communication must be c) robust under credible environment and d)	Compliant

N290.14-15 Clause	PSR2 Review	Compliant or Gap
	compliant with a recognized and appropriate standard for digital communication. For c), N-STI-69000-10013, "Computer System Requirements and Design" [B.16-11], requires that the environmental conditions for qualification be established. N-PROC-MP-0049, "Procurement of Software and Products Containing Software" [B.16-14], Section 1.6.1 ensures supplier tests for target installation. For d) digital communication protocol would be identified in the specification for procurement as a standard practice. Hence, neither of the changes relating to A.13 'communication' are gaps.	
6.3 Failure analysis requirements	Heading only	Compliant
6.3.1 For Category 1 applications, failure modes shall be identified and analyzed for the candidate product, using a rigorous technique (e.g., FMEA, fault tree analysis (FTA), Hazard and operability analysis (HAZOP), Systems theoretic process analysis (STPA), etc.). **Note:** *For guidance on failure analysis techniques recognized standards such as the following may be used:* a) *IEC 62502 for event tree analysis;* b) *IEC 60812 for failure mode and effects analysis;* c) *IEC 61882 for hazard and operability studies;* d) *IEC 61025 for fault tree analysis; and* e) *EPRI 3002000509 for hazard analysis.*	There are currently no Category I applications in use at Pickering. However, this requirement is reflected in N-STI-69000-10013, "Computer System Requirements and Design" [B.16-11], Section 5.3.23, 6.3.4.1(f) and CE-1001-STD, "CANDU Computer Systems Engineering Centre of Excellence Standard for Software Engineering of Safety Critical Software" [B.16-28], Section 4.4.1. Failure analysis requirements are cited in the 2012 version of the standard referenced in OPG governance (e.g., N-PROC-MP-0049, "Procurement of Software and Products Containing Software" [B.16-14], Appendix B). The 2015 version has only introduced the assessment techniques and references.	N/A
6.3.2 For candidate products that are determined to perform Category 1 functions, or where candidate product complexity warrants, more than one complementary technique should be used.	See 6.3.1	N/A

N290.14-15 Clause	PSR2 Review	Compliant or Gap
6.3.3 For Category 2 and 3 applications, failure modes should be identified and analyzed for the candidate product, using a rigorous technique (e.g., FMEA, FTA, HAZOP, STPA, etc.).	See 6.3.1	Compliant
6.4 Digital item qualification activities	Heading only	N/A
6.4.1 Pre-developed software	The qualification approaches for pre-developed and custom software are specified per either procedure N-PROC-MP-0049, "Procurement of Software and Products Containing Software" [B.16-14] or N-PROC-MP-0099, "Development of Real-time Process Computing Systems" [B.16-17]. These procedures and associated forms specify the qualification approaches for the two types of software under the applicable Standards. Note, the 2007 version of the standard and its application is referenced as required in OPG governance.	Compliant
6.4.1.1 General Each pre-developed software item shall be assessed using one of the methods for qualification specified in Clause 6.4.1.2 to ensure reliable performance of the software's safety-related functions. **Notes:** **1)** *It is intended that the assessment specified in this Clause will help ensure that the software does not contain bugs or flaws that could impair the software's ability to perform its function in the intended application or to handle predictable hardware and system interface errors.* **2)** *Table 2 provides guidance for selecting a method for qualification based on the category of the application.*	See 6.4.1	Compliant
6.4.1.2 Methods for qualification	Heading only	N/A
6.4.1.2.1 Recognized program	See 6.4.1	Compliant

N290.14-15 Clause	PSR2 Review	Compliant or Gap
6.4.1.2.2 Mature product method	See 6.4.1	Compliant
6.4.1.2.3 Proof through testing The proof through testing method may be applied to pre-developed software digital items, and that are to be used in Category 3 applications. This method may be used when either Item a) or b) below is true: …	See 6.4.1. This clause is a new addition to the 2015 version of the standard. It is intended to provide more flexibility for pre-developed software and does not impose more stringent requirements.	Compliant
6.4.1.2.4 Preponderance of evidence The preponderance of evidence method may be applied to pre-developed software digital items to be used in Category 1, 2, or 3 applications. This method shall consist of one or more of the following elements. The following elements shall be considered for this method:…	See 6.4.1	Compliant
6.4.2 Custom software	Heading only	N/A
6.4.2.1 General Custom software shall be developed in accordance with an appropriate standard or guideline as listed in Table 3.	See 6.4.1	Compliant
6.4.2.2 Documentation All documents required by the appropriate standard or guideline shall be available so that conformance can be confirmed.	See 6.4.1	Compliant
6.4.3 Digital hardware	As identified in Clause 1.1 above, Digital Hardware is a new addition to the standard which is addressed via OPG programs and design standards.	Compliant
6.4.3.1 General Hardware requirements shall be specified consistent with CSA N290.8. **Notes:** **1)** *For the purpose of qualifying digital hardware, the relationship of this Standard to CSA N290.8 is that the requirements for the digital hardware are specified in accordance with CSA N290.8, and this Standard identifies assessments to be conducted to verify compliance with the stated requirements.* **2)** *For the purpose of supplying digital hardware this Standard assumes that the supplier is qualified to supply the scope of*	Note, N290.8 is generic and applicable to developing specifications for any nuclear component. It is not specific to digital hardware. N-PROC-MP-0090, "Modification Process" [B.16-15] is applicable to digital hardware design and changes. If a digital component contains software N-PROC-MP-0049, "Procurement of Software and Products Containing Software" [B.16-14], is initiated. N-INS-69000-10002, "Computer Spare Parts and Other Electronic Components	Compliant

N290.14-15 Clause	PSR2 Review	Compliant or Gap
work consistent with the requirements in CSA N286.	Acquisition By Computers and Control Design Department" [B.16-22], details the managed process for procuring hardware and interfaces with N-PROC-MP-0098, "Procurement Engineering Activities" [B.16-29], which ensures that the Quality Control and Quality Assurance requirements for hardware are adhered to. Hardware is specified in accordance with N-STI-69000-10013, "Computer System Requirements and Design" [B.16-11].	
6.4.3.2 Hardware assessment	Heading only	N/A
6.4.3.2.1 Environmental tolerance assessment An environmental tolerance assessment shall be conducted against the requirements specified in the technical specification…	This is addressed in the specification developed in accordance with, N-STI-69000-10013, "Computer System Requirements and Design" [B.16-11], Section 5.3.13.	Compliant
6.4.3.2.2 Electromagnetic immunity and emissions assessment An electromagnetic immunity and emissions assessment shall be conducted against the requirements specified in the technical specification…	See 6.4.3.2.1	Compliant
6.4.3.2.3 Seismic tolerance assessment A seismic tolerance assessment shall be conducted against the requirements specified in the technical specification…	See 6.4.3.2.1	Compliant
6.4.3.3 Hardware engineering process assessment	Heading only	N/A
6.4.3.3.1 Hardware testing techniques assessment A hardware testing techniques assessment shall be conducted...	These activities are conducted by Procurement Engineering, N-PROC-MP-0098, "Procurement Engineering Activities" [B.16-29], combined with vendor surveillance. The Computer Engineering department establishes the requirements and design and is involved with the acceptance of test/qualification results. There are several procurement specifications e.g., N-TSI-60XXX and standards N-STI-60XXX, relating to I&C and digital hardware components, e.g., N-STI-60407-10003, "Specification for OPG	Compliant

N290.14-15 Clause	PSR2 Review	Compliant or Gap
	Radiated Susceptibility Field Test" [B.16-30]. N-TSI-60458-10000, "Digital Process Controllers" [B.16-31], Section 1.2 provides an example of applicable standards and specifications. Additionally, this specification provides manufacturer accountabilities.	
6.4.3.3.2 Hardware design process assessment A hardware design process assessment shall be conducted. The following shall be considered: a) use of automated design tools; b) change control and configuration management of the design; and c) design practices to improve reliability, robustness, and fault tolerance…	See 6.4.3.3.1	Compliant
6.4.3.3.3 Product manufacturing process methods assessment A manufacturing process methods assessment shall be conducted.	See 6.4.3.3.1	Compliant
6.4.3.4 Third party certification assessment All third party certifications shall be assessed to determine if they can be used in part or entirely to satisfy the assessments identified in Clause 6.4.3…	There are no specific requirements relating to third party certification identified for hardware. However, providing evidence of third party testing (i.e., certification) is done as part of the standard I&C procurement. Also, third party certification for software is explicitly addressed in N-PROC-MP-0049, "Procurement of Software and Products Containing Software" [B.16-14].	Compliant
6.4.4 Software engineering tool Software engineering tools used for the configuration, maintenance of digital items, or the development of custom software shall be qualified in accordance with Annex D.	This is a new requirement introduced in the 2015 version of the Standard. Annex D is Mandatory. This requirement is addressed in Section 1.1 of N-PROG-MP-0006, "Software" [B.16-8], *"(c) Except as above, for software that meets the definition of Software Engineering Tool, the classification of the target application (i.e., RTPC or SESA*	Compliant

N290.14-15 Clause	PSR2 Review	Compliant or Gap
	software) is used to determine the applicable standards and procedures." Hence, because the 'tool' is qualified to the same extent as the application, the new requirement does not impose any additional requirements.	
6.5 Qualification results	Heading only	N/A
6.5.1 The qualification of a candidate product shall be documented in a qualification report. **Note:** *The qualification report may consist of one or more documents.* **6.5.2** The identification of all digital items within the candidate product undergoing qualification and the context of how the candidate product fits into the overall system of which it is a component shall be clearly identified. **6.5.3** The qualification report shall document the failure analysis assessment when required along with the mitigation strategies for the deficiencies identified during the assessment. **6.5.4** The qualification report shall identify the safety category or categories of the safety-related functions to be performed by the candidate product. **6.5.5** The qualification report shall identify the candidate product, including the versions of all software and hardware within the candidate product. Where software is used, the target platform and key aspects of the configuration shall be specified (see Clause 6.5.13b)). **6.5.6** The qualification report shall provide the rationale and supporting evidence used to justify the individual categorization of functions of individual components of the candidate product (see Clause 5.2.3). **6.5.7** The qualification report shall have an assessment of each of the qualification	The "Qualifications Results" section of the standard has been extensively revised from the R2012 to the 2015 version of N290.14. N-FORM-10409, "Software Quality Assurance Requirements" [B.16-23] and N-PROC-MP-0049 [B.16-14], "Procurement of Software and Products Containing Software", specify that CSA N290.14 must be complied with. Although the section is revised, for the most part the individual clauses have just been modified or moved to improve clarity. However, Clause 6.5.13 identifies new requirements for software and hardware combined in a candidate product. As discussed in Clause 1.1 above, there are applicable OPG standards, including N-STI-69000-10013, "Computer System Requirements and Design" [B.16-11], which outlines requirements for software and hardware design (e.g., design constraints and margins) and N-STI-69000-10005, "Category II/III Software Testing" [B.16-12], which identifies requirements for integration testing.	Compliant

N290.14-15 Clause	PSR2 Review	Compliant or Gap
concerns listed in Clause 6.2, including all relevant arguments and qualification evidence.		
6.5.8 The qualification report shall document the qualification results of each digital item identified in Clause 6.1. **6.5.9** The qualification result for each identified digital item shall conclude one of the following: a) The digital item in the candidate product is qualified for the intended use, subject to the limits identified by Clause 6.5.13. b) The digital item in the candidate product is not qualified for the intended use. c) The digital item in the candidate product is conditionally qualified for the intended use once identified activities (e.g., further analysis or testing) are successfully completed, and subject to the limits identified by Clause 6.5.13. **6.5.10** The qualification report shall document and summarize the overall qualification conclusion of the candidate product based on the results of the digital item(s) being evaluated. **6.5.11** The qualification report shall document personnel qualification for the individuals who have been party to the assessment of the candidate product and completion of the qualification report. **6.5.12** The report shall identify the qualification method for the digital items within the candidate product as per Clause 6.4.1.2. **6.5.13** When the software and hardware in the candidate product is qualified for the identified use, limits shall be identified and documented to specify the candidate product operating envelope within which it is acceptable to use the candidate product. Limits can include identification of a) limits of the candidate product's functionality; b) limits of the intended safety-related functions of the product; c) assumptions about the system in which the candidate product is to be used;		

N290.14-15 Clause	PSR2 Review	Compliant or Gap
d) limits of the platform and operating conditions; e) configuration parameters and the limits imposed on their values; f) interface constraints; g) user-configured safety-related functions and their appropriate documentation; h) performance and security limits; and i) life-time limiting components… **6.5.14** When a candidate product has minor customizations, the qualification report shall document the nature of the customizations, and the results of any complementary testing and the impact on the qualification conclusions.		
7 Maintaining qualification All software and hardware changes shall be documented and evaluated to determine the effect on the existing qualification. If such changes cannot be demonstrated to have no significant effect on the existing qualification, re-qualification of the component to the original criteria shall be conducted…	Per N-PROC-MP-0100, "Maintenance of Real-Time Process Computing Systems" [B.16-32], a change control procedure that is in conformance with ECC is to be defined for each computer-based system containing software. Requirements are identified in N-STI-69000-10001, "Software Maintenance" [B.16-33]. Section 3.3.5 specifies the configuration management requirements. Section 3.4 details the requirements for a Software Maintenance Plan (SMP). Section 3.0 refers to N-STI-69000-10019, "Configurable Software for Real-Time Process Computing Systems" [B.16-34], for the interpretation of the requirements within the standards and issues when using configurable software.	Compliant

B.16.3 Compliance Summary for Pickering PSR2

There is one PSR2 CSA N290.14-15 gap relating to Safety Factor 1 (Plant Design):

1. Correspondence with the CNSC [B.16-7] identifies all of the software application qualifications for software Categories 1, 2 and 3 from January 1, 2007 to the time of the correspondence (June 2016). However, an evaluation of legacy Real-Time Process Computing applications with respect to the requirements of N290.14-15 for Categories 1, 2 and 3 software has not been performed. Therefore, this has been identified as a PSR2 gap.

B.16.4 References

[B.16-1] CSA Standard, N290.14-15, *Qualification of Digital Hardware and Software for Use in Instrumentation and Control Applications for Nuclear Power Plants*, November 2015.

[B.16-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.16-3] CSA Standard, N290.14-07 (R2012), *Qualification of Pre-developed Software for Use in Safety-Related Instrumentation and Control Applications for Nuclear Power Plants*, July 2007.

[B.16-4] CSA Impact Statement, *Notification of CSA N290.14; Product: New Edition; Product Designation: CSA N290.14; Previous Edition: N290.14-07*, Date not provided.

[B.16-5] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.16-6] OPG Procedure, N-PROC-MP-0049 R004, *Procurement of Software and Products Containing Software*, May 2009.

[B.16-7] OPG Letter, N-CORR-00531-18111, B. Duncan and B. McGee to M. Santini and H. Khouaja, *Darlington and Pickering NGS – Response to CNSC Staff's Request for a Desktop Review of Qualification of Pre-developed Software of Digital Instrumentation and Control Systems at Ontario Power Generation*, June 14, 2016.

[B.16-8] OPG Program, N-PROG-MP-0006 R009, *Software*, April 2015.

[B.16-9] OPG Program, N-PROG-MP-0009 R011, *Design Management*, January 2015.

[B.16-10] OPG Program, N-PROG-MP-0001 R014, *Engineering Change Control*, January 2015.

[B.16-11] OPG Standard, N-STI-69000-10013 R001, *Computer System Requirements and Design*, November 2009.

[B.16-12] OPG Standard, N-STI-69000-10005 R004, *Category II/III Software Testing*, January 2016.

[B.16-13] OPG Program, N-PROG-MP-0005 R005, *Configuration Management*, June 2012.

[B.16-14] OPG Procedure, N-PROC-MP-0049 R009, *Procurement of Software and Products Containing Software*, June 2016.

[B.16-15] OPG Procedure, N-PROC-MP-0090 R013, *Modification Process*, July 2016.

[B.16-16] OPG Form, N-FORM-10959 R016, *Design Scoping Checklist*, June 2016.

[B.16-17] OPG Procedure, N-PROC-MP-0099 R003, *Development of Real-time Process Computing Systems*, March 2014.

[B.16-18] OPG Standard, N-STI-69000-10000 R004, *Software Categorization*, May 2016.

[B.16-19] OPG Training and Qualification Description, N-TQD-403-00001 R011, *Nuclear Engineering Support Personnel Training and Qualification Description*, April 2015.

[B.16-20] OPG Qualification Guide, N-QG-403-00008 R009, *Nuclear Design Engineering Qualification Guide*, March 2015.

[B.16-21] OPG Qualification Guide, N-QG-403-00012 R006, *Nuclear Components Engineering Qualification Guide*, August 2015.

[B.16-22] OPG Instruction, N-INS-69000-10002 R000, *Computer Spare Parts and Other Electronic Components Acquisition by Computers and Control Design Department*, March 2009.

[B.16-23] OPG Form, N-FORM-10409 R004, *Software Quality Assurance Requirements*, July 2014.

[B.16-24] OPG Standard, N-STI-69000-10003 R003, *Category II / III Software Design*, December 2009.

[B.16-25] OPG Form, N-FORM-10408 R003, *Software Procurement Planning*, March 2010.

[B.16-26] OPG Procedure, N-PROC-MP-0103 R003, *Security for Real-Time Process Computing Systems*, March 2015.

[B.16-27] OPG Guideline, N-DG-60407-10000 R000, *Guidelines for Electromagnetic Compatibility Test*, October 2011.

[B.16-28] CE-1001-STD R003, *CANDU Computer Systems Engineering Centre of Excellence, Standard for Software Engineering of Safety Critical Software*, November 2013.

[B.16-29] OPG Procedure, N-PROC-MP-0098 R007, *Procurement Engineering Activities*, June 2014.

[B.16-30] OPG Standard, N-STI-60407-10003 R000, *Specification for OPG Radiated Susceptibility Field Test*, October 2011.

[B.16-31] OPG Specification, N-TSI-60458-10000 R001, *Digital Process Controllers*, January 2006.

[B.16-32] OPG Procedure, N-PROC-MP-0100 R004, *Maintenance of Real-Time Process Computing Systems*, March 2014.

[B.16-33] OPG Standard, N-STI-69000-10001 R004, *Software Maintenance*, January 2016.

[B.16-34] OPG Standard, N-STI-69000-10019 R000, *Configurable Software for Real-Time Process Computing Systems*, November 2012.

B.17 CSA N291-15, "Requirements for Safety-Related Structures for Nuclear Power Plants"

B.17.1 Background

The following, paraphrased from the preface and scope of CSA N291-15 [B.17-1], provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *The purpose of CSA N291 is to specify the requirements for the material, analysis & design, construction, fabrication, inspection, examination and aging management of safety-related structures constructed of structural steel, reinforced concrete, and reinforced masonry. The minimum design requirements specified in this Standard follow the requirements of CSA A23.3, CSA S16, CSA S304.1, and the National Building Code of Canada.*
>
> *The standard provides a general definition of safety-related structures and ensures that the design and analysis of safety-related structures cover static and dynamic loads, and the loads, load factors, load combinations, and safety criteria cover service loads and abnormal/environmental loads.*

All of CSA N291-15 is directly relevant to Safety Factor 1 (Plant Design). CSA N291-15 includes clauses pertaining to in-service examinations (Clause 7) and aging management (Clause 9) that are also relevant to Safety Factor 2 (Actual Condition of SSCs) and Safety Factor 4 (Aging).

CSA N291 is identified in Appendix E.1 of the R04 Pickering Licence Conditions Handbook (LCH) [B.17-2] as "Guidance or Criteria". As indicated in Section 6.1 of the LCH, CSA N291 provides additional recommendations and guidance for development of the design program. As indicated in Section 7.1 of the LCH, CSA N291 provides additional recommendations and guidance for the development of inspection programs for balance of plant.

CSA N291-15 is the second edition of this standard, and supersedes the previous edition published in 2008 under the title "Requirements for Safety-Related Structures for CANDU Nuclear Power Plants" [B.17-3]. The noteworthy changes introduced as part of this revision are summarized in [B.17-4][15] and detailed in Section B.17.2.2 below.

Prior to the 2015 edition, Update No.1 to the 2008 edition was released in 2011 to incorporate a small number of minor clarifications [B.17-5]. Update No. 2 was issued in 2014 but only made an editorial change to a table column heading [B.17-6].

The results of PSR1 N291 reviews (Pickering A Return to Service assessments, and Pickering and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.17.2.1. As identified in Reference [B.17-7], the Pickering PSR2 review of CSA N291-15 is an Incremental Review. PSR2 Incremental Review includes an assessment of the intent of recent changes to the Law,

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[15] Note: The impact statement indicates a date of release of March 2016. However, the summary of changes have been compared to the second edition of CSA N291 (N291-15) and it is confirmed that this impact statement is meant to apply to this revision.

Regulation, Code or Standard (L/R/C/S) on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.
- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.17.2 Compliance Assessment for Pickering PSR2

B.17.2.1 Application of PSR1 Reviews

The versions of N291 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

The first edition of this standard was issued in March 2008 after the completion of the code reviews for the Pickering B ISR and Pickering A Return to Service. Therefore, no PSR1 code review was performed for Pickering NGS against CSA N291.

Darlington NGS

A clause-by-clause (non-PROL) intent review of N291-08 [B.17-3] was performed as part of the Darlington ISR, as documented in Reference [B.17-8]. Gaps were identified against fifteen clauses related to requirements for: anchorages, concrete coating inside containment structures, slenderness ratio of columns and tension members, embedded parts supporting piping or equipment designed for CSA N285 series and reinforcement detailing for joints, special detailing for other system components that interface with walls, walls and wall openings in concrete structures, and in-service examinations.

These gaps were grouped into six ISR Issues and documented in the Final ISR Report [B.17-9] as follows:

ISR Issue D071 - Coatings and Coverings Within Containment System

ISR Issue D283 - Design Requirements for Steel Safety-Related Structures

ISR Issue D284 - Load Factors for Metallic Embedded Parts in Safety-Related Building Structures

ISR Issue D285 - Seismic Design Provisions for Concrete Safety-Related Structures

ISR Issue D299 - Anchorage Requirements for Safety-Related Structures

ISR Issue D300 - Inspection Requirements for Safety-Related Structures.

With the exception of ISR Issue D300, all of the above were determined to have low safety significance and were dispositioned as Acceptable Deviations with No Further Action Required [B.17-9].

In addition, as summarized in Reference [B.17-10], two additional gaps against three clauses in CSA N291-08 were identified as part of the resolution of CNSC comments on the code review [B.17-8], as follows:

ISR Issue D401 - Load Factors for the Design of Nuclear Safety Related Structures

ISR Issue D411 - Approval of In-Service Examination Plans.

These issues were also determined to have low safety significance and were dispositioned as Acceptable Deviations with No Further Action Required, as documented in Reference [B.17-11]. The applicability to Pickering of the resolution of the Darlington ISR issues described above is discussed in Section B.17.2.2 below.

As described in the Final ISR Report [B.17-9], and reflected in the Darlington Global Assessment Report [B.17-12], D300 was identified as an ISR issue requiring action. This captured the finding that compliance with CSA N291-08 (specifically clauses 7.3.2.1, 7.3.2.2, 7.3.2.3, 7.3.3.1 and 7.3.3.2) could not be demonstrated due to no evidence being located to indicate that Darlington NGS had an active program for in-service examination that meets these requirements. In Reference [B.17-13], it was noted that subsequent to completion of the Darlington ISR code review against CSA N291-08, OPG had developed an aging management plan for non-containment structures. The recommended resolution per Reference [B.17-13] was as follows:

Initiate action(s) to establish a program for life-cycle management of safety-related structures in accordance with the requirements of N291-08.

By establishing an in-service examination program, OPG will ensure compliance with the requirements of CSA N291-08 and provide assurance that DNGS safety-related structures will continue to meet the design intent until the post-refurbishment end of plant life.

In Reference [B.17-10], OPG reported that in response to this issue, Periodic Inspection Program (PIP) plans had been developed for all safety-related structures at Darlington and that they will be implemented as part of the Integrated Aging Management Program [B.17-14], with schedules for inaugural inspections to be included in the Darlington ISR Integrated Implementation Plan (IIP), and that this response was accepted by the CNSC.

In the Darlington ISR IIP [B.17-15], ISR Issue D300 was assigned IIP Item number IIP-OI 038 with the following description:

There is a need to conduct regular in-service examinations of safety-related structures for evidence of degradation. The structures covered include:

-Those that support, house or protect nuclear safety systems,

-Components of structures required for the safe operation and reactor shutdown, and

-Facilities for storage of irradiated fuel and other radioactive waste material.

The Darlington ISR IIP [B.17-15] discussion of IIP-OI 038 listed the structures for which PIP plans had been prepared.

The applicability to Pickering of Darlington ISR gap D300 (IIP-OI 038) is discussed in Section B.17.2.2 below.

As part of the Darlington ISR, a code refresh review [B.17-16] was conducted against N291-08 (including Update No. 1) which found that the changes in the requirements are minor and reflect improvements for clarity or for formatting purposes. In conclusion, the review did not identify any additional gaps relative to the changes made in this update.

B.17.2.2 Application of Post PSR1 Reviews

In terms of applicability of the Darlington CSA N291-08 compliance assessment ([B.17-8] and [B.17-10]) and corresponding gap dispositions ([B.17-9] and [B.17-11]) to Pickering Units 1,4 and 5-8, it is noted that the Darlington conclusions are in general supported by making reference to Darlington-specific design documents. While the expectation is that similar conclusions would apply for Pickering, this cannot be confirmed without performing a more detailed review to identify the corresponding Pickering-specific design references.

A compliance review against CSA N291-08 [B.17-3] was not undertaken as a part of previous PSR1 reviews for Pickering NGS and the following incremental review has been performed against the Darlington CSA N291-08 compliance assessment. This incremental review includes high level review elements for portions of the Darlington compliance assessment that could not be readily applied to Pickering NGS due to station-specific differences. In the review below, the degree of conformance with clauses or groups of clauses in the L/R/C/S is assessed for Pickering NGS by reference to supporting evidence stating whether the intent of the requirements stipulated in the L/R/C/S is met.

Where high level review was determined to be necessary, Compliances and Gaps were identified, where required, as defined below:

- Compliance: Compliance indicates that the intent of the safety requirement is met.
- Gap: A Gap indicates that the intent of the safety requirement is not met.

CSA N291-08 Clause	PSR2 Review	Compliant or Gap
0 Preface	There are no requirements specified. The preface describes the purpose of CSA N291-08.	N/A
1 Scope	There are no requirements specified. This section sets the context for CSA N291-08.	N/A

CSA N291-08 Clause	PSR2 Review	Compliant or Gap
2 Reference Publications	There are no requirements specified. This section describes the publications that CSA N291-08 refers to.	N/A
3 Definitions	There are no requirements specified. This section defines various words, phrases, or acronyms, used in CSA N291-08.	N/A
4 General Requirements This section of CSA N291-08 defines the basis for jurisdictional boundaries between safety-related structures and structural components/parts designed and fabricated in accordance with other Codes and Standards. In addition, this section establishes requirements for the owner/licensee to establish and maintain an overall quality assurance program and establish requirements for: a) Design; b) Performance; c) Quality Assurance; d) In-Service Monitoring, Surveillance, and Examination; and e) Decommissioning.	In the PSR1 assessment [B.17-8], Darlington was deemed to be compliant with requirements related to jurisdictional boundaries based on a listing of safety-related buildings and structures being provided in a Nuclear Safety Design Guide. The equivalent document for Pickering NGS is P-LIST-06937-00001, "Pickering A and B List of Safety Related Systems" [B.17-17]. In the PSR1 assessment [B.17-8], Darlington was deemed to be compliant with requirements for establishing a quality assurance program based on the original quality assurance manual for the Darlington site. The establishment of a quality assurance program is a programmatic requirement for OPG Nuclear, which is common to both Darlington and Pickering NGS. These requirements are established through N-CHAR-AS-0002 [B.17-18], "Nuclear Management System", which spans all aspects of the business.	Compliant
5 Materials, Construction, and Operating Exposure Conditions	There are no requirement specified. The corresponding content in the document is a section heading.	N/A
5.1 Structural Steel This section specifies the material and product Standards and specifications for structural steel and fasteners. In addition, this section specifies requirements for materials and products to be identified and marked in accordance with CSA Standard S16.	In the PSR1 assessment [B.17-8], design documentation containing references to applicable Standards for structural steel was used to demonstrate Darlington's compliance with this section of N291-08. Pickering safety-related structures were designed in accordance with applicable Standards and specifications at that time, which meets the intent of this section of N291-08. Due to Pickering 1,4 and Pickering 5-8 being designed and constructed at different times, different Standards were used to design the corresponding safety-related structures, as described below.	Compliant

CSA N291-08 Clause	PSR2 Review	Compliant or Gap
	• Pickering 1,4 structures were designed in accordance with the National Building Code of Canada (NBCC) and associated material specifications [B.17-19]. Where applicable, structure-specific requirements were included in the corresponding Design Manual (References [B.17-21] to [B.17-37]). • CSA S16 is listed as one of the applicable standards for steel structures in the Pickering 5-8 Plant Level Topical Design Basis Document [B.17-20].	
5.2 Concrete and Grout This section specifies the following requirements for the use of concrete and grout materials: • Applicable Standards, including deterioration mechanisms. • Maximum concrete temperatures during normal operating or post-accident conditions. • Upper and lower limits for concrete placing temperatures and grout placing temperatures. • Maximum temperature differential allowed when placing fresh concrete against existing concrete in walls and suspended slabs. • Material requirements for concrete and grout, including cement, water, and aggregate.	In the PSR1 assessment [B.17-8], design documentation containing references to CSA A23.1, CSA A23.2, CSA A3000, and CSA A179 was used to demonstrate Darlington's compliance with this section of N291-08. Pickering safety-related structures were designed in accordance with applicable Standards and specifications at that time, which meet the intent of this section of N291-08. Due to Pickering 1,4 and Pickering 5-8 being designed and constructed at different times, different Standards were used to design the corresponding safety-related structures, as described below. • Pickering 1,4 structures were designed in accordance with the NBCC and associated material specifications [B.17-19]. Where applicable, structure-specific requirements were included in the corresponding Design Manual (References [B.17-21] to [B.17-37]). • CSA A23.1 is listed as one of the applicable Standards for concrete and grout in the Pickering 5-8 Plant Level Topical Design Basis Document [B.17-20].	Compliant

CSA N291-08 Clause	PSR2 Review	Compliant or Gap
• Testing requirements for concrete and grout.		
5.3 Reinforcing Steel This section specifies the requirements for reinforced steel, including: • Tensile test requirements; • Bend test requirements; • Provision of certified mill test reports; and • Identification/marking requirements for welded splices.	In the PSR1 assessment [B.17-8], design documentation containing references to the CSA G30 series of Standards for reinforcing steel was used to demonstrate Darlington's compliance with this section of N291-08. Pickering safety-related structures were designed in accordance with applicable Standards and specifications at that time, which meet the intent of this section of N291-08. Due to Pickering 1,4 and Pickering 5-8 being designed and constructed at different times, different Standards were used to design the corresponding safety-related structures, as described below. • Pickering 1,4 structures were designed in accordance with the NBCC and associated material specifications [B.17-19]. Where applicable, structure-specific requirements were included in the corresponding Design Manual (References [B.17-21] to [B.17-37]). • CSA G30.3, CSA G30.5, and CSA G30.12 are listed as applicable Standards for reinforcing steel in the Pickering 5-8 Plant Level Topical Design Basis Document [B.17-20].	Compliant
5.4 Pre-Stressing Steel This section specifies the requirements for materials used in the pre-stressing reinforcing systems of concrete structures. Specific requirements identified in this clause include: • Responsibilities of the owner/licensee to establish material qualification testing and conformance test requirements and the respective acceptance criteria. • Provision of certified mill test reports.	In the PSR1 assessment [B.17-8], design documentation containing references to CSA A23.1 was used to demonstrate Darlington's compliance with this section of N291-08. Pickering safety-related structures were designed in accordance with the applicable Standards and specifications at that time, which meet the intent of this section of N291-08. Due to Pickering 1,4 and Pickering 5-8 being designed and constructed at different times, different Standards were used to design the corresponding safety-related structures, as described below. • Pickering 1,4 structures were designed in accordance with the NBCC and associated materials specifications [B.17-19]. Where applicable, structure-specific requirements were included in the corresponding Design Manual (References [B.17-21] to [B.17-37]). • CSA A23.1 is listed as one of the applicable Standards for concrete and grout in the	Compliant

CSA N291-08 Clause	PSR2 Review	Compliant or Gap
	Pickering 5-8 Plant Level Topical Design Basis Document [B.17-20].	
5.5 Masonry This section specifies requirements for masonry materials used in safety-related structures.	In the PSR1 assessment [B.17-8], design documentation containing references to CSA Standard CAN3-A371-M84 was used to demonstrate Darlington's compliance with this section of N291-08. Pickering safety-related structures were designed in accordance with the applicable Standards and specifications at that time, which meet the intent of this section of N291-08. Due to Pickering 1,4 and Pickering 5-8 being designed and constructed at different times, different Standards were used to design the corresponding safety-related structures, as described below. • Pickering 1,4 structures were designed in accordance with the NBCC and associated material specifications [B.17-19]. Where applicable, structure-specific requirements were included in the corresponding Design Manual (References [B.17-21] to [B.17-37]). • CSA A371 is listed as one of the applicable Standards for the design and construction of masonry structures in the Pickering 5-8 Plant Level Topical Design Basis Document [B.17-20].	Compliant
5.6 Stainless Steel This section specifies requirements for stainless steel used in safety-related structures. This includes: • Requirements for use of electrodes; and • Requirements for welding procedures.	In the PSR1 assessment [B.17-8], technical specifications containing references to applicable Standards for the use of stainless steel were used to demonstrate Darlington's compliance with this section of N291-08. Pickering safety-related structures were designed in accordance with the applicable Standards and specifications at that time, which meet the intent of N291-08. • Pickering 1,4 structures were designed in accordance with the NBCC and associated material specifications [B.17-19]. Where applicable, structure-specific requirements were included in the corresponding Design Manual (References [B.17-21] to [B.17-37]). • The NBCC and CSA W47.1 are listed as applicable Standards for the use of stainless steel in the Pickering 5-8 Plant Level Topical Design Basis Document [B.17-20].	Compliant

CSA N291-08 Clause	PSR2 Review	Compliant or Gap
5.7 Anchorage *Anchorage shall meet the requirements of N287.2.*	There are no new requirements specified in this clause. N287.2 requirements are being assessed separately under PSR2, which includes consideration of the applicability of previous gaps identified in the PSR1 assessment for Darlington [B.17-8].	N/A
5.8 Other Materials *Materials other than those specified in this Standard may be used, provided that material tests are conducted to the satisfaction of the engineer.*	The PSR1 assessment for Darlington [B.17-8] concluded that this clause represents a general requirement where it is not necessary to assess compliance. This is a programmatic conclusion that also applies to Pickering.	N/A
5.9 Coatings This clause specifies requirements for: • Application of coatings to steel inside and outside containment structures; and • Application of coatings to concrete inside and outside containment structures.	In the PSR1 assessment [B.17-8], references to design documentation were used to demonstrate Darlington's compliance with this section of N291-08. Requirements specific to this section fall into two categories: • Application of coatings to structures inside containment shall be in accordance with CSA N287.2. • Application of coatings to structures outside of containment shall be in accordance with CSA S16. N287.2 requirements are being assessed separately as a part of PSR2, which includes consideration of the applicability of previous gaps identified in the PSR1 assessment for Darlington [B.17-8]. Thus, the scope of this assessment is limited to the requirements of CSA S16. Pickering safety-related structures were designed in accordance with the applicable Standards and specifications at that time, which meet the intent of N291-08. Due to Pickering 1,4 and Pickering 5-8 being designed and constructed at different times, different Standards were used to design the corresponding safety-related structures, as described below. • Pickering 1,4 structures were designed in accordance with the NBCC and associated material specifications [B.17-19]. Where applicable, structure-specific requirements were included in the corresponding Design Manual (References [B.17-21] to [B.17-37]). • CSA S16 is listed as one of the applicable Standards in the Pickering 5-8 Plant Level Topical Design Basis Document [B.17-20].	Compliant

CSA N291-08 Clause	PSR2 Review	Compliant or Gap
6 Analysis and Design	There are no requirements specified. The corresponding content in the document is a section heading.	N/A
6.1 General This section identifies applicable Standards for the design and analysis of safety-related structures. In addition, for structures subjected to complex dynamic loads such as drop and impact loads, test and/or analysis may be used.	In the PSR1 assessment [B.17-8], references to design documentation were used to demonstrate Darlington's compliance with this section of N291-08. Pickering safety-related structures were designed and analyzed in accordance with the applicable Standards and specifications at that time, which meet the intent of this section of N291-08. Due to Pickering 1,4 and Pickering 5-8 being designed and constructed at different times, different Standards were used to design the corresponding safety-related structures, as described below. • Pickering 1,4 structures were designed and analyzed in accordance with the NBCC [B.17-19]. Structure-specific requirements regarding dynamic loads are provided in the corresponding Design Manuals (References [B.17-21] to [B.17-37]). • CSA S16 (structural steel), CSA A23.3 (concrete structures) and CSA A371 (masonry structures) are listed as applicable Standards in the Pickering 5-8 Plant Level Topical Design Basis Document [B.17-20]. Structure-specific requirements regarding dynamic loads are provided in the corresponding Design Manuals (References [B.17-38] to [B.17-55]).	Compliant
6.2 Symbols This clause defines symbols associated with loads, internal moments, and forces.	There are no requirements specified. The corresponding content in the document is a listing of symbols that does not impose any requirements on safety-related structures.	N/A
6.3 Load and Load Factors, Load Combinations, and Limit States This section identifies requirements related to loads considered in the design and analysis of safety-related structures. Specific requirements include: • Loads considered in the design.	In the PSR1 assessment [B.17-8], references to design documentation were used to demonstrate Darlington's compliance with this section of N291-08. As part of the PSR1 assessment [B.17-8], discrepancies were identified between load factors documented in station Design Manuals and prescribed load factors in this section of N291-08. ISR Issue D401 was created to determine the safety-significance of these discrepancies [B.17-11]. As noted in Section B.17.2.1 of this document, ISR Issue D401 was determined to have a low safety significance and was dispositioned as an acceptable deviation with no further action required.	Compliant

CSA N291-08 Clause	PSR2 Review	Compliant or Gap
• Load combinations considered in the design. • Design of structures for sufficient strength and stability. • Design of structures to meet serviceability requirements.	Pickering safety-related structures were designed and analyzed in accordance with the applicable Standards at that time, which meet the intent of N291-08. Given that Pickering meets the intent of N291-08, it is appropriate to apply the assessment of the safety significance of ISR Issue D401 to Pickering. A new gap is not required for PSR2. Due to Pickering 1,4 and Pickering 5-8 being designed and constructed at different times, different Standards were used to design the corresponding safety-related structures, as described below. • Pickering 1,4 structures were designed and analyzed in accordance with the NBCC [B.17-19]. Structure-specific requirements for load factors and combinations were included in the corresponding Design Manuals (References [B.17-21] to [B.17-37]). • Pickering 5-8 structures were designed and analyzed in accordance with the Standards listed in Reference [B.17-20]. Structure-specific requirements for load factors and load combinations were included in the corresponding Design Manuals (References [B.17-38] to [B.17-55]).	
6.4 Seismic Design This section identifies requirements for the seismic analysis of safety-related structures, including: • Requirements for dynamic analysis; • Applicable Standards for seismic analysis; • Specification of live load reduction in computation of seismic forces; and • Requirements to avoid the interaction of elements to seismic motion.	In the PSR1 assessment [B.17-8], design documentation containing references to the NBCC and CSA CAN3-N289.3 was used to demonstrate Darlington's compliance with this section of N291-08. Pickering safety-related structures were designed and analyzed in accordance with the applicable Standards at that time, which meet the intent of N291-08. • Pickering 1,4 structures were initially designed and analyzed in accordance with the NBCC [B.17-19], with supplemental requirements included in the corresponding Design Manuals (References [B.17-21] to [B.17-37]). Subsequently, a seismic assessment of Pickering A was performed to confirm safe operation following a credible seismic event. Relevant documentation from this work includes References [B.17-56] to [B.17-58]. • Pickering 5-8 was designed to be seismically qualified, as described in Section 8.1 of the Pickering B Separation Philosophy & Common Mode Events Topical Design Basis Document [B.17-59]. Additionally, Section 8.2 of [B.17-59] includes references to the NBCC and N289 series of Standards. A complete listing of	Compliant

CSA N291-08 Clause	PSR2 Review	Compliant or Gap
	structures with design basis requirements for seismic events is documented in Section 8.6.3 of [B.17-59].	
6.5 Foundations This section identifies requirements for the design and analysis of foundations, including: • Applicable Standards; • Requirement that safety-related structures shall not be built on strata that are potentially subject to liquefaction; and • Analysis of effects of fluctuating ground water in settlement analysis and evaluation of soil capacity.	In the PSR1 assessment [B.17-8], design documents containing references to the NBCC were used to demonstrate Darlington's compliance with this section of N291-08. Pickering safety-related structures were designed and analyzed in accordance with the applicable Standards at that time, which meet the intent of N291-08. • Pickering 1,4 structures were designed and analyzed in accordance with the NBCC [B.17-19]. • Pickering 5-8 structures were designed and analyzed in accordance with the NBCC [B.17-20]. In the PSR1 assessment, requirements related to liquefaction were determined to be not applicable as there was no evidence that any safety-related systems were built on strata potentially subject to liquefaction [B.17-8]. This conclusion also applies to Pickering NGS as the corresponding Design Manuals (References [B.17-21] to [B.17-55]) do not contain any evidence that safety-related structures were built on strata potentially subject to liquefaction.	Compliant
6.6 Steel Structures This section identifies requirements for the design of safety-related steel structures, including: • Applicable Standards; • Use of plastic analysis in the design process; • Requirements for bolted connections; • Slenderness ratios for structural steel members. • Requirement for fasteners to be installed with locking devices where connections are subjected to	In the PSR1 assessment [B.17-8], design documents containing references to CSA S16 were used to demonstrate Darlington's compliance with this section of N291-08. As part of the PSR1 assessment [B.17-8], discrepancies were identified between the slenderness ratios specified in this section of N291-08 and CSA S16. ISR Issue D283 was created to determine the safety-significance of these discrepancies [B.17-9]. As noted in Section B.17.2.1 of this document, ISR Issue D283 was determined to have a low safety significance and was dispositioned as an acceptable deviation with no further action required. Pickering safety-related structures were designed and analyzed in accordance with the applicable Standards at that time, which meet the intent of N291-08. Given that Pickering meets the intent of N291-08, it is appropriate to apply the assessment of the safety significance of Issue D283 to Pickering. A new gap is not required for PSR2. Due to Pickering 1,4 and Pickering 5-8 being designed and constructed at different times, different Standards were used to	Compliant

CSA N291-08 Clause	PSR2 Review	Compliant or Gap
dynamic loads and/or vibration. • Specification of load factors for steel components supporting piping or equipment.	design the corresponding safety-related structures, as described below. • Pickering 1,4 structures were designed and analyzed in accordance with the NBCC [B.17-19]. General requirements from the NBCC were supplemented by structure-specific requirements contained in the corresponding Design Manuals (References [B.17-21] to [B.17-37]). • CSA S16 is listed as an applicable Standard for steel structures in the Pickering 5-8 Plant Level Topical Design Basis Document [B.17-20]. General requirements from S16 were supplemented by structure-specific requirements contained in the corresponding Design Manuals (References [B.17-38] to [B.17-55]).	
6.7 Concrete Structures This section identifies requirements for the design of safety-related concrete structures, including: • Applicable Standards; • Consideration of cracked section properties for thermally-induced loads; • Design requirements for metallic embedded parts for building structures; • Load factors for metallic embedded parts supporting piping or equipment. • Requirements for welded joints.	In the PSR1 assessment [B.17-8], design documentation containing references to CSA A23.3, CSA S16, and CSA W59 was used to demonstrate Darlington's compliance with this section of N291-08. As part of the PSR1 assessment [B.17-8], discrepancies were identified between the load factors documented in station Design Manuals and the prescribed load factors in this section of N291-08. Specifically, the discrepancies identified related to load factors for metallic embedded parts supporting piping or equipment designed in accordance with the CSA N285 series of Standards. ISR Issue D284 was created to determine the safety-significance of these discrepancies [B.17-9]. As noted in Section B.17.2.1 of this document, ISR Issue D284 was determined to have a low safety significance and was dispositioned as an acceptable deviation with no further action required. Pickering safety-related structures were designed and analyzed in accordance with the applicable Standards at that time, which meet the intent of N291-08. Given that Pickering meets the intent of N291-08, it is appropriate to apply the assessment of the safety significance of ISR Issue D284 to Pickering. A new gap is not required for PSR2. Due to Pickering 1,4 and Pickering 5-8 being designed and constructed at different times, different Standards were used to design the corresponding safety-related structures, as described below. • Pickering 1,4 structures were designed in accordance with the NBCC [B.17-19]. General	Compliant

CSA N291-08 Clause	PSR2 Review	Compliant or Gap
	requirements from the NBCC were supplemented by structure-specific requirements documented in the corresponding Design Manuals (References [B.17-21] to [B.17-37]). • CSA A23.3, CSA S16.1, and CSA W59.1 are all listed as applicable Standards in the Pickering 5-8 Plant Level Topical Design Basis Document [B.17-20].	
6.8 Masonry Structures This section identifies requirements for the design of safety-related masonry structures, including high-density concrete block walls. Requirements identified in this clause include: • Applicable Standards; • Restrictions on the use of unreinforced masonry structural elements.	In the PSR1 assessment [B.17-8], design documentation containing references to CSA Standard CAN3-S304-M84 was used to demonstrate Darlington's compliance with this section of N291-08. Pickering safety-related structures were designed in accordance with the applicable Standards at that time, which meet the intent of N291-08. Due to Pickering 1,4 and Pickering 5-8 being designed and constructed at different times, different Standards were used to design the corresponding safety-related structures, as described below. • Pickering 1,4 structures were designed in accordance with the NBCC [B.17-19]. • CSA S304 is listed as an applicable Standard for masonry structures in the Pickering 5-8 Plant Level Topical Design Basis Document [B.17-20]. In the PSR1 review, requirements related to the use of unreinforced masonry structural elements were determined to be not applicable as there was no evidence that unreinforced masonry elements were used in any safety-related structures [B.17-8]. This conclusion also applies to Pickering NGS as the corresponding Design Manuals (References [B.17-21] to [B.17-55]) do not contain any evidence that unreinforced masonry elements were used in any safety-related structures.	Compliant
6.9 Displacement of Structures This section specifies limitations on the maximum displacements of structures or buildings.	In the PSR1 assessment [B.17-8], design documents containing references to the NBCC were used to demonstrate Darlington's compliance with this section of N291-08. Pickering safety-related structures were designed in accordance with the applicable Standards at that time, which meet the intent of N291-08. • Pickering 1,4 structures were designed and analyzed in accordance with the NBCC [B.17-19].	Compliant

CSA N291-08 Clause	PSR2 Review	Compliant or Gap
	• Pickering 5-8 structures were designed and analyzed in accordance with the NBCC [B.17-20].	
6.10 Concrete Structures with Non-Metallic or Metallic Liners This section identifies design requirements for metallic and non-metallic liners in safety-related concrete structures. • Applicable Standards; • Restrictions on use of rigid non-metallic liners as a primary leakage barrier. • Requirements to maintain uniform stress levels across the liner plate. • Requirements for water treatments and purification • Requirements for pool-type irradiated fuel storage bays.	In the PSR1 assessment [B.17-8], Design Manuals for the Irradiated Fuel Bay (IFB) were used to demonstrate Darlington's compliance with this section of N291-08. Equivalent documentation exists for the Pickering IFBs, which meets the intent of N291-08. The corresponding Design Manuals for Pickering 1,4 are References [B.17-28], [B.17-60], and [B.17-61]. The corresponding Design Manuals for Pickering 5-8 are References [B.17-46] and [B.17-62].	Compliant
7 Construction Inspection and In-Service Examination	There are no requirements specified. The corresponding content in the document is a section heading.	N/A
7.1 General This section identifies programmatic requirements related to construction inspections and in-service examinations, which include: • Responsibility of the engineer to establish a construction inspection and in-service examination program for all	Requirements for in-service examinations are assessed under Section 7.3. The assessment of Section 7.1 is limited to construction inspection programs. Pickering structures were initially constructed, inspected, and tested in accordance with the quality assurance program in effect at the time, which meets the intent of N291-08. Pickering 1,4 safety-related structures were designed, fabricated, constructed, inspected, and tested in accordance with Parts 4 through 8 of the NBCC [B.17-19]. General requirements from the NBCC were supplemented by structure-specific requirements documented in the corresponding Design Manuals (References [B.17-21] to [B.17-37]).	Compliant

CSA N291-08 Clause	PSR2 Review	Compliant or Gap
safety-related structures. • Responsibility of the designer to identify critical components of structures requiring additional construction examinations and/or in-service examinations. • Specification of qualification requirements for personnel performing construction inspections and in-service examinations.	Pickering 5-8 safety-related structures were designed, fabricated, constructed, inspected, and tested in accordance with Parts 4 through 8 of the NBCC [B.17-65]. General requirements from the NBCC were supplemented by structure-specific requirements contained in the corresponding Design Manuals (References [B.17-38] to [B.17-55]).	
7.2 Construction Inspection This section identifies inspection and testing requirements applicable to the construction, installation, and fabrication of parts and components of safety-related structures. Applicable structures and components include: • Steel structures; • Welded joints; • Bolted joints; • Concrete and grout materials; • Reinforcing materials; • Arc-welded and mechanical splices; • Embedded parts; • Anchors; • Steel liners;	Requirements for in-service examinations are assessed under Section 7.3. The assessment of Section 7.2 is limited to construction inspection programs. Pickering structures were initially constructed, inspected, and tested in accordance with the quality assurance program in effect at the time, which meets the intent of N291-08. Pickering 1,4 safety-related structures were designed, fabricated, constructed, inspected, and tested in accordance with Parts 4 through 8 of the NBCC [B.17-19]. General requirements from the NBCC were supplemented by structure-specific requirements documented in the corresponding Design Manuals (References [B.17-21] to [B.17-37]). Pickering 5-8 safety-related structures were designed, fabricated, constructed, inspected, and tested in accordance with Parts 4 through 8 of the NBCC [B.17-65]. General requirements from the NBCC were supplemented by structure-specific requirements contained in the corresponding Design Manuals (References [B.17-38] to [B.17-55]).	Compliant

CSA N291-08 Clause	PSR2 Review	Compliant or Gap
• Reinforced masonry structures.		
7.3 In-Service Examination This section specifies requirements for in-service examinations of safety-related structures and their components during the life of the plant. Key aspects of the in-service examination program include: • Definition of activities that comprise the program; • Considerations for the extent of examination and basis of comparison; • Examination frequency for safety-related structures; • Inspections of all structural components for safety-related structures following any abnormal/environmental conditions.	In the PSR1 assessment [B.17-8], design documentation containing references to CSA N287.7 was used to demonstrate Darlington's compliance with this section of N291-08. Additionally, references to operating documentation that would be used in response to a seismic event were used to demonstrate Darlington's compliance with requirements for inspections of structural components following abnormal plant conditions. There are separate activities being performed as a part of PSR2 to assess N287.7 requirements. Thus, assessment of this code is not discussed further in the context of N291-08. Pickering Containment structures (Reactor Building, Pressure Relief Duct, Vacuum Building) have full rigorous in-service inspection programs. As part of the PSR1 assessment [B.17-8], it was identified there was no evidence of an in-service examination program for safety-related structures outside of containment. These structures are not covered by N287.7 but are included in the scope of N291-08. ISR Issue D300 identified the need to address incomplete in-service examination plans for these structures. As noted in Section B.17.2.1 herein, Issue D300 was determined to require resolution. As discussed separately in the text immediately following this summary table, assessment of the applicability of ISR Issue D300 to Pickering operation beyond 2020 has been identified as a gap. This issue is identified as a PSR2 gap in OPG Report P-REP-03680-00024 R000, "Pickering 5-8 Continued Operations Plan Review in Support of PNGS Periodic Safety Review 2 (PSR2)" [B.17-69]. A duplicate gap under CSA N291-15 will not be created. For requirements specific to the inspection of structural components following abnormal plant conditions, the rationale provided for Darlington's response to a seismic event, through the use of Seismic Abnormal Incident Manuals (AIMs), also applies to Pickering NGS. The corresponding procedures for Pickering are References [B.17-63] and [B.17-64].	Gap **COP Gap relating to COP Action #31 [B.17-69]**
8 Seismic and Dynamic Considerations	There are no requirements specified. The corresponding content in the document is a section heading.	N/A

<table>
<tr><th>CSA N291-08 Clause</th><th>PSR2 Review</th><th>Compliant or Gap</th></tr>
<tr><td>8.1 General
This section specifies general requirements related to the consideration of seismic and dynamic loads in the design of safety-related structures.
Structures subject to seismic ground motion shall be designed elastically.
Identification of applicable Standards.
Restrictions on use of plain concrete in safety-related structures for structural purposes.</td><td>In the PSR1 assessment [B.17-8], design documentation containing references to CSA N289.2, CSA A23.3, and CSA S16 was used to demonstrate Darlington's compliance with this clause.
N289.2 requirements are being assessed under a separate code review that is being performed in support of PSR2. The scope of the assessment for this section of N291-08 is limited to CSA A23.3 and CSA S16.
Pickering safety-related structures were designed and analyzed in accordance with the applicable Standards at that time, which meet the intent of this section of N291-08. Due to Pickering 1,4 and Pickering 5-8 being designed and constructed at different times, different Standards were used to design the corresponding safety-related structures, as described below.
Pickering 1,4 structures were initially designed and analyzed in accordance with the NBCC [B.17-19], with supplemental requirements included in the corresponding Design Manuals (References [B.17-21] to [B.17-37]). Subsequently, a seismic assessment of Pickering A was performed to confirm safe operation following a credible seismic event. Relevant documentation from this work includes References [B.17-56] to [B.17-58].
CSA A23.3 and CSA S16 are both listed as applicable Standards in the Pickering 5-8 Plant Level Topical Design Basis Document [B.17-20].</td><td>Compliant</td></tr>
<tr><td>8.2 Analysis
Seismic and design analysis shall be performed in accordance with Clause 6.4.</td><td>There are no new requirements specified in this clause. The content of Section 8.2 is limited to a cross-reference to Section 6.4.</td><td>N/A</td></tr>
<tr><td>8.3 Design Provisions for Concrete Structures
This section specifies requirements for the seismic design and reinforcement detailing of concrete structural members, including the identification of applicable Standards.
These requirements apply to:</td><td>In the PSR1 assessment [B.17-8], design documentation containing references to CSA A23.3 was used to demonstrate Darlington's compliance with this section of N291-08.
As part of the PSR1 assessment [B.17-8], it was identified that N291-08 imposed more stringent requirements than what was imposed by the Standards in effect at the time Darlington was designed and constructed. These requirements were specifically for concrete safety-related structures outside of containment. ISR Issue D285 was created to determine the safety significance of the more stringent requirements imposed by N291-08 [B.17-9]. As noted</td><td>Compliant</td></tr>
</table>

CSA N291-08 Clause	PSR2 Review	Compliant or Gap
• Beams; • Columns; • Walls; • Openings; and • Joints.	in Section B.17.2.1 of this document, ISR Issue D285 was determined to have a low safety significance and was dispositioned as an acceptable deviation with no further action required. The assessment of Issue D285 was based on the existence of alternative methods used to demonstrate adequate assurance of Darlington's capability to withstand seismic events [B.17-9]. Similar work has been performed at Pickering NGS (References [B.17-56] to [B.17-59]) which demonstrates the station's capability to withstand a seismic event. Thus, it is appropriate to apply the assessment of the safety significance of ISR Issue D285 to Pickering. A new gap is not required for PSR2. Pickering safety-related structures were designed in accordance with the applicable Standards at that time, which meet the intent of this section of N291-08. Due to Pickering 1,4 and Pickering 5-8 being designed and constructed at different times, different Standards were used to design the corresponding safety-related structures, as described below. • Pickering 1,4 structures were designed in accordance with the NBCC [B.17-19]. Where applicable, structure-specific requirements are specified in the corresponding Design Manuals (References [B.17-21] to [B.17-37]). • CSA A23.3 is listed as an applicable Standard for the design of concrete structures in the Pickering 5-8 Plant Level Topical Design Basis Document [B.17-20].	
Annex A Special Provisions for Impulse and Impact Effects for Concrete Structures	Provides information and does not establish any requirements.	N/A
Annex B Waiting Periods for Placing Fresh Concrete Adjacent to Hardened Concrete	Provides information and does not establish any requirements.	N/A
Annex C Occupational Health and Safety Requirements for Access to Normally Inaccessible Structures	Provides information and does not establish any requirements.	N/A

Although the Pickering B ISR did not explicitly consider CSA N291-08, the Pickering Units 5-8 Continued Operations Plan (COP) [B.17-66] identified a completed action (Action #31 - F14-4.2 in Appendix A of [B.17-66]) that addresses CSA N291-08 in relation to the same issue associated with Darlington ISR Issue D300, discussed in Section B.17.2.1 above. Specifically:

> *Include the periodic inspection programs and LCMPs [Life Cycle Management Plans] for the safety-significant civil structures that are under the scope of CSA N291-08, but not covered by the N287.7 standard.*

Action #31 from the Pickering Units 5-8 COP [B.17-66] was closed in the context of Pickering operation up to 2020. This action involved submission of Periodic Inspection Plans and Life Cycle Management Plans for a number of safety-significant civil structures to address fitness for service "to end of mission time" (which will need to be extended for Pickering operation past 2020). The need to revisit Action #31 in the context of Pickering operation beyond 2020 relates specifically to CSA N291 and has been identified as a PSR2 gap in OPG Report P-REP-03680-00024 R000, "Pickering 5-8 Continued Operations Plan Review in Support of PNGS Periodic Safety Review 2 (PSR2)" [B.17-69]. This gap will consider both Pickering Units 1,4 and Pickering Units 5-8. A duplicate gap under CSA N291-15 will not be created.

Following the Darlington ISR, a code-over-code review [B.17-70] was conducted against N291-08 (including Updates No. 1 and No. 2). This review identified no significant technical changes to the requirements and, therefore, no impact on the findings of the reviews that had been performed previously.

The preceding discussion summarizes assessments that were performed in relation to CSA N291-08 (including Updates No. 1 and No. 2). For the purpose of PSR2, review against code edition CSA N291-15 is required. The CSA Impact Statement notification for N291-15 [B.17-3] identifies the following noteworthy changes from the previous edition (paraphrased):

1. *The word CANDU was removed from the title of the standard and technology neutral language was used.*
2. *The type of safety-related structure covered by this standard (Clause 1.1) was clarified.*
3. *Clarified the definition for safety-related structures, added new definitions and removed obsolete ones (Clause 3).*
4. *Sub-clauses under new Clause 5.2.3 clarify the requirements for "Concrete temperature at placing" and "Waiting Periods" and introduce new requirements for "Internal Concrete temperature for Mass Concrete".*
5. *Requirements for welding of reinforcing bars were added through new Clause 5.3.1.2 that refers to CSA W186.*
6. *The snow load requirement in Sub-clause 6.3.1(a)(iv) was updated to 100-year occurrence to be distinguished from the NBCC 2010, which uses 50-year occurrence.*
7. *The requirements on limiting strain conditions for safety-related structures were clarified:*

a. Sub-clause 6.3.3.3 was revised to provide clarification on the design of safety-related structures.

b. Annex A was revised to clarify the provisions for impulse and impact effects for concrete structures.

8. Seismic design requirements for safety-related structures were clarified:

a. Clause 6.4 on Seismic design was moved and combined with Clause 8 "Seismic analysis, design and dynamic considerations".

b. Clause 8 was revised to align with the design and detailing requirements of CSA A23.3.

9. Sub-clause 6.5.2.2 was added to supplement the requirements in Sub-clause 6.6.2.2 of CSA N291-08 (Sub-clause 6.5.2.1 in the 2015 edition of the standard) such that the design of bolted connections in members that are part of the seismic load resisting system are not governed by a brittle limit state.

10. Sub-clause 6.6.2.5 (6.5.2.5 in the 2015 edition of the standard) and Table 4 were updated to provide clarification on the load factors for steel components supporting piping or equipment which are designed to CSA N285.

11. Sub-clauses 6.6.3 and 6.6.4 consolidate and clarify the requirements for temperature limitations and loads induced by thermal effects in concrete structures.

12. Sub-clause 6.7.3 (6.6.5 in the 2015 edition of the standard) and Table 4 were updated to provide clarification on the load factors for metallic embedded parts supporting piping or equipment which are designed to CSA N285.

13. Sub-clause 6.6.6 was added to provide the requirements for anchorage to concrete.

14. Sub-clause 7.2.4.3.3 was updated to be applied to reinforcing bars partially embedded in concrete only. The standard was previously written to require inspection of all bent bars.

15. Clause 9 Aging Management was added to identify the requirements for aging management of safety-related structures.

16. Test method references were updated for Table 1 and Table 2: Clearer pointers to the reference standards were provided including the specific test methods

17. Annex C was removed as it is not part of this standard.

With respect to the changes listed above, Items 1, 2, 3, 7a, 7b, 8a, 10, 11, 12, 16 and 17 are either editorial in nature or provide additional clarification or guidance that does not introduce new requirements or modify previously existing requirements.

Item 4 is related to construction of concrete structures and the new recommendation it presents (for the internal temperature of mass concrete) is intended to additionally limit the effects of heat from hydration (e.g., cracks in the concrete during its curing). The requirements for mass concrete are specified in the referenced standards CSA A23.1-14/A23.2-14 [B.17-71]. The change recommends a best practice that applies during construction that is not safety significant for an existing structure, for which confirmation of no excessive cracking is achieved through other means such as in-service examinations.

Item 5 sets a requirement for the rebar splicing materials to comply with CSA W186-M1990 (R2012) [B.17-72]. Although this requirement is not in CSA N291-08, CSA W186-M1990 (R2007) [B.17-73] is referenced in it, making the latter code the governing document for rebar splicing. This and the fact that CSA W186-M1990 (R2012) and CSA W186-M1990 (R2007) are identical indicates that the change is for clarification and consequently it does not impact safety.

Item 6 introduces higher snow loads due to the change in requirement to design for a 100 year occurrence event instead of a 50-year occurrence event. With reference to the note under Sub-clause 6.3.1(a)(iv), the increase in snow load is 22%. This yields a snow load value of 2.14 kPa that is not safety significant in relation to the snow load value of 2.1 kPa that was used for the design of Pickering, as reflected in [B.17-74].

Item 8b removes the requirements of Clause 8.3 "Design provisions for concrete structures", which is now less restrictive. Hence, the change has no safety impact.

Item 9 imposes new requirements for bolted connections in members that are part of the seismic load resisting system. This item may have safety significance and requires further assessment since Pickering structures were not explicitly designed to meet these requirements. This is identified as **PSR2 CSA N291-15 Gap #1**.

Item 13 establishes explicit design requirements for concrete anchorage. In CSA N291-08, anchorage requirements were stated in general terms in Sub-clause 6.7.3.1, by making reference to Annex D of CSA A23.3. Corresponding requirements appear in Sub-Clause 6.6.6 of the 2015 code edition, which continues to refer to Annex D of CSA A23.3 for concrete anchorage requirements. Some new requirements are stated regarding non-safety related structures, post-installed anchors, operating temperature and the use of adhesive anchors that are either covered by other existing requirements, reflect standard OPG design practice or are not applicable to the current Pickering design. Therefore, there is no PSR2 gap.

Item 14 implements changes that make the requirements less restrictive than CSA N291-08, hence, it has no safety impact.

Item 15 introduces the new Clause 9 "Aging Management" (with sub-clauses) containing new requirements that are not in CSA N291-08. Clauses 9.1 and 9.2 are programmatic in nature. Clauses 9.3 and 9.4 require the effects of aging of safety related structures to be considered in the design by including safety margins. Clause 9.5 calls for design support on the related inspection and testing activities. Although these clauses do not provide specific criteria, they may be safety-significant. Further assessment is required since Pickering structures were not explicitly designed to meet these requirements and meeting these requirements could have significance for operation of the Pickering station beyond 2020. This is identified as **PSR2 CSA N291-15 Gap #2**.

The majority of the clauses in N291-15 relate to the design and construction of structures that have already been built and, as such, meeting these requirements is generally not impacted by the possibility of continued operation of Pickering beyond 2020. The only clauses that may be safety significant and that could have significance for operation of the Pickering station beyond 2020 are as follows:

- Clause 6.1.1(b) specifies that the analysis and design of safety-related structures shall include consideration of the design service life. Clause 6.9.2.1.4 specifies that non-metallic liner and joint sealant material shall have a minimum service life equal to the plant service life where repair, re-coating or replacement is not possible. These need to be assessed taking into consideration that operation of Pickering NGS beyond 2020 may exceed the service life assumed in the analysis and design. This is identified as **PSR2 CSA N291-15 Gap #3**.

- Clause 7.3 pertains to in-service examination, which is addressed by the PSR2 gap identified in OPG Report P-REP-03680-00024 R000 [B.17-69] relating to submission of Periodic Inspection Plans and Life Cycle Management Plans for a number of safety-significant civil structures to address fitness for service "to end of mission time" (which will need to be extended for Pickering operation past 2020).

- Clause 9 pertains to aging management, which is addressed by **PSR2 CSA N291-15 Gap #2**.

It is worth noting that CSA N291-15 references a number of other codes and standards for which PSR2 code reviews have been performed. The following codes are referenced in CSA N291-15 and were reviewed as part of the Pickering PSR2.

Code or Standard Number	Code or Standard Title
CSA N285.0-12	General Requirements for Pressure Retaining Systems and Components in CANDU Nuclear Power Plants
CSA N285.4-14	Periodic Inspection of CANDU Nuclear Power Plant Components
CSA N285.5-13	Periodic Inspection of CANDU Nuclear Power Plant Containment Components
CSA N285.6 Series-12	Material Standards for Reactor Components for CANDU Nuclear Power Plants
CSA N286-12	Management System Requirements for Nuclear Power Plants
CSA N287.1-14	General Requirements for Concrete Containment Structures for Nuclear Power Plants
CSA N287.2-08	Material Requirements for Concrete Containment Structures for Nuclear Power Plants
CSA N287.3-14	Design Requirements for Concrete Containment Structures for Nuclear Power Plants
CSA N287.5-11	Examination and Testing Requirements for Concrete Containment Structures for Nuclear Power Plants

Code or Standard Number	Code or Standard Title
CSA N287.7-08	In-Service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants
CSA N289.1-08	General Requirements for Seismic Design and Qualification of CANDU Nuclear Power Plants
CSA N289-2-10	Ground Motion Determination for Seismic Qualification of Nuclear Power Plants
CSA N289.3-10	Design Procedures for Seismic Qualification of Nuclear Power Plants
CNSC REGDOC-2.6.3	Aging Management
NBCC 2010	National Building Code of Canada

B.17.3 Compliance Summary for Pickering PSR2

There are three PSR2 CSA N291-15 gaps which relate to Safety Factor 1 (Plant Design):

1. Clause 6.5.2.2 of CSA N291-15 imposes new requirements for bolted connections in members that are part of the seismic load resisting system. Pickering NGS structures were not explicitly designed to meet these requirements and this is therefore identified as a PSR2 gap.

2. Clause 9 of CSA N291-15 contains new requirements related to aging management (including design provisions to account for aging) that are not in CSA N291-08 and that may have significance for operation of Pickering beyond 2020. Pickering structures were not explicitly designed to meet these requirements and this is therefore identified as a PSR2 gap.

3. Clauses 6.1.1(b) and 6.9.2.1.4 of CSA N291-15 state requirements for aspects of the design that are specifically based on the plant service life. Pickering structures were not explicitly designed or assessed in relation to the requirements of these clauses for operation beyond 2020. This is identified as a PSR2 gap.

There is also one PSR2 gap for CSA N291 related to submission of Periodic Inspection Plans and Life Cycle Management Plans for a number of safety-significant civil structures to address fitness for service "to end of mission time" (which will need to be extended for Pickering operation beyond 2020). The gap is related to Safety Factor 4 (Aging). As discussed in Section B.17.2.2, this issue is identified as a PSR2 gap in OPG Report P-REP-03680-00024 R000, "Pickering 5-8 Continued Operations Plan Review in Support of PNGS Periodic Safety Review 2 (PSR2)". Therefore, a duplicate gap has not been created under CSA N291-15.

B.17.4 References

[B.17-1] CSA Standard N291-15, *Requirements for Safety-Related Structures for Nuclear Power Plants*, November 2015.

[B.17-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 23, 2015, [B.17-3] CSA Standard N291-08, *Requirements for Safety-Related Structures for CANDU Nuclear Power Plants*, March 2008.

[B.17-4] CSA Impact Statement for Public Review, *Product: New Edition; Product Designation: CSA N291; Date of Release: March 2016*, Date not provided.

[B.17-5] CSA Standard N291-08 Update No.1, *Requirements for Safety-Related Structures for CANDU Nuclear Power Plants*, June 2011.

[B.17-6] CSA Standard N291-08 Update No.2, *Requirements for Safety-Related Structures for CANDU Nuclear Power Plants*, September 2014.

[B.17-7] OPG Report P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.17-8] OPG Report, NK38-REP-03680-10111 R000, *Review of CSA Standard N291-08 Requirements for Safety-Related Structures for CANDU Nuclear Power Plants for Darlington Integrated Safety Review,* October 14, 2011.

[B.17-9] OPG Report, NK38-REP-03680-10104 R000, *Darlington NGS Integrated Safety Review (ISR) - Final ISR Report,* October 18, 2011.

[B.17-10] OPG Report, NK38-REP-03680-10111-ADD-001 R000, *Addendum to the CAN/ CSA N291-08 Code Review Report for Darlington ISR,* January 16, 2014.

[B.17-11] OPG Report, NK38-REP-03680-10104-ADD-001 R000, *Darlington NGS Integrated Safety Review (ISR) - Final ISR Report Addendum 01,* June 27, 2013.

[B.17-12] OPG Report, NK38-REP-03680-10186 R000, *Darlington NGS - Global Assessment Report (GAR),* November 14, 2011.

[B.17-13] OPG Report, NK38-REP-03680-10104-ADD-002 R000, *Darlington NGS Integrated Safety Review (ISR) - Final ISR Report Addendum 002,* November 22, 2013.

[B.17-14] OPG Program, N-PROG-MP-0008 R006A, *Integrated Aging Management,* October 2015.

[B.17-15] OPG Report, NK38-REP-03680-10185, *Darlington NGS Integrated Implementation Plan (IIP),* April 30, 2015.

[B.17-16] OPG Report, NK38-REP-03680-10159 R000, *Code Refresh Review of CSA-N291-08-UPD1 Requirements for Safety-Related Structures for CANDU Nuclear Power Plants for DNGS ISR,* July 25, 2013.

[B.17-17] OPG List, P-LIST-06937-00001 R000, *Pickering A and B List of Safety Related Systems,* February 2011.

[B.17-18] OPG Charter, N-CHAR-AS-0002 R018, *Nuclear Management System,* March 2015.

[B.17-19] AECL Assessment Document, 44RS-00531-ASD-001 R004, *Review of Pickering 'A' Design Against Current Codes and Standards,* November 2000.

[B.17-20] OPG Design Basis Document, NK30-DBD-01310-00001 R000, *Plant Level Topical Design Basis Document,* October 1999.

[B.17-21] Ontario Hydro Design Manual, NA44-21000 R000, *Pickering Nuclear Generating Station Design Manual – Reactor Building – General,* December 1972.

[B.17-22] Ontario Hydro Design Manual, NA44-21040 R000, *Pickering Nuclear Generating Station Design Manual – Reactor Building Floor Loadings,* April 1969.

[B.17-23] Ontario Hydro Design Manual, NA44-21149 R000, *Pickering Nuclear Generating Station Design Manual – Reactor Building Dome,* February 1970.

[B.17-24] Ontario Hydro Design Manual, NA44-21240 R000, *Pickering Generating Station Design Manual – Reactor Building Interior Walls and Floors,* June 1970.

[B.17-25] Ontario Hydro Design Manual, NA44-21250 R000, *Pickering Generating Station Design Manual – Reactor Building Structural Steel,* September 1969.

[B.17-26] Ontario Hydro Design Manual, NA44-21300.1 R000, *Pickering Generating Station Design Manual – Calandria Vault – General,* February 1971.

[B.17-27] Ontario Hydro Design Manual, NA44-21300.2 R000, *Pickering Generating Station Design Manual – Calandria Vault Structure*, September 1969.

[B.17-28] Ontario Hydro Design Manual, NA44-21500 R000, *Pickering Generating Station Design Manual – Spent Fuel Storage and Receiving Bays,* March 1972.

[B.17-29] Ontario Hydro Design Manual, NA44-22000 R000, *Pickering Generating Station A Design Manual – Powerhouse & Reactor Auxiliary Bay,* January 1992.

[B.17-30] Ontario Hydro Design Manual, NA44-22500 R000, *Pickering Generating Station Design Manual – Turbine Block,* March 1969.

[B.17-31] Ontario Hydro Design Manual, NA44-23000.1 R000, *Pickering Generating Station Design Manual – Intake Structures,* February 1969.

[B.17-32] Ontario Hydro Design Manual, NA44-23000.2 R000, *Pickering Generating Station Design Manual – C.W. System Supply and Discharge,* February 1969.

[B.17-33] Ontario Hydro Design Manual, NA44-23000.3 R000, *Pickering Generating Station Design Manual – Screenhouse Structural,* March 1969.

[B.17-34] Ontario Hydro Design Manual, NA44-24140 R000, *Pickering Generating Station Design Manual – Service Wing Floor Loadings,* October 1969.

[B.17-35] OPG Design Manual, NA44-25000 R002, *Pickering Design Manual – Vacuum Building,* April 2005.

[B.17-36] OPG Design Manual, NA44-25200 R001, *Pickering Design Manual – Pressure Relief Duct,* April 2005.

[B.17-37] OPG Design Manual, NA44-29800 R000, *Pickering Generating Station Heavy Water Upgrading Plant Design Manual,* November 1971.

[B.17-38] Ontario Hydro Design Manual, NK30-20000 R001, *Pickering Generating Station B Design Manual – Buildings General,* December 1988.

[B.17-39] Ontario Hydro Design Manual, NK30-21000 R002, *Pickering Generating Station B Design Manual – Reactor Building – General,* February 1989.

[B.17-40] Ontario Hydro Design Manual, NK30-21040 R002, *Pickering Generating Station B Design Manual – Reactor Building Floor Loadings,* September 1988.

[B.17-41] Ontario Hydro Design Manual, NK30-21140 R002, *Pickering Generating Station B Design Manual – Reactor Building Foundation and Perimeter Wall,* November 1988.

[B.17-42] Ontario Hydro Design Manual, NK30-21149 R002, *Pickering Generating Station B Design Manual – Reactor Building Dome,* December 1988.

[B.17-43] Ontario Hydro Design Manual, NK30-21240 R002, *Pickering Generating Station B Design Manual – Reactor Building Interior Walls and Floors,* November 1988.

[B.17-44] Ontario Hydro Design Manual, NK30-21250 R002, *Pickering Generating Station B Design Manual – Reactor Building Structural Steel,* December 1988.

[B.17-45] Ontario Hydro Design Manual, NK30-21300 R001, *Pickering Generating Station B Design Manual – Calandria Vault Structure,* August 1982.

[B.17-46] Ontario Hydro Design Manual, NK30-21500 R001, *Pickering Generating Station B Design Manual – Irradiated Fuel Storage Bay,* August 1982.

[B.17-47] Ontario Hydro Design Manual, NK30-22040 R000, *Pickering Generating Station B Design Manual – Powerhouse Floor Loadings,* March 1980.

[B.17-48] Ontario Hydro Design Manual, NK30-22041 R000, *Pickering Generating Station B Design Manual – Powerhouse Column Foundations,* April 1980.

[B.17-49] Ontario Hydro Design Manual, NK30-22050 R003, *Pickering Generating Station B Design Manual – Powerhouse Superstructure,* December 1992.

[B.17-50] Ontario Hydro Design Manual, NK30-22500 R000, *Pickering Generating Station B Design Manual – Turbine Block,* December 1977.

[B.17-51] Ontario Hydro Design Manual, NK30-23000.1 R000, *Pickering Generating Station B Design Manual – Intake Structures,* December 1977.

[B.17-52] Ontario Hydro Design Manual, NK30-23000.2 R000, *Pickering Generating Station B Design Manual – CW System Supply and Discharge,* December 1977.

[B.17-53] Ontario Hydro Design Manual, NK30-23000.3 R000, *Pickering Generating Station B Design Manual – Screenhouse – Structural,* January 1978.

[B.17-54] Ontario Hydro Design Manual, NK30-25200 R002, *Pickering Generating Station B Design Manual – Pressure Relief Duct*, December 1988.

[B.17-55] Ontario Hydro Design Manual, NK30-23910 R001, *Pickering Generating Station B Design Manual – Emergency Power and Water Building,* July 1982.

[B.17-56] Ontario Hydro Report, NA44-REP-02004-0073 Sht: 0002 R000, *Seismic Assessment of Pickering A Nuclear Generating Station Summary Report Volume 2 of 7,* February 1998.

[B.17-57] OPG Report, NA44-REP-02004-00002 R001, *Pickering NGS A Seismic Success Path Addendum Including the Safe Shutdown Equipment List,* August 2013.

[B.17-58] OPG Design Guide, NA44-DG-03650-00001 R003, *Seismic Design Guide for Seismic Qualification of Pickering NGS A Success Path Structures, Systems, and Components,* February 2012.

[B.17-59] OPG Report, NK30-DBD-03600-00001 R000, *Pickering B Separation Philosophy & Common Mode Events Topical Design Basis Document,* June 2000.

[B.17-60] Ontario Hydro Design Manual, NA44-34400 R001, *Pickering Generating Station Design Manual – Spent Fuel Storage Bay Cooling and Purification System,* September 1973.

[B.17-61] OPG Design Manual, NK30-DM-35710-00001 R000, *Pickering Design Manual – Auxiliary Irradiated Fuel Bay Facility Cooling and Purification System,* November 2013.

[B.17-62] OPG Design Manual, NK30-DM-34400-00001 R001, *Pickering Design Manual – Irradiated Fuel Storage Bay Cooling and Purification System,* June 2013.

[B.17-63] OPG Abnormal Incident Manual, NA44-AIM-014-09013-06 R018, *Abnormal Incidents Manual – Seismic Event,* February 2016.

[B.17-64] OPG Abnormal Incident Manual, NK30-AIM-058-09013-06 R027, *Abnormal Incidents Manual – Seismic/Common Mode Event,* February 2016.

[B.17-65] OPG Report, NK30-REP-03680-00001 R000, *Pickering NGS-B Integrated Safety Review – Plant Design Safety Factor,* August 2007.

[B.17-66] OPG Plan, NK30-PLAN-00531-00001 R005, *Pickering 5-8 Continued Operations Plan,* November 24, 2015.

[B.17-67] CSA Standard, N287.7-08 (R2013), *In-Service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants, May 2008; Update No.1,* September 2010.

[B.17-68] OPG Report, P-REP-03680-00004 R000, *Pickering NGS Periodic Safety Review 2: Code and Standard Reviews for Safety Factors 2 (Actual Condition of SSCs), 3 (Equipment Qualification) and 4 (Aging),* July 13, 2016.

[B.17-69] OPG Report, P-REP-03680-00024 R000, *Pickering 5-8 Continued Operations Plan Review in Support of PNGS Periodic Safety Review 2 (PSR2),* January 2017.

[B.17-70] OPG Report, N-REP-00590-00002 R000, *Code-Over-Code Review Report: CSA N291-08 (R2013) Including Update No.1 2011 and Update No.2 2014 for the Year 2014,* November 11, 2014.

[B.17-71] CSA Standard, A23.1-14/A23.2-14, *Concrete Materials and Methods of Concrete Construction/Test Methods and Standard Practices for Concrete*, August 2014.

[B.17-72] CSA Standard, W186-M1990 (R2012), *Welding in Reinforcing Bars in Reinforced Concrete Construction*, December 1990.

[B.17-73] CSA Standard, W186-M1990 (R2007), *Welding in Reinforcing Bars in Reinforced Concrete Construction*, December 1990.

[B.17-74] OPG Report, NK30-SR-01320-00002 R004, "*Pickering B Safety Report - Part 2*", Table 2-1 (Design Weather Data), October 2012.

[B.17-75] CSA Standard, A23.3-14, *Design of Concrete Structures,* June 2014.

B.18 CSA N285.6 Series-12, "Material Standards for Reactor Components for CANDU Nuclear Power Plants"

B.18.1 Background

The following, paraphrased from the scope of the CSA N285.6 section of the CSA N285.0/N285.6 Series-12 (including Updates No. 1 and No. 2) [B.18-1], provides a brief overview of the purpose of this standard and the requirements expressed therein:

The purpose of the CSA N285.6 series is to provide material, testing, and fabrication requirements for various CANDU reactor components and materials including:

- *Pressure tubes for use in CANDU fuel channels.*
- *Seamless zirconium alloy tubing for reactivity control units.*
- *Annealed seamless zirconium alloy tubing for liquid injection shutdown system (LISS) nozzles.*
- *Thin-walled, large-diameter zirconium alloy tubing.*
- *Non-destructive examination criteria for zirconium alloys.*
- *Zirconium alloy design data.*
- *Martensitic stainless steel for fuel-channel end fittings.*
- *Materials for supports for pressure-retaining items.*
- *Nickel-based alloy wire for fuel-channel spacers.*
- *Zirconium alloy wire.*
- *Zirconium alloy bars and rods for reactivity control units.*
- *Zirconium alloy sheet, strip and plate for reactivity control units.*

CSA N285.6 Series is directly relevant to Safety Factor 1 (Plant Design).

CSA N285.6 Series is identified in Appendix E.1 of the R04 Pickering Licence Conditions Handbook [B.18-2] as "Guidance or Criteria".

Since 2008, the CSA N285.6 Series has been published with CSA N285.0. The CSA N285.0/N285.6 Series-12 is the second edition of the CSA N285.0/N285.6 Series as it supersedes the previous series published in 2008. The N285.6-12 portion also supersedes previous editions of CSA N285.6 published in 2005 and 1988.

The preface to the N285.0/N285.6 Series-12 edition indicates that "CSA N285.6.5 was withdrawn in the 2005 edition of the CSA N285.6 Series as the material that it covered (heat-treated Zr-2.5Nb-0.5 Cu wire for fuel-channel spacers) is no longer used for new spacers."

The CSA N285.0/N285.6 Series-12 Impact Statement [B.18-3] provides a "Summary of significant changes from the previous edition" which identifies several changes to the Standard as described in Section B.18.2.2.

The results of PSR1 CSA N285.6 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.18.2. As identified in Reference [B.18-4], the Pickering PSR2 review of the CSA N285.6 Series-12 is an Incremental Review. PSR2 Incremental Review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.
- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.18.2 Compliance Assessment for Pickering PSR2

B.18.2.1 Application of PSR1 Reviews

The versions of the N285.6 Series (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

OPG Report NK30-REP-03680-00001 R000 [B.18-5] performed a code review of CSA N285.6-05 against OPG documentation (typically AECL specifications), and stated:

> *The conclusion of the review is that adoption of this standard has no impact on the original design basis of the Pickering B plant.*
>
> *No additional action is required to establish compliance with the Standard CSA N285.6-05 of the Pickering 'B' SOE [Safe Operating Envelope] systems after the planned refurbishment has been completed.*

N285.6-05 was comprised of ten standards:

- N285.6.1-05 Pressure tubes for use in CANDU fuel channels

- N285.6.2-05 Seamless zirconium alloy tubing for reactivity control units
- N285.6.3-05 Annealed seamless zirconium alloy tubing for liquid injection shutdown system (LISS) nozzles
- N285.6.4-05 Thin-walled, large diameter zirconium alloy tubing
- N285.6.6-05 Non-destructive examination criteria for zirconium alloys
- N285.6.7-05 Zirconium alloy design data
- N285.6.8-05 Martensitic stainless steel for fuel-channel end fittings
- N285.6.9-05 Materials for supports for pressure-retaining items
- N285.6.10-05 Nickel-based alloy wire for fuel-channel spacers
- N285.6.11-05 Zirconium alloy wire

OPG Report NK30-REP-03680-00001 R000 [B.18-5] excluded N285.6.1, 6.3, 6.6, 6.7, 6.8, 6.10, and 6.11 from code review as they cover requirements for fabrication, properties, and inspection of components that were planned to be replaced as part of Pickering B refurbishment with components meeting the latest versions of the standards. N285.6.9 was not assessed in detail based on an assessment that concluded that Pickering B generally meets the requirements of N285.6.9-05 [B.18-5].

N285.6.2-05 and N285.6.4-05 were reviewed clause-by-clause and 13 gaps were found which were classified as Acceptable Deviations (AD). Three ADs were dispositioned with having no safety significance, while additional assessment was required for 10 of the gaps [B.18-6]. These 10 ADs were addressed as part of the Final ISR report [B.18-7], in attached report [B.18-8]. The dispositions were provided in Appendix A.10 of [B.18-8] and A.10 Supplement 1-209. This assessment concluded that Pickering had an effective Reactor Assembly Aging Management Strategy and Plan and that there were no operational or aging related issues that would require components to be replaced. This combined with demonstrated compliance with the original design specifications, provides assurance that these gaps are acceptable.

These gaps relating to N285.6.2-05 and N286.5.4-05 are also applicable to Pickering Units 1,4. The disposition to the gaps as outlined above is equally applicable to Pickering 1,4, given the very similar design and material specifications between the units. Further detail regarding the plant specific designs and material is provided in Section B.18.2.2 below.

<u>Pickering Units 1,4</u>

The Pickering A Basis for Return to Service [B.18-9] identified CAN/CSA-N285.6-88 for review.

An assessment of CSA N285.6-88 was conducted in 2000 and documented [B.18-10]. This assessment performed clause-by-clause reviews of the nine standards in CAN/CSA-N285.6-88 (N285.6.1 to 6.9) against OPG documentation (typically AECL specifications). The review concluded that CSA N285.6.1 had no gaps; N285.6.2 had 3 ADs; N285.6.4 had 2 ADs; and

N285.6.7/6.8/6.9 had no gaps. Standards N285.6.3/6.5 are not applicable to Pickering Units 1,4 and Standard N285.6.6 is referenced by the other N285.6 standards whose evaluations demonstrate compliance with N285.6.6.

This comprehensive assessment against the 1988 version of the standards provides assurance that Pickering 1,4 complied with the original specifications and had no safety significant compliance gaps in 2000. However, the N285.6 series has subsequently been revised in 2005, 2008 and 2012. Hence, it is necessary to revisit the assessments performed to determine what changes have been made to the code, the impact and safety significance. This is addressed in Section B.18.2.2 below.

Darlington NGS

OPG Report NK38-REP-03680-10033 R000 [B.18-11] documents a review of CSA N285.6-08 against OPG documentation (typically AECL specifications) for the Darlington ISR, which identified 29 gaps against all of the 2008 versions of the standards listed above under the Pickering Units 5-8 review, except for N285.6.9-08. The report assessed some of these gaps to be minor and concluded the following:

> *The current version of CSA standard N285.6-08 "Material standards for reactor components for CANDU power plants", which contains ten individual standards, is the third edition which was published in June 2008. This series of material standards has been reviewed Clause-by-Clause and a number of gaps and deviations from the standards have been identified. A detailed analysis presented in the compliance discussion of each Clause presented in Appendix C indicates that the various noted deviations have little to no significant impact on the specified material property from being achieved in the fabricated components installed in the Darlington plant.*
>
> *Overall, the conclusion of the review is that adoption of this standard may have some impact on the original design basis of the Darlington plant due to the following factors associated with identified gaps:*
>
> *(a) The Limit of 0.0005 wt% hydrogen concentration in finished/autoclaved pressure tube material in Clause 7.9 of CSA N285.6.1-08;*
>
> *(b) The application UT-1 Standard requirements in Clause 7.1 of CSA N285.6.6-08 to seamless zirconium alloy reactivity control unit tubing;*
>
> *(c) The hydrostatic testing requirement for seamless small-diameter tubes in Clause 5.3 of CSA N285.6.2-08;*
>
> *(d) The application of UT-1 Standard requirements in Clause 7.1 of CSA N285.6.6-08 to seamless LISS nozzle tubing;*
>
> *(e) The application of UT-6 Standard requirements in Clause 7.6 of CSA N285.6.6-08 to seam-welded, thin-walled large-diameter tubes;*
>
> *(f) The axial and transverse yield strength requirements in Clause 7.1 and 7.2 of CSA N285.6.4-08 for seam-welded, thin-walled, large-diameter tubes;*

(g) The application of V-1 Standard requirements in Clause 4.1 of CSA N285.6.6-08 to seamless small diameter tubes;

(h) The application of V-2 Standard requirements in Clause 4.2 of CSA N285.6.6-08 to seam-welded, thin-walled large-diameter tubes;

(i) The application of V-4 Standard requirements in Clause 4.4 CSA N285.6.6-08 to pressure tubes;

(j) The noted differences in design data for Young's modulus, shear modulus and thermal expansion coefficient in the Darlington NGS design manual and those in Tables 1-4 in CSA N285.6.7-08;

(k) Methods and practices relating to chemical analysis of type 403 stainless steel end fitting material in Clause 5.2 of CSA N285.6.8-08;

(l) The noted differences in design data for thermal expansion coefficient and thermal conductivity in the Darlington NGS design manual and those in Annex A of CSA N285.6.8-08.

The notes, as shown in the affected clause of CSA N285.6-08 in Appendix C, indicate that the respective AECL specification provides more specific or more stringent requirements than the corresponding CSA Standard. These notes provide more detailed information than is available from the current Standard and should prove useful if fabrication of new components is required for refurbishment purposes.

OPG Report NK38-REP-03680-10104 R000 [B.18-12], "Darlington NGS Integrated Safety Review (ISR) - Final ISR Report", packages the N285.6-08 gaps into 3 issues, and classifies them all as Acceptable Deviations with no further action required, with the following statements:

Issue D021: Cleaning Solution Composition Limits [B.18-13]

In-service monitoring and inspections, along with planned component replacements and inspections, provide assurance that reactor components will continue to meet the design intent.

Issue D022: Non-destructive Testing of Reactor Components [B.18-14]

The variation from the modern requirements has no significant impact on the specified material properties and components identified in this issue.

Issue D023: Material Properties of Reactor Components [B.18-15]

The identified gaps are against the updated and new clauses from CSA N285.6-Series 08. Some critical components specified in the gaps (pressure tubes, calandria tubes, and end fittings) will be replaced as part of the refurbishment scope according to NK38-PLAN-01060-10003 R006. The design, fabrication, examination, testing and inspection of the replacement components will be in accordance with the latest versions of modern applicable codes and standards.

In addition, reactor components are subject to stringent aging management and inspection requirements as per N-PLAN-01060-10003 R006. Section 2.4 of N-PLAN-01060-10008 R000, "Reactor Components & Structures Life Cycle Management Plan: Technical Basis Document", discussed various degradation mechanisms and inspection requirements for reactivity mechanism Guide Tube (GT) assemblies in detail for the current as well as refurbished life of the reactor.

OPG Report NK38-REP-03680-10201 R001 [B.18-16], "ISR Open Issues and Acceptable Deviations – Adequacy Review", confirms the re-categorization of Darlington issues D021/D022/D023 as Acceptable Deviations with no further action required.

Subsequent to the review of N285.6-08, a Code Refresh review of CSA N285.6-12 against N285.6-08 was performed for Darlington in OPG Report NK38-REP-03680-10139 R000 [B.18-17] which concluded:

Review of the 2008 versus 2012 (including the September 2013 update, Update No.1) Code editions did not identify any significant changes. Any revisions identified involved relaxation in the code requirements. Consequently, the gaps identified in the original Code Review Report remain unchanged and the update of CSA N285.6-12 series of standards has no impact on Darlington's Integrated Safety Review.

As noted above, no code to code comparisons were made between the three versions of N285.6 used for Pickering A Return to Service or the Pickering B and Darlington ISRs (i.e., N285.6-88, -05, and -08). Therefore, Section B.18.2.2 below assesses and compares the differences between the plant designs and component materials used.

B.18.2.2 Application of Post PSR1 Reviews

As discussed above, a Code Refresh review of the changes in CSA N285.6-12 against N285.6-08 was performed as part of the Darlington ISR. In addition, a code-over-code review was conducted to evaluate changes between the 2012 versions of the standards and Update 1. The changes are documented in N-REP-00590-0520105, "Code-over-Code Report: CSA N285.0/CSA N285.6 Series For Year 2014" [B.18-18]. This code update only included changes to N285.6.6-12 which are summarized as follows:

1. Clause 4.3.2: clarification on inspection requirements for external and internal surfaces of tubular products.

2. Clause 4.3.3: acceptance criteria for "inspection tooling scratches" changed to "linear type scratches" and clarified wording to align with Non-Destructive Examination terminology and requirements.

The review in reference [B.18-18] concluded that these changes only provided additional clarification.

Update No. 2 (2014) to CSA N285.0/N285.6 included the following changes to N285.6 Series-12, paraphrased from the Impact Statement for Publication [B.18-3]:

1. *Two new material standards for materials used in Reactivity Control Units (RCUs) that were not covered by the existing standards were added, as follows:*

    a. *N285.6.12-14 Zirconium alloy bars and rods for reactivity control units*

    b. *N285.6.13-14 Zirconium alloy sheet, strip and plate for reactivity control units*

2. *A correction to the referenced clause in clauses 7.3.3 of standard N285.6.6-12.*

Hence, the above assessments support the conclusion that no significant changes to the previously existing content in N285.6 occurred between 2008 (Darlington ISR version) and 2012 version (used for this evaluation), including Updates No. 1 (2013) and Update No. 2 (2014). The two new standards introduced in Update No. 2 are addressed under their respective sub-sections below.

Although the three OPG plants (Darlington, Pickering 1,4 and Pickering 5,8) have significant differences in design and component ages, there are many similarities between the plants in terms of material specifications for reactor components relating to N285.6. Pickering 1,4 units underwent a Fuel Channel Replacement Program starting in the mid-1980s, and both Pickering 1,4 and Pickering 5-8 have made several changes to reactor in-core components, e.g., completely replacing in-core flux monitor assemblies. In addition, a combination of maintenance and inspections have resulted in replacement of several additional in-core components (e.g., Fuel Channels, Calandria Tubes, Garter Springs). These activities provide assurance of fitness for service and assurance that degradation mechanisms are known and monitored [B.18-19]. Further, all units have active Life Cycle Management Plans (LCMPs) for reactor components and structures [B.18-20], and fuel channels [B.18-21].

The CSA standard material requirements are implemented in Technical Specifications that are used for procurement. These specifications augment existing requirements usually specified by American Society for Testing of Materials (ASTM) or American Society of Mechanical Engineers (ASME). Hence, the approach to this incremental review is to compare the associated material/component specifications for each plant against the CSA requirements.

The sub-sections below summarize the applicable specifications for each plant (Pickering 1,4; Pickering 5-8 and Darlington) and assesses their level of compliance against the current CSA N285.6 series of material requirements. The Darlington references are identified so that they can be compared against the Pickering specifications.

<u>CSA N285.6.1-12: Pressure Tubes for Use in CANDU Fuel Channels</u>

CSA N285.6.1-12 covers the manufacturing and property requirements for seamless, cold-worked and stress-relieved Zr-2.5Nb pressure tubes. The three stages of fabrication are: (1) the preparation of ingots by consumable electrode melting and hot forging of these ingots into billets; (2) the preparation of the billets into seamless shells, and the cold working of the extruded shells into tubes; (3) the cleaning, caustic washing, autoclaving, straightening, and inspection of the tubes. A similar set of technical specifications was used in the fabrication and final processing of the pressure tubes for each plant. These are identified below:

Pickering 1,4 [B.18-10]	Pickering 5-8 [B.18-27][B.18-28]	Darlington [B.18-11]
Unit 1 • TS-44-31110-17 R0 [B.18-22] (manufacturing), based on TS-XX[16]-31110-5 R7 [B.18-24] • TS-XX-31110-4 R6 [B.18-23] (cleaning and stress relief) **Unit 4** • TS-XX-31110-5 R9 [B.18-26] (manufacturing) • TS-XX-31110-4 R6 [B.18-23] (cleaning and stress relief)	• TS-XX-31110-5 R3 [B.18-25] (manufacturing) **Unit 5** • TS-XX-31110-4 R1 per [B.18-27] (cleaning and stress relief) **Units 6,7,8** • TS-XX-31110-8 R1 per [B.18-27] (cleaning and stress relief)	• TS-XX-31110-5 R7 [B.18-24] (manufacturing) • TS-XX-31110-4 R5 [B.18-29] (cleaning and stress relief)

The Darlington review performed and detailed a comparative review of material and mechanical properties between the various specifications, compiled in Tables 3 and 4 of [B.18-11]. The clause by clause review of this standard for Darlington [B.18-11] identified the following gaps:

1. Clause 4.5, Limits on halogen and sulphur contents in final cleaning solutions for pressure tubes.

2. Clause 7.9, Limit on hydrogen concentration in autoclaved pressure tube material.

3. Clause 7.11, Visual inspection of pressure tubes to CSA N285.6.6-08 Standard V-4.

Gap 1 arises because the AECL specifications used for Darlington [B.18-24], [B.18-29] do not include any limits on halogen and sulphur contents in the cleaning solutions. Gap 2 is the result of the new requirement in Clause 7.9 which limits the measured hydrogen concentration in autoclaved pressure tube material to a maximum of 0.0005 wt % (5 ppm). Gap 3 arises because the 2008 Standard in Clause 7.11(a) limits the scratch length to a maximum of 25 mm whereas the original AECL specification allows a maximum length of 150 mm (note, this is now addressed in Clause 4.3.3 of the 2012 version of N285.6.6-12). For Material/manufacturing requirements per TS-XX-31110-5 above, Pickering Unit 1 and Darlington are the same (R7 - 1979). For Pickering Unit 4, a more recent revision (R9 -1988) was used. This new version is more comprehensive and has some tighter specifications. Therefore, Clause 7.11 would not result in a gap for Pickering Unit 4. For Pickering 5-8, an older version of the specification was used (R3-1977). The revisions between 1977 and 1981 were relatively minor as the manufacturing process for the installed pressure tubes was nearly identical, with the most significant change being that a post-heat treat quenching requirement was added [B.18-30]. This is not a requirement of the CSA standard and does not result in an additional gap. Additionally, per Figure 3 of [B.18-30], the comparison of measured mechanical properties and corrosion data demonstrates minimal impact.

[16] 'XX' is included in the actual document number. The specific plant of application is identified in the document Revision and/or Issue # summary.

For pressure tube finishing, including inspection, TS-XX-31110-4 was used for Pickering Units 1,4; Pickering Unit 5 and Darlington. Pickering Units 6,7,8 used TS-XX-31110-8 which was subsequently superseded by TS-XX-31110-4 R5 [B.18-29]. The Darlington review of Clause 7.11 [B.18-11] is also applicable to these specifications, and the above gap 3 is also applicable to Pickering 5-8.

Based on a comparison of the various specifications against the standard, it is concluded that the issues identified in the Darlington ISR review that led to Darlington gaps are also applicable to Pickering and that there are no additional Pickering specific gaps. These Darlington gaps were evaluated in [B.18-13], [B.18-14] and [B.18-15] with resolutions provided. All gaps were confirmed to be Acceptable Deviations with low safety significance, identifying that on-going surveillance per the Fuel Channel LCMP [B.18-20] provides adequate assurance of continued fitness for service. These conclusions are also applicable to Pickering and therefore the Darlington issues are not PSR2 gaps.

As already noted, Darlington completed a clause by clause review against the 2008 version of the standard and also determined that there are no significant incremental requirements between the 2008 and 2012 versions of the standard.

There are no Pickering PSR2 gaps for CSA N285.6.1-12.

CSA N285.6.2-12: Seamless Zirconium Alloy Tubing for Reactivity Control Units

CSA N285.6.2-12 addresses the manufacturing and property requirements for seamless, zirconium alloy tubing used in the fabrication of reactivity control units. The AECL specifications that applied to the manufacture are identified below.

Pickering 1,4 [B.18-10]	Pickering 5-8 [B.18-5]	Darlington [B.18-11]
• NP-M-653 R1 [B.18-31]	• TS-XX-31700-6 R0 [B.18-32]	• TS-XX-31700-6 R0 [B.18-32]

A list of seamless Zircaloy-2 tubing used for reactivity units is presented in Table 5 of [B.18-11]. Tabulations of the mechanical property and chemical specification requirements according to the above-mentioned AECL specification for Darlington, relevant ASTM Standards and N285.6.2-08 requirements for Zircaloy-2 tubing used for reactivity units are presented in Tables 4 and 6 of [B.18-11]. Pickering 1,4 used NP-M-653 R1 which references ASTM B350. Since N285.6.2-12 now references ASTM 350, the chemical properties are consistent with the standard.

The Darlington clause by clause review of this standard identified the following gaps in meeting the requirements of CSA N285.6.2-08:

1. Clause 3.5: Limits on halogen and sulphur contents in final cleaning solutions for seamless reactivity control unit tubing.

2. Clause 5.3: Hydrostatic testing of seamless small-diameter tubes.

3. Clause 5.4: Visual inspection of seamless small-diameter tubes to CSA N285.6.6-0

Standard V-1.

4. Clause 5.5: Ultrasonic inspection of seamless small-diameter tubes to CSA N285.6.6-08 Standard UT-1.

Gap 1 arises because the specified maximum limits for halogen and sulphur are higher than the limits specified in Clause 3.5 of N285.6.2-08. However, this gap in meeting the specified limits is considered minor, and should have no impact on the fabricated seamless Zircaloy-2 tubing meeting the requirements for the reactivity control unit application [B.18-11]. Gap 2 arises because Reference [B.18-31] does not specify hydrostatic testing for seamless small diameter tubing and the requirements in Clause 5.3 of the Standard are not met. Gap 3 relates to no explicit specification of boroscope or videoscope methods required by Clause 4.1.2 of N285.6.6-12. Gap 4 arises because Reference [B.18-31] does not include an ultrasonic inspection for seamless reactivity control unit tubing. It should be noted that these gaps were also identified in the Pickering B ISR [B.18-5] and the gaps are also applicable to PSR2. As noted, in Section B.18.2.1, above, Pickering 1,4 assessed NP-M-653 Rev. 1 against the 1988 version of CSA N285.6.2 and issues associated with Darlington gaps 2, 3 and 4 were are also identified. No issues relating to cleaning solutions were identified, however, given that the tighter specification now exists, it can be surmised that this would have also been a gap for the older Pickering 1,4 components.

Therefore, based on a comparison of the various specifications against the standard, it is concluded that the gaps identified for Darlington are also applicable to Pickering and there are no additional Pickering specific gaps. These gaps were evaluated in [B.18-13] and [B.18-14] with resolutions provided. All gaps were confirmed to be Acceptable Deviations with low safety significance, identifying that on-going surveillance per the component LCMP [B.18-20] provides adequate assurance of continued fitness for service. These conclusions are also applicable to Pickering and therefore the Darlington issues are not PSR2 gaps.

As already noted, Darlington has done a clause by clause review against the 2008 version of the standard and determined that there are no significant incremental requirements between the 2008 and 2012 versions of the standard.

There are no Pickering PSR2 gaps for CSA N285.6.2-12.

CSA N285.6.3-12: Annealed Seamless Zirconium Alloy Tubing for Liquid Injection Shutdown Systems (LISS) Nozzles

CSA N285.6.3-12 covers the manufacturing and property requirements for seamless zirconium alloy tubing used in fabrication of LISS nozzles. The applicable AECL specifications are identified below. Note, there is no LISS system in Pickering 1,4.

Pickering 1,4	Pickering 5-8	Darlington [B.18-11]
• Not Applicable to Pickering 1,4	• TS-XX-31761-5 R0 [B.18-33]	• TS-XX-31761-5 R1 [B.18-34] (Refers to ASTM B350-73)

Tabulations of the mechanical property and chemical specification requirements according to the above-mentioned AECL specification for Darlington, relevant ASTM Standards and N285.6.3-08 requirements for Zircaloy-2 tubing used for LISS nozzles are presented in Tables 4 and 6 of [B.18-11]. A review of Pickering 5-8 history dockets [B.18-35] and [B.18-36] against the ASTM B 350 requirements in Table 6 of [B.18-11] confirms that all chemical specifications were the same as or more stringent than the ASTM requirements.

The Darlington review identified the following gaps in meeting the requirements of CSA N285.6.3-08:

1. Clause 3.5: Limits on halogen and sulphur contents in final cleaning solutions for seamless LISS nozzle tubing.

2. Clause 5.1: Ultrasonic inspection of seamless LISS nozzle tubing to CSA N285.6.6-08 Standard UT-1.

3. Clause 5.3: Liquid Penetrant examination procedure for seamless LISS nozzle tubing.

Gap 1 relates to Section 5.7.3 of Reference [B.18-34] where the limits are higher than the limits specified in Clause 3.5 of N285.6.3-08. A review of [B.18-33] confirms that this issue is also applicable to Pickering 5-8. Similarly, gap 2 is applicable to Pickering 5-8, requiring an explicit procedure or acceptance criteria for the ultrasonic inspection that is in accordance with Standard UT-1 in Clause 5.1 of N285.6.6-08. Gap 3 is also applicable to Pickering 5-8 where the liquid penetrant inspection method is not specified in [B.18-33].

Therefore, based on a comparison of the various specifications against the standard, it is concluded that the gaps identified for Darlington are also applicable to Pickering and there are no additional Pickering specific gaps. These gaps were evaluated in [B.18-13] and [B.18-14] with resolutions provided. All gaps were confirmed to be Acceptable Deviations with low safety significance, identifying that on-going surveillance per the component LCMP [B.18-20] provides adequate assurance of continued fitness for service. These conclusions are also applicable to Pickering 5-8 and therefore the Darlington issues are not PSR2 gaps.

Darlington has done a clause by clause review against the 2008 version of the standard and determined that there are no significant incremental requirements between the 2008 and 2012 versions of the standard.

There are no Pickering PSR2 gaps for CSA N285.6.3-12.

CSA N285.6.4-12: Thin-Walled, Large-Diameter Zirconium Alloy Tubing

CSA N285.6.4-12 covers the manufacturing and property requirements for thin-walled, large-diameter tubes that are used for calandria tubes and guide tubes for reactivity control mechanisms. The AECL technical specifications used for fabrication of the tubes are identified below:

Pickering 1,4 [B.18-10]	Pickering 5-8	Darlington [B.18-11]
• NP-M-601 R1 (Calandria Tubes) [B.18-37] • NP-M-653 R1 (Guide Tubes) [B.18-31]	• TS-XX-31230-2 R4 (Calandria Tubes) [B.18-38] • TS-XX-31700-6 R0 (Guide Tubes) [B.18-31]	• TS-38-31231-1 R1 (Calandria Tubes) [B.18-39] • TS-38-31700-14 R0 (Guide Tubes) [B.18-40]

The mechanical property and chemical specification requirements for the above Darlington specifications are presented in Tables 4 and 7 [B.18-11], respectively. The chemical composition of materials used for Pickering 5-8 is the same as those used for Darlington. Pickering 1,4 assessed the NP-M-601 R1 specification against N285.6.4-88 in Reference [B.18-10]. The review compared the NP-M-601 R1, ASTM B350 requirements for Alloy R60802 against N285.6.4-88 and showed all requirements to be met. The same conclusion was applicable to NP-M-653 R1 which also references ASTM B350. N285.6.6-12 now references ASTM 350, hence, chemical properties for Pickering materials are consistent with the standard.

The Darlington clause-by-clause review identified the following gaps in meeting the requirements of CSA N285.6.4-08:

1. Clause 3.5: Limits on halogen and sulphur contents in final cleaning solutions for seam-welded thin-walled large-diameter tubes.

2. Clause 7.1 Transverse yield strength requirements for seam-welded, thin-walled large-diameter tubes.

3. Clause 7.2: Axial yield strength requirements for seam-welded, thin-walled large-diameter tubes.

4. Clause 7.5 Visual inspection of thin-walled, large-diameter tubes to CSA N285.6.6-8 Standard V-2.

5. Clause 7.6: Ultrasonic inspection of thin-walled, large-diameter tubes to CSA N285.6.6-08 Standard UT-6.

Gap 1 relates to the specified maximum limits for halogen and sulphur and has already been identified and addressed above. This gap is equally applicable to all Pickering units.

Gap 2 arises because the transverse room temperature 0.2% yield strength specified in Section 5.4.3 of [B.18-39] is lower than that specified in Clause 7.1 of N285.6.4-08. Similarly gap 3 relates to the axial room temperature 0.2% yield strength specified in Section 5.4.3 of Reference [B.18-39] being lower than that specified in Clause 7.1 of N285.6.4-08. The Pickering 1,4 specifications identified above are below the limits in the standard for both Calandria Tubes and Guide Tubes. The Pickering 5-8 specifications are equivalent to those in the standard for Calandria Tubes and lower for Guide Tubes. For Pickering 1,4, the ISR review against Clause 5.3.2.2 of the N285.6.4-88 review [B.18-10] indicated that history docket test results are also below the standard requirements. However, this was assessed as being acceptable in [B.18-10] when accounting for actual wall thickness and material strengthening due to irradiation. Incorporating these factors would result in the standard requirements being exceeded. Since Pickering 5-8 has the same specified Guide Tube mechanical properties as Pickering 1,4, the same Pickering 1,4 conclusions are applicable to Pickering 5-8. Additionally, the N285.6.7-12 (*Zirconium Alloy Design Data*) yield strength requirement to be used for N285.6.4 finished material is 240 MPa, which is lower than the 320 MPa in N285.6.4-12. All of the Pickering specifications meet the 240 MPa requirement. Given the above, the Pickering components satisfy the requirements of the standard and are therefore acceptable. Hence, gaps 2 and 3 above are not PSR2 gaps for Pickering.

Gap 4 relates to visual surface examination of thin-walled, large-diameter tubes not explicitly requiring boroscope or videoscope inspections. Gap 5 arises because the ultrasonic inspection specified did not meet the requirements of Standard UT-6 in N285.6.6-08. These gaps are also applicable to Pickering.

Therefore, based on a comparison of the various specifications against the standard, it is concluded that gaps 1, 4 and 5 identified for Darlington are also applicable to Pickering and there are no additional Pickering specific gaps. These gaps were evaluated in [B.18-13] and [B.18-14] with resolutions provided. All gaps were confirmed to be Acceptable Deviations with low safety significance, identifying that on-going surveillance per the component LCMP [B.18-20] provides adequate assurance of continued fitness for service. These conclusions are also applicable to Pickering and therefore the Darlington issues are not PSR2 gaps.

As already noted, Darlington has done a clause by clause review against the 2008 version of the standard and determined that there are no significant incremental requirements between the 2008 and 2012 versions of the standard.

There are no Pickering PSR2 gaps for CSA N285.6.4-12.

CSA N285.6.6-12: Non-Destructive Examination Criteria for Zirconium Alloys

CSA N285.6.6-12 covers the non-destructive examination criteria for zirconium alloys. AECL technical specifications relevant for this Standard are those already discussed in the sections above for the reviews of N285.6.1, N285.6.2, N285.6.3 and N285.6.4. The reviews of those standards have already been addressed above. Hence, no further review is required and no PSR2 gaps have been identified.

CSA N285.6.7-12: Zirconium Alloy Design Data

CSA N285.6.7-12 addresses zirconium alloy design data and there is only one clause in this standard. The minimum tensile properties specified in N285.6.1, N285.6.2, N285.6.3 and N285.6.4 presented in Table 1 of the current Standard are compared with the original AECL specifications tabulated in Table 4 of [B.18-11]. Reference [B.18-30] provides the yield and ultimate strengths for all Pickering unit pressure tubes (P1,4 and P5-8). In all cases the requirements in Table 1 of N285.6.7-12 are met. Pickering 5-8 tube stress intensity is included in the Fuel Channel Design Manual, Figure 13 of [B.18-27]. This figure demonstrates that the allowable stresses in Table 1 of N285.6.7-12 for the R60901 Alloy are met for the temperatures.

For N285.6.2-12, N285.6.3-12 and N285.6.4-12 each of the applicable component's mechanical strength requirements have been assessed in their respective code reviews above. Per [B.18-10], all other physical properties are included in References 14, 15 and 16 of that report.

There are no Pickering PSR2 gaps for CSA N285.6.7-12.

CSA N285.6.8-12: Martensitic Stainless Steel for Fuel-Channel End Fittings

CSA N285.6.8-12 covers martensitic stainless steel forged blanks used in fuel channel end fittings. The AECL technical specifications used in the fabrication of the stainless steel forged blanks and the end fittings are identified below.

Pickering 1,4 [B.18-10]	Pickering 5-8 [B.18-27]	Darlington [B.18-11]
• AECL NPE-610 per [B.18-19] (original Unit 1 only) • TS-XX-31120-2 R2 [B.18-42] (Referenced in [B.18-10]) • TS-XX-31120-4 R2 (blanks) [B.18-41]	• TS-XX-31120-2 R2 [B.18-42] • TS-XX-31120-4 R0 (blanks) [B.18-43]	• TS-XX-31120-3 R1 [B.18-44] • TS-XX-31120-4 R2 (blanks) [B.18-41]

Tabulations of material property requirements for AECL specification TS-XX-31120-4 for blanks, relevant ASTM Standards and N285.6.8-08 requirements for type 403 stainless steel fuel channel end fittings, are presented in Table 9 and 10 of [B.18-11], respectively. The Darlington clause by clause review of this standard identified the following gaps in meeting the requirements of CSA N285.6.8-08:

1. Clause 4.3: Minimum temperature for tempering of type 403 SS end fitting blanks;
2. Clause 5.2: Methods and practices for chemical analysis of type 403 SS end fitting material;
3. Annex A design data.

Gap 1 relates to the specified heat treatment of blanks. Section 4.5 of [B.18-41] calls for a minimum tempering temperature of 593°C as opposed to the minimum temperature of 600°C specified in Clause 4.3 of N285.6.6-08. This is the same requirement in the 2012 version of the standard and this gap is also applicable to Pickering 1,4 and 5-8. However, this gap is not considered to have a significant impact since the mechanical properties in the AECL specification and the CSA Standard are the same [B.18-11]. Gap 2 arises because Sections 4.3 and 5.2 of [B.18-41] do not specify explicitly what methods and practices relating to the chemical analysis of the type 403 SS end fitting material are to be followed as required by Clause 5.2 of N285.6.6-08 and -12. This gap is also applicable to Pickering. Gap 3 relates to Darlington design data being slightly different than that in Annex A of the standard. The specifications in Annex A have not changed between the 2008 and 2012 versions of the standard and are also applicable to Pickering. However, the Pickering 1,4 code review [B.18-10] against the 1988 standard identified the units to be indirectly compliant, citing the Design Specification as being sufficient. Hence, Darlington gap 3 is only applicable to Pickering 5-8.

Therefore, based on a comparison of the various specifications against the standard, it is concluded that the gaps identified for Darlington are all applicable to Pickering, with exception of gap 3, which is not applicable to Pickering 1,4. There are no additional Pickering specific gaps. These three gaps were evaluated in [B.18-15] with resolutions provided. The gaps were confirmed to be Acceptable Deviations with low safety significance, identifying that on-going surveillance per the reactor component LCMP [B.18-20] provides adequate assurance of continued fitness for service. These conclusions are also applicable to Pickering and therefore the Darlington issues are not PSR2 gaps.

Therefore, there are no Pickering PSR2 gaps for CSA N285.6.8-12.

CSA N285.6.9-12: Materials for Supports for Pressure-Retaining Items

CSA N285.6.9-12 covers the material for supports for pressure-retaining items in nuclear power plants, and includes the materials permitted by Article NF-2000, in ASME Section III. This applies to all pressure-retaining nuclear systems. Article NF-2000 establishes general requirements for materials for use in supports to meet the strength specifications required by ASME/ASTM. The topics covered in the standard address requirements for material certification, removal of sample coupons from the various fabricated forms of the material and testing procedures to verify material properties.

The N285.6.9 standard did not exist when either Pickering plant was designed or built. Assessments were performed for Pickering A Return to Service [B.18-10] and Pickering B ISR [B.18-5] against the 1988 and 2005 versions of N285.6.9 respectively. Both reviews identified that the stations were indirectly compliant with the requirements by virtue of their compliance with the ASME requirements. The assessments demonstrate that compliance with the ASME code requirements imposed the applicable ASTM requirements. There have not been any significant changes to this standard since the Pickering 1,4 or Pickering 5,8 code reviews have been done as part of PSR1. Therefore, there are no PSR2 gaps relating to CSA N285.6.9-12.

CSA N285.6.10-12: Nickel-Based Alloy Wire for Fuel-Channel Spacers

CSA N285.6.10-12 covers nickel-based alloy wire for fuel channel spacers/garter springs. The AECL technical specifications used in the fabrication of the garter spring type fuel channel annulus spacers are identified below. Note, Pickering 5-8 did not use Inconel X-750 spacers for most of the original installation. However, several Unit 7 and Unit 8 fuel channels were replaced pre-service with tight fitting X-750 spacers. Fuel Channel replacements have also used the X-750 spacers.

Pickering 1,4 [B.18-10]	Pickering 5-8	Darlington [B.18-11]
• TS-XX-31160-3 R2 [B.18-45] (Refers to ASTM B351-79)	• Used Zirconium alloy per withdrawn CSA N285.6.5-05 • TS-XX-31160-3 R2 [B.18-45], used for limited number of specific Unit 7 and 8 channels and for channel replacements.	• TS-XX-31160-3 R2 [B.18-45] (Refers to ASTM B351-79)

The Clause-by-Clause review of this standard for Darlington identified only one gap in meeting the requirements of CSA N285.6.10-08:

1. Clause 3.6: Limits on halogen and sulphur content in cleaning solutions for X-750 spacer wire.

The gap arises because Section 5.5 of Reference [B.18-45] does not contain any specific requirements for halogen and sulphur contents in the final cleaning solution. Given that Pickering Units 1,4 and Units 7,8 used the same specification, this Darlington gap is also applicable. However, this gap in meeting the requirement of Clause 3.6 of N285.6.10-08 was considered minor and to have no significant impact on the nickel-based alloy wire meeting the design requirements for the spacer application in the Darlington reactors [B.18-11]. There are no additional Pickering 1,4 gaps relating to this standard.

This gap was evaluated in [B.18-13] with a resolution provided and confirmed to be an Acceptable Deviation with low safety significance. This conclusion is also applicable to Pickering and therefore the Darlington issue is not a PSR2 gap.

There are no Pickering PSR2 gaps for CSA N285.6.10-12.

CSA N285.6.11-12: Zirconium Alloy Wire

CSA N285.6.11-12 covers zirconium alloy wire used as filler wire in the fabrication of seam-welded, thin-walled, large-diameter tubes and as the girdle wire in the current design annulus fuel channel spacers. Zirconium wire is used in the fabrication of seam-welded, thin-walled, large-diameter Zircaloy-2 tubes and in the fabrication of garter springs. This standard did not exist at the time Pickering 1,4 or 5-8 were designed and built. There were no specific specifications for zirconium wire; rather it is used as required in the respective component specifications. Examples of specifications are included in the table below.

Pickering 1,4 [B.18-10]	Pickering 5-8	Darlington [B.18-11]
• TS-XX-31160-3 R2 (spacers) [B.18-45] (Refers to ASTM B351-79)	• TS-XX-31160-1 R5 (spacers) [B.18-46] • TS-XX-31230-2 R4 (Calandria Tubes) [B.18-35] • TS-XX-31700-6 R0 (Guide Tubes) [B.18-31]	• TS-XX-31160-3 R2 (spacers) [B.18-45] (Refers to ASTM B351-79) • TS-38-31231-1 R1 (Calandria Tubes) [B.18-39] • TS-38-31700-14 R0 (Guide Tubes) [B.18-40]

The material requirements in CSA N285.6.11-12 are the same as those in N285.6.4-12 requiring Zirconium wire properties to meet ASTM B351 Grade R60802 or R60804. The ingot elemental composition is the same as those requirements in N285.6.4-12. Table 8 of [B.18-11] compares the chemical specifications in N285.6.11-08 (same as 2012) to ASTM and the AECL specifications. Specifications for Oxygen, Lead, Tantalum and Vanadium are not provided in the ASTM and AECL requirements. However, this deviation was identified in the Pickering B ISR [B.18-5] for Clause 4.2 of CSA N285.6.4-05 and in the Darlington review of CSA N285.6.11-08 [B.18-11] and dispositioned as not having any consequential impact on the mechanical design.

The Darlington clause-by-clause review of this standard identified only one gap in meeting the requirements of CSA N285.6.11-08:

1. Clause 3.3: Limits on halogen and sulphur contents in cleaning solutions for zirconium alloy wire.

This gap arises because the specified maximum limits for halogen and sulphur in cleaning solutions are higher than the limits specified in Clause 3.3 of N285.6.11-12. This Darlington issue is also applicable to Pickering. However, this gap in meeting the specified limits is minor and should have no impact on the Zircaloy-2 wire meeting the design requirements for the filler wire per [B.18-11].

This gap was evaluated in [B.18-13] with a resolution provided and confirmed to be an Acceptable Deviation with low safety significance. This conclusion is also applicable to Pickering and therefore the Darlington issue is not a PSR2 gap.

There are no Pickering PSR2 gaps for CSA N285.6.11-12.

CSA N285.6.12-14: Zirconium Alloy Bars and Rods for Reactivity Control Units and CSA N285.6.13-14: Zirconium Alloy Sheet, Strip and Plate for Reactivity Control Units

The introduction of the two new standards (N285.6.12-14/.6.13-14) relating to zirconium material properties used in new reactivity mechanisms does not introduce any incremental requirements that have not already been addressed in the above standard reviews related to reactivity control mechanisms. This is because these standards were established to codify existing AECL specifications in CSA requirements and to document the acceptability of materials that are currently in use and that have ASTM specifications for the material. In both standards, the ASTM specifications are augmented with additional specific composition requirements. These same requirements exist in other CSA N285.6-12 standards and are already assessed above. The standards also cross-reference the specific inspection requirements that are also included in the component specifications and N286.6.6-12 standard that is also addressed above. Hence, the introduction of these two new standards has no impact on the existing Pickering plant and there are no Pickering PSR2 gaps.

B.18.3 Compliance Summary for Pickering PSR2

There are no PSR2 gaps for CSA N285.6 Series-12. Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with CSA N285.6 Series-12.

B.18.4 References

[B.18-1] CSA Standard N285.0-12/N285.6 Series-12, *General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants / Material Standards for Reactor Components for CANDU Nuclear Power Plants*, August 2012; Update No. 1: September 2013; Update No. 2: November 2014.

[B.18-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.18-3] CSA Impact Statement for Publication, *Product: Amendment No. 2; Product Designation: CSA N285.0-12/285.6 Series-12 Amendment No. 2*; Date of Release: November 2014, Date not provided.

[B.18-4] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.18-5] OPG Report, NK30-REP-03680-00001 R000, *Pickering NGS-B Integrated Safety Review – Plant Design Safety Factor*, August 2007.

[B.18-6] CNSC Letter, e-Docs # 3256609, OPG File No. NK30-CORR-00531-04876, T.E. Schaubel to D.P. McNeil, *Pickering NGS-B – Integrated Safety Review (ISR) – CNSC Review of Acceptable Deviations and Discrepancies for the Plant Design Safety Factor Report*, June 27, 2008.

[B.18-7] OPG Report, NK30-REP-03680-00015 R000, *Pickering NGS-B Integrated Safety Review (ISR) – Final ISR Report,* August 2009.

[B.18-8] OPG Report, NK30-REP-03680-00016 R000, *OPG Response to CNSC Comments on Pickering NGS-B Integrated Safety Review – Plant Design, Safety Analysis, Safety Performance, Ageing and Equipment Qualification Safety Factors and Discrepancy Resolutions,* (OPG Confidential), September 2009.

[B.18-9] OPG Letter, NA44-CORR-00531-00381, R.J. Strickert to J.S.C Tong, *Pickering A – Updated Basis for Return to Service Document*, April 20, 2001.

[B.18-10] AECL Assessment Document, 44RS-00531-ASD-001 Rev. 04, *Review of Pickering A Design Against Current Codes and Standards*, November 2000.

[B.18-11] OPG Report, NK38-REP-03680-10033 R000, Review of CAN/CSA-N285.6-08 Series (June 2008), *Material Standards for Reactor Components for CANDU NPP for Darlington ISR*, August 2011.

[B.18-12] OPG Report, NK38-REP-03680-10104 R000, *Darlington NGS Integrated Safety Review (ISR) – Final ISR Report*, October 2011.

[B.18-13] OPG Report, NK38-REP-00770-0417587 R0, *Nuclear Refurbishment Issue Resolution Form – Darlington, Cleaning Solution Composition Limits – Issue D021*, May 2011.

[B.18-14] OPG Report, NK38-REP-00770-0417588 R0, *Nuclear Refurbishment Issue Resolution Form – Darlington, Nondestructive Testing of Reactor Components – Issue D022*, May 2011.

[B.18-15] OPG Report, NK38-REP-00770-0417589 R0, *Nuclear Refurbishment Issue Resolution Form – Darlington, Material Properties of Reactor Components – Issue D023*, April 2011.

[B.18-16] OPG Report, NK38-REP-03680-10201 R001, *ISR Open Issues and Acceptable Deviations – Adequacy Review*, October 2014.

[B.18-17] OPG Report, NK38-REP-03680-10139 R000, *Code Refresh Review of CSA-N285.6-12 Material Standards for Reactor Components for CANDU NPP*, February 2014.

[B.18-18] OPG Report, N-REP-00590-0520105 R001, *Code-Over-Code Review Report: CSA-N285.0/CSA N285.6 Series for the Year 2014*, April 2015.

[B.18-19] OPG Plan, N-PLAN-01060-10008 R000, *Reactor Components & Structures Life Cycle Management Plan: Technical Basis Document*, June 2010.

[B.18-20] OPG Plan, N-PLAN-01060-10003 R013, *Reactor Components & Structures Life Cycle Management Plan*, October 2015.

[B.18-21] OPG Plan, N-PLAN-01060-10002 R016, *Fuel Channels Life Cycle Management Plan*, October 2015.

[B.18-22] AECL Technical Specification, TS-44-31110-17 Rev. 0, *Pickering Nuclear Generating Station 'A' LSFCR Program, Cold Worked Zirconium – 2.5 Wt% Niobium Extruded and Drawn Pressure Tubes with Thick End*, February 1984.

[B.18-23] AECL Technical Specification, TS-XX-31110-4 Rev. 6, *Cleaning and Stress Relief of Cold Worked Zirconium-2.5 wt% Niobium Pressure Tubes*, no date.

[B.18-24] AECL Technical Specification, TS-XX-31110-5 Rev. 7, *Cold Worked Zirconium-2.5 wt% Niobium Extruded and Drawn Pressure Tubes*, June 1979.

[B.18-25] AECL Technical Specification, TS-XX-31110-5 Rev. 3, *Cold Worked Zirconium-2.5 wt% Niobium Extruded and Drawn Pressure Tubes*, June 1977.

[B.18-26] AECL Technical Specification, TS-XX-31110-5 Rev. 9, *Cold Worked Zirconium-2.5 wt% Niobium Extruded and Drawn Pressure Tubes*, December 1988.

[B.18-27] OPG Design Manual, NK30-DM-31100-00001 R000, *Fuel Channels,* April 2006.

[B.18-28] OPG History Docket, NK30-HDOC-31110-114685 Volume 7, *Ontario Hydro P.O. 14-31303-24, AECL Requisition 30RN31110-1, Pickering Generating Station B Units 5, 6, 7, 8*, January 1980.

[B.18-29] AECL Technical Specification, TS-XX-31110-4 Rev. 5, *Cleaning and Stress Relief of Cold Worked Zirconium-2.5 Wt% Niobium Pressure Tubes*, September 1986.

[B.18-30] OPG Report, NA44-REP-31100-00020 R001, *Fuel Channel Pressure Tube Integrated Material Surveillance Program for Pickering A – Technical Basis*, February 2016.

[B.18-31] AECL Specification, NP-M-653 R1 (File 44-31710), *Seamless and Seam Welded Zircaloy-2 Tube for the Reactivity Control Rods of Pickering Generating Station,* November 1966.

[B.18-32] AECL Technical Specification, TS-XX-31700-6 Rev. 0, *Seamless and Seam Welded Zircaloy-2 Tubes for Control Rods*, March 1974.

[B.18-33] AECL Technical Specification, TS-XX-31761-5 Rev. 0, *Annealed Zircaloy-2 Seamless Tubing for Poison Injection System Nozzle Assembly*, September 1976.

[B.18-34] AECL Technical Specification, TS-XX-31761-5 Rev. 1, *Annealed Zircaloy-2 Seamless Tubing for Liquid Injection Shutdown System Nozzle Assembly*, November 1977.

[B.18-35] OPG History Docket, NK30-HDOC-31761-111051 R0, *Ontario Hydro Pickering Generating Station "B" Unit #5 Reactivity Control Units Liquid Injection S/D/ Unit Injection Nozzle,* December 12, 1977.

[B.18-36] OPG History Docket, NK30-HDOC-31761-111050 R0, *Ontario Hydro Pickering Generating Station "B" Unit #7 Reactivity Control Units Liquid Injection S/D/ Unit Injection Nozzle,* March 30, 1979.

[B.18-37] AECL Specification, NP-M-601 R1 (File 44-31230), *Seam Welded Zircaloy - 2 Calandria Tubes for the Pickering Generating Station,* July 1966.

[B.18-38] AECL Technical Specification, TS-XX-31230-2 Rev. 4, *Seam Welded Zircaloy-2 Calandria Tubes*, March 1974.

[B.18-39] AECL Technical Specification, TS-38-31231-1 Rev. 1, *Calandria and End Shields, Calandria Tube Assemblies, Calandria Tube - Manufacture*, September 1979.

[B.18-40] AECL Technical Specification, TS-38-31700-14 Rev. 0, *Darlington Nuclear Generating Station 'A', Seam Welded Zircaloy-2 Tubes for Control Rods*, December 1981.

[B.18-41] AECL Technical Specification, TS-XX-31120-4 Rev. 2, *Stainless Steel Blanks for Fuel Channel End Fitting Bodies*, June 1980.

[B.18-42] AECL Technical Specification, TS-XX-31120-2 Rev. 2, *Fuel Channel End Fitting Assemblies*, December 1981.

[B.18-43] AECL Technical Specification, TS-XX-31120-4 Rev. 0, *Fuel Channel End Fitting Stainless Steel Blanks*, April 1975.

[B.18-44] AECL Technical Specification, TS-XX-31120-3 Rev. 1, *Fuel Channel End Fittings*, April 1978.

[B.18-45] AECL Technical Specification, TS-XX-31160-3 Rev. 2, *Fuel Channels – Garter Spring Type Annulus Spacers Inconel X-750*, January 1985.

[B.18-46] AECL Technical Specification, TS-XX-31160-1 Rev. 5, *Fuel Channel Annulus Garter Spring Wire*, September 1982.

B.19 ASME B31.1 (2014), "Power Piping"

B.19.1 Background

The following excerpts paraphrased from the Scope of ASME B31.1-14 [B.19-1] provide a brief overview of the purpose of the standard and the requirements expressed therein:

> *ASME B31.1 has been developed considering the needs for applications that include piping typically found in electric power generating stations, in industrial and institutional plants, geothermal heating systems, and central and district heating and cooling systems.*
>
> *ASME B31.1 prescribes requirements for the design, material, fabrication, erection, test, inspection, operation, and maintenance of power piping systems. Piping as used in this Code includes pipe, flanges, bolting, gaskets, valves, pressure-relieving valves/ devices, fittings, and the pressure containing portions of other piping components, whether manufactured in accordance with Standards listed in Table 126.1 or specially designed. It also includes hangers, supports and other equipment necessary to prevent overstressing the pressure containing components.*

ASME B31.1 is relevant to Safety Factor 1 (Plant Design).

ASME B31.1 is identified in Appendix E.1 of the R04 Pickering Licence Conditions Handbook [B.19-2] as "Guidance or Criteria".

ASME B31.1-14 [B.19-1] supersedes numerous previous editions of B31.1, most recently issued in 2012, 2010 and 2007. The origins of ASME B31.1 date back to the 1920s, with the first official edition published in 1935. The American Society of Mechanical Engineers [B.19-3] identifies the key changes to ASME B31.1-14 which are discussed under Section B.19.2.2 below. Also discussed in Section B.19.2.2, OPG has established the CNSC accepted process of using the Reedy Engineering ASME Code Reconciliation Report to ensure ongoing compliance with ASME B31.1, and, in parallel, there is a yearly examination of any incremental (Code-over-Code) review requirements which ensure that significant technical changes to B31.1 will be applied, as appropriate, for modifications to existing Pickering NGS power piping going forward.

The results of PSR1 ASME B31.1 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.19.2. As identified in Reference [B.19-4], the Pickering PSR2 review of ASME B31.1 (2014) is an Incremental Review. PSR2 Incremental Review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.
- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.19.2 Compliance Assessment for Pickering PSR2

B.19.2.1 Application of PSR1 Reviews

The versions of ASME B31.1 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

No review was performed against ASME B31.1 as part of the Pickering B ISR. OPG Report NK30-REP-03680-00001 R000, "Pickering NGS-B Integrated Safety Review - Plant Design Safety Factor" [B.19-5] did not review ASME B31.1. Nevertheless, OPG Report N-REP-01903.1-10000 R017, "ASME Code Reconciliation for Material, Parts and Components" [B.19-6] identifies all the significant changes in ASME B31.1 up to and including the 2014 edition. N-REP-01903.1-10000 R017, known as the Reedy Report, is used as a reference at OPG when designing pressure boundary repairs, replacements and modifications. As will be discussed under Section B.19.2.2 below, OPG has established the process of using the Reedy Report to ensure compliance with ASME pressure boundary codes and this process has been accepted by the CNSC. As discussed in Section B.19.2.2, the Engineering Change Control (ECC) process ensures that significant technical changes to B31.1 will be applied, as appropriate, for modifications going forward in accordance with OPG governance. Application of B31.1-14 to the as-built design is also discussed under Section B.19.2.2. The conclusion is that there are no PSR2 gaps for Pickering Units 5-8 compliance with ASME B31.1-14.

Pickering Units 1,4

OPG Report NA44-REP-00584.1-10001 R000, "Pickering A ASME Codes Reconciliation - Pickering A Return to Service" [B.19-7] was prepared to:

> *Provide a reconciliation of the ASME Code requirements between the ASME Section VIII and B31.1 1992 Editions, and Section III 1995 Edition to the latest Code 1998 Editions including the 1999 Addenda.*

The ASME Codes Reconciliation found no major impacts from the review against the 1992 edition of ASME B31.1 (including 1999 Addenda). OPG Letter NA44-CORR-00531-00190 R000, "ASME Codes Reconciliation - System and Component Modifications (Pickering NGS-A, Units 1-4)" [B.19-8] states:

> *For modifications, including associated equipment and material, CNSC indicated their acceptance of a case-by-case, code specific, reconciliation. In response, OPG has developed a master ASME codes reconciliation report, which addresses all code changes that impact Pickering A systems' design, equipment, and material. This codes reconciliation report (i.e., OPG Report No. NA44-REP-00584.1-10001, Rev. 00, 30 August 2000), entitled "Pickering A ASME Codes Reconciliation," is enclosed for your information and use….*

The code reconciliation report was based primarily on a technical report, prepared by Reedy Engineering, entitled "ANSI/ASME Code Reconciliation for Replacement Material, Parts, and Components," Rev. 9, 7 January 2000. In addition, various ASME code editions were reviewed and compared; including some that were not addressed in the Reedy report. Significant code changes were reviewed to determine if they would have a major impact on the design basis, such that the code change could require a design change.

Potential major impacts were then reviewed against the current design practices, associated with the Pickering A project design modification packages. The conclusion was that all identified potential major impacts would have a minimal affect on the current design. As a result, modifications to systems and components, which were designed to ASME Boiler and Pressure Vessel Code Section III 1995 edition, Section VIII 1992 edition, and B31.1 1992 edition, are considered equivalent to those designed to the latest code editions.

Based on the above, Pickering Units 1,4 were demonstrated to meet the intent of the 1992 edition of B31.1 as part of Pickering A Return to Service. The conclusions drawn in NA44-CORR-00531-00190 [B.19-8] reflected the then current design of Pickering Units 1,4 against the 1992 edition of B31.1, and those conclusions are not impacted by Pickering NGS operation past 2020. Pickering NGS compliance against the latest version, B31.1-14, is addressed below.

As previously discussed, OPG Report N-REP-01903.1-10000 R017 [B.19-6] identifies all the significant changes in ASME B31.1 up to and including the 2014 edition. OPG has established the CNSC accepted process of using N-REP-01903.1-10000 R017 to demonstrate compliance with ASME pressure boundary codes. As discussed under Section B.19.2.2 below, the ECC process ensures that significant technical changes to B31.1 will be applied, as appropriate, for modifications going forward in accordance with OPG governance. Application of B31.1-14 to the as-built design is also discussed under Section B.19.2.2. The conclusion is that there are no PSR2 gaps for Pickering Units 1,4 compliance with ASME B31.1-14.

Darlington NGS

OPG Reports NK38-REP-03680-10026 R000, "Review of ASME B31.1 (December 2007), Power Piping for Darlington Integrated Safety Review" [B.19-9] and NK38-REP-03680-10134 R000, "Code Refresh Review of ASME B31.1-2012, Power Piping, for DNGS ISR" [B.19-10] document clause-by-clause reviews against the 2012 and 2007 editions of ASME B31.1. NK38-REP-03680-10026 R000 [B.19-9] stated that: "The B31.1 code structure and requirements did not undergo substantial alterations from the time of the construction license award to the Darlington station [1981], however, some of the technical requirements listed therein were subject to refinements and changes."

Given the ASME Code Reconciliation process discussed earlier (which is discussed further in Section B.19.2.2 below), the above Darlington ISR reviews have not been assessed further for applicability to Pickering NGS.

B.19.2.2 Application of Post PSR1 Reviews

N-CORR-00590-0454242 R001, "Code-over-Code Review of B31.1 Power Piping - 2012 over 2010" [B.19-11] reviewed the 2012 version of B31.1 and concluded that there was "no impact on existing installations".

According to ASME [B.19-3], the key changes in the 2014 edition of B31.1 include:

1. *Now mandatory rules for nonmetallic piping and piping lined with nonmetals;*
2. *Cold-forming rules for creep strength enhanced ferritic steels have been incorporated into Chapter V Fabrication, Assembly, and Erection;*
3. *Chapter VII Operation and Maintenance revised to include requirements for:*
    a. *Documentation of dynamic events*
    b. *Piping and pipe-support maintenance programs and personnel requirements;*
4. *Preheat and post-weld heat treatment rules have been reformatted into tabular form for clarity.*

With respect to the changes to B31.1-14, OPG Report, N-REP-00590-00006 R000, "Code-over-Code Review Report: ASME B31.1 Paragraph 127 to 132.7 for the Year 2014" [B.19-12] performed a review of the welding aspects of the 2014 edition of B31.1 against the 2012 edition (paragraphs 127 to 132.7 in both versions), and stated:

> *OPG's existing WPS'* [Welding Procedure Specifications] *were found to be compliant with the new requirements of the 2014 edition of ASME B31.1. No requalification is required as a result of the findings of this Code-Over-Code review.*

OPG Report, N-REP-04800-10001 R006, "ASME Code Summary Report for Reconciliation of Welding Procedure Specifications" [B.19-13] records the changes in ASME Code requirements for B31.1-14 as they pertain to welding and brazing and which may affect compliance with welding and brazing procedure specifications.

OPG Report, N-REP-00590-00001 R000, "Code-over-Code Review Report: ASME Power Piping B31.1 for the Year 2014" [B.19-14] performed a review of B31.1-14 against the 2012 version (except for the welding aspects covered under OPG Report N-REP-00590-00006 R000 [B.19-12] discussed above) and identified the four significant changes below:

- Item 4 paragraph 105.3 - Non-Metallic Pipe: Rules for non-metallic pipe were in non-mandatory Appendix III in the 2012 version of B31.1. There are now mandatory rules to cover non-metallic pipe in B31.1-14.
- Item 11 paragraph 122.8.2(I) - Toxic Fluids (Gas or Liquid): New requirements were added for the vent lines of Toxic fluids (gases or liquids).

- Item 13 paragraph 124.1.2 - Lower Temperature Limits: There are new requirements for lower temperature limits in Chapter III, "Materials". The 2012 edition of B31.1 was less prescriptive, stating only "The designer shall give consideration to the possibility of brittle fracture at low service temperature."

- Item 22 Appendix D, Table D-1 – General note added on the validity of stress intensification and flexibility factor data: B31.1-14 is more restrictive than the 2012 edition since "the limit of $D_o/t_n \leq 100$ applies to all configurations, not just concentric reducers".

The mitigation instituted for each finding above was to include the changes in N-LIST-00590-00001 R002, "List of Significant Technical Changes from Code-over-Code Review" [B.19-15]. All modifications require review of this document as identified in N-FORM-10959 R016, "Design Scoping Checklist" [B.19-16], as per N-PROC-MP-0090 R014, "Modification Process" [B.19-17] (N-FORM-10959 R016 Section 2.19 requires that a review of N-LIST-00590-00001 R002 "shall be completed to determine if any code change improvement actions apply to the modification"). As a result, significant technical changes to B31.1 will be applied, as appropriate, for modifications to existing Pickering NGS power piping going forward in accordance with OPG governance. The ECC process, together with a yearly examination of any incremental (Code-over-Code) review requirements due to changes to B31.1 per OPG Guideline, N-GUID-00590-00002 R001, "Code over Code Review - Guideline" [B.19-18], ensure that that any applicable design changes made to Pickering NGS comply with the latest edition of B31.1.

As previously discussed, OPG Report N-REP-01903.1-10000 R017, "ASME Code Reconciliation for Material, Parts and Components" [B.19-6] identifies all the significant changes in ASME B31.1 up to and including the 2014 edition. N-REP-01903.1-10000 R017 is used as a reference at OPG when designing pressure boundary repairs, replacements and modifications. With respect to the retroactive application of B31.1 to the as-built design at Pickering NGS, N-REP-01903.1-10000 R017 states:

> *This report verifies that ASME Section III, Division 1, Section VIII, Division 1, B31.1, and B31.7 have not made any design changes that depend on corresponding changes to material, fabrication, examination, testing or quality assurance requirements that change the original design basis of any equipment. The significant design changes since 1955 are identified, and methodology and detailed examples are given for selecting provisions from later Editions and Addenda to these Codes.*

OPG Letter NK30-CORR-00531-06324 R000, "Pickering NGS-B - CNSC staff assessment of OPG's 2011 Continued Operations Plan (Action Item 2010-8-05 (2461)) and path forward" [B.19-19] discusses OPG Report N-REP-01903.1-10000 R017 in relation to Pickering B ISR gaps identified for other ASME codes (Boiler Pressure Vessel Code), and states: "Pickering B will continue to follow the practice of designing pressure boundary repairs/replacements to code of construction and referencing the Reedy Report to ensure compliance to current standards. Inspection and testing practices continue to comply with N285.4-05 and 285.5-M90, per our PROL [Power Reactor Operating Licence]." The CNSC considered this response to be satisfactory [B.19-19].

With respect to OPG staff use of N-REP-01903.1-10000 R017 [B.19-6], OPG Memorandum N-CORR-01903.1-0381196, "Information on Reedy ASME Code Reconciliation Report and on Rapid

Access (RA) Database" [B.19-20] provides "basic information on what is contained in the ASME Code Reconciliation Report and in the Rapid Access (RA) database prepared by Reedy Engineering and how to access it. This information will help to support individuals who are required to use the earlier editions of ASME codes or sections thereof as a part of their job function and are especially useful for staff involved in performing reconciliations of different code editions." Further, OPG Guideline N-GUID-01913.11-10000 R002, "ASME Code Effective Date Reconciliation Guidelines" [B.19-21] provides information assisting OPG Design Engineers in: a) applying the code reconciliation process in accordance with OPG Instruction N-INS-01913.11-10020 R003, "Preparation of ASME Code Reconciliation for Pressure Boundary Items" [B.19-22]; b) determining the Required Code Effective Date of the design, and defining the item Code Effective Date; c) performing formal reconciliation on N-FORM-10911 R002, "Item Code Effective Date Reconciliation" [B.19-23] associated with the instruction N-INS-01913.11-10020 R003; and d) recognition of pre-reconciled items, which may be accepted without performing case-by-case reconciliation.

Based on the above, Pickering NGS meets the intent of ASME B31.1-14 since: a) OPG has established the CNSC accepted ASME Code Reconciliation process of using the Reedy Report to ensure ongoing compliance with B31.1, and, in parallel, b) there is a yearly examination of any incremental (Code-over-Code) review requirements which ensure that significant technical changes to B31.1 will be applied, as appropriate, for modifications to existing Pickering NGS power piping going forward.

B.19.3 Compliance Summary for Pickering PSR2

There are no PSR2 gaps for ASME B31.1-14 [B.19-1]. Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with ASME B31.1-14.

B.19.4 References

[B.19-1] ASME B31.1-2014, *Power Piping*, August 2014.

[B.19-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.19-3] *ASME - STANDARDS - Power Piping B31.1 – 2014*, Retrieved November 11, 2016, https://www.asme.org/products/codes-standards/b311-2014-power-piping.

[B.19-4] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.19-5] OPG Report, NK30-REP-03680-00001 R000, *Pickering NGS-B Integrated Safety Review – Plant Design Safety Factor*, August 2007.

[B.19-6] OPG Report, N-REP-01903.1-10000 R017, *ASME Code Reconciliation for Material, Parts and Components*, November 18, 2015.

[B.19-7] OPG Report, NA44-REP-00584.1-10001 R000, *Pickering A ASME Codes Reconciliation – Pickering A Return to Service*, August 2000.

[B.19-8] OPG Letter, NA44-CORR-00531-00190 R000, *ASME Codes Reconciliation - System and Component Modifications (Pickering NGS-A, Units 1-4),* October 6, 2000.

[B.19-9] OPG Report, NK38-REP-03680-10026 R000, *Review of ASME B31.1 (December 2007), Power Piping for Darlington Integrated Safety Review*, August 2011.

[B.19-10] OPG Report, NK38-REP-03680-10134 R000, *Code Refresh Review of ASME B31.1-2012, Power Piping, for DNGS ISR*, July 2013.

[B.19-11] OPG Memorandum, N-CORR-00590-0454242 R001, *Code-over-Code Review of B31.1 Power Piping – 2012 over 2010*, March 2013.

[B.19-12] OPG Report, N-REP-00590-00006 R000, *Code-over-Code Review Report: ASME B31.1 Para. 127 to 132.7 for the Year 2014*, November 2014.

[B.19-13] OPG Report, N-REP-04800-10001 R006, *ASME Code Summary Report for Reconciliation of Welding Procedure Specifications*, May 26, 2016.

[B.19-14] OPG Report, N-REP-00590-00001 R000, *Code-over-Code Review Report: ASME Power Piping B31.1 for the Year 2014*, November 2014.

[B.19-15] OPG List, N-LIST-00590-00001 R002, *List of Significant Technical Changes from Code-Over-Code Review*, August 2015.

[B.19-16] OPG Form, N-FORM-10959 R016, *Design Scoping Checklist*, June 2016.

[B.19-17] OPG Procedure, N-PROC-MP-0090 R014, *Modification Process*, October 14, 2016.

[B.19-18] OPG Guideline, N-GUID-00590-00002 R001, *Code over Code Review - Guideline*, August 15, 2016.

[B.19-19] OPG Letter, NK30-CORR-00531-06324 R000, *Pickering NGS-B - CNSC staff assessment of OPG's 2011 Continued Operations Plan (Action Item 2010-8-05 (2461)) and path forward*, June 19, 2012.

[B.19-20] OPG Memorandum, N-CORR-01903.1-0381196, *Information on Reedy ASME Code Reconciliation Report and on Rapid Access (RA) Database*, March 31, 2011.

[B.19-21] OPG Guideline, N-GUID-01913.11-10000 R002, *ASME Code Effective Date Reconciliation Guidelines*, January 19, 2016.

[B.19-22] OPG Instruction, N-INS-01913.11-10020 R003, *Preparation of ASME Code Reconciliation for Pressure Boundary Items*, April 14, 2014.

[B.19-23] OPG Form, N-FORM-10911 R002, *Item Code Effective Date Reconciliation*, November 2010.

B.20 ASME BPVC (2015), "Boiler and Pressure Vessel Code"

B.20.1 Background

The following excerpts paraphrased from ASME BPVC 2015 [B.20-1] provide a brief overview of the purpose of this standard and the requirements expressed therein:

> *The purpose of ASME BPVC is to regulate the design and construction of boilers and pressure vessels. The standard defines the requirements for the construction, inspection, and materials of boilers, pressure vessels, and nuclear components. It also provides guidance for the care and operation of boilers.*
>
> *The standard has been broken up into the following 12 sections:*
>
> - *Section I – Rules for Construction of Power Boilers*
> - *Section II – Materials*
> - *Section III – Rules for Construction of Nuclear Facility Components*
> - *Section IV – Rules for Construction of Heating Boilers*
> - *Section V – Non-Destructive Examination [NDE]*
> - *Section VI – Recommended Rules for the Care and Operation of Heating Boilers*
> - *Section VII – Recommended Guidelines for the Care of Power Boilers*
> - *Section VIII – Rules for Construction of Pressure Vessels*
> - *Section IX – Welding, Brazing and Fusing Qualifications*
> - *Section X – Fiber-Reinforced Plastic Pressure Vessels*
> - *Section XI – Rules for In-service Inspection of Nuclear Power Plant Components*
> - *Section XII – Rules for Construction and Continued Service of Transport Tanks*

BPVC Sections III and VIII are addressed by Pickering PSR2. Although power/heating boilers are applicable to Pickering NGS, they are not safety related and therefore Sections I, IV and VII do not apply to PSR2. Section II is a "Service Section" to the other BPVC Sections, providing material specifications for ferrous materials and requirements for chemical and mechanical properties, heat treatment and manufacture; therefore, Section II is only relevant to the manufacturing of components. Sections VI, X and XII do not apply to Pickering NGS. Sections

V, IX and XI were not assessed as part of PSR1 for either Darlington or Pickering NGS. Similar to the PSR1 assessment of ASME BPVC, only Sections III and VIII are addressed by PSR2.[17]

ASME BPVC is relevant to Safety Factor 1 (Plant Design).

ASME BPVC is identified in Appendix E.1 of the R04 Pickering Licence Conditions Handbook [B.20-5] as "Guidance or Criteria". However, ASME BPVC Section III is an indirect licence requirement, since compliance with CSA N285.0 (which calls up BPVC Section III) is a licence requirement as indicated in Appendix C.1 of the Licence Conditions Handbook [B.20-5].

ASME BPVC 2015 [B.20-1] supersedes numerous previous editions of the BPVC, most recently issued in 2013, 2010 and 2007. The origins of ASME BPVC date back to the early 1900's, with the first official edition published in 1914.

As discussed in Section B.20.2 below, OPG has established the CNSC accepted process of using the Reedy Engineering ASME Code Reconciliation Report to ensure ongoing compliance with BPVC Sections III and VIII, and, in parallel, there is a yearly examination of any incremental (Code-over-Code) review requirements which ensures that significant technical changes to BPVC will be applied, as appropriate, for modifications to existing Pickering NGS pressure vessels going forward.

The results of PSR1 ASME BPVC reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.20.2. As identified in Reference [B.20-6], the Pickering PSR2 review of ASME BPVC (2015) is an Incremental Review. PSR2 Incremental Review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.

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[17] For BPVC Section V, NDE required for the periodic inspection of nuclear components is performed in accordance with N-STD-MA-0021 R001, "Non-Destructive Examination" [B.20-2] to ensure that examinations performed meet code and regulatory requirements (periodic inspection is performed as a licensing requirement in compliance with CSA N285.4, N285.5 and N287.7). Additional ASME Section V specific examinations are applied by the pressure boundary program (e.g., I-IP-04163-XXXXX series of inspection and maintenance procedures). Welding, brazing and fusing (BPVC Section IX) is addressed via N-PROG-MA-0013 R009, "Welding" [B.20-3] which establishes controlled processes and standardized welding practices to make sound welds that meet safety, structural integrity, code, and licensing requirements, as stipulated in CSA N285.0. Section XI is addressed by periodic inspection of pressure-retaining components and their supports which is performed in accordance with CSA N285.4, per OPG Program N-PROG-MA-0017 R008, "Component and Equipment Surveillance" [B.20-4]. In addition, a yearly examination of any incremental (Code-over-Code) requirements due to changes to ASME BPVC Sections V and IX is undertaken by OPG, per OPG Guideline N-GUID-00590-00002 R001, "Code-over-Code Review – Guideline" [B.20-32].

- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.20.2 Compliance Assessment for Pickering PSR2

B.20.2.1 Application of PSR1 Reviews

The versions of the ASME BPVC (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

OPG Report NK30-REP-03680-00001 R000, "Pickering NGS-B Integrated Safety Review - Plant Design Safety Factor" [B.20-7] performed a clause-by-clause review of the 2001 edition of ASME BPVC Section III, up to and including the 2003 Addenda, and stated:

> *The majority of the gaps identified in the course of the review of Plant Design Safety Factor are associated with Section III of the ASME B&PV Code. Since the design and commissioning of the Pickering NGS-B, the ASME B&PVC Section III has evolved with a number of addenda and revisions which have resulted in some design criteria becoming more restrictive. A preliminary review has revealed that none of the gaps identified during the review poses a risk to operating plant and it is expected that these gaps will be successfully resolved following detailed review of the ASME B&PVC requirements. Accordingly, the Pickering B design is deemed to be in an Acceptable Deviation compliance with the ASME B&PVC Section III.*

OPG Letter NK30-CORR-00531-04739 R000, "Pickering NGS-B Integrated Safety Review - Discrepancy Resolution" [B.20-8] proposed the following disposition to the CNSC regarding Issue 1-359 (which amalgamated 134 Discrepancies):

> *ASME Section III - Significant Changes to Design Requirements, NX-2000 Revisions That May Affect Use of Earlier Materials: Perform case-by-case evaluations of ASME Boiler & Pressure Vessel Code Section III code changes, identify and disposition gaps in accordance with accepted OPG gap disposition process.*

The CNSC accepted the categorization of Issue 1-359 as a Discrepancy in CNSC Letter NK30-CORR-00531-04876 R000, "Pickering NGS-B - Integrated Safety Review (ISR) - CNSC Review of Acceptable Deviations and Discrepancies for the Plant Design Safety Factor Report" [B.20-9]. CNSC Letter NK30-CORR-00531-05008 R000, "Pickering NGS-B Discrepancy Resolution for Ageing, Safety Analysis, Emergency Planning, Environment, Plant Design and Management Safety Areas" [B.20-10] and OPG Report NK30-REP-03680-00016 R000, "OPG Response to CNSC Comments on Pickering NGS-B Integrated Safety Review - Plant Design, Safety Analysis, Safety Performance, Ageing and Equipment Qualification Safety Factors and Discrepancy

Resolutions" [B.20-11] acknowledged CNSC agreement with the planned Issue 1-359 disposition process.

Issue 1-359 was later closed in OPG Letter NK30-CORR-00531-06324 R000, "Pickering NGS-B - CNSC staff assessment of OPG's 2011 Continued Operations Plan (Action Item 2010-8-05 (2461)) and path forward" [B.20-12] on the basis that "OPG has established the process of using the Reedy Report to ensure compliance with ASME B&PVC Section III". OPG Report N-REP-01903.1-10000 R017, "ASME Code Reconciliation for Material, Parts and Components" [B.20-13], known as the Reedy Report, identifies all the significant changes in BPVC Sections III and VIII up to and including the 2015 edition. N-REP-01903.1-10000 R017 is used as a reference by OPG when designing pressure boundary repairs, replacements and modifications. NK30-CORR-00531-06324 R000 addressed N-REP-01903.1-10000 in relation to the Pickering B ISR BPVC Section III gaps, and stated:

> *The OPG practice for designing pressure boundary repairs/replacements, is to design to code of construction. All modifications are designed in compliance to ASME B&PVC Section III 2007, per our licence and per N-PROC-MP-0082-R06 "Design Registration"…*
>
> *Pickering B will continue to follow the practice of designing pressure boundary repairs/replacements to code of construction and referencing the Reedy Report to ensure compliance to current standards. Inspection and testing practices continue to comply with N285.4-05 and 285.5-M90, per our PROL [Power Reactor Operating Licence].*

CNSC staff agreed with the resolution, and stated that they considered the response to be satisfactory [B.20-12]. This is also applicable to BPVC Section VIII, since N-REP-01903.1-10000 R017 [B.20-13] addresses compliance with both BPVC 2015 Sections III and VIII. As discussed under Section B.20.2.2 below, the Engineering Change Control (ECC) process ensures that significant technical changes to ASME BPVC 2015 will be applied, as appropriate, for modifications going forward in accordance with OPG governance. Pickering NGS compliance against ASME BPVC 2015 edition, including application to the as-built design, is discussed further under Section B.20.2.2. The conclusion is that there are no PSR2 gaps for Pickering Units 5-8 compliance with Sections III and VIII of ASME BPVC 2015.

<u>Pickering Units 1,4</u>

OPG Report NA44-REP-00584.1-10001 R000, "Pickering A ASME Codes Reconciliation - Pickering A Return to Service" [B.20-14] was prepared to:

> *Summarize the study for an evaluation of the ASME Code Changes for the Pickering A Return to Service Project from the ASME VIII 1992 and ASME III 1995 Edition to the 1999 Addenda … and their impact on the design basis as per the CSA N285.0-95 requirements.*

The ASME Codes Reconciliation found no major impacts from the review against ASME Section III 1995 edition and ASME Section VIII 1992 edition (including the 1999 Addenda). OPG Letter NA44-CORR-00531-00190 R000, "ASME Codes Reconciliation - System and Component Modifications (Pickering NGS-A, Units 1-4)" [B.20-15] states:

For modifications, including associated equipment and material, CNSC indicated their acceptance of a case-by-case, code specific, reconciliation. In response, OPG has developed a master ASME codes reconciliation report, which addresses all code changes that impact Pickering A systems' design, equipment, and material. This codes reconciliation report (i.e., OPG Report No. NA44-REP-00584.1-10001, Rev. 00, 30 August 2000), entitled "Pickering A ASME Codes Reconciliation," is enclosed for your information and use….

The code reconciliation report was based primarily on a technical report, prepared by Reedy Engineering, entitled "ANSI/ASME Code Reconciliation for Replacement Material, Parts, and Components," Rev. 9, 7 January 2000. In addition, various ASME code editions were reviewed and compared; including some that were not addressed in the Reedy report. Significant code changes were reviewed to determine if they would have a major impact on the design basis, such that the code change could require a design change.

Potential major impacts were then reviewed against the current design practices, associated with the Pickering A project design modification packages. The conclusion was that all identified potential major impacts would have a minimal effect on the current design. As a result, modifications to systems and components, which were designed to ASME Boiler and Pressure Vessel Code Section III 1995 edition, Section VIII 1992 edition, and B31.1 1992 edition, are considered equivalent to those designed to the latest code editions.

Based on the above, Pickering Units 1,4 compliance with ASME Section III 1995 edition and ASME VIII 1992 edition was demonstrated as part of Pickering A Return to Service. These conclusions are not impacted by Pickering NGS operation past 2020.

As discussed previously, OPG Report N-REP-01903.1-10000 R017 [B.20-13] identifies all the significant changes in BPVC Sections III and VIII up to and including the 2015 edition. OPG has established the process of using N-REP-01903.1-10000 R017 to demonstrate compliance with ASME BPVC, and this process has been accepted by the CNSC. As discussed under Section B.20.2.2 below, the ECC process ensures that significant technical changes to ASME BPVC 2015 will be applied, as appropriate, for modifications going forward in accordance with OPG governance. Pickering NGS compliance against ASME BPVC 2015 edition, including application to the as-built design, is discussed further under Section B.20.2.2. The conclusion is that there are no PSR2 gaps for Pickering Units 1,4 compliance with Sections III and VIII of ASME BPVC 2015.

Darlington NGS

OPG Reports NK38-REP-03680-10028 R000, "Review of ASME BPVC Section VIII Design and Fabrication of Pressure Vessels for Darlington ISR" [B.20-16] and NK38-REP-03680-10133 R000, "Code Refresh Review of ASME Boiler and Pressure Vessel Code, Section VIII (July 2007 to July 2013) Rules for Construction of Pressure Vessels" [B.20-17] addressed Darlington NGS compliance with the 2007 and 2013 editions of BPVC Section VIII, respectively. OPG Reports NK38-REP-03680-10027 R000, "Review of ASME BPVC Section III Rules for Construction of NPP Components for Darlington ISR" [B.20-18] and NK38-REP-03680-10132 R000, "Code Refresh Review of ASME Boiler and Pressure Vessel Code, Section III (July 2007 to July 2013) Rules for

Construction of Nuclear Facility Components" [B.20-19] addressed compliance for the 2007 and 2013 editions of BPVC Section III. OPG Reports NK38-REP-03680-10027 ADD-001 R000, "Addendum to the ASME Boiler and Pressure Vessel Section III Code Review Report for Darlington ISR" [B.20-20] and NK38-REP-03680-10028 ADD-001 R000, "Addendum to the ASME Boiler Sec VIII Code Review Report for Darlington ISR" [B.20-21] document CNSC staff comments and OPG responses related to the ASME BPVC Section VIII and III findings.

Given the ASME Code Reconciliation process discussed previously (which is discussed further in Section B.20.2.2 below), the above Darlington ISR reviews have not been assessed further for applicability to Pickering NGS.

B.20.2.2 Application of Post PSR1 Reviews

With respect to the changes to the 2015 edition of BPVC Sections III and VIII, the following Code-over-Code reviews were recently performed by OPG:

- OPG Report, N-REP-00590-0557392 R001, "Code-over-Code Review Report: ASME Section III NCA for the Year 2015" [B.20-22];
- OPG Report, N-REP-00590-00009 R001, "Code-over-Code Review Report: ASME III/1 Subsection NB- 2013 Edition Over 2015 Edition for the Year 2015" [B.20-23];
- OPG Report, N-REP-00590-00010 R001, "Code-over-Code Review Report: ASME III/1 Subsection NC- 2013 Edition Over 2015 Edition for the Year 2015" [B.20-24]
- OPG Report, N-REP-00590-00011 R001, "Code-over-Code Review Report: ASME III/1 Subsection ND- 2013 Edition Over 2015 Edition for the Year 2015" [B.20-25];
- OPG Report, N-REP-00590-00012 R000, "Code-over-Code Review Report: ASME III/1 Subsection NE- 2013 Edition Over 2015 Edition for the Year 2015" [B.20-26];
- OPG Report, N-REP-00590-00013 R001, "Code-over-Code Review Report: ASME III/1 Subsection NF- 2013 Edition Over 2015 Edition for the Year 2015" [B.20-27]; and
- OPG Report, N-REP-00590-0564946 R002, "Code-over-Code Review Report: ASME VIII/1-2015 Edition - ASME VIII/1-2013 Edition UG-125 to UG-140 Overpressure Protection for the Year 2015" [B.20-28].

A number of significant changes were identified. The mitigation instituted for each finding is to include the changes in N-LIST-00590-00001 R002, "List of Significant Technical Changes from Code-over-Code Review" [B.20-29]. All modifications require review of this document as identified in N-FORM-10959 R016, "Design Scoping Checklist" [B.20-30], as per N-PROC-MP-0090 R014, "Modification Process" [B.20-31] (N-FORM-10959 R016 Section 2.19 requires that a review of N-LIST-00590-00001 R002 "shall be completed to determine if any code change improvement actions apply to the modification"). As a result, significant technical changes to the 2015 edition of ASME BPVC Sections III and VIII will be applied, as appropriate, for modifications to existing Pickering NGS pressure vessels going forward in accordance with OPG governance. The ECC process, together with a yearly examination of any incremental (Code-over-Code) review requirements due to changes to ASME BPVC per OPG Guideline, N-GUID-

00590-00002 R001, "Code-over-Code Review - Guideline" [B.20-32], ensure that that any applicable design changes made to Pickering NGS comply with the latest edition of the BPVC.

As discussed previously, OPG Report N-REP-01903.1-10000 R017 [B.20-13] identifies all the significant changes in BPVC Sections III and VIII up to and including the 2015 edition. N-REP-01903.1-10000 R017 is used as a reference at OPG when designing pressure boundary repairs, replacements and modifications. With respect to the retroactive application of BPVC Sections III and VIII to the as-built design at Pickering NGS, N-REP-01903.1-10000 R017 states:

> *This report verifies that ASME Section III, Division 1, Section VIII, Division 1, B31.1, and B31.7 have not made any design changes that depend on corresponding changes to material, fabrication, examination, testing or quality assurance requirements that change the original design basis of any equipment. The significant design changes since 1955 are identified, and methodology and detailed examples are given for selecting provisions from later Editions and Addenda to these Codes.*

As discussed previously, OPG Letter NK30-CORR-00531-06324 R000, "Pickering NGS-B - CNSC staff assessment of OPG's 2011 Continued Operations Plan (Action Item 2010-8-05 (2461)) and path forward" [B.20-12] identifies that "OPG has established the process of using the Reedy Report to ensure compliance with ASME B&PVC Section III". CNSC staff agreed with OPG, and stated that they considered this response to be satisfactory [B.20-12]. This is also applicable to ASME BPVC Section VIII, since N-REP-01903.1-10000 R017 [B.20-13] addresses changes to both Sections III and VIII.

With respect to OPG staff use of N-REP-01903.1-10000 R017 [B.20-13], OPG Memorandum N-CORR-01903.1-0381196, "Information on Reedy ASME Code Reconciliation Report and on Rapid Access (RA) Database" [B.20-33] provides "basic information on what is contained in the ASME Code Reconciliation Report and in the Rapid Access (RA) database prepared by Reedy Engineering and how to access it. This information will help to support individuals who are required to use the earlier editions of ASME codes or sections thereof as a part of their job function and are especially useful for staff involved in performing reconciliations of different code editions." Further, OPG Guideline N-GUID-01913.11-10000 R002, "ASME Code Effective Date Reconciliation Guidelines" [B.20-34] provides information assisting OPG Design Engineers in: a) applying the code reconciliation process in accordance with OPG Instruction N-INS-01913.11-10020 R003, "Preparation of ASME Code Reconciliation for Pressure Boundary Items" [B.20-35]; b) determining the Required Code Effective Date of the design, and defining the item Code Effective Date; c) performing formal reconciliation on N-FORM-10911 R002, "Item Code Effective Date Reconciliation" [B.20-36] associated with the instruction N-INS-01913.11-10020; and d) recognition of pre-reconciled items, which may be accepted without performing case-by-case reconciliation.

Based on the above, Pickering NGS meets the intent of ASME BPVC 2015 Sections III and VIII since: a) OPG has established the CNSC accepted ASME Code Reconciliation process of using the Reedy Report to ensure ongoing compliance with BPVC Sections III and VIII, and, in parallel, b) there is a yearly examination of any incremental (Code-over-Code) review requirements which ensures that significant technical changes to BPVC will be applied, as appropriate, for modifications to existing Pickering NGS pressure vessels going forward.

B.20.3 Compliance Summary for Pickering PSR2

There are no PSR2 gaps for ASME BPVC (2015) [B.20-1]. Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with ASME BPVC (2015).

B.20.4 References

[B.20-1] ASME BPVC, *Boilers and Pressure Vessel Code*, 2015.

[B.20-2] OPG Standard, N-STD-MA-0021 R001, *Non-Destructive Examination*, July 30, 2015.

[B.20-3] OPG Program, N-PROG-MA-0013 R009, *Welding*, April 29, 2015.

[B.20-4] OPG Program, N-PROG-MA-0017 R008, *Component and Equipment Surveillance*, June 1, 2015.

[B.20-5] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.20-6] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.20-7] OPG Report, NK30-REP-03680-00001 R000, *Pickering NGS-B Integrated Safety Review – Plant Design Safety Factor*, August 2007.

[B.20-8] OPG Letter, NK30-CORR-00531-04739 R000, D.P. McNeill and P.F. Tremblay to T.E. Schaubel, *Pickering NGS-B Integrated Safety Review – Discrepancy Resolution*, April 17, 2008.

[B.20-9] CNSC Letter, e-Docs # 3256609, OPG File No. NK30-CORR-00531-04876 R000, T.E. Schaubel to D.P. McNeil, *Pickering NGS-B – Integrated Safety Review (ISR) – CNSC Review of Acceptable Deviations and Discrepancies for the Plant Design Safety Factor Report*, June 27, 2008.

[B.20-10] CNSC Letter, e-Docs # 3297754, OPG File No. NK30-CORR-00531-05008 R000, T.E. Schaubel to D.P. McNeil, *Pickering NGS-B Discrepancy Resolution for Ageing, Safety Analysis, Emergency Planning, Environment, Plant Design and Management Safety Areas*, November 03, 2008.

[B.20-11] OPG Report, NK30-REP-03680-00016 R000, *OPG Response to CNSC Comments on Pickering NGS-B Integrated Safety Review – Plant Design, Safety Analysis, Safety Performance, Ageing and Equipment Qualification Safety Factors and Discrepancy Resolutions*, September 2009.

[B.20-12] OPG Letter, NK30-CORR-00531-06324 R000, *Pickering NGS-B - CNSC staff assessment of OPG's 2011 Continued Operations Plan (Action Item 2010-8-05 (2461)) and path forward*, June 19, 2012.

[B.20-13] OPG Report, N-REP-01903.1-10000 R017, *ASME Code Reconciliation for Material, Parts and Components*, November 18, 2015.

[B.20-14] OPG Report, NA44-REP-00584.1-10001 R000, *Pickering A ASME Codes Reconciliation – Pickering A Return to Service*, August 2000.

[B.20-15] OPG Letter, NA44-CORR-00531-00190 R000, *ASME Codes Reconciliation - System and Component Modifications (Pickering NGS-A, Units 1-4),* October 6, 2000.

[B.20-16] OPG Report, NK38-REP-03680-10028 R000, *Review of ASME Boiler And Pressure Vessel Code Section VIII (July 2007), Design and Fabrication of Pressure Vessels for Darlington Integrated Safety Review*, August 2011.

[B.20-17] OPG Report, NK38-REP-03680-10133 R000, *Code Refresh Review of ASME Boiler and Pressure Vessel Code, Section VIII (July 2007 to July 2013) Rules for Construction of Pressure Vessels*, January 2014.

[B.20-18] OPG Report, NK38-REP-03680-10027 R000, *Review of ASME Boiler Pressure Vessel Code, Section III (July 2007), Rules for Construction of Nuclear Power Plant Components for Darlington ISR*, August 2011.

[B.20-19] OPG Report, NK38-REP-03680-10132 R000, *Code Refresh Review of ASME Boiler and Pressure Vessel Code, Section III (July 2007 to July 2013) Rules for Construction of Nuclear Facility Components*, January 2014.

[B.20-20] OPG Report, NK38-REP-03680-10027-ADD-001 R000, *Addendum to the ASME Boiler and Pressure Vessel Section III Code Review Report for Darlington ISR*, January 2014.

[B.20-21] OPG Report, NK38-REP-03680-10028-ADD-001 R000, *Addendum to the ASME Boiler Sec VIII Code Review Report for Darlington ISR*, January 2014.

[B.20-22] OPG Report, N-REP-00590-0557392 R001, *Code-over-Code Review Report: ASME Section III NCA for the Year 2015*, January 2016.

[B.20-23] OPG Report, N-REP-00590-00009 R001, *Code-over-Code Review Report: ASME III/1 Subsection NB- 2013 Edition Over 2015 Edition for the Year 2015*, January 29, 2016.

[B.20-24] OPG Report, N-REP-00590-00010 R001, *Code-over-Code Review Report: ASME III/1 Subsection NC- 2013 Edition Over 2015 Edition for the Year 2015*, January 29, 2016.

[B.20-25] OPG Report, N-REP-00590-00011 R001, *Code-over-Code Review Report: ASME III/1 Subsection ND- 2013 Edition Over 2015 Edition for the Year 2015*, January 29, 2016.

[B.20-26] OPG Report, N-REP-00590-00012 R000, *Code-over-Code Review Report: ASME III/1 Subsection NE- 2013 Edition Over 2015 Edition for the Year 2015*, November 12, 2015.

[B.20-27] OPG Report, N-REP-00590-00013 R001, *Code-over-Code Review Report: ASME III/1 Subsection NF- 2013 Edition Over 2015 Edition for the Year 2015*, January 29, 2016.

[B.20-28] OPG Report, N-REP-00590-0564946 R002, *Code-over-Code Review Report: ASME VIII/1-2015 Edition - ASME VIII/1-2013 Edition UG-125 to UG-140 Overpressure Protection For The Year 2015*, March 23, 2016.

[B.20-29] OPG List, N-LIST-00590-00001 R002, *List of Significant Technical Changes from Code-Over-Code Review*, August 2015.

[B.20-30] OPG Form, N-FORM-10959 R016, *Design Scoping Checklist*, June 2016.

[B.20-31] OPG Procedure, N-PROC-MP-0090 R014, *Modification Process*, October 14, 2016.

[B.20-32] OPG Guideline, N-GUID-00590-00002 R001, *Code-over-Code Review - Guideline*, August 15, 2016.

[B.20-33] OPG Memorandum, N-CORR-01903.1-0381196, *Information on Reedy ASME Code Reconciliation Report and on Rapid Access (RA) Database*, March 31, 2011.

[B.20-34] OPG Guideline, N-GUID-01913.11-10000 R002, *ASME Code Effective Date Reconciliation Guidelines*, January 19, 2016.

[B.20-35] OPG Instruction, N-INS-01913.11-10020 R003, *Preparation of ASME Code Reconciliation for Pressure Boundary Items*, April 14, 2014.

[B.20-36] OPG Form, N-FORM-10911 R002, *Item Code Effective Date Reconciliation*, November 2010.

B.21 CSA B51-14, "Boiler, Pressure Vessel, and Pressure Piping Code"

B.21.1 Background

The following excerpt paraphrased from the Preface of CSA B51-14 (including Update No. 1) [B.21-1] provides a brief overview of the purpose of the standard and the requirements expressed therein:

> *The purpose of CSA B51 is to promote safe design, construction, installation, operation, inspection, testing, and repair practices. The requirements defined within the standard are split into three parts:*
>
> - *Part 1 contains the requirement for boilers, pressure vessels, pressure piping, and fittings.*
> - *Part 2 contains the requirements for high-pressure cylinders for the on-board storage of natural gas, blends of natural gas and hydrogen (hydrogen blends), and hydrogen as fuels for automotive vehicles.*
> - *Part 3 contains requirements for compressed natural gas and hydrogen refuelling station pressure piping systems and ground storage vessels.*

CSA B51 Parts 2 and 3 are not relevant to Pickering NGS and not addressed in PSR2. As discussed in the PSR2 Basis Document [B.21-2], the scope of the Structures, Systems and Components (SSCs) within the PSR2 review encompasses Pickering NGS safety related systems, with a focus on Systems Important to Safety (SIS) and Safe Operating Envelope (SOE) Systems.

CSA B51 is relevant to Safety Factor 1 (Plant Design).

CSA B51 is identified in Appendix E.1 of the R04 Pickering Licence Conditions Handbook [B.21-3] as "Guidance or Criteria". However, B51 is an indirect licence requirement, since compliance with CSA N285.0, "General Requirements for Pressure Retaining Systems and Components in CANDU Nuclear Plants" (which calls up B51) is a licence requirement as indicated in Appendix C.1 of the Licence Conditions Handbook [B.21-3].

CSA B51-14 is the eighteenth edition of this standard, and supersedes numerous previous editions of B51, most recently issued in 2009, 2003 and 1995. The Preface to CSA B51-14 (including Update No. 1) [B.21-1] identifies the significant technical changes to the standard, which are discussed in Section B.21.2.2.

The results of PSR1 CSA B51 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.21.2. As identified in Reference [B.21-2], the Pickering PSR2 review of CSA B51-14 (including Update No. 1) [B.21-1] is an Incremental Review. PSR2 Incremental Review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.

- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.21.2 Compliance Assessment for Pickering PSR2

B.21.2.1 Application of PSR1 Reviews

The versions of CSA B51 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

OPG Report NK30-REP-03680-00001 R000, "Pickering NGS-B Integrated Safety Review - Plant Design Safety Factor" [B.21-4] documents a clause-by-clause review of CSA B51-03 [B.21-5] Part 1 against SOE/SIS systems, and found:

> *The compliance assessment covers Part 1 of the B51 standard which has been specified for pressure retaining components. The review set out to identify gaps in the design of the SOE/SIS systems due to revisions in the B51-1975 standard, which applied during the design and construction of Pickering B, compared to B51-03, which is the most recent revision of the standard. The clause-by-clause review found no gaps in requirements that would require a design change to be made to the existing Pickering B SOE/SIS systems to satisfy B51-03.*
>
> *No additional action is required to establish compliance with the Standard B51-03 of the Pickering 'B' SOE/SIS systems after refurbishment has been completed.*

NK30-REP-03680-00001 R000 [B.21-4] classified the following three CSA B51-03 [B.21-5] clauses as Acceptable Deviations, and the rest of the clauses as compliant or not applicable/relevant:

- Clause 4.2.1, which relates to registry of fittings intended for use on boilers, pressure vessels and piping;

- Clause 7.5.1.3, which relates to requirements on minimum diameter and volume of the blow off vessel for a coil-tube boiler having a capacity of less than 1200 litres; and

- Clause 7.5.2.1, which specifies access opening requirements to facilitate internal inspection and permit cleaning.

At the CNSC's request, Clause 4.2.1 was reclassified as a Discrepancy as documented in OPG Report NK30-REP-03680-00015 R000, "Pickering NGS-B Integrated Safety Review (ISR) - Final ISR Report" [B.21-6]. The remaining two Acceptable Deviations were addressed in NK30-REP-

03680-00015 R000 [B.21-6], with no further action required. The CNSC accepted reclassification of Clause 4.2.1 as an Acceptable Deviation in CNSC Letter NK30-CORR-00531-05312 R000, "Pickering NGS-B Integrated Safety Review (ISR) - CNSC Staff Response to OPG's Comment Disposition Report" [B.21-7], which states:

> *OPG continues to believe that this item should be categorized as Acceptable Deviation and provides the following additional supporting justification. Clause 4.2.1 in B51-03 applies to the manufacturer of fittings, and thus has no impact on current installed fittings in the plant systems. Registration every 10 years is a requirement of the manufacturer to ensure that their QA meets requirements and that they have a valid certificate of authorization for the fabrication of these fittings. This enables OPG to ensure that applicable new fittings coming into the plant comply with the code clause.*

Per OPG Letter NK30-CORR-00531-05578, "Pickering NGS-B - Follow-up to the Closure of the Integrated Safety Review (ISR) Study" [B.21-8], there were no unaddressed findings relating to CSA B51-03 compliance upon closure of the Pickering B ISR project. None of the Acceptable Deviations identified above are impacted by Pickering NGS operation beyond 2020. In addition, the findings and dispositions are generally applicable to Pickering Units 1,4, given the similarities in inspection, testing and registration requirements, as well as design and material specifications, between the units for boilers, pressure vessels, pressure piping, and fittings.

Based on the above, there are no PSR2 gaps associated with previous Pickering B ISR reviews against CSA B51-03 [B.21-5]. Pickering NGS compliance with subsequent editions of CSA B51 is addressed below.

<u>Pickering Units 1,4</u>

With respect to review against CSA B51 as part of Pickering A Return to Service, subsection VII.1.3, "Codes and Standards for Return to Service" of OPG Letter NA44-CORR-00531-00381 R000, "Pickering A - Updated Basis for Return to Service Document" [B.21-9], made the following statement:

> *For modifications, CSA N285.0-95 and CSA B51-95 shall be followed with specific conditions as specified in the AECB's approval letter.*

Part 6, "Evaluation of Code Class Requirements Against CSA Standards N285.0 and N285.2" of AECL Assessment Document 44RS-00531-ASD-001 Rev. 04, "Review of Pickering A Design Against Current Codes and Standards" [B.21-10] noted that:

> *The requirement to comply with the CSA N285.0-95 is stated in the Pickering 'A' Power Reactor Operating License (PROL). The PROL, in part, requires OPG to "design, modify, repair, test, inspect and otherwise perform work on vessels, systems, piping, fittings, components and supports according to the technical requirements of CSA Standard N285.0-95 (and B51-95)."*

Alignment with CSA B51-95 [B.21-11] was further addressed in OPG Letter NA44-CORR-00531-00190 R000, "ASME Codes Reconciliation - System and Component Modifications (Pickering NGS-A, Units 1-4)" [B.21-12], which stated:

For modifications, including associated equipment and material, CNSC indicated their acceptance of a case-by-case, code specific, reconciliation. In response, OPG has developed a master ASME codes reconciliation report, which addresses all code changes that impact Pickering A systems' design, equipment, and material. This codes reconciliation report (i.e., OPG Report No. NA44-REP-00584.1-10001, Rev. 00, 30 August 2000), entitled "Pickering A ASME Codes Reconciliation", is enclosed…

The codes reconciliation report reflects the overall results of our evaluation, and shows that the Pickering A systems and components will, in general, satisfy the requirements of CSA N285.0-95 and B51-95.

Section 5.6 of the Pickering PROL Renewal Application [B.21-13], which was prepared in 2012, confirms alignment at the time with CSA B51-03 [B.21-5]:

The N-MAN-01913.11-10000, Pressure Boundary Program Manual [B.21-14]*, describes the program used to control the quality of pressure boundary activities at OPG Nuclear facilities and stations. It complies with CSA N285.0, General Requirements for Pressure Retaining Systems and Components in CANDU Nuclear Power Plants, and CSA B51, Boiler, Pressure Vessel, and Pressure Piping Code. Pressure boundary requirements for all states of work, from design through to installation and testing, are implemented through OPG Nuclear governing documents.*

Pickering has implemented CSA N285.0-2008 Update 1 and B51-03 for pressure boundary activities at site in 2010. The OPG QA Manual, along with associated Standards and Procedures, were revised to comply with 2008 Edition of N285.0 with Update 1 and 2003 edition of B51.

After a successful survey by the Technical Standards & Safety Authority (TSSA) demonstrating pressure boundary processes to be in compliance with the Pressure Boundary Program Manual, Certificates of Authorization (C of A) for Pressure Boundary activities were renewed for another three years…

Section 2.1 of OPG Manual N-MAN-01913.11-10000 R016, "Pressure Boundary Program Manual" [B.21-14], which was most recently updated in 2015, states:

This Pressure Boundary (PB) Program Manual (hereinafter referred to as the Manual) complies with the applicable rules and quality requirements contained in the Canadian Standards Association (CSA) N285.0 and CSA B51 standards for Class 1, 1C, 2, 2C, 3, 3C, 4, and 6 systems and items, as applicable. The Code Effective Date for activities initiated after October 30, 2013 has been established under the terms of the PROLs and WFOLs [Waste Facility Operating Licence] as follows: - CSA N285.0-08 with Update 1, Update 2, Annex K, Annex M - CSA B51-09 with Update 1…

Based on the above, OPG Governance, which is applicable across OPG's nuclear fleet, currently requires compliance with CSA B51-09 (including Update No. 1) [B.21-15]. As a result, there are no PSR2 gaps for Pickering Units 1,4 or Units 5-8 compliance with CSA B51-09 (including Update No. 1). Pickering NGS compliance with subsequent editions of CSA B51 is addressed below.

Darlington NGS

OPG Report NK38-REP-03680-10029 R000, "Review of CAN/CSA-B51-03 (R2007) (March 2003), Boiler, Pressure Vessel, and Pressure Piping Code for Darlington Integrated Safety Review" [B.21-16] performed a clause-by-clause review against CSA B51-03 [B.21-5] Part 1, and concluded:

> *The CAN/CSA B51-03 conformity assessment of nine Darlington NGS pressure-retaining systems was found to show full compliance with the Standard.*

OPG Report NK38-REP-03680-10029-ADD-001 R000, "Addendum to the CAN/CSA-B51-03 Code Review Report for Darlington ISR" [B.21-17] documents CNSC staff comments and OPG responses related to OPG Report NK38-REP-03680-10029 R000 [B.21-16]. OPG responses to CNSC comments 024/025/026 were deemed acceptable, with no further action required.

OPG Report NK38-REP-03680-10104-ADD-001 R000, "Darlington NGS ISR - Final ISR Report Addendum" [B.21-18] noted six gaps against CSA B51-09, which were grouped into three ISR Issues (D483/D488/D489) and classified as Acceptable Deviations. These Acceptable Deviations were related to long-standing Darlington NGS concessions against B51. OPG Report NK38-REP-03680-10201 R001, "ISR Open Issues and Acceptable Deviations – Adequacy Review" [B.21-19] discussed Issues D483 and D488, which are both labelled 'CSA B51-09 Hydrostatic Test Concession', and Issue D489 which is labelled 'CSA B51-09 Defective Pipe Concession'. The concessions were classified as Acceptable Deviations and ranked as having very low safety significance. All three of the associated concessions were specific to Darlington and thus are not applicable to Pickering PSR2. (Note: Per the PSR2 Basis Document [B.21-2], PSR2 Safety Factor Reports include a separate review of the Pickering Licence Conditions Handbook [B.21-3] for any impacts of Pickering NGS operation beyond 2020 on: a) OPG commitments previously made to the CNSC, and b) exemptions granted by the CNSC).

OPG Report NK38-REP-03680-10135 R000, "Code Refresh Review of CSA B51-09-UPD1, Boiler, Pressure Vessel, and Pressure Piping Code for Darlington ISR" [B.21-20] prepared a clause-by-clause review of CSA B51-09 (including Update No. 1) [B.21-15] versus CSA B51-03 [B.21-5], and concluded:

> *The changes made in CSA B51-09 UPD 1 relative to CSA B51-03 (R2007) included very minor changes, mostly for clarification purposes. The review of the changed clauses in this code refresh report confirms that OPG Nuclear governance continues to be compliant with the requirements of CSA B51-09 UPD 1.*

The Darlington ISR results therefore confirm that OPG Governance, which is applicable across OPG's nuclear fleet, is compliant with the requirements of CSA B51-09 (including Update No. 1) [B.21-15]. Pickering NGS compliance with the latest edition, CSA B51-14 (including Update No. 1) [B.21-1], is addressed in Section B.21.2.2 below.

B.21.2.2 Application of Post PSR1 Reviews

As discussed earlier, Pickering Units 1, 4 and 5-8 have demonstrated compliance against CSA B51-95 [B.21-11] and B51-03 [B.21-5], respectively, and the Units 5-8 B51-03 conclusions are generally applicable to Units 1,4 given the similarities in inspection, testing and registration requirements, as well as design and material specifications, as discussed in Section B.21.2.1. Similarly, although the three OPG plants (Darlington, Pickering Units 1,4 and Pickering Units 5-8) have differences in age, there are many similarities between the plants in terms of inspection, testing and registration requirements (as well as material specifications and design) of boilers, pressure vessels, pressure piping, and fittings. Therefore, the Darlington ISR review against CSA B51-09 (including Update No. 1) [B.21-15], which identified no safety significant findings, is also generally applicable to Pickering NGS. Furthermore, OPG Governance is compliant with the requirements of CSA B51-09 (including Update No. 1) [B.21-15] as discussed above.

The current edition of the standard is CSA B51-14 (including Update No. 1) [B.21-1], which supersedes CSA B51-09 (including Update No. 1) [B.21-15]. The Preface of CSA B51-14 (including Update No. 1) [B.21-1] provides the following information on the changes to this most recent version of the standard:

> *This Standard has undergone substantial technical and editorial revisions since the previous edition in 2009. Significant changes to Part 1 include the following:*
>
> *(a) The following definitions have been added to Clause 3:*
>
> *(i) air heater coil;*
>
> *(ii) historical steam boiler;*
>
> *(iii) owner;*
>
> *(iv) pressure relief device (PRD); and*
>
> *(v) thermal expansion relief valve.*
>
> *(b) General requirements: Clauses 4.1.1, 4.1.4, 4.2.4, 4.2.8, 4.6.5, 4.7.1, 4.7.2 Note 1, 4.8.2, 4.8.3, 4.9.3 Note, and 4.10 have been updated, and Clauses 4.1.9, 4.1.10, 4.2.5, and 4.2.6 have been added.*
>
> *(c) Boilers and related components: Clauses 6.2.1 and 6.7 have been updated, and Clauses 6.2.2 and 6.5.1 Note 3 have been added.*
>
> *(d) Pressure vessels — applicable codes and standards: Clauses 7.1.2 and 7.1.4 have been added.*
>
> *(e) Piping and fittings: Clause 8.1(c) has been updated.*
>
> *(f) Repairs and alterations: Clause 11.5 has been added.*
>
> *(g) Pressure relief devices: Clause 12 has been added.*
>
> *(h) In-service inspection: Clause 13 has been added.*

(i) *Categories of fittings: Table 1 Note 2 has been updated.*

(j) *Maximum servicing intervals: Table 5 has been added.*

(k) *The sample forms in Annex D have been updated to include cast aluminum.*

(l) *Requirements for a crush test for manifold vessels have been added to Annex G (Clause G.3.7).*

(m) *Provisions applicable to overpressure protection devices have been revised (Annex H).*

(n) *Requirements for historical boilers have been added (Annex I).*

(o) *Requirements regarding the use of finite elements analysis (FEA) to support a pressure equipment design submission have been added (Annex J).*

OPG Report N-REP-00590-0555896 R001, "Code-over-Code Review Report: CSA B51 for the Year 2015" [B.21-21] performed a Code-over-Code review of CSA B51-14 (including Update No. 1) [B.21-1] versus B51-14 [B.21-22], and stated:

There are no significant technical changes to CSA B51-14 Update 1.

OPG Report N-REP-00590-0520103 R000, "Code-over-Code Review Report: CSA B51 for the Year 2014" [B.21-23] performed a Code-over-Code review of CSA B51-14 [B.21-22] versus B51-09 (including Update No. 1) [B.21-15], and stated:

The following clauses have been identified as having significant technical changes and will be included in N-LIST-00590-00001, List of Significant Technical changes from Code-Over-Code Review.

Clause	***Description***
4.1.9	*New Clause. Requirements for Division 2 or 3 vessels regarding registration submissions, contents of design specifications, sale of vessels and change of ownership.* *Remarks & Mitigation Plans to Address Significant Changes - There are no Pressure Vessels constructed to ASME Section VIII Division 2 or 3 in OPG, however, this new requirement will be included in the revision to the N-LIST-00590-00001, List of Significant Technical Changes From Code-Over-Code Review.*
4.1.10	*New Clause. Addresses use of FEA [Finite Element Analysis] used to support design of pressure equipment.* *Remarks & Mitigation Plans to Address Significant Changes - This new requirement will be included in the revision to the N-LIST-059000001, List of Significant Technical Changes From Code-Over-Code Review.*
7.1.2	*New Clause. Mandates use of ASME Section VIII, Division 1, Appendix FF for [design of] pressure vessels incorporating quick-actuating closure.*

Clause	Description
	Remarks & Mitigation Plans to Address Significant Changes - This new requirement will be included in the revision to the N-LIST-0059000001, List of Significant Technical Changes From Code-Over-Code Review.
7.1.4	*New Clause. Addresses pressure vessels designed for cyclic service to ASME Section VIII, division 2 or 3.* *Remarks & Mitigation Plans to Address Significant Changes - There are no Pressure Vessels constructed to ASME Section VIII Division 2 or 3 in OPG, however, this new requirement will be included in the revision to the N-LIST-00590-00001, List of Significant Technical Changes From Code-Over-Code Review.*
12.2.1 (12.2.1.1 and 12.2.1.2)	*New clause. Addresses design of overpressure protection.* *Remarks & Mitigation Plans to Address Significant Changes - Not required. OPG over-pressure protection is in compliance with requirements of the licence and CSA N285.0 which envelopes the requirements of Clause 12.*
12.2.2 (12.2.2.1 through 12.2.2.10)	*New clause. Addresses installation of pressure relief devices.* *Remarks & Mitigation Plans to Address Significant Changes - Not required. OPG over-pressure protection is in compliance with requirements of the licence and CSA N285.0 which envelopes the requirements of Clause 12.*

The mitigation instituted for the CSA B51-14 [B.21-22] clauses identified above was to include significant technical changes in the next revision of N-LIST-00590-00001 R002, "List of Significant Technical Changes from Code-over-Code Review" [B.21-24]. All modifications require review of this document as identified in N-FORM-10959 R016, "Design Scoping Checklist" [B.21-25], as per N-PROC-MP-0090 R014, "Modification Process" [B.21-26] (N-FORM-10959 R016 Section 2.19 requires that a review of N-LIST-00590-00001 R002 "shall be completed to determine if any code change improvement actions apply to the modification"). In addition, there is a yearly examination of any incremental (Code-over-Code) review requirements which ensures that significant technical changes to B51 will be applied going forward, per OPG Guideline N-GUID-00590-00002 R001, "Code over Code Review - Guideline" [B.21-27]. As a result, significant technical changes will be applied, as appropriate, for future modifications to existing Pickering NGS installations in accordance with OPG governance. Further, compliance against the above-mentioned clauses is not required (or not applicable) to Pickering NGS, except for Clauses 4.1.10 and 7.1.2 which relate to design-support activities for pressure equipment/vessels as discussed below:

- Clause 4.1.10 of CSA B51-14 (including Update No. 1) [B.21-1] states: "Finite element analysis (FEA) may be used to support pressure equipment design where the configuration is not covered by the available rules in the code of construction…. Note: The designer should check with the regulatory authority to confirm that use of FEA is acceptable." Per Annex J of CSA B51-14, the intent of this clause is to identify potential compliance demonstration alternatives when pressure equipment design configurations are not covered by the available rules in the ASME code. In these situations, Annex J of B51-14 specifies the protocol to be followed. Clause 4.1.10 does not require that FEA be performed to prove a design; rather, it states what the requirements are if it is performed. Furthermore, although FEA was not available or utilized during initial design of Pickering NGS pressure piping and fittings, requisite safety margins were integrated into the design at the time to account for analytical uncertainties. As a result, Clause 4.1.10 does not impact the original design basis and is not safety significant.

- Clause 7.1.2 mandates the use of ASME Boiler Pressure Vessel Code (BPVC) Section VIII, Division 1, Appendix FF for the design of pressure vessels incorporating quick-actuating closure. As discussed in the separate PSR2 review of ASME BPVC, Pickering NGS meets the intent of BPVC Section VIII. OPG Report N-REP-01903.1-10000 R017, "ASME Code Reconciliation for Material, Parts and Components" [B.21-28], known as the Reedy Report, identifies all the significant changes in BPVC Sections III and VIII (including Division 1) up to and including the latest 2015 edition. As discussed in the PSR2 BPVC review, OPG Letter NK30-CORR-00531-06324 R000, "Pickering NGS-B - CNSC staff assessment of OPG's 2011 Continued Operations Plan (Action Item 2010-8-05 (2461)) and path forward" [B.21-29] states that "OPG has established the process of using the Reedy Report to ensure compliance with ASME B&PVC Section III" in response to CNSC questions on Pickering B ISR Issue 1-359 (which relates to resolution of significant changes to BPVC that may impact the original design basis of the plant). CNSC staff agreed with OPG, and stated that they considered the response in NK30-CORR-00531-06324 R000 [B.21-29] to be satisfactory. This is also applicable to ASME BPVC Section VIII, since N-REP-01903.1-10000 R017 [B.21-28] addresses changes to both Sections III and VIII. Pickering NGS meets the intent of ASME BPVC Section VIII and there is therefore no PSR2 gap associated with compliance with CSA B51 Clause 7.1.2.

Based on the above, application of CSA B51-14 (including Update No. 1) [B.21-1] results in no findings that need to be addressed for Pickering NGS, unless modifications are made in the future. The Engineering Change Control (ECC) process (which includes review of N-LIST-00590-00001 R002 [B.21-24]), together with a yearly examination of any incremental (Code-over-Code) review requirements due to changes to CSA B51 (per OPG Guideline N-GUID-00590-00002 R001 [B.21-27]), ensures that that any applicable design changes made to Pickering NGS comply with the latest version of B51 going forward. Therefore, there are no PSR2 gaps for Pickering NGS compliance with CSA B51-14 (including Update No. 1) [B.21-1].

B.21.3 Compliance Summary for Pickering PSR2

There are no PSR2 gaps for CSA B51-14 (including Update No. 1) [B.21-1]. Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with CSA B51-14 (including Update No. 1).

B.21.4 References

[B.21-1] CSA Standard B51-14, *Boiler, Pressure Vessel, and Pressure Piping Code*, January 2014; Update No. 1, December 2014.

[B.21-2] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.21-3] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.21-4] OPG Report, NK30-REP-03680-00001 R000, *Pickering NGS-B Integrated Safety Review – Plant Design Safety Factor*, August 2007.

[B.21-5] CSA Standard B51-03 (R2007), *Boiler, Pressure Vessel, and Pressure Piping Code*, March 2003; Update No. 1, 2007.

[B.21-6] OPG Report, NK30-REP-03680-00015 R000, *Pickering NGS-B Integrated Safety Review (ISR) - Final ISR Report,* August 2009.

[B.21-7] CNSC Letter, NK30-CORR-00531-05312, *Pickering NGS-B Integrated Safety Review (ISR) - CNSC Staff Response to OPG's Comment Disposition Report*, e-Docs # 3507572, March 4, 2010.

[B.21-8] OPG Letter, NK30-CORR-00531-05578, *Pickering NGS-B – Follow-up to the Closure of the Integrated Safety Review (ISR) Study*, June 23, 2010.

[B.21-9] OPG Letter, NA44-CORR-00531-00381, *Pickering A - Updated Basis for Return to Service Document*, April 20, 2001.

[B.21-10] AECL Assessment Document, 44RS-00531-ASD-001 Rev. 04, *Review of Pickering A Design Against Current Codes and Standards*, November 2000.

[B.21-11] CSA Standard B51-95, *Boiler, Pressure Vessel, and Pressure Piping Code*, January 31, 1995.

[B.21-12] OPG Letter, NA44-CORR-00531-00190 R000, *ASME Codes Reconciliation - System and Component Modifications (Pickering NGS-A, Units 1-4),* October 6, 2000.

[B.21-13] OPG Letter, P-CORR-00531-03719 R000, *Application for Renewal of Pickering Nuclear Generating Station Power Reactor Operating Licence*, July 4, 2012.

[B.21-14] OPG Manual, N-MAN-01913.11-10000 R016, *Pressure Boundary Program Manual*, February 13, 2015.

[B.21-15] CSA Standard B51-09 including Update No. 1, *Boiler, Pressure Vessel, and Pressure Piping Code*, March 2009.

[B.21-16] OPG Report, NK38-REP-03680-10029 R000, *Review of CAN/CSA-B51-03 (R2007) (March 2003), Boiler, Pressure Vessel, and Pressure Piping Code for Darlington Integrated Safety Review*, June 2011.

[B.21-17] OPG Report, NK38-REP-03680-10029-ADD-001 R000, *Addendum to the CAN/CSA-B51-03 Code Review Report for Darlington ISR*, January 2014.

[B.21-18] OPG Report, NK38-REP-03680-10104-ADD-001 R000, *Darlington NGS Integrated Safety Review (ISR) - Final ISR Report Addendum*, June 2013.

[B.21-19] OPG Report, NK38-REP-03680-10201 R001, *ISR Open Issues and Acceptable Deviations - Adequacy Review*, October 2014.

[B.21-20] OPG Report, NK38-REP-03680-10135 R000, *Code Refresh Review of CSA B51-09-UPD1, Boiler, Pressure Vessel, and Pressure Piping Code for Darlington ISR*, July 2013.

[B.21-21] OPG Report, N-REP-00590-0555896 R001, *Code-over-Code Review Report: CSA B51 for the Year 2015*, November 2015.

[B.21-22] CSA Standard B51-14, *Boiler, Pressure Vessel, and Pressure Piping Code*, January 2014.

[B.21-23] OPG Report, N-REP-00590-0520103 R000, *Code-over-Code Review Report: CSA B51 for the Year 2014*, December 2014.

[B.21-24] OPG List, N-LIST-00590-00001 R002, *List of Significant Technical Changes from Code-over-Code Review*, August 2015.

[B.21-25] OPG Form, N-FORM-10959 R016, *Design Scoping Checklist*, June 2016.

[B.21-26] OPG Procedure, N-PROC-MP-0090 R014, *Modification Process*, October 14, 2016.

[B.21-27] OPG Guideline, N-GUID-00590-00002 R001, *Code-over-Code Review - Guideline*, August 15, 2016.

[B.21-28] OPG Report, N-REP-01903.1-10000 R017, *ASME Code Reconciliation for Material, Parts and Components*, November 18, 2015.

[B.21-29] OPG Letter, NK30-CORR-00531-06324 R000, *Pickering NGS-B - CNSC staff assessment of OPG's 2011 Continued Operations Plan (Action Item 2010-8-05 (2461)) and path forward*, June 19, 2012.

B.22 NFPA 20 (2016), "Standard for the Installation of Stationary Pumps for Fire Protection"

B.22.1 Background

The following excerpts paraphrased from NFPA 20 (2016) [B.22-1] provide a brief overview of the purpose of this standard and the requirements expressed therein:

> *The purpose of NFPA 20 is to provide a reasonable degree of protection for life and property from fire through selection and installation requirements for stationary pumps for fire protection based upon sound engineering principles, test data, and field experience. NFPA 20 criteria address the full range of issues and apply to all types of pumps including those for high-rise buildings, centrifugal, vertical shaft turbine-type, and positive displacement.*
>
> *The standard deals with the selection and installation of pumps supplying liquid for private fire protection. The scope includes liquid supplies; suction, discharge, and auxiliary equipment; power supplies, including power supply arrangements; electric drive and control; diesel engine drive and control; steam turbine drive and control; and acceptance tests and operation.*

NFPA 20 is relevant to Safety Factor 1 (Plant Design).

Compliance with NFPA 20 is not a licence requirement for Pickering NGS (PROL 48.02/2018), although it is referred to in the R04 Pickering Licence Conditions Handbook [B.22-2]. It may be considered an indirect licence requirement as CSA N293-07, "Fire Protection for CANDU Nuclear Power Plants" [B.22-3] is a licence requirement and refers to NFPA 20 as follows:

> *5.1.3 Where specific design or operational requirements are not addressed in this Standard, the NBCC [National Building Code of Canada], or the NFCC [National Fire Code of Canada], good engineering practice shall apply and, where appropriate, recognized Standards (such as those of the National Fire Protection Association [NFPA]) shall be used.*
>
> *7.3.2.2 Fire pumps shall be provided in accordance with this Standard and NFPA 20.*

NFPA 20 (2016) is the twentieth edition of the standard, and supersedes the previous edition published in 2013. Other recent editions include changes in 2010 and 2007. The Preface of the current 2016 edition of NFPA 20 [B.22-1] provides information on recent changes which are discussed in Section B.22.2.2.

The results of PSR1 NFPA 20 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.22.2. As identified in Reference [B.22-4], the Pickering PSR2 review of NFPA 20 (2016) is an Incremental Review. PSR2 Incremental Review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.

- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.22.2 Compliance Assessment for Pickering PSR2

B.22.2.1 Application of PSR1 Reviews

The versions of NFPA 20 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

No review was performed against NFPA 20 as part of the Pickering B ISR. OPG Report NK30-REP-03680-00001 R000, "Pickering NGS-B Integrated Safety Review - Plant Design Safety Factor" [B.22-5] discusses NFPA 20 in terms of other standards requiring general compliance with NFPA standards (e.g., CSA N293-07 [B.22-3] which states: "…where appropriate, recognized Standards (such as those of the National Fire Protection Association [NFPA]) shall be used"). Reviews against NFPA 20 were also not performed as part of the 2000 or 2010 Pickering B Fire Protection Code Compliance Review (CCR) assessments ([B.22-6], [B.22-7] respectively).

Although a review was not completed against NFPA 20 for the Pickering B ISR or CCRs, a review was not required since the fire protection headers at Pickering Units 5-8 are supplied by the station High Pressure Service Water (HPSW) pumps and not dedicated fire pumps (per Design Manuals NK30-DM-71400-00001 R006, "Fire Protection System" [B.22-8] and NK30-DM-71340-00001 R002, "High Pressure Service Water System" [B.22-9]). Therefore, NFPA 20 is not applicable to Pickering Units 5-8.

Pickering Units 1,4

AECL Report 44RS-00531-AB-001 Rev. 01, "Methodology for Review of Pickering A Design Against Current Regulations and Standards" [B.22-10] lists NFPA 20 as a "Code and Standard typically used for CANDU Nuclear Generating Stations". No review was performed, since Pickering A did not have dedicated fire pumps at the time. Similarly, reviews were not performed as part of the 2011 or 2000 Pickering A Fire Protection CCR assessments ([B.22-11], [B.22-12] respectively).

Per OPG Letter NA44-CORR-00531-06269 R000, "Pickering "A" - Installation of Diesel Engine Driven Fire Pumps (MEC 91665)" [B.22-13], the fire protection headers at Pickering A were originally supplied by the station HPSW pumps. During the determination of the proposed end state for the Units 2 and 3 HPSW systems (following the OPG decision not to restart these units), it was identified that the capacity of the Class III HPSW systems on Units 1 and 4 was not sufficient to supply the combined Class III HPSW loads and station fire water requirements under a Loss of Bulk Electrical Supply scenario if the Units 2 and 3 Service Water systems were shut down. Following a detailed review of options to satisfy the combined firewater / Class III demand, four new diesel firewater pumps were installed to supply the full Units 1,4 firewater load. The four pumps, which are located in the Pickering A Screenhouse pump pit area, are each designed to supply 50% of design water flow rate and are started sequentially on falling pressure with a time delay to allow the running pump(s) to recover fire header pressure before an additional pump is started ([B.22-14], [B.22-15]). Monthly and yearly Safety Related Systems Tests are performed on the pumps as outlined in [B.22-16] and [B.22-17].

Per Section 2.6 of the Modification Design Requirements (MDR) for the Pickering Units 1,4 firewater pumps [B.22-18]: "The new fire pump system shall meet the performance requirements specified in NFPA-20". Similar statements are made in the MDR with respect to the Pickering Units 1,4 firewater pumps meeting NFPA 20 mandated piping, civil design, instrumentation and control, periodic inspection and commissioning requirements, per Sections 2.1, 2.3, 2.4, 2.14 and 2.17 of [B.22-18]. Analogous statements are also made in the Design Plan for the installation of the fire pumps [B.22-19]. Per OPG Drawing NA44-DRAW-71410-10005 R000, "Pickering NGS A Screenhouse Diesel Fire Pumps Flow Diagram" [B.22-20], the 2003 version of NFPA 20 was used for the installation of the Pickering Units 1,4 firewater pumps. Pickering Units 1,4 compliance against the more recent (i.e., 2007, 2010, 2013 and 2016) versions of NFPA 20 is addressed under Section B.22.2.2 below.

Darlington NGS

OPG Reports NK38-REP-03680-10050 R000, "Review of NFPA-20, Standard for the Installation of Stationary Pumps for Fire Protection - 2007 Edition for Darlington ISR" [B.22-21] and NK38-REP-03680-10127 R001, "Gap Analysis to NFPA 20 2007 Edition, for DNGS Fire Protection Booster Pumps for Darlington ISR" [B.22-22] were prepared to assess compliance against the 2007 version of NFPA 20 for the Darlington ISR. A follow-up Code Refresh review of the 2013 version of NFPA 20 was also prepared in OPG Report NK38-REP-03680-10182 R000, "Code Refresh of NFPA 20, Standard for the Installation of Stationary Pumps for Fire Protection, 2013 Edition" [B.22-23]. Thirteen minor gaps were identified. OPG Report NK38-REP-03680-10104-ADD-002 R000, "Darlington NGS ISR - Final ISR Report Addendum 002" [B.22-24] provided updates to issues, classifications, justifications, and plans for the gaps identified in the above reviews. OPG Report NK38-REP-03680-10201 R001, "ISR Open Issues and Acceptable Deviations - Adequacy Review" [B.22-25] undertook a systematic approach to confirm previously discussed justification for the gaps being Acceptable Deviations and to confirm the adequacy of the safety improvements proposed for open issues. All issues were classified as Acceptable Deviations with a very low safety significance and no follow-up required.

The Darlington ISR review findings are not discussed further due to limited station-specific applicability to the Pickering Unit 1,4 firewater pumps.

B.22.2.2 Application of Post PSR1 Reviews

The baseline for Pickering Units 1,4 NFPA 20 compliance is the 2003 version of the standard. Therefore, the changes to NFPA since 2003 are discussed below.

The following information on recent revisions to the standard is taken from the current 2016 edition of NFPA 20 [B.22-1]:

> *For the 2007 edition, requirements for variable speed drives were refined, requirements for break tanks were added, and component replacement testing tables were included.*
>
> *The 2010 edition included a new chapter on fire pumps for high-rise buildings. Requirements for pumps arranged in series were also added to the general requirements chapter. Chapter 11 of the standard was reorganized.*
>
> *The 2013 edition clarified and added new requirements for water mist positive displacement pumping units. Chapter 5 of the standard was reorganized. Limited service controller requirements were revised, and the component replacement table was removed.*
>
> *The 2016 edition of NFPA 20 provides new requirements for pumps in series relative to protection of control wiring, status signals, and communications. NFPA 20 recognizes the potential use of multistage, multiport pumps in fire suppression systems and provides requirements specific to that application. Break tank criteria have been removed and are now in accordance with NFPA 22, Standard for Water Tanks for Private Fire Protection. A new annex, Annex C, has been added to provide guidance on controller security where a controller is connected to the Internet. New requirements have been added to address use of an automatic fuel maintenance system with a diesel fire pump installation. In addition, protection criteria for both a diesel fire pump room and an electric fire pump room are defined in Chapter 4.*

With respect to the 2007 NFPA 20 additions, variable speed drives are applicable to electric pumps. The firewater pumps for Pickering Units 1,4 are diesel pumps, and therefore, these changes are not applicable. For changes relating to break tanks, the current 2016 edition of NFPA 20 [B.22-1] states "Break tank criteria have been removed and are now in accordance with NFPA 22, "Standard for Water Tanks for Private Fire Protection"." Component replacement testing tables were also removed in subsequent editions of NFPA 20 and are now in accordance with NFPA 25, "Standard for the Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems". As noted in NFPA 20 2016 [B.22-1], acceptance test criteria for replacement of critical path components of a fire pump installation were already part of NFPA 20 2003.

With respect to the 2010 NFPA 20 additions, fire pumps for high-rise buildings are not applicable to Pickering NGS. As per NA44-DRAW-71410-10005 R000, "Pickering NGS A Screenhouse Diesel Fire Pumps Flow Diagram" [B.22-26], the two sets of fire pumps in the East/West Screenhouse pump pits are arranged in parallel. Therefore, the revised 2010 NFPA general requirements associated with pumps arranged in series are also not applicable to Pickering NGS.

OPG Memorandum N-CORR-00590-0477036 [B.22-27] assessed the changes between the 2013 and 2010 versions of NFPA 20 and identified that all changes were either editorial or were not significant technical changes. Therefore, there are no significant changes that are relevant to the Pickering Units 1,4 firewater pumps.

OPG Report N-REP-00590-00014 R001, "Code-over Code Review Report: Code over Code Review for NFPA 20-2016 Vs. NFPA 20-2013" [B.22-28] assessed the changes between the 2016 and 2013 versions of NFPA 20 and identified ten significant technical changes to be added to OPG List N-LIST-00590-00001 R002, "List of Significant Technical Changes from Code-over-Code Review" [B.22-29]. As noted in N-LIST-00590-00001 R002 [B.22-29]:

> *Changes identified as Significant Technical Changes that do not require immediate compliance or review of OPG existing installations are listed in this document. Projects for modifications to existing installations or new installations shall use the listed Significant Technical Changes, as appropriate.*

Nine of the changes identified in N-REP-00590-00014 R001 [B.22-28] are new requirements for series fire pumps and are not applicable to Pickering Units 1,4 firewater pumps as discussed earlier. The tenth change is a new requirement for circulation relief valve setpoint pressure and is related to NFPA 20 2016 Clause 4.12.1.1 that states: "Where an electric variable speed pressure limiting controller is installed, the automatic circulation relief valve shall be set to a minimum of 5 psi (0.34 bar) below the operation set pressure." This clause is applicable to electric pumps in order to control the speed of the pump. The firewater pumps for Pickering Units 1,4 are diesel pumps, and therefore, this clause does not apply.

Based on the above, there are no significant changes from the 2003 version of NFPA 20 that have an impact on nuclear safety. As a result, Pickering Units 1,4 firewater pumps meet the intent of NFPA 20 (2016) [B.22-1].

B.22.3 Compliance Summary for Pickering PSR2

There are no PSR2 gaps for NFPA 20 (2016) [B.22-1]. Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with NFPA 20 (2016).

B.22.4 References

[B.22-1] NFPA 20, *Standard for the Installation of Stationary Pumps for Fire Protection (2016 Edition)*, 2016

[B.22-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.22-3] CSA Standard, N293-07, *Fire Protection for CANDU Nuclear Power Plants*, February 2012.

[B.22-4] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.22-5] OPG Report, NK30-REP-03680-00001 R000, *Pickering NGS-B Integrated Safety Review – Plant Design Safety Factor*, August 2007.

[B.22-6] OPG Report, NK30-REP-71400-10001 R000, *Fire Protection Code Compliance Review Pickering Nuclear Generating Station B*, May 2000.

[B.22-7] OPG Report, NK30-REP-71400-10001 R001, *Fire Protection Code Compliance Review Pickering Nuclear Generating Station B*, November 23, 2010.

[B.22-8] OPG Design Manual, NK30-DM-71400-00001 R006, *Fire Protection System*, January 2016.

[B.22-9] OPG Design Manual, NK30-DM-71340-00001 R002, *High Pressure Service Water System,* December 2011.

[B.22-10] AECL Report, 44RS-00531-AB-001 Rev. 01, *Methodology for Review of Pickering A Design Against Current Regulations and Standards*, November 2000.

[B.22-11] OPG Report, NA44-REP-71400-10001 R001, *Pickering Nuclear Generating Station "A" Fire Protection Code Compliance Review*, March 2011.

[B.22-12] OPG Report, NA44-REP-71400-10001 R000, *Pickering Nuclear Generating Station "A" Fire Protection Code Compliance Review*, April 2000.

[B.22-13] OPG Letter, NA44-CORR-00531-06269 R000, *Pickering "A" – Installation of Diesel Engine Driven Fire Pumps (MEC 91665)*, February 23, 2010.

[B.22-14] OPG Operating Manual, NA44-OM-014-71400-04.06 Rev. 014, *Screenhouse Fire Pumps*, May 2016.

[B.22-15] OPG Design Manual, NA44-DM-71400-00002 R000, *Fire Protection Systems (Water),* February 2014.

[B.22-16] Pickering NGS Safety Related System Test, NA44-SRS-P-097 R010, *Diesel Fire Pumps Weekly Test*, July 2016.

[B.22-17] Pickering NGS Safety Related System Test, NA44-SRS-P-098 R001, *Diesel Fire Pumps Annual Test*, October 2015.

[B.22-18] OPG Modification Design Requirements, NA44-MDR-71400-00002 R000, *Modification Design Requirements for Installing Two New Dedicated Diesel Fire Pumps*, May 22, 2007.

[B.22-19] OPG Design Plan, NA44-DP-71400-00009 R002*, Pickering NGS A – Design Plan – Fire Protection – Installation of Dedicated Diesel Fire Pumps*, February 11, 2010.

[B.22-20] OPG Drawing, NA44-DRAW-71410-10005 R000, *Pickering NGS A Screenhouse Diesel Fire Pumps Flow Diagram*, February 2010.

[B.22-21] OPG Report, NK38-REP-03680-10050 R000, *Review of NFPA-20, Standard for the Installation of Stationary Pumps for Fire Protection – 2007 Edition for Darlington Integrated Safety Review*, August 2011.

[B.22-22] OPG Report, NK38-REP-03680-10127 R001, *Gap Analysis to NFPA 20 2007 Edition, for DNGS Fire Protection Booster Pumps for Darlington Integrated Safety Review*, March 2013.

[B.22-23] OPG Report, NK38-REP-03680-10182 R000, *Code Refresh of NFPA 20, Standard for the Installation of Stationary Pumps for Fire Protection, 2013 Edition*, December 2013.

[B.22-24] OPG Report, NK38-REP-03680-10104-ADD-002 R000, *Darlington NGS Integrated Safety Review (ISR) - Final ISR Report Addendum 002*, November 2013.

[B.22-25] OPG Report, NK38-REP-03680-10201 R001, *ISR Open Issues and Acceptable Deviations – Adequacy Review*, October 2014.

[B.22-26] OPG Drawing, NA44-DRAW-71410-10005 R000, *Pickering NGS A Screenhouse Diesel Fire Pumps Flow Diagram*, February 2010.

[B.22-27] OPG Memorandum, N-CORR-00590-0477036 P, *Code-over-Code Review of NFPA 20, Standard for the Installation of Private Service Mains and Their Appurtenances – 2010 Edition over the 2010 Edition Including Amendments 1 and 2* [sic – actually 2013 Edition over 2010 Edition, and Title is incorrect], October 08, 2013.

[B.22-28] OPG Report, N-REP-00590-00014 R001, *Code-over Code Review Report: Code over Code Review for NFPA 20-2016 Vs. NFPA 20-2013*, March 2016.

[B.22-29] OPG List, N-LIST-00590-00001 R002, *List of Significant Technical Changes From Code-over-Code Review*, August 2015.

B.23 NFPA 24 (2016), "Standard for the Installation of Private Fire Service Mains and Their Appurtenances"

B.23.1 Background

The following excerpts paraphrased from NFPA 24 (2016) [B.23-1] provide a brief overview of the purpose of this standard and the requirements expressed therein:

> *The purpose of NFPA 24 is to provide a reasonable degree of protection for life and property from fire through installation requirements for private fire service main systems based on sound engineering principles, test data, and field experience. The standard helps to ensure water supplies are available in a fire emergency. NFPA 24 provides minimum requirements for the installation of private fire service mains and their appurtenances, which includes supplying automatic sprinkler systems, open sprinkler systems, water spray fixed systems, foam systems, private hydrants, monitor nozzles or standpipe systems with reference to water supplies, and hose houses.*
>
> *NFPA 24 defines the requirements that govern water supplies, valves, hydrants, hose houses and equipment, master streams, underground and aboveground piping, and hydraulic calculations.*

NFPA 24 is relevant to Safety Factor 1 (Plant Design).

Compliance with NFPA 24 is not a licence requirement for Pickering NGS (PROL 48.02/2018), although it is referred to in the R04 Pickering Licence Conditions Handbook [B.23-2]. It may be considered an indirect licence requirement as CSA N293-07, "Fire Protection for CANDU Nuclear Power Plants" [B.23-3] is a licence requirement and refers to NFPA 24 as follows:

> *5.1.3 Where specific design or operational requirements are not addressed in this Standard, the NBCC [National Building Code of Canada], or the NFCC [National Fire Code of Canada], good engineering practice shall apply and, where appropriate, recognized Standards (such as those of the National Fire Protection Association [NFPA]) shall be used.*
>
> *7.3.2.3 The fire protection water distribution system shall be provided in accordance with NFPA 24.*

It is noted that OPG has been actively working on risk reduction plans to address higher risk areas, including internal fires at Pickering Units 1,4, with the objective of reducing Pickering Units 1,4 risk for both Severe Core Damage Frequency and Large Release Frequency. These risk reduction plans include further refinements to the fire Probabilistic Safety Assessments (PSAs) ([B.23-4], [B.23-5]) to address potential over-conservatisms in the existing results, as well as potential physical plant and operational changes intended to reduce overall risk. The fire PSA results have highlighted the increased importance of fire to site risk at Pickering NGS, and as a result this NFPA 24 review focuses on identifying any nuclear safety significant findings that may impact fire water supplies. P-LIST-71400-00001 R000, "Application of CSA N293-07 to Structures, Systems and Components for Pickering Nuclear" [B.23-6] was prepared by OPG to identify the Structures, Systems and Components (SSCs) within the protected area that are exempt from the application of N293-07, and the structures outside the protected area that need to align with N293-07. P-LIST-71400-00001 R000 has been utilized in this NFPA 24 review to identify fire water supply credits at Pickering NGS. P-LIST-71400-00001 R000 also identifies the exempted SSCs which have no impact on systems availability to achieve and maintain safe shutdown, as per the Fire Safe Shutdown ([B.23-7], [B.23-8]) and Fire Hazard ([B.23-9], [B.23-10]) Analyses.

NFPA 24 (2016) is the fifteenth edition of this standard, and supersedes the previous edition published in 2013. Other recent editions include changes in 2010 and 2007. The Preface of the current 2016 edition of NFPA 24 [B.23-1] provides information on recent changes which will be discussed in Section B.23.2.2.

The results of PSR1 NFPA 24 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.23.2. As identified in Reference [B.23-11], the Pickering PSR2 review of NFPA 24 (2016) is an Incremental Review. PSR2 Incremental Review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- <u>Compliance:</u> Compliance indicates that the change in the safety requirement, per the topical review, is met.
- <u>Gap</u>: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.23.2 Compliance Assessment for Pickering PSR2

B.23.2.1 Application of PSR1 Reviews

The versions of NFPA 24 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

<u>Pickering Units 5-8</u>

No review was performed against NFPA 24 as part of the Pickering B ISR. OPG Report NK30-REP-03680-00001 R000, "Pickering NGS-B Integrated Safety Review - Plant Design Safety Factor" [B.23-12] discusses NFPA 20 in terms of other standards requiring general compliance with NFPA standards (e.g., CSA N293-07 [B.23-3] which states: "…where appropriate, recognized Standards (such as those of the National Fire Protection Association [NFPA]) shall be used").

Although a review against NFPA 24 was not prepared as part of the Pickering B ISR, reviews against NFPA 24 were performed as part of the 2000 and 2010 Pickering B Fire Protection Code Compliance Review (CCR) assessments ([B.23-13], [B.23-14] respectively). Per Section 2.4 of

OPG Report NK30-REP-71400-10001 R001, "Fire Protection Code Compliance Review Pickering Nuclear Generating Station B" [B.23-14], the 1970 version of NFPA 24 was reviewed as part of the Pickering B CCR. NFPA 24 was also reviewed in the context of the CCR review [B.23-14] of the 1969 version of NFPA 11, "Foam Extinguishing Systems" and 1972 version of NFPA 13, "Standard for the Installation of Sprinkler Systems" which also specify requirements for underground mains.

Section 3.7 of NK30-REP-71400-10001 R001 [B.23-14] describes the deviations identified against NFPA 24 as follows (text in italics is taken verbatim):

- NFPA 1970 Section 3601: *Deviation - Yard post indicator valves are not provided with industry-standard signs, which indicate section or portion of underground system controlled as required by code (Deviation # 04801). Technical Resolution - Current fire safety standards require that identification signs be provided to indicate the valve function and what it controls. The existing tags provide identification however the tags are difficult to read in some cases. Also, the tags do not readily indicate what section of the underground system is controlled by the valve. It is common practice in industrial facilities to identify valves by painting their identification number either on the valve post or a sign attached to the bumper post. Of the valves inspected, most were provided with a tag, however a general review of the valve identification should be made. Recommended Action - Valves in the exterior fire main loop should be inspected for proper identification and legibility of information. Action Status - Resolved… No further action required.*

- NFPA 1970 Section 3303: *Deviation - Yard post indicator valves at PNGS B are not installed at the code required 36 inches above finish grade (Deviation #'s 03201 and 09919)). Technical Resolution – The main intent of the requirement is to ensure that the top of the valve is at a reasonable height to permit correct and easy operation of the valve. The height difference will not prevent the operation of the valve. Disposition – Noncompliance - Acceptable. Recommended Action - No action required.*

- NFPA 1970 Section 3601: *Deviation - Yard post indicator valves at PNGS B are not secured in the open position as required by code (Deviation # 13301). Technical Resolution - To ensure that water is available at hydrants, hose stations and foam systems when required by emergency responders, the yard post indicator valves [PIVs] controlling the water supply to hydrants and systems on the site must remain open at all times. Where electronic supervision of control valves is not provided, such valves should be locked in the open position or sealed and inspected on a monthly basis. Recommended Action - Post indicator valves that are not electronically supervised should be locked in the open position or sealed and checked on a monthly basis. Action Status - Outstanding… Yard PIV's serving fire water systems should be either locked in the open position with chains or locks or equipped with supervisory monitoring switches connected to the station fire alarm systems.*

Based on the above, Deviations # 04801, 03201 and 09919 have been resolved. These resolutions are not impacted by Pickering NGS operation past 2020. Outstanding Deviation # 13301, which applies to Pickering Units 1,4 as well as Units 5-8, is currently in progress with locks installed on the majority of the affected valves. As discussed earlier, OPG List P-LIST-

71400-00001 R000 [B.23-6] identifies the SSCs within the protected area that are exempt from the application of N293-07, and the structures outside the protected area that need to follow N293-07. Exempted SSCs "have no impact on systems availability to achieve and maintain safe shutdown as per the fire safe shutdown analysis" [B.23-6]. Based on P-LIST-71400-00001 R000, there are a number of SSCs in the yard which directly support plant operation and which are defined as being "related to nuclear safety". As a result, fire water supply to these SSCs is a credited safety function. Since Deviation # 13301 is not yet complete, this deviation is nuclear safety significant and is identified as **PSR2 NFPA 24 Gap #1**.

Pickering Units 5-8 compliance against the 2016 version of NFPA 24 is addressed under Section B.23.2.2 below.

Pickering Units 1,4

AECL Report 44RS-00531-AB-001 Rev. 01, "Methodology for Review of Pickering A Design Against Current Regulations and Standards" [B.23-15] lists NFPA 24 as a "Code and Standard typically used for CANDU Nuclear Generating Stations". A review against NFPA 24 was not performed as part of Pickering A Return to Service. Reviews were also not performed against NFPA 24 as part of the 2000 or 2011 Pickering A Fire Protection CCR assessments ([B.23-16], [B.23-17] respectively). As a result, there is no "baseline" NFPA 24 compliance assessment available for Pickering Units 1,4. Since compliance has not been formally documented, this is identified as a PSR2 gap (**PSR2 NFPA 24 Gap #2**).

The safety significance of this PSR2 gap is discussed in more detail under Section B.23.2.2 below. In particular, Pickering A Firewater Pipe Replacement Project 13-80069 has replaced various sections of the buried fire protection piping in the north and south yards in the Pickering A un-zoned area. This proactive modification was undertaken to replace the existing grey cast iron piping, which was prone to failure, with PVC piping. As identified in OPG Project Charter NA44-PCH-71450-00001 R000, "Project Charter Buried Fire Pipe Replacement" [B.23-18], the main goals of the project are to eliminate isolation of major fire suppression systems due to buried pipe failures, minimize future failures of the buried grey cast iron pipe, and limit any future failures to lower consequence areas. The new portions of piping will be installed in accordance with NFPA 24. Given that the scope of the project includes the highest risk portions of piping, the consequence of failure of the remaining portions of the piping is considered to be low.

Darlington NGS

OPG Reports NK38-REP-03680-10051 R000, "Review of NFPA-24, Standard for the Installation of Private Service Mains and Their Appurtenances - 2007 Edition for Darlington Integrated Safety Review" [B.23-19] and NK38-REP-03680-10176 R002, "Gap Analysis to NFPA 24 2007 Edition, for DNGS Fire Protection Water Supply and Distribution System for Refurbishment" [B.23-20] identified Darlington NGS gaps against the 2007 version of NFPA 24. OPG Report NK38-REP-03680-10181 R000, "Code Review Refresh of NFPA 24, Standard for the Installation of Private Service Mains and Their Appurtenances, 2013 Edition" [B.23-21] reviewed the changes to NFPA 24 from the 2007 edition to the 2013 edition and identified three additional gaps. OPG Report NK38-REP-03680-10201 R001, "ISR Open Issues and Acceptable Deviations

- Adequacy Review" [B.23-22] later classified all NFPA 24 gaps as Acceptable Deviations with a very low safety significance and no follow-up required.

These gaps are not applicable to Pickering NGS; however, the safety significance of the findings is generally applicable to Pickering NGS, as discussed under Section B.23.2.2 below.

B.23.2.2 Application of Post PSR1 Reviews

For Pickering Units 5-8, the baseline for NFPA 24 compliance is the 1970 version of the standard. The following information on recent revisions to the standard is taken from the Preface of the current 2016 edition of NFPA 24 [B.23-1]:

> *In 1953, on recommendation of the Committee on Standpipes and Outside Protection, the two standards (NFPA 24 and NFPA 25) were completely revised and adopted as NFPA 24. Amendments were made leading to separate editions in 1955, 1959, 1962, 1963, 1965, 1966, 1968, 1969, 1970, 1973, 1977, 1981, 1983, and 1987.*
>
> *The 1992 edition included amendments to further delineate the point at which the water supply stops and the fixed fire protection system begins. Minor changes were made concerning special topics such as thrust restraint and equipment provisions in valve pits.*
>
> *The 1995 edition clarified requirements for aboveground and buried piping. Revisions were made to provide additional information regarding listing requirements, signage, valves, valve supervision, hydrant outlets, system attachments, piping materials, and thrust blocks. User friendliness of the document was also addressed.*
>
> *The 2002 edition represented a complete revision of NFPA 24. Changes included reorganization and editorial modifications to comply with the Manual of Style for NFPA Technical Committee Documents. Additionally, all of the underground piping requirements were relocated into a new Chapter 10.*
>
> *The 2007 edition was revised in five major areas: Chapter 10 was editorially updated and minor technical changes were made. In addition, newly established leakage test criteria, well as updated requirements for thrust blocks and restrained joints were added to Chapter 10. Two annexes were new to this edition: Annex C, Recommended Practice for Fire Flow Testing, and Annex D, Recommended Practice for Marking of Hydrants. These two annexes were developed based on the 2002 edition of NFPA 291.*
>
> *The 2010 edition was revised in three major areas: the provisions for location and identification of fire department connections [FDCs], valves controlling water supply, and protection of service mains entering the building.*
>
> *The 2013 edition of NFPA 24 included clarifications on the requirements for running piping under buildings, including annex figures depicting clearances. The Contractors Material and Test Certificate for Underground Piping (Figure 10.10.1) was modified to include confirmation that the forward flow test of the backflow preventer had been conducted. A provision requiring the automatic drip valve to be located in an accessible location that permits inspections in accordance with NFPA 25 was also added.*

NFPA 24 underwent a structural rewrite for the 2016 edition. The hydrant definitions have been clarified to describe the type of hydrant in question, as opposed to describing when and where they would be used. The valve arrangement requirements have been rewritten for clarity, and annex figures added to provide figures that are consistent with NFPA 13. The title of Chapter 6 has been changed from "Valves" to "Water Supply Connections," to better describe the material covered within the chapter. Revisions to Section 6.1 better call out the permitted exceptions to indicating valves and permit nonlisted tapping sleeve and valve assemblies in connections to municipal water supplies. The center of hose outlet measurements have been updated to include clear minimum and maximum values for the location of the outlet, along with the appropriate measurement for a hose house installation. The steel underground piping references have been removed from the table in Chapter 10 since steel pipe is required to be listed other than in the FDC line. A statement also has been added to allow underground fittings to be used above ground to transition to aboveground piping.

With respect to the significance of recent changes to the 2013 and 2016 editions of NFPA 24, OPG Memorandum N-CORR-00590-0477037, "Code-over-Code Review of NFPA 24, Standard for the Installation of Private Service Mains and Their Appurtenances - 2013 Edition over the 2010 Edition" [B.23-23] identified no significant technical changes and no mitigation requirements. OPG Report N-REP-00590-00015 R001, "Code-over Code Review Report: Code over Code Review for NFPA 24-2016 VS NFPA 24-2013" [B.23-24] concluded:

> *With respect to our review of the changes to the NFPA 24, 2016 Edition, no significant technical changes were identified which would impact the use of NFPA 24 at OPG nuclear generating stations. As such, no mitigation plans are proposed.*

However, N-REP-00590-00015 R001 identified two significant technical changes to be added to N-LIST-00590-00001 R002, "List of Significant Technical Changes from Code-over-Code Reviews" [B.23-25] for use in future modifications or new-build. These new requirements relate to the installation of fire mains under a building, and a new requirement for backfill. As noted in N-LIST-00590-00001 R002 [B.23-25]:

> *Changes identified as Significant Technical Changes that do not require immediate compliance or review of OPG existing installations are listed in this document. Projects for modifications to existing installations or new installations shall use the listed Significant Technical Changes, as appropriate.*

All modifications require review of N-LIST-00590-00001 R002, as identified in N-FORM-10959 R016, "Design Scoping Checklist" [B.23-26], as per N-PROC-MP-0090 R014, "Modification Process" [B.23-27] (N-FORM-10959 R016 Section 2.19 requires that a review of N-LIST-00590-00001 R002 "shall be completed to determine if any code change improvement actions apply to the modification"). Significant technical changes will be applied in accordance with the most recent versions of NFPA 24, as appropriate, for modifications to existing Pickering NGS installations going forward in accordance with OPG governance. Therefore, the Engineering Change Control (ECC) process, together with a yearly examination of any incremental (Code-over-Code) review requirements due to changes to NFPA 24 [B.23-28], ensure that any design changes made to the Pickering NGS firewater system comply with the latest version of NFPA 24 going forward. Furthermore, per the R04 Pickering Licence Conditions Handbook [B.23-2], any changes that have the potential to impact fire protection are assessed for compliance with CSA N293-07 and, if required, an external third party review is performed and the results submitted to the CNSC. Per OPG Report N-REP-09076-10006 R000, "Review of Design Requirements of CSA N293-07" [B.23-29], the process presently implemented by OPG in the ECC process for modifications which have the potential to impact fire safety is: a) all modifications will be screened and those having potential for fire impact will receive a detailed review, b) those modifications screened as having potential to impact fire safety will be submitted to a qualified third party for review, and c) the qualified third party must not be in the same management and financial operation as the design organization.

Nevertheless, for Pickering Units 5-8 the baseline for NFPA 24 compliance is the 1970 version of the standard, and Pickering Units 1,4 have not been previously assessed against NFPA 24. Although recent changes to the 2013 and 2016 versions of NFPA 24 will be addressed in any firewater system design changes going forward, existing (as-built) Pickering Units 1,4 and 5-8 fire service mains compliance has not been formally documented against the most recent versions of NFPA 24. Further, there have been a large number of significant changes since the Pickering Units 5-8 fire service mains were last reviewed against NFPA 24 (in 1970), including (as mentioned above) the 2002 edition which "represented a complete revision of NFPA 24". Since Pickering NGS has not demonstrated compliance with the 2016 version of NFPA 24, this has been identified as a PSR2 gap (**PSR2 NFPA 24 Gap #2**) as discussed earlier.

With respect to the retroactive application of NFPA 24 to Pickering NGS, it is generally not practicable to make design changes to the fire service mains without rebuilding it (or parts of it). Furthermore, if a fundamental change in understanding occurs that could have a negative impact on safety, this is addressed in a timely fashion through Industry Operating Experience (OPEX) and associated design changes as required. For example, OPG Letter NA44-CORR-00531-07592 R000, "Submission of Fire Protection Independent Third Party Code Compliance Review for Pickering A Firewater Pipe Replacement Project 13-80069" [B.23-30] and associated OPG Report NA44-REP-71400-00034 R000, "Pickering A Buried Piping Third Party Review" [B.23-31] document the recent Pickering A Buried Fire Piping Replacement Project that has replaced various sections of the buried fire protection piping in the north and south yards in the Pickering A un-zoned area. This proactive modification was undertaken to replace the existing grey cast iron piping, which was prone to failure, with PVC piping. Section 2.0 of NA44-DCS-71450-00001 R000, "Detailed Commissioning Specification for 13-80069 PA Buried Fire Pipe Replacement" [B.23-32] states:

1. *The new underground piping shall be completely flushed before tie-in to fire protection system piping in accordance with NFPA 24.*

2. *Hydrostatic testing shall be performed in accordance with NFPA 24.*

As discussed earlier, OPG Project Charter NA44-PCH-71450-00001 R000 [B.23-18] states that the main goals of the project are to eliminate isolation of major fire suppression systems due to buried pipe failures, minimize future failures of the buried grey cast iron pipe and limit any future failures to lower consequence areas. Given that the scope of the project includes the highest risk portions of piping, the consequence of failure of the remaining portions of the piping is considered to be low.

Finally, as per NK30-CORR-00531-06032 R000, "Pickering B - Response to CNSC Action Item 20118-2289 - Type II Inspection of Fire Protection Water Supply Systems" [B.23-33], buried fire protection systems are included in the scope of N-PROC-MA-0088 R003, "Buried Piping Program Requirements" [B.23-34]. System piping is required to be tested every 5 years in accordance with NFPA 25, "Standard for the Inspection, Testing and Maintenance of Water-Based Fire Protection Systems". The requirement involves conducting flow tests of system piping to determine the internal condition of the buried piping. Pickering NGS exceeds the minimum testing requirement, and tests fire protection piping annually per Pickering Unit 018 Nuclear Operating Procedure P-OP-71450-0001 R004, "PNGS Semi-Annual Yard Fire Hydrant Inspection/Flow Test and Annual Hydrant Isolating Valve Maintenance" [B.23-35].

B.23.3 Compliance Summary for Pickering PSR2

There are two PSR2 gaps for NFPA 24 (2016) [B.23-1] which relate to Safety Factor 1 (Plant Design):

1. For OPG Report NK30-REP-71400-10001 R001, "Fire Protection Code Compliance Review Pickering Nuclear Generating Station B", there is an outstanding issue (Deviation # 13301) which relates to NFPA 24 1970 Section 3601: "Yard post indicator valves at PNGS B are not secured in the open position as required by code" (and which applies to Pickering Units 1,4 as well as Units 5-8). Work to resolve this deviation is currently in progress with locks installed on the majority of the affected valves. Based on OPG List P-LIST-71400-00001 R000, there are a number of Structures, Systems and Components (SSCs) in the yard which directly support plant operation and which are defined as being "related to nuclear safety". As a result, fire water supply to these SSCs is a credited safety function. Deviation # 13301 is not yet complete. Therefore, this has been identified as a PSR2 gap.

2. For Pickering Units 5-8 the baseline for NFPA 24 compliance is the 1970 version of the standard. Pickering Units 1,4 have not been previously assessed against NFPA 24. Although recent changes to the 2013 and 2016 versions of NFPA 24 will be addressed in any firewater system design changes going forward (as a result of Code-over-Code reviews performed for NFPA 24), compliance has not been formally documented for Pickering Units 1,4 or Units 5-8 against the most recent versions of NFPA 24. Furthermore, there have been a large number of significant changes to NFPA 24 since 1970, including the 2002 edition which "represented a complete revision of NFPA 24". Since Pickering NGS has not demonstrated compliance with the 2016 version of NFPA 24, this has been identified as a PSR2 gap. It is noted that OPG is proactively replacing portions of the firewater piping in accordance with NFPA 24, under the Pickering A Firewater Pipe Replacement Project 13-80069.

B.23.4 References

[B.23-1] NFPA 24 (2016 Edition), *Standard for the Installation of Private Fire Service Mains and Their Appurtenances*, 2016.

[B.23-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.23-3] CSA Standard, N293-07, *Fire Protection for CANDU Nuclear Power Plants*, February 2012.

[B.23-4] OPG Report, NA44-REP-03611-00038 R000, *Pickering NGS A Probabilistic Risk Assessment (PRA) – Internal Fire Report*, April 2014.

[B.23-5] OPG Report, NK30-REP-03611-00012 R000, *Pickering NGS B Probabilistic Risk Assessment - Internal Fire Report*, December 2012.

[B.23-6] OPG List, P-LIST-71400-00001 R000, *Application of CSA N293-07 to Structures, Systems and Components for Pickering Nuclear*, July 16, 2009.

[B.23-7] OPG Report, NA44-REP-71400-00023 R000, *Fire Safe Shutdown Analysis - Pickering A Nuclear Generating Station*, April 5, 2012.

[B.23-8] OPG Report, NK30-REP-71400-00001 R002, *Fire Safe Shutdown Analysis - Pickering B Nuclear Generating Station*, October 5, 2011.

[B.23-9] OPG Report, NA44-REP-71400-10003 R001, *Fire Hazard Assessment - Pickering A Nuclear Generating Station*, April 30, 2012.

[B.23-10] OPG Report, NK30-REP-71400-10002 R002, *Fire Hazard Assessment - Pickering B Nuclear Generating Station*, November 23, 2011.

[B.23-11] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.23-12] OPG Report, NK30-REP-03680-00001 R000, *Pickering NGS-B Integrated Safety Review – Plant Design Safety Factor*, August 2007.

[B.23-13] OPG Report, NK30-REP-71400-10001 R000, *Fire Protection Code Compliance Review Pickering Nuclear Generating Station B*, May 2000.

[B.23-14] OPG Report, NK30-REP-71400-10001 R001, *Fire Protection Code Compliance Review Pickering Nuclear Generating Station B*, November 23, 2010.

[B.23-15] AECL Report, 44RS-00531-AB-001 Rev. 01, *Methodology for Review of Pickering A Design Against Current Regulations and Standards*, November 2000.

[B.23-16] OPG Report, NA44-REP-71400-10001 R000, *Pickering Nuclear Generating Station "A" Fire Protection Code Compliance Review*, April 2000.

[B.23-17] OPG Report, NA44-REP-71400-10001 R001, *Pickering Nuclear Generating Station "A" Fire Protection Code Compliance Review*, March 2011.

[B.23-18] OPG Project Charter, NA44-PCH-71450-00001 R000, *Project Charter Buried Fire Pipe Replacement*, April 23, 2012.

[B.23-19] OPG Report, NK38-REP-03680-10051 R000, *Review of NFPA-24, Standard for the Installation of Private Service Mains and Their Appurtenances – 2007 Edition for Darlington Integrated Safety Review*, August 2011.

[B.23-20] OPG Report, NK38-REP-03680-10176 R002, *Gap Analysis to NFPA 24 2007 Edition, for DNGS Fire Protection Water Supply and Distribution System for Refurbishment*, June 2013.

[B.23-21] OPG Report, NK38-REP-03680-10181 R000, *Code Review Refresh of NFPA 24, Standard for the Installation of Private Service Mains and Their Appurtenances, 2013 Edition*, December 2013.

[B.23-22] OPG Report, NK38-REP-03680-10201 R001, *ISR Open Issues and Acceptable Deviations – Adequacy Review*, October 2014.

[B.23-23] OPG Memorandum, N-CORR-00590-0477037, *Code-over-Code Review of NFPA 24, Standard for the Installation of Private Service Mains and Their Appurtenances – 2013 Edition over the 2010 Edition*, October 09, 2013.

[B.23-24] OPG Report, N-REP-00590-00015 R001, *Code-over Code Review Report: Code over Code Review for NFPA 24-2016 VS NFPA 24-2013*, February 2016.

[B.23-25] OPG List, N-LIST-00590-00001 R002, *List of Significant Technical Changes From Code-over-Code Review*, August 2015.

[B.23-26] OPG Form, N-FORM-10959 R016, *Design Scoping Checklist*, June 2016.

[B.23-27] OPG Procedure, N-PROC-MP-0090 R014, *Modification Process*, October 14, 2016.

[B.23-28] OPG Guideline, N-GUID-00590-00002 R001, *Code over Code Review - Guideline*, August 15, 2016.

[B.23-29] OPG Report, N-REP-09076-10006 R000, *Review of Design Requirements of CSA N293-07*, April 12, 2007.

[B.23-30] OPG Letter, NA44-CORR-00531-07592 R000, *Submission of Fire Protection Independent Third Party Code Compliance Review for Pickering A Firewater Pipe Replacement Project 13-80069*, March 16, 2016.

[B.23-31] OPG Report, NA44-REP-71400-00034 R000, *Pickering A Buried Piping Third Party Review*, January 8, 2016.

[B.23-32] OPG Detailed Commissioning Specification, NA44-DCS-71450-00001 R000, *Detailed Commissioning Specification for 13-80069 PA Buried Fire Pipe Replacement*, July 2016.

[B.23-33] OPG Letter, NK30-CORR-00531-06032 R000, *Pickering B – Response to CNSC Action Item 20118-2289 – Type II Inspection of Fire Protection Water Supply Systems*, September 23, 2011.

[B.23-34] OPG Procedure, N-PROC-MA-0088 R003, *Buried Piping Program Requirements*, April 2015.

[B.23-35] Pickering Unit 018 Nuclear Operating Procedure, P-OP-71450-0001 R004, *PNGS Semi-Annual Yard Fire Hydrant Inspection/Flow Test and Annual Hydrant Isolating Valve Maintenance*, January 2014.

B.24 CNSC REGDOC-2.5.2 (2014), "Design of Reactor Facilities: Nuclear Power Plants"

B.24.1 Background

The following paraphrase from the purpose and scope of CNSC REGDOC-2.5.2 (2014) [B.24-1] provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *REGDOC-2.5.2 sets out the requirements of the Canadian Nuclear Safety Commission (CNSC) for the design of new water-cooled nuclear power plants (NPPs, or plants). It establishes a set of comprehensive design requirements and guidance that are risk-informed and align with accepted national and international codes and practices.*
>
> *REGDOC-2.5.2 deals with a wide variety of topics related to the design of new NPPs. To the extent practicable, this document is technology-neutral with respect to water-cooled reactors, and includes requirements and guidance for:*
>
> - *establishing the safety goals and objectives for the design*
> - *utilizing safety principles in the design*
> - *applying safety management principles*
> - *designing structures, systems and components (SSCs)*
> - *interfacing engineering aspects, plant features and facility layout*
> - *integrating safety assessments into the design process*
>
> *To a large degree, REGDOC-2.5.2 represents the CNSC's adoption of the principles set forth in the International Atomic Energy Agency (IAEA) document SSR-2/1, Safety of Nuclear Power Plants: Design, and the adaptation of those principles to align with Canadian practices.*
>
> *It is recognized that specific technologies may use alternative approaches. If a design other than a water-cooled reactor is to be considered for licensing in Canada, the design is subject to the safety objectives, high-level safety concepts and safety management requirements associated with this regulatory document. However, the CNSC's review of such a design will be undertaken on a case-by-case basis.*
>
> *Conventional industrial safety is addressed only from a high-level perspective, with a focus on design requirements that are related to nuclear safety.*

REGDOC-2.5.2 (R2014) is directly relevant to Safety Factors 1 (Plant Design), 5 (Deterministic Safety Analysis), 6 (Probabilistic Safety Assessment) and 7 (Hazard Analysis).

Compliance with REGDOC-2.5.2 is not currently a licence requirement Pickering NGS (in accordance with PROL 48.02/2018) [B.24-2] per the R04 Pickering Licence Conditions Handbook (LCH) [B.24-3].

CNSC REGDOC-2.5.2 is a first edition (version 1) and supersedes RD-337, Design of New Nuclear Power Plants [B.24-4], and consolidates the updated requirements and guidance related to the design of NPPs set out in draft RD-337 version 2, Design of Nuclear Power Plants [B.24-5], and draft GD-337, Guidance for the Design of New Nuclear Power Plants [B.24-6]. The following details on the changes presented by REGDOC-2.5.2 are obtained from the CNSC publication notice [B.24-7]:

> *The amendments to RD-337 included in this REGDOC ensure alignment with current national and international codes and practices, such* [as] *the principles set forth in the IAEA document SSR-2/1, Safety of Nuclear Power Plants: Design, with adoption of these principles to Canadian practices. Key changes adopted from the IAEA document include the added requirement for the design to explicitly consider the construction phase and expansion of the plant's fundamental safety functions to include cooling of all fuel (not just the core).*
>
> *REGDOC-2.5.2 implements recommendations from the CNSC Fukushima Task Force Report* [B.24-8] *including improved requirements for spent fuel storage, new requirements for mobile equipment, and more comprehensive coverage for design extension conditions.*

As identified in Reference [B.24-9], the Pickering PSR2 review of CNSC REGDOC-2.5.2 (2014) is an Incremental Review. PSR2 Incremental Review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.
- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.24.2 Compliance Assessment for Pickering PSR2

As noted above, CNSC REGDOC-2.5.2 supersedes RD-337. In general, REGDOC-2.5.2 builds on the previous requirements in RD-337 while also introducing additional guidance for new plants. For the most part, clause numbering in REGDOC-2.5.2 is the same as that in RD-337, with any new requirements being assigned new sub-clause numbers. Both RD-337 and REGDOC-2.5.2 are aligned with the high level requirements in IAEA NS-R-1 and its superseding document, IAEA SSR-2/1 R0. For these reasons, PSR1 reviews for both RD-337 (Darlington) and IAEA NS-R-1 (Pickering B) are largely applicable and relevant to the PSR2 REGDOC-2.5.2 review. However, there are no directly applicable code reviews from Pickering A Return to Service.

The approach to this compliance assessment in Section B.24.2.1 is to review the results from the Pickering B PSR1 clause-by-clause review conducted against IAEA NS-R-1 and assess their impact on Pickering 1,4 and PSR2. The results from the Darlington PSR1 clause-by-clause review conducted against RD-337 are then assessed for potential impact on Pickering 5-8 and Pickering 1,4 for PSR2.

Section B.24.2.2 provides a review of Pickering against REGDOC-2.5.2. Gaps that were identified in the Darlington PSR1 RD-337 review are also identified and applicability to Pickering PSR2 addressed. Although, there are significant programmatic and high level design similarities between the Darlington and Pickering plants, there are enough design differences that Darlington conclusions cannot be readily applied to Pickering. For this reason, Section B.24.2.2 additionally performs a high level review of all the REGDOC-2.5.2 clauses with respect to their impact on Pickering PSR2.

There is significant overlap between requirements in REGDOC-2.5.2 and several other CNSC and CSA standards that are being assessed as part of PSR2. For any REGDOC-2.5.2 gaps against clauses that have the same requirements as those in another code review, the corresponding gap(s) from the other review will be identified rather than initiating a REGDOC-2.5.2 gap. If there are similar clauses, but REGDOC-2.5.2 introduces requirements beyond those in the other code reviews for which there is a Gap, a new REGDOC-2.5.2 gap is identified.

B.24.2.1 Application of PSR1 Reviews

The versions of REGDOC-2.5.2 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

<u>Pickering Units 5-8</u>

REGDOC-2.5.2 did not exist when the Pickering B Integrated Safety Review (ISR) was conducted. Pickering B conducted a compliance review against IAEA NS-R-1 as part of the Plant Design Safety Factor report [B.24-10]. The complete set of NS-R-1 Discrepancies and Acceptable Deviations are documented in the final ISR report [B.24-11] in Appendices C and D respectively. The Discrepancies have been extracted and summarized in the table below.

Pickering B ISR IAEA NS-R-1 Discrepancies

NS-R-1 Clause	Pickering B ISR Gap #	Topic	Comments
2.10 5.31 6.54 6.57 6.59 6.61 6.63	1-428 1-430 1-431 1-432 1-433 1-434 1-435	Severe Accidents	Addressed in Gap 6-393 below

NS-R-1 Clause	Pickering B ISR Gap #	Topic	Comments
6.69 6.42 A-3	1-436 1-458 1-464		
6.37 2.10 5.31 6.42 6.54 6.57 6.59 6.61 6.63 6.69	6-393	Severe Accident (SA) Management – Station specific program for Pickering B is being developed	Addressed in PSR1 Safety Analysis Safety Factor review below and in Section B.24.2.2.
5.12	1-429	Fire-Fighting Systems	Addressed in Gap 1-643 below
	1-643	A design review of CSA N293-07 has not been performed.	PSR2 has conducted a code review against N293-12.
5.14 5.17	1-440 1-440	Probabilistic Safety Assessment (PSA) External Events	Addressed in Gap 6-391 below
5.73	1-468	PSA Approach and Methodology	Addressed in Gap 6-391 below
5.73	6-391	Risk from External Events	Addressed in PSR1 Safety Analysis Safety Factor review below and in Section B.24.2.2.
5.24	1-441	Design Limits related to Defense in Depth	Addressed in PSR2 Safety Factor 1 Report for Plant Design and in Section B.24.2.2.
5.70	1-456 5-388 5-386	Deterministic Safety Analysis Modelling - Legacy Safety Analysis Codes, Analytical Methods	Gap 5-388 and 5-386 are addressed in PSR1 Safety Analysis Safety Factor review below and in Section B.24.2.2.
5.71	1-482	Deterministic Safety Analysis approach	Captured in 5-388 above
6.46	1-483	Containment Strength	Captured in 5-388 and 6-393
6.66	1-484	Containment – control and clean-up of atmosphere	Captured in 5-388 and 6-393
5.24, C.2.0	5-373	Ability to shutdown for Anticipated Operational Occurrences (AOOs)	Addressed in PSR1 Safety Analysis Safety Factor review below and in Section B.24.2.2.

NS-R-1 Clause	Pickering B ISR Gap #	Topic	Comments
5.31	6-394	Level 2 Probabilistic Risk Assessment (PRA)	Addressed in 6-393 above
6.54 6.57 6.59	6-398	Capability of Penetrations for and structures for SAs	Addressed in 6-393 above
6.63	6-400	Containment heat removal following a SA	Addressed in 6-393 above

In addition to these Discrepancies, Appendix D – Acceptable Deviations (ADs) in [B.24-11] of the final ISR report was reviewed. There were twelve Acceptable Deviations identified against IAEA NS-R-1 (Clauses 1.7, 3.8, 5.31, 5.34, 5.35, 5.37, 5.38, 5.41, 6.1, 6.64, 6.67, and 6.9). These ADs and their dispositions have been reviewed for applicability to Pickering Units 1,4. In each case the disposition provided for Pickering Units 5-8 is also applicable to Pickering Units 1,4. Therefore, there are no PSR2 gaps resulting from the Pickering B ISR NS-R-1 ADs.

The results of the NS-R-1 review from reference [B.24-10] relating to the Plant Design Safety Factor review are summarized below:

> *It was determined that most of the clauses of the IAEA Safety Standard NS-R-1 fall under the "Direct Compliance" or "Acceptable Deviation" categories. Nine clauses have been categorized as "Discrepancy" all of which fall under the severe accident analysis subject area. All nine clauses were identified as "Discrepancy" in the "Pickering NGS B - Integrated Safety Review – Safety Analysis Review" (Reference 8). This review did not identify any additional clauses categorized as "Discrepancy".*

As noted in the table above, the Safety Analysis Safety Factor review [B.24-12] highlighted the following gaps (in *italics*), which are discussed further in the pages that follow:

> ***Hydrogen behaviour in containment and mitigation***
>
> *As per Generic Action Item 88G02, there is debate regarding control of long-term hydrogen due to radiolysis and corrosion. Short-term hydrogen control is provided by the PLHIS (Post LOCA [Loss of Coolant Accident] Hydrogen Ignition System). This issue is considered to be an Acceptable Deviation since PLHIS has been environmentally qualified for a mission time of 30 days, and would therefore likely be available for hydrogen mitigation in the short term following a LOCA+LOECI [Loss of Emergency Coolant Injection]. This issue is also identified as a gap against requirements from NS-R-1 (Clause 6.64) and CNSC document R-7 (Clause 3.10.2).*

This gap concerning long-term control of hydrogen has been addressed for Pickering with the installation of Passive Autocatalytic Recombiners (PARs) and completion of the Fukushima Action Items [B.24-13]. Hence, it is not a PSR2 gap.

Qualification of safety analysis codes used in the Safety Report

Many legacy codes have not undergone complete verification and validation and these may need to be used to support safety assessments for life extension. Resolution of this issue may require gap assessments to be done to show that analysis results remain conservative. This issue is also identified as an Acceptable Deviation against requirements from NS-R-1 (Clause 5.70).

This issue is considered to be a gap for PSR2. However, this issue was included within the scope of an issue that was identified as a gap against REGDOC-2.4.1 "Deterministic Safety Analysis" [B.24-14]. Therefore, the above issue related to safety analysis codes is not an incremental PSR2 gap for REGDOC-2.5.2.

Anticipated Operational Occurrences

The concept of anticipated operational occurrences (AOOs) is relatively new in the Canadian licensing context for Safety Analysis… The more formal establishment of AOOs in the licensing framework appears in the draft CNSC Regulatory Document S-310.

This issue is considered to be a gap for PSR2. However, this issue is already included under a Deterministic Safety Analysis REGDOC-2.4.1 [B.24-14] methodology gap and is, therefore, not an incremental PSR2 gap for REGDOC-2.5.2.

Probabilistic Safety Assessment

As a result of the limitations in scope of the existing PRA, deficiencies have been identified relating to development and use of Probabilistic Risk Assessments. In terms of modern codes and standards, OPG does not currently meet the intent of several IAEA requirements or clauses in S-294 (e.g., the PBRA [Pickering B Risk Assessment] currently does not include seismic and tornado events, or fire, as initiating events). However, S-294 allows assessment of events by other methods. OPG has addressed these events via hazard assessments and design (i.e., fire and seismic - refer to Appendix C, Section C.3.0).

It can be concluded that OPG and the Pickering B PRA are reasonably well aligned with the standards and requirements for modern PRAs.

While the Level-1 PBRA analysis is fully acceptable and usable, OPG is currently pursuing improvements to the Level 2 analysis.

This issue was addressed by OPG via the S-294 compliance program. However, a PSR2 Gap has been identified against REGDOC-2.4.2 Probabilistic Safety Analysis [B.24-15]. In addition, there are new requirements in REGDOC-2.5.2 which are discussed and addressed in Section B.24.2.2 below.

Application of Single Failure Criterion

The 'single failure' criteria as defined by the IAEA (NS-R-1 & NS-G-1.2) is different than that established by the CNSC in Regulatory Documents R-7, R-8 and R-9. NS-R-1 states the following:

> *"5.37. Compliance with the criterion shall be considered to have been achieved when each safety group has been shown to perform its safety function when the above analyses are applied, under the following conditions:*
>
> *(1) any potentially harmful consequences of the [Postulated Initiating Event (PIE)] for the safety group are assumed to occur; and*
>
> *(2) the worst permissible configuration of safety systems performing the necessary safety function is assumed, with account taken of maintenance, testing, inspection and repair, and allowable equipment outage times.*

Clearly stated acceptance criteria, objectives and safety goals are demonstrated to be met in the current Safety Report and the PBRA. From a design perspective, the single failure criterion is reflected in the CNSC Requirements for Special Safety Systems (R7, R8 & R9). These requirements do not specify that this criterion be applied at the limit of operation. For Pickering B, the approach has been to respond to address instances of no redundancy in systems important to safety with high priority per the impairments procedures. If the limit of operation is exceeded (an additional failure) and the safety function is not available, then immediate repair or controlled shut down is required.

The application of Single Failure Criterion was identified as a gap (Issue 1-443) in the NS-R-1 reviews and classified as an Acceptable Deviation. The CNSC requested this Acceptable Deviation be reclassified as a Discrepancy in [B.24-11] and OPG provided additional risk based rationale to support maintaining this gap as an Acceptable Deviation:

OPG provides the following additional information to support the categorization of this item as Acceptable Deviation. OPG recently initiated a new project, the Probabilistic Safety Assessment Project, in order to meet the operating license conditions to perform for OPG stations a level 2 Probabilistic Safety Assessment in accordance with the CNSC Standard S-294, Probabilistic Safety Assessment (PSA) for Nuclear Power Plants. The Pickering NGS-B Risk Assessment will be updated. Following the established practice, the methodology and the specific "risk assessment tools" to be used for the assessment will be discussed and agreed upon with the CNSC staff, prior to performing the actual assessment.

The CNSC continued to maintain that this issue was a PSR1 discrepancy. Subsequently, this issue was transferred to the Pickering B Continued Operations Plan [B.24-16], where it was closed on the basis that the Level 1 and Level 2 PSA demonstrated that the existing plant design satisfied all safety requirements. The applicability of the gap resolution and subsequent disposition is considered to be equally applicable to Pickering Units 1,4, where the PSAs have similarly been updated. Hence this issue is not a gap for PSR2.

In summary, there were four general areas where there were PSR1 gaps relating to Pickering B PSR1 IAEA NS-R-1 review. These were:

- Severe Accident Management and Assessment, including hydrogen mitigation and instrumentation for SA management – this gap is now closed.
- Fire-fighting and CSA N293 compliance – this gap is now closed.
- Deterministic Safety Analysis methodology (including treatment of AOOs, Code Qualification, Treatment of Human Initiating Events) - this is a PSR2 Gap and addressed in PSR2 REGDOC-2.4.1 review.
- Probabilistic Safety Assessment (including external events) - this is a PSR2 Gap and addressed in PSR2 REGDOC-2.4.2 review.

In addition, gaps relating to the following two clauses remained open or the CNSC requested re-categorization as discrepancies per [B.24-11], Appendix D:

- Validation of field actions for Design Basis Accidents (DBAs) (Clause 5.30).
  This issue has been addressed with the implementation of CNSC G-323 [B.24-17] at Pickering. Reference [B.24-18] identifies the assessments conducted to validate the credited field actions. Reference [B.24-19] documents the CNSC's acceptance of Pickering's Minimum Shift Complement and closes the related CNSC Action Items.

  OPG has conducted this analysis in accordance with G323 - Ensuring the Presence of Sufficient Qualified Staff at Class I Nuclear Facilities - Minimum Staff Complement and G-278 Verification and Validation. The analysis methodology, analysis and…

- Auxiliary Services (Clause 5.41)
  This issue has been addressed in Appendix D of [B.24-11] and maintained as an AD. In addition, the PSR2 review of CSA N290.0 has a similar clause relating to Special Safety Systems. The assessment for the clause in CSA N290.0 [B.24-20] is applicable to other safety-related systems. Hence, this is not a gap for PSR2.

The above gaps are resolved and do not represent PSR2 gaps.

It is concluded that all Discrepancies identified as part of the IAEA NS-R-1 review for the Pickering B ISR are captured as part of other PSR2 Safety Factor and Code Reviews, or were addressed as part of the ISR and Integrated Implementation Plan development and follow up activities. Due to their programmatic nature, all of the Pickering 5-8 issues and their resolutions identified above are also applicable to Pickering 1,4.

<u>Pickering Units 1,4</u>

REGDOC-2.5.2 did not exist when the Pickering A return to Service ISR was conducted and there were no assessments conducted against its predecessors (CNSC RD-337, S-337) or IAEA basis documents (e.g., NS-R-1).

Darlington NGS

REGDOC-2.5.2 did not exist when the Darlington ISR was conducted. Darlington conducted a review against CNSC RD-337 [B.24-21] in August 2011 and a subsequent review against IAEA NS-R-1 [B.24-22] in September 2011. As part of the code refresh activities, a review against IAEA SSR-2/1 was performed [B.24-23] in February 2014.

As noted in B.24.1, REGDOC-2.5.2 was an update of RD-337 to reflect Operating Experience (OPEX) from the Fukushima event and to align with IAEA SSR-2/1. However, per [B.24-23], the IAEA SSR-2/1 review did not identify any gaps incremental to those identified against IAEA NS-R-1. Hence, a combination of the review results from the RD-337 and IAEA NS-R-1 reviews provides a comprehensive basis for evaluating the relative compliance with REGDOC-2.5.2.

The results of the RD-337 review for Darlington are summarized below:

> *This review finds that the Darlington NGS-A design is compliant with the majority of the requirements of RD-337. Out of a total of 133 clauses of RD-337 which were assessed for compliance, only 37 gaps were found. The numbers of gaps, in brackets, are in the areas of:*
>
> - *Plant Design Provisions (10)*
> - *Severe Accidents / Beyond Design Basis Accidents (8)*
> - *Security (6)*
> - *Deterministic Safety Analysis (5)*
> - *Fire (4)*
> - *Probabilistic Safety Analysis (2)*
> - *Decommissioning (1)*
> - *Periodic Inspection Programs (1)*

The following identifies the issues from the Darlington ISR that required resolution and that were based on gaps resulting from the RD-337 code review [B.24-21] compiled in [B.24-24], and their applicability to Pickering PSR2.

D011 Changes to In-service Examination and Testing Requirements for Concrete Containment Structures

> *Clause 7.15.2 of CNSC RD-337 provides a list of requirements for design to facilitate inspection of civil structures. Gap 01470 states that DNGS is not fully compliant with the requirements of CAN/CSA-N287.7-08 based on the Gaps identified in NK38-REP-03680-10061.*

Although this issue is specific to containment structures, it refers to a related Issue D300 which is applicable to non-containment safety-related structures:

> *The gaps included in this Issue concern the requirements to conduct regular in-service examinations of safety-related structures for evidence of degradation. The structures covered include i) those that support, house or protect nuclear safety systems, ii) components of structures required for the safe operation or reactor shutdown, and iii) facilities for storage of irradiated fuel and other radioactive waste material. The Code review found that DNGS does not have an active program for in-service examination of safety-related structures, other than the periodic inspection programs for the concrete containment structures covered in the CAN/CSA-N287 series, or for the pressure-retaining systems and components covered in CAN/CSA-N285.0.*

Specific to inspection requirements for non-containment safety-related structures, the above gap also identifies issues with these structures. This is addressed in further detail under REGDOC-2.5.2 Clause 7.15.2 in Section B.24.2.2 below.

D013 Long Term Control of Hydrogen in Containment

> *Gap 01480 requires that the design of the nuclear power plant shall provide systems to control the release of fission products, hydrogen, oxygen, and other substances into the reactor containment as necessary, to:*
>
> *1. Reduce the amount of fission products that might be released to the environment during an accident; and,*
>
> *2. Prevent deflagration or explosion that could put the integrity or leak tightness of the containment envelope at risk.*
>
> *Gap 01481 describes the relationship between coatings inside containment and post-accident conditions inside containment (i.e., the production of hydrogen from metal corrosion). The gap states that no evidence exists to confirm that post-accident conditions inside containment were considered when choosing the type and quantity of coating for civil structures and steel lined reactor structures.*

These gaps were identified against Clauses 8.6.8 and 8.6.10 of CNSC RD-337. As noted above in the Pickering ISR review, treatment of Hydrogen does not represent a gap for PSR2. The gap relating to coatings is addressed under Issue D072 below.

D025 Darlington Risk Assessment (DARA) Scope and Completion

> *This ISR Issue consists of a total of ten (10) ISR Gaps from three (3) Code Review Reports.*
>
> - *Seven (7) Gaps are from NK38-REP-03680-10007, "Review of CNSC S-294 (April 2005) Probabilistic Safety Assessment (PSA) For Nuclear Power Plants For Darlington Integrated Safety Review".*

- *One (1) Gap is from NK38-REP-03680-10088 "Review of IAEA GS-R-2 (November 2002) Preparedness and Response for a Nuclear or Radiological Emergency for Darlington Integrated Safety Review".*

- *Two (2) Gaps are from NK38-REP-03680-10109, "Review of CNSC RD-337 (September 2008) Design of New Nuclear Power Plants for Darlington Integrated Safety Review"*

D025 Gaps from NK38-REP-03680-10109, "Review of CNSC RD-337 (September 2008) Design of New Nuclear Power Plants for Darlington Integrated Safety Review" Gap 01461 was declared against Clause 7.6.1 of CNSC RD-337 which requires consideration of Common Cause Failures. The Code Review Report, NK38-REP-03680-10109, found that a formal Common Cause Failures evaluation using probabilistic methods has not been performed.

Gap 01488 was declared against Clause 9.5 of CNSC RD-337 which requires a Probabilistic Safety Assessment in accordance with CNSC S-294. The Code Review Report, NK38-REP-03680-10109, found that a Darlington Risk Assessment fully compliant with CNSC S-294 has not yet been issued.

The gaps identified in this Issue have subsequently been addressed in the DARA update [B.24-25]. Similar to Darlington, the Pickering PSAs have been updated to address the same gaps identified in this issue. Hence, this issue is not a PSR2 gap for Pickering. Note that a separate review of REGDOC-2.4.2 has been performed as part of PSR2.

D027 Severe Accident and Beyond Design Basis Accident (BDBA) Analysis/ SAMG [Severe Accident Management Guidelines]

This ISR issue consists of a total of 17 ISR Gaps from two (2) Code Review Reports.

- *Two (2) gaps are from NK38-REP-03680-10102, "Review of CNSC RD-310 (February 2008) Safety Analysis for Nuclear Power Plants".*

- *Fifteen (15) Gaps are from NK38-REP-03680-10021, "Review of IAEA NS-G-1.2 (January 2002) Safety Assessment and Verification for Nuclear Power Plants For Darlington Integrated Safety Review".*

Although these gaps did not result from the RD-337 code review, the subject of accident analysis in Issue D027 was also identified in Clause 7.3.4 of the RD-337 review [B.24-21], and hence is included here. This issue is addressed in further detail specific to Pickering under the REGDOC-2.5.2 review in Section B.24.2.2 below.

D028 Systematic Analysis of Anticipated Operational Occurrences (AOOs)

Gap 01457 was declared against Clause 7.3.2 of CNSC RD-337 which contains various requirements for Anticipated Operational Occurrences, including: "The design includes provisions such that releases to the public following an Anticipated Operational Occurrence do not exceed the dose acceptance criteria". The Code Review Report, *NK38-REP-03680-10109, found that Anticipated Operational Occurrences have not been analysed.*

REGDOC-2.5.2 has the same requirement in Clause 7.3.2 and this issue is applicable to Pickering, however, it is being addressed under PSR2 REGDOC-2.4.1 (Deterministic Safety Analysis) review.

D030 Identification and Classification of Events per CNSC RD-310

This ISR issue consists of a total of nine (9) ISR Gaps from two (2) Code Review Reports:

- Five (5) ISR Gaps are from NK38-REP-03680-10102, "Review of CNSC RD-310 (February 2008) Safety Analysis for Nuclear Power Plants"
- Four (4) ISR Gaps are from NK38-REP-03680-10109, "Review of CNSC RD-337 (September 2008) Design of New Nuclear Power Plants for Darlington Integrated Safety Review"

Gap 01452 was declared against Clause 4.2.1 of CNSC RD-337 which requires that the committed whole-body dose for average members of the critical groups who are most at risk, at or beyond the site boundary is calculated in the deterministic safety analysis for a period of 30 days after the analyzed event. This dose is less than or equal to the dose acceptance criteria of 0.5 millisievert for any Anticipated Operational Occurrence or 20 millisieverts for any Design Basis Accidents. The Code Review Report, NK38-REP-03680-10109, found that activities designed to achieve compliance with CNSC RD-310 and map CNSC C-006 dose limits to new dose limits are not yet complete.

Gap 01454 was declared against Clause 6.4 of CNSC RD-337 which requires that the design includes provisions for the prevention and mitigation of radiation exposures resulting from Design Basis Accidents and Beyond Design Basis Accidents. The design also ensures that potential radiation doses to the public from Anticipated Operational Occurrences and Design Basis Accidents do not exceed dose acceptance criteria provided in subsection 4.2.1. The calculated overall risk to the public from all plant states meets the safety goals in subsection 4.2.2. The Code Review Report, NK38-REP-03680-10109, found that activities designed to achieve compliance with CNSC RD-310 and map CNSC C-006 dose limits to new dose limits are not yet complete.

Gap 01460 was declared against Clause 7.4.3 of CNSC RD-337 which requires that combinations of randomly occurring individual events that could credibly lead to Anticipated Operational Occurrences, Design Basis Accidents, or Beyond Design Basis Accidents are considered in the design. Such combinations are identified early in the design phase, and are confirmed using a systematic approach. The Code Review Report, NK38-REP-03680-10109, found that for certain low power states, normal operation, and for events from operating experience, the DNGS safety analysis may not comply and referred to Clause 5.2.1 of CNSC RD-310.

Gap 01487 was declared against Clause 9.4 of CNSC RD-337 which describes the purpose of deterministic safety analysis and references CNSC RD-310. The Code

Review Report, NK38REP-03680-10109, found that, based on the existence of Gaps with CNSC RD-310, such as those found to exist in the DNGS Design guides for events to be analysed, acceptance criteria, methods, and assumptions, this clause was also a Gap.

This issue of identification and classification of events from the Darlington ISR is applicable to Pickering NGS however it will be addressed as part of the review PSR2 CNSC REGDOC-2.4.1 (Deterministic Safety Analysis) review.

D040 Human Factors Design

This issue relates to Clauses 7.21 (Gap 1471) and 8.10.1 (Gap 1484) for Human Factors relating to the original Darlington Design. Additionally, this issue addresses gaps identified against IAEA NS-R-1 and SSR-2/1. All issues were closed in the Issue Resolution form [B.24-26]. This issue is addressed in further detail specific to Pickering under Clause 7.21 in Section B.24.2.2 below.

D260 Human Factors - Annunciation Improvements

This issue is documented in [B.24-27] and a summary of the issue is provided in the final ISR report [B.24-24]. The RD-337 clauses relating to this issue are 7.21 and 8.10.1 [B.24-21] and these are the same clauses in REGDOC-2.5.2. The Darlington gaps and issue relating to annunciations were closed based on the assessments in and referred to in [B.24-27]. Because of differences between the Pickering and Darlington annunciation system designs, the annunciation issue is not directly applicable to Pickering. This issue is addressed in further detail specific to Pickering under Clauses 7.21 and 8.10.1 in Section B.24.2.2 below.

D063 Seismic Qualification – General

Gap 01469 was declared against Clause 7.13.1 of CNSC RD-337 which requires that seismic fragility levels should be evaluated for Structures, Systems and Components important to safety, by analysis or, where possible, by testing. The Code Review Report, NK38-REP-03680-10109, found that seismic fragility assessments / Probabilistic Risk Assessment are currently in progress and therefore DNGS is currently not considered to be in compliance with Clause 7.13.1.

The Seismic PSAs have been completed for Pickering. However, as detailed in Section B.24.2.2 for Clause 7.13.1 of REGDOC-2.5.2 below, it has not been confirmed that the required margin between the Design Basis Earthquake (DBE) and Review Level Earthquake (RLE) meets the requirement for a new plant.

D068 Severe Accident and Beyond Design Basis Accident (BDBA) Design/SAMG

Grouped with D143 [B.24-28], Severe Accident and Beyond Design Basis Accident (BDBA) Program/SAMG, the gaps in these issues are as follows:

Gap 01456 was declared against Clause 7.2 of CNSC RD-337 which requires that the design authority establishes the plant design envelope, which comprises the design basis and complementary design features. Complementary design features address the performance of the plant in Beyond Design Basis Accidents, including selected *severe accidents. The Code Review Report, NK38-REP-03680-10109, found that design features to cater to severe accidents were not part of the original design.*

Gap 01458 was declared against Clause 7.3.4 of CNSC RD-337 which contains various specific design requirements for Beyond Design Basis Accidents and Severe Accidents. The Code Review Report, NK38-REP-03680-10109, declared a Gap because a Severe Accident Management Guidelines program has not been fully implemented at DNGS.

Gap 01463 was declared against Clause 7.8 of CNSC RD-337 which requires that equipment credited to operate during Beyond Design Basis Accident and severe accident states is assessed for its capacity to perform its intended function under the expected environmental conditions. A justifiable extrapolation of equipment behaviour may be used to provide assurance of operability, and is typically based on design specifications, environmental qualification testing, or other considerations. The Code Review Report, NK38-REP-0368010109, found that OPG has provisions in place to qualify equipment, however, a Gap was declared because a Severe Accident Management Guidelines program has not been fully implemented at DNGS.

Gap 01464 was declared against Clause 7.9.1 of CNSC RD-337 which requires that the design includes provision of instrumentation to monitor plant variables and systems over the respective ranges for normal operation, Anticipated Operational Occurrences, Design Basis Accidents, and Beyond Design Basis Accidents, in order to ensure that adequate information can be obtained on plant status. The Code Review Report, NK38-REP-03680-10109, found that DNGS complies for Anticipated Operational Occurrences and Design Basis Accidents, however, a Gap was declared because a Severe Accident Management Guidelines program has not been fully implemented at DNGS.

Gap 01465 was declared against Clause 7.9.3 of CNSC RD-337 which requires that instrumentation and recording equipment is such that essential information is available to support plant procedures during and following accidents by facilitating decisions in accident management. The Code Review Report, NK38-REP-03680-10109, found that it is expected that the existing Post Accident Radiation Monitoring instrumentation will provide substantial monitoring capability following any severe accident, thereby enabling suitable emergency response actions to be taken. However, a Gap was declared because a Severe Accident Management Guidelines program has not been fully implemented at DNGS.

Gap 01479 was declared against Clause 8.6.8 of CNSC RD-337 which requires consideration of pressure differentials and control of hydrogen in the design of containment internal structures. The Code Review Report, NK38-REP-03680-10109, found that these issues have been considered to some extent, however, a Gap was declared because a Severe Accident Management Guidelines program has not been fully implemented at DNGS.

Gap 01482 was declared against Clause 8.6.12 of CNSC RD-337 which requires consideration of the ability of the containment system to withstand loads associated with Severe Accidents as part of the design. The Code Review Report, NK38-REP-

03680-10109, found that the current design meets some of these requirements, however, a Gap was declared because a Severe Accident Management Guidelines program has not been fully implemented at DNGS.

As discussed for the Pickering B ISR IAEA NS-R-1 review above, all issues relating to SA and SAMG program have been closed per [B.24-13]. Also, refer to the PSR2 code review for CNSC REGDOC-2.3.2, which concludes that all gaps relating to the above clauses have been addressed.

D070 Consideration of Decommissioning During Design Phase

Clause 5.68 of IAEA NS-R-1, "Safety of Nuclear Power Plants: Design" requires that at the design stage, special consideration shall be given to the incorporation of features that will facilitate decommissioning and dismantling of the plant. Special attention should be paid to the choice of materials to minimize radioactive waste, access capabilities that may be necessary, and radioactive storage facilities. Gap 00781 identified that during original design phase of DNGS, no special consideration was given to decommissioning of the nuclear power plant.

Clause 7.24 of CNSC RD-337, "Design of New Nuclear Power Plants" mirrors the above requirement. It states that the future decommissioning and dismantling activities shall be taken into account, such that:

1. Materials are selected for the construction and fabrication of the plant components and structures with the intent of minimizing quantities of radioactive waste and assisting decontamination.

2. Plant layout is designed to facilitate access for decommissioning or dismantling activities; and 3. Consideration is given to the future potential requirements for storage of radioactive waste generated as a result of new facilities being built, or existing facilities being expanded.

Gap 01476 identified that during the design phase of DNGS no special consideration was given to the ultimate decommissioning and dismantling of the plant.

This issue was closed as an Acceptable Deviation in the final ISR report [B.24-24], and given there is a low safety impact, this is not a PSR2 gap.

D071 Coatings and Coverings

The gaps in this Issue are as follows:

Clause 6.67 in IAEA NS-R-1 requires the coverings and coatings for components and structures within the containment system shall be carefully selected, and their methods of application specified, to ensure fulfillment of their safety functions and to minimize interference with other safety functions in the event of deterioration of coverings and coatings. Gap 00795 identified that the OPG design specification does not meet all of these requirements.

Clause 8.6.11 of CNSC RD-337 requires that the coverings and coatings for components and structures within the containment system shall be carefully selected, and their methods of application specified, to ensure fulfillment of their safety functions and to minimize post-accident interference with other safety functions or accident mitigation systems. In addition, the choice of materials inside containment should take into account the impact of post-accident containment conditions that may affect containment performance and integrity and fission product release.

Gap 01481 states that there is no evidence that documents L-970, "Darlington GS A Sealing and Painting Specifications" and NK38-DM-21200 R001 "Reactor Building Internal Structure" considered the impact of post-accident containment conditions.

This issue has been closed as an Acceptable Deviation in the final ISR report [B.24-24] with the justification provided therein, noting that no further action is required. In addition, this issue was classified as an Acceptable Deviation for the Pickering B ISR, with supporting rationale provided. The rationale provided was equally applicable to Pickering Units 1,4. Hence, this issue does not result in a PSR2 gap for Pickering.

D072 Single Failure Criterion

Gap 01462 was declared against Clause 7.6.2 of CNSC RD-337 which contains various high level requirements for the single failure criterion. The Code Review Report, NK38-REP-03680-10109 [B.24-21], found that there are some instances of single failure vulnerabilities in systems other than the Special Safety Systems which have not been addressed.

This issue was classified as an Acceptable Deviation for the Pickering B IAEA NS-R-1 review as identified above. No incremental issues have been identified for the Darlington code reviews that would invalidate the rational for the Acceptable Deviation classification at Pickering B. The rationale provided for the Acceptable Deviation classification is also applicable to Pickering Units 1,4, hence, this does not represent a PSR2 gap.

D143 Severe Accident and Beyond Design Basis Accident (BDBA) Program/ SAMG

This issue is included with issue D068 above. No PSR2 gap has been identified.

D279 Cabling between Main Control Room and Secondary Control Area

This ISR Issue is made up of 2 ISR Gaps from NK38-REP-03680-10109, "Review of RD-337 Design of New Nuclear Power Plants for Darlington Integrated Safety Review".

Clause 8.10.1 of CNSC RD-337, "Design of New Nuclear Power Plants", requires that cabling for the instrumentation and control equipment in the Main Control Room is arranged such that a fire in the Secondary Control Room cannot disable the equipment in the Main Control Room. Gap # 01484 identifies that there are gaps with the requirements of CAN/CSA-N293-07, "Fire Protection for CANDU Nuclear Power Plants".

Clause 8.10.2 of CNSC RD-337 requires that cabling for the instrumentation and control equipment in the Secondary Control Room is arranged such that a fire in the Main Control Room cannot disable the equipment in the Secondary Control Room.

Gap # 01485 identifies that there are gaps with the requirements of CAN/CSA-N293-07.

Issue D279 [B.24-29] relating to RD-337 Clause 8.10.1 and 8.10.2 was subsequently moved to Issue D430 [B.24-30]. The issue at Darlington was due to a postulated fire event in the Common Secondary Control Area (CSCA) in which it was conservatively assumed that control of Group 2 systems, including Containment was lost. These gaps have subsequently been closed with the CNSC's acceptance of the Darlington Fire Hazard Assessment (FHA) and Fire Safe Shutdown Analysis (FSSA) as noted in [B.24-30].

At Pickering 5-8, the same fire event is not applicable as there is no CSCA and Group 2 systems are not used for process control during normal operation. Pickering 5-8 does have Unit Emergency Control Centers (UECC) in each Unit. In accordance with [B.24-31], Pickering 5-8 is designed such that all outputs from active safety systems to the control room are suitably buffered such that wires can be open or short circuited without disabling protective functions or causing common mode failures. Additionally, per the Pickering 5-8 FHA [B.24-32] and FSSA [B.24-33]), there is no impact on control of Containment (or any other safety systems) that prevents safe shutdown for a fire event in either the Main Control Room (MCR) or UECCs. Pickering 1,4 has remote monitoring and reactor trip capability in the Shutdown System Enhancement (SDSE) Instrument Rooms. These rooms are remote from the MCR and instrumentation and equipment is physically isolated from the MCR.

Therefore, neither of the gaps identified above are applicable to Pickering.

D276 Fire Protection

All gaps relating to fire protection were closed in the RD-337 review, noting that any issue will be addressed in CSA N293 and fire code reviews. The PSR2 CSA N293-12 [B.24-34] review is being addressed separately, and therefore is not repeated here.

<u>Additional RD-337 Gaps</u>

There were four additional gaps relating to CNSC RD-337 compliance that were either closed with a rationale or reclassified as acceptable deviations. These are:

- Clause 8.2.2 Accommodation of Primary Heat Transport (PHT) shrinkage for multi-unit shutdown (D277) [B.24-35]. This is not an issue for Pickering, as there are no credits for non-unit D_2O identified per the Pickering Units 1,4 and Pickering Units 5-8 Heat Transport System (HTS) Operational Safety Requirements (OSRs) [B.24-36] and [B.24-37]. Hence, this is not a PSR2 gap.

- Clause 8.3.3 Turbine Orientation (Issue D278) [B.24-38] was closed for Darlington noting that Turbine Failures were analyzed in the Safety Report and Dose Class limits were met. Turbine Failures at Pickering are addressed in the PSA Hazard Screening addressed in the PSR2 Hazard Analysis Safety Factor report
  The D278 issue resolution is also applicable to Pickering. Hence, this is not a PSR2 gap.

- Clause 8.8 and 8.112 Fire Water / Emergency Heat Removal System (EHRS) Interconnection (Issue D225) [B.24-39]. This is not an issue for Pickering. Pickering Units 5-8 has an Emergency Water Supply (EWS) supply to the HTS. Pickering Units 1,4 has a manual flowpath (33350-V478) that supplies to the HTS from firewater or Emergency Mitigating Equipment (EME) water. The gap was also related to fire coincident with LOCA relying on a common Emergency Service Water System. This is not an issue for Pickering where fire water is supplied by Diesel pumps (Pickering Units 1,4) or High Pressure Service Water (Pickering Units 5-8). Hence, this is not a PSR2 gap.

- Clause 8.11.2 relates to the Off-Gas System not being in-service (Issue D280) [B.24-40]. The D280 issue resolution applies equally to Pickering where the Off-Gas Management System is no longer in use [B.24-41]. Hence, this is not a PSR2 gap.

In summary, the above assessment has reviewed all Darlington PSR1 RD-337 gaps and issues and considered their applicability to Pickering PSR2. Additionally, all gaps and issues resulting from the Darlington PSR1 IAEA NS-R-1 review have been included in and addressed in the RD-337 issue discussions above.

B.24.2.2 Application of Post PSR1 Reviews

Per the preface of REGDOC-2.5.2, this document is primarily designed for new nuclear facilities,

> …*For existing facilities: The requirements contained in this document do not apply unless they have been included, in whole or in part, in the licensing basis.*

Per [B.24-42], OPG and the nuclear industry systematically reviewed the REGDOC-2.5.2 (draft) and provided comments. Many of the comments related to clarity and application. Notwithstanding the above, the approach for this review is to assume all clauses to be applicable to the existing plant and to assess compliance even if the clause may not actually be applicable or practicable for a mature nuclear power plant.

The table below details the additional incremental review of REGDOC-2.5.2 for Pickering. It identifies new additions or significant changes from RD-337 that have been introduced in REGDOC-2.5.2. In addition, clauses that were RD-337 gaps for Darlington and are considered applicable are included with the corresponding REGDOC-2.5.2 clause and the Gap is addressed in the context of Pickering PSR2.

REGDOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
1. Purpose	No requirements	n/a
2. Scope	No requirements	n/a
3. Relevant Legislation Clause identifies high level nuclear regulatory framework and establishes the Nuclear Safety and Control Act (NSCA) requirements.	A compliance assessment for Pickering PSR2 has been performed per [B.24-9] and demonstrates that Pickering is compliant.	**Compliant**

REGDOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
4. Safety Objectives and Concepts	Heading	n/a
4.1 General Nuclear Safety Objectives High level radiological protection, nuclear safety and environmental objectives.	No explicit requirements	n/a
4.2 Application of Technical Safety Objectives	Heading	n/a
4.2.1 Dose acceptance criteria *The committed whole-body dose for average members of the critical groups who are most at risk, at or beyond the site boundary, shall be calculated in the deterministic safety analysis for a period of 30 days after the analyzed event.* *This dose shall be less than or equal to the dose acceptance criteria of:* *1. 0.5 millisievert (mSv) for any AOO or* *2. 20 mSv for any DBA*	Clause 4.2.1 had a gap and Issue in the DSA review for Darlington RD-337 review [B.24-21]. OPG has a plan for the implementation of REGDOC-2.4.1 Deterministic Safety Analysis for Pickering. REGDOC-2.4.1 has the same requirements for AOO analysis and dose limits and OPG will address these as part of the compliance framework. However, for a new plant, demonstration that the requirements are met is mandatory. Therefore, this is a PSR2 gap relating to new plant requirements.	**PSR2 REGDOC-2.5.2 Gap #1** (Deterministic Safety Analysis) Safety Factor 5 (Deterministic Safety Analysis)
4.2.2 Safety goals … two qualitative safety goals have been established: *Individual members of the public* *Societal risks* *Core damage frequency* *10^{-5} per reactor year.* *Small release frequency* *a release to the environment of more than 10^{15} becquerels of iodine-131 shall be less than 10^{-5} per reactor year. A greater release may require temporary evacuation of the local population.* *Large release frequency* *a release to the environment of more than 10^{14} becquerels of cesium-137 shall be less than 10^{-6} per reactor year. A greater release may require long term relocation of the local population…*	This clause requires that all event frequencies should be summed and core damage frequency be $<10^{-5}$ yrs/yr. This is a limit as opposed to a goal. The OPG governance uses 10^{-4} yrs/yr as the Safety Goal [B.24-43]. Therefore, this is a PSR2 gap relating to new plant requirements.	**PSR2 REGDOC-2.5.2 Gap #2** (Probabilistic Safety Assessment) Safety Factor 6 (Probabilistic Safety Analysis)
4.2.3 Safety Analysis *To demonstrate achievement of the safety objectives, a comprehensive hazard analysis (HA), a deterministic safety analysis (DSA), and a*	These are high level objectives. Reviews of DSA, PSA and HA are being addressed as part of the Safety Factor reviews for PSR2 under	Addressed in other PSR2 Safety Factor and code reviews.

REGDOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
probabilistic safety assessment (PSA) are carried out. These analyses identify all sources of exposure, in order to evaluate potential radiation doses to workers at the plant and to the public, and to evaluate potential effects on the environment. The safety analyses examine plant performance for: *1. Normal operation;* *2. Anticipated operational occurrences;* *3. Design basis accidents; and* *4. Beyond design basis accidents (BDBAs), including event sequences that may lead to a severe accident.*	Safety Factors 5, 6 and 7 respectively. A PSR2 review of REGDOC-2.4.1, "Deterministic Safety Analysis" [B.24-14] is being performed. A PSR2 review of REGDOC 2.4.2, "Probabilistic Safety Assessment (PSA) for Nuclear Power Plants" [B.24-15] is being performed. Hence, these objectives are not addressed in this review.	PSR2 REGDOC-2.4.1 Gap PSR2 REGDOC-2.4.2 Gap
4.2.4 Accident Mitigation and Management *The design shall include provisions to limit radiation exposure in normal operation and AOOs to ALARA [As Low As Reasonably Achievable] levels, and to minimize the likelihood of an accident that could lead to the loss of normal control of the source of radiation. However, given that there is a remaining probability that an accident may occur, measures shall be taken to mitigate the radiological consequences of accidents….*	The high level objectives in this clause are explicitly addressed in a separate review of the requirements for accident management. This review has been performed as part of the PSR2 review of REGDOC-2.3.2, "Accident Management" [B.24-44].	**Compliant** Addressed in other PSR2 code reviews.
4.3 Safety concepts *This section details the high level objectives relating to Defence in depth and establishes the five levels of defence in depth and that they shall be independent to the extent practicable. Physical barriers between the layers of defence. Operational limits and conditions, to clearly establish the limits of the different levels and finally they enforce the Interface of safety with security and safeguards.*	Defence in depth is embodied in "Nuclear Management System", N-CHAR-AS-0002, [B.24-45]. Defense in Depth is partially addressed in PSR2 Plant Design Safety Factor 1 Report, under Review Task 4. The 'Procedures' Safety Factor 11 review addresses the operational procedures for the first two levels of Defense in depth. Operational limits and conditions are addressed in the PSR2 Safe Operating Envelope (SOE) review, CSA Standard N290.15-10 (R2015), "Requirements for the Safe Operating Envelope of Nuclear Power Plants" [B.24-46]. Levels 3, 4 and 5 of Defense in depth are addressed in PSR2 code reviews of REGDOC-2.3.2, "Accident Management" [B.24-44] and REGDOC-2.10.1, "Nuclear Emergency	**Compliant** Addressed in other PSR2 Safety Factor and code reviews.

REGDOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
	Preparedness and Response" [B.24-47]. In addition, an integrated assessment of 'defense-in-depth' will be done as part of the Global Assessment per [B.24-9]. Security and safeguards are not in the PSR2 scope.	
5. Safety Management in Design This section details the role of the Design Authority, design process management, quality assurance and use of Operating Experience and research.	The PSR2 review for Safety Factor 10 Report "Organization, Management System, and Safety Culture", addresses the concepts and requirements in this clause. Additionally, a separate PSR2 review of OPEX has been performed as part of PSR2 Safety Factor 9, "Use of Experience from Other Nuclear Power Plants and Research Findings". OPG Nuclear Program, N-PROG-MP-0007 [B.24-48], "Conduct of Engineering", implements a series of programs, standards, and procedures for performing engineering in a consistent manner across OPG Nuclear. The Design Authority roles and responsibilities reside with the Chief Nuclear Engineer (CNE), who prescribes (1) the overall requirements for the Conduct of Engineering program, (2) the scope, development, and implementation of Engineering programs, and (3) the manner in which design activities are performed. The CNE may delegate, within specified limits and controls, station specific Engineering and Design Authority responsibilities as detailed in, "Engineering and Design Authority" [B.24-49]. In addition, the overall design program is addressed in the review of CSA N286, "Management System Requirements for Nuclear Power Plants", [B.24-50], which has been performed as part of PSR2.	**Compliant** Addressed in other PSR2 Safety Factor and code reviews.

REDGOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
6. Safety Requirements	Heading	n/a
6.1 Application of Defence in Depth *The design of an NPP shall incorporate defence in depth. The levels of defence in depth shall be independent to the extent practicable. Defence in depth shall be achieved at the design phase through the application of design provisions specific to the five levels of defence.* This section provides more detail regarding the five layers of defence in depth.	This clause is unchanged from that in RD-337 addressed in the Darlington PSR1 review [B.24-21]. That review is generic to OPG Nuclear and is fully applicable to Pickering. As noted in Clause 4.3 above, Defense in depth is partially addressed in PSR2 Plant Design Safety Factor 1 Report, under Review Task 4. In addition, an integrated assessment of 'Defense-in-depth' will be done as part of the Global Assessment per [B.24-9].	**Compliant**
6.2 Safety Functions *The NPP design shall provide adequate means to:* *1. maintain the plant in a normal operational state* *2. ensure the proper short-term response immediately following a PIE* *3. facilitate the management of the plant in and following DBAs and DECs [Design Extension Conditions].* …	Normal operation of the plant is within the envelope established in the Operating Manuals. This is addressed in the "Procedures" Safety Factor 11, Review Task #6: "Procedures for Normal, Abnormal and Emergency Conditions" review. The Operating Policies and Principles [B.24-51] and [B.24-52], in conjunction with the SOE establish the operating envelope within the Safety Analysis. The Pickering 1,4 [B.24-56] and 5-8 [B.24-57], Safety Reports – Part 3, identify the response of the plant to PIEs. The systems and equipment required to ensure the plant remains within it analyzed envelope are included in the Safety Related System lists [B.24-58]. The importance of the specific systems is addressed in PSR2 Plant Design Safety Factor 1 Report, under Review Task 3, "List of SSCs Important to Safety". The safety functions for dealing with DECs are addressed in the beyond design basis accident management program [B.24-59].	**Compliant**

REGDOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
	The guidelines for managing mitigating systems and equipment for BDBA events are documented in [B.24-60] and [B.24-61]. Pickering is compliant with this clause.	
3.3 Accident Prevention and Plant Safety Characteristics The design shall apply the principles of defence in depth to minimize sensitivity to PIEs. Following a PIE, the plant is rendered safe by: 1. inherent safety features 2. passive safety features 3. specified procedural actions 4. action of control systems 5. action of safety systems 6. action of complementary design features (CDFs)	This clause is unchanged from that in RD-337 addressed in the Darlington PSR1 review [B.24-21], with the exception of the sixth item relating to CDFs, which is new. The Darlington review is generic and fully applicable to Pickering. The addition relating to the CDFs is addressed in the PSR2 review relating to REGDOC-2.3.2 "Accident Management" [B.24-44]. Pickering is compliant with this clause.	**Compliant**
6.4 Radiation protection and acceptance criteria *The design shall include provisions for the prevention and mitigation of radiation exposures resulting from DBAs and DECs.* *The design shall also ensure that potential radiation doses to the public from AOOs and DBAs do not exceed dose acceptance criteria provided in section 4.2.1. The calculated overall risk to the public shall meet the safety goals in section 4.2.2.*	Clause 6.4 was a gap and Issue for DSA for Darlington RD-337 review [B.24-21]. See 4.2.1 for discussion above. Compliance with AOO dose limits will be addressed through the REGDOC-2.4.1 compliance framework.	**PSR2 REGDOC-2.5.2 Gap #1** (Deterministic Safety Analysis) Safety Factor 5 (Deterministic Safety Analysis)
6.5 Exclusion Zone *The design shall include adequate provision for an appropriate exclusion zone. The appropriateness of the exclusion zone shall be based on several factors, including:* *1. evacuation needs* *2. land usage needs* *3. security requirements* *4. environmental factors*	The exclusion zone and requirements for maintaining it are identified in the LCH, Section 1.4 [B.24-3]. Section 10 of [B.24-3] explicitly addresses Pickering's requirements in the areas of environmental monitoring and controls. This also addresses land use within the exclusion zone. Evacuation requirements are identified in the Consolidated Nuclear Emergency Plan [B.24-62]. Pickering is compliant with this clause.	**Compliant**

REGDOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
6.6 Facility layout *The facility layout shall take into account PIEs to enhance protection of SSCs important to safety.* *The design shall take into account the interfaces between the safety, security and safeguards provisions of the NPP and other aspects of the facility layout, such as:* …	This clause details security provision requirements and the interface with safety-related systems and structures. Security and Safeguards are not included in this scope of this review.	n/a
6.6.1 Requirements for multiple units *The design shall take due account of challenges to multiple units at a site. Specifically, the risk associated with common-cause events affecting more than one unit at a time shall be considered.*	This is a new clause that was not previously included in RD-337. The requirements in this clause have been addressed in the REGDOC-2.3.2, "Accident Management" [B.24-44] PSR2 review. Pickering is compliant with this clause.	**Compliant** Addressed in other PSR2 code review.
4. General Design Requirements	Heading	n/a
7.1 Safety classification of structures, systems and components *The design authority shall classify SSCs using a consistent and clearly defined classification method. The SSCs shall then be designed, constructed, and maintained such that their quality and reliability is commensurate with this classification.* *In addition, all SSCs shall be identified as either important or not important to safety. The criterion for determining safety importance is based on:* *1. safety function(s) to be performed* *2. consequence(s) of failure* *3. probability that the SSC will be called upon to perform the safety function* *4. the time following a PIE at which the SSC will be called upon to operate, and the expected duration of that operation.* *SSCs important to safety shall include:* *1. safety systems* *2. complementary design features* *3. safety support systems* *4. other SSCs whose failure may lead to safety concerns (e.g., process and control systems)* *Appropriately designed interfaces shall be provided between SSCs of different classes in order to minimize the risk of having SSCs less important to*	This clause was compliant for the Darlington RD-337 review. Pickering uses the same approach to classification as Darlington and is also compliant. However, the introduction of Design Extension Conditions (DEC) and Complementary Design Features (CDFs) in this clause is a new requirement. The systems and equipment required to ensure the plant remains within its analyzed envelope are included in the Safety Related System lists [B.24-58] and those credited in the Safety Report are addressed in the SOE [B.24-63]. The importance of the specific systems is addressed in PSR2 Plant Design Safety Factor 1 Report, under Review Task 3, "List of SSCs Important to Safety". The safety functions for dealing with DECs are addressed in the beyond design basis accident management program [B.24-59]. The guidelines for managing mitigating systems and equipment for BDBA events are	**Compliant** Addressed in other PSR2 Safety Factor review.

REGDOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
safety adversely affecting the function or reliability of SSCs of greater importance….	documented in [B.24-60] and [B.24-61]. Pickering is compliant and has established provisions for DECs and CDFs [B.24-64].	
7.2 Plant design envelope *The design authority shall establish the plant design envelope, which comprises all plant states considered in the design: normal operation, AOOs, DBAs and DECs, as shown in figure 1.*	DECs are a new concept being introduced. However, both BDBA and DECs have been addressed for Pickering as detailed in Clause 7.1 immediately above.	**Compliant**
7.3 Plant states The plant states are grouped into four states: 1) Normal Operation 2) Anticipated Operational Occurances 3) Design Basis Accidents 4) Design Extension Conditions, including Severe Accidents Items 1 and 3 are addressed here , while Items 2 and 4 have incremental new requirements relevant to Pickering that merit further discussion below:	Normal operation of the plant is within the envelope established in the Operating Manuals. This is addressed in the "Procedures" Safety Factor 11, Review Task #6: "Procedures for Normal, Abnormal and Emergency Conditions" review. The Operating Policies and Principles [B.24-51] and [B.24-52], in conjunction with the SOE [B.24-63], establish the operating envelope within the Safety Analysis for Design Basis Accidents. DECs, as a subset of BDBAs, are a new concept being introduced in item 4. This new addition and the remainder of the clause is addressed in Clause 7.3.4 below.	**Compliant**
7.3.2 Anticipated operational occurrences *The design shall include provisions such that releases to the public following an AOO do not exceed the dose acceptance criterion provided in section 4.2.1.* *The design shall also provide that, to the extent practicable, SSCs not involved in the initiation of an AOO shall remain operable following the AOO.* *The response of the plant to a wide range of AOOs shall allow safe operation or shutdown, if necessary, without the need to invoke provisions beyond Level 1 defence in depth or, at most, Level 2.* *…*	This was clause 7.3.2 for the RD-337 review and was a gap for DSA for RD-337 [B.24-21] at Darlington. As identified above (under 4.2.1 above), this is a PSR2 Gap relating to AOOs.	**PSR2 REGDOC-2.5.2 Gap #1** (Deterministic Safety Analysis) Safety Factor 5 (Deterministic Safety Analysis)

REGDOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
7.3.4 Design extension conditions *The design authority shall identify the set of design-extension conditions (DECs) based on deterministic and probabilistic methods, operational experience, engineering judgment and the results of research and analysis. These DECs shall be used to further improve the safety of the NPP by enhancing the plant's capabilities to withstand, without significant radiological releases, accidents that are either more severe than DBAs or that involve additional failures…*	DECs is new concept that was not in RD-337. The Fukushima Action Item (FAI) process has addressed this concept per [B.24-13] and [B.24-53], for Pickering. A Standard and Guidelines have been established for BDBA program management [B.24-59] and [B.24-65]. Complementary Design Feature (CDF) design, management and surveillance [B.24-64], [B.24-60], [B.24-61] have been established. Equipment Important to Emergency Response (EITER) is managed through a new managed process [B.24-67]. Pickering is compliant with this requirement.	**Compliant**
7.3.4.1 Severe accidents within design extension conditions *The design shall be balanced such that no particular design feature or event makes a dominant contribution to the frequency of severe accidents, taking uncertainties into account. Early in the design process, the various potential barriers to core or fuel degradation shall be identified, and features that can be incorporated to halt core or fuel degradation at those barriers shall be provided.* *The design shall also identify the equipment to be used in the management of severe accidents including equipment that is available onsite and offsite…* *For DECs with severe core damage, the containment shall maintain its role as a leak-tight barrier for a period that allows sufficient time for the implementation of offsite emergency procedures following the onset of core damage. Containment shall also prevent uncontrolled releases of radioactivity after this period…* *Consideration shall be given to the prevention of recriticality following severe accidents.*	The SAMG program is addressed under the BDBA management standard [B.24-59]. All of the identified issues relating to BDBA mitigation equipment (EME) and procedures have been addressed. The design of the BDBA and SAMG response has addressed these issues. The requirement for containment to maintain a leak tight barrier for a period of time was not an explicit requirement in BDBA/SA mitigation. This represents a PSR2 Gap. A combination of multi-unit safety analysis and implementation of additional mitigating provisions to support containment integrity for RLCs (e.g., enhancements such as Phase 2 EME) are expected to address this gap. However, this is identified as a PSR2 gap specific to Containment leak tightness. Consideration of criticality is addressed in SAMG, specifically relating to any strategies that add	**PSR2 REGDOC-2.5.2 Gap #3** (Containment Leak Tightness for DECs) Safety Factor 1 (Plant Design)

REGDOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
	D_2O (See SAG2 for constraints) to the reactor. There is no PSR2 gap.	
7.4 Postulated Initiating Events (PIEs) The requirements for systematic identification and classification of PIEs are identified. REGDOC-2.4.1 (DSA) and REGDOC-2.4.2 (PSA) are to be used to identify the PIEs. The events should include: 1) Internal events 2) External events 3) Combination of events Items 2 and 3 have incremental new requirements relevant to Pickering that merit further discussion below:	The internal events are identified in the Pickering Safety Reports [B.24-54] and [B.24-55]. Additionally, identification and quantification of PIEs in the PSAs are performed in accordance with Reference [B.24-69]. The PIEs for Pickering 1,4 are included in Reference [B.24-70], while those for Pickering 5-8 are included in [B.24-71].	**Compliant**
7.4.2 External hazards *All natural and human-induced external hazards that may be linked with significant radiological risk shall be identified. External hazards which the plant is designed to withstand shall be selected, and classified as DBAs or DECs…*	Clause 7.4.2 was a gap and Issue in DSA for RD-337 for Darlington [B.24-21]. However, hazard screening has now been conducted as part of the Pickering PSA program. This is addressed in the PSR2 Hazard Analysis Safety Factor report. The hazard screening and identification for Pickering 1,4 is included in Reference [B.24-72] while that for Pickering 5-8 is in [B.24-73]. Therefore, this is not a PSR2 gap.	**Compliant**
7.4.3 Combination of events *Combinations of randomly occurring individual events that could credibly lead to AOOs, DBAs, or DECs shall be considered in the design. Such combinations shall be identified early in the design phase, and shall be confirmed using a systematic approach.* *Events that may result from other events, such as a flood following an earthquake, shall be considered to be part of the original PIE.*	This was clause 7.4.3 in RD-337 and was a gap for DSA [B.24-21] for Darlington. The review identified that not all initial states or initiating events were considered. The PSAs already consider combinations of events. Similarly, the Hazard Screening assessments have considered combinations. The DSA in the Safety Report assumes random failure of select mitigating functions. Notwithstanding, as noted for the Darlington RD-337 review, there remains a gap relating to identification and treatment of event combinations. This is a PSR2 gap, however, it is already captured within the scope of the REGDOC-2.4.1 review and gap.	Addressed in other PSR2 code review. PSR2 REGDOC-2.4.1 Gap

REDGOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
7.5 Design rules and limits *The design authority shall specify the engineering design rules for all SSCs. These rules shall comply with appropriate accepted engineering practices…* *The design of complementary design features should be such that they are effective for fulfilling the actions credited in the safety analysis, with a reasonable degree of confidence. Other SSCs that are credited for DECs should also meet this expectation.*	The design rules are governed by the "Design Management' [B.24-74] and the "Engineering Change Control" (ECC) [B.24-75] programs. These programs ensure designs rules and accepted engineering practices are adhered to. Rules for DECs are new requirements that were not in RD-337. However, they have been established and are addressed fully in ECC and specific governance and guidance [B.24-59] and [B.24-64].	**Compliant**
7.6 Design for Reliability *All SSCs important to safety shall be designed with sufficient quality and reliability to meet the design limits. A reliability analysis shall be performed for each of these SSCs. Where possible, the design shall provide for testing to demonstrate that the reliability requirements will be met during operation. The safety systems and their support systems shall be designed to ensure that the probability of a safety system failure on demand from all causes is lower than 10^{-3}.* …	The failure probability of 10^{-3} in the REGDOC-2.5.2 context applies to on-demand failure of a 'safety system' which is synonymous with a SIS per the REGDOC-2.5.2 definition. The target reliabilities/unavailabilities for the Systems Important to Safety (SIS) are included in Table 2.1-1 of the 2015 Annual Reliability Report [B.24-76]. For the Special Safety Systems, all Pickering 1,4 and Pickering 5-8 systems meet the target, with exception of Pickering 1,4 ECI, which has a target of $2x10^{-3}$ yrs/yr. This is compliant with the RD/RG-98 for an existing plant, however, it is greater than the requirement for a new plant. In addition, the other SIS standby safety support and safety-related support systems generally do not meet the on-demand reliability requirement for a new plant. This is therefore a PSR2 gap.	**PSR2 REGDOC-2.5.2 Gap #4** (On-Demand Reliability of Safety Systems) Safety Factor 1 (Plant Design)
7.6.1 Common-cause failures *The potential for common-cause failures (CCFs) of items important to safety shall be considered in determining where to apply the principles of separation, diversity and independence so as to achieve the necessary reliability. Such failures could simultaneously affect a number of different items important to safety. The event or cause could be a design deficiency, a manufacturing deficiency, an operating or maintenance error, a natural phenomenon, a human-induced event, or an*	Clause 7.6.1 was a gap for RD-337 [B.24-21] in the Darlington review. For Darlington, this issue was addressed under the Darlington A Risk Assessment (DARA) update program. Similarly, for Pickering, the PSA update has included and addressed CCFs. Pickering is compliant with this requirement.	**Compliant**

REDGOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
unintended cascading effect from any other operation or failure within the plant.		
7.6.2 Single-failure criterion *All safety groups shall function in the presence of a single failure. The single-failure criterion requires that each safety group can perform all safety functions required for a PIE in the presence of any single component failure, as well as:* *1. all failures caused by that single failure* *2. all identifiable but non-detectable failures, including those in the non-tested components* *3. all failures and spurious system actions that cause (or are caused by) the PIE…*	Clause 7.6.2 was a gap for RD-337 [B.24-21] for Darlington. However, it was classified as an Acceptable Deviation with the rationale provided as detailed in Section B.24.2.1 of this PSR2 assessment, above. This clause was also an IAEA NS-R-1 gap and Acceptable Deviation for Pickering 5-8 in PSR1. The rationale provided for classifying this requirement as an Acceptable Deviation for Pickering 5-8 is also applicable to Pickering 1,4. Pickering is compliant with this requirement.	**Compliant**
7.6.3 Fail-safe Design *The concept of fail-safe design shall be incorporated, as appropriate, into the design of systems and components important to safety.*	This is a general principle and good design practice that has been incorporated into the design of systems and components important to safety as part of design, to the extent practicable. Failure positions and states (e.g., open, close, as-is, de-energized, etc.) of components due to loss of power, control and/or air for SIS are generally documented in the Auxiliary Service Failures (Section 8) of the Operating Manuals, e.g., reference [B.24-77]. Where fail-safe cannot be achieved or unsafe failures occur, annunciations are provided so that the failure can be readily detected and corrected. Pickering is compliant with this requirement.	**Compliant**
7.6.4 Allowance for equipment outages *The design shall include provisions for adequate redundancy, reliability, and effectiveness, to allow for online maintenance and online testing of systems important to safety.*	There is an extensive set of on-line testing and maintenance/surveillance predefines for Safety Systems. These are identified in the SOE Compliance Tables in accordance with [B.24-63] and in the system reliability models as identified in Appendix F of the annual reliability report [B.24-76]. Credited testing is identified in Appendix G of [B.24-76].	**Compliant**

REDGOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
	This demonstrates that the designs include adequate provisions for surveillance and Pickering is compliant with these requirements.	
7.6.5 Shared Systems This clause establishes the requirements relating to sharing of safety and process functions. There are also two sub-clauses relating to sharing Safety System instrumentation and the sharing of SSCs important to safety between reactors. Sharing of instrumentation is addressed here while sharing between units is addressed below.	There are several cases where sharing safety and process functions (including instrumentation) exists at both Pickering 1,4 and Pickering 5-8 (e.g., Boilers and Steam Reject Valves (SRVs), Moderator, Class III and II power, etc.). In all cases the safety function standards and rules apply and suitable isolation between the process control and safety functions is provided. Pickering is compliant with these requirements.	**Compliant**
7.6.5.2 Sharing of SSCs between reactors *SSCs important to safety shall typically not be shared between two or more reactors. In exceptional cases when SSCs are shared between two or more reactors, such sharing shall exclude safety systems and turbine generator buildings that contain high-pressure steam and feedwater systems, unless this contributes to enhanced safety. If sharing of SSCs between reactors is arranged, then the following requirements shall apply:* *1. safety requirements shall be met for all reactors during operational states, DBAs and DECs* *2. in the event of an accident involving one of the reactors, orderly shutdown, cool down, and removal of residual heat shall be achievable for the other reactor(s)…*	This sub-clause has a new requirement that sharing of safety systems and the turbine generator building not be permitted. ECI and Negative Pressure Containment are shared between units at Pickering. Additionally, the Turbine Buildings are shared between units on each station. If either common ECI or Containment are unavailable, all affected units are considered impaired and must shutdown within specified time limits, hence minimizing the risk of a coincidental DBA. Environmental conditions in the common turbine building have been assessed and credited provisions have been protected to ensure the ability to shutdown/control, cool and monitor remains available on non-accident units. Notwithstanding the above, sharing of safety systems and having a common turbine building for multiple units is a PSR2 gap.	**<u>PSR2 REGDOC-2.5.2 Gap #5</u>** (Sharing of Safety Systems and Turbine Hall) Safety Factor 1 (Plant Design)

REGDOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
7.7 Pressure-retaining structures, systems and components *All pressure-retaining SSCs shall be protected against overpressure conditions, and shall be classified, designed, fabricated, erected, inspected, and tested in accordance with established standards. For DECs, relief capacity shall be sufficient to provide reasonable confidence that pressure boundaries credited in severe accident management will not fail….* *Pressure-retaining components whose failure will affect nuclear safety shall be designed to permit inspection of their pressure boundaries throughout the design life…* In addition to the above, the guidance section of the clause provides additional considerations with regards to Leak-before-break system design.	The requirements for pressure retaining components and inspections are addressed in the CSA N285 series of code reviews as part of PSR2, included under the Plant Design Safety Factor 1 review. Inspection requirements and condition monitoring are addressed under Safety Factor 2 (Actual Conditions of SSCs) and Safety Factor 4 (Aging) codes and standards reviews [B.24-68]. Heat Transport System overpressure protection requirements for DECs has been addressed as part of the Fukushima Action Items, FAI 1.1.1 [B.24-53]. Further, boiler relief capacity for DECs is bounded by the relief capacity for DBAs initiating from high power. Pickering is compliant with this requirement.	**Compliant** Addressed in other PSR2 Safety Factor and code reviews.
7.8 Equipment environmental qualification *The design shall include an equipment environmental qualification (EQ) program. Development and implementation of this program shall ensure that the following functions can be carried out:* … *Equipment and instrumentation credited to operate during DECs shall be demonstrated, with reasonable confidence, to be capable of performing their intended safety function(s) under the expected environmental conditions. A justifiable extrapolation of equipment and instrumentation behaviour may be used to provide assurance of operability, and is typically based on design specifications, environmental qualification testing, or other considerations.*	This same issue was addressed in Clause 7.8 for the RD-337 [B.24-21] review for Darlington. A gap was identified relating to the absence of qualification for BDBA conditions. As discussed in Section B.24.2.1 for Darlington, Issues D072 and D143 have been closed for both Darlington and for Pickering as part of the FAI process [B.24-13]. The DBA qualification is addressed in a PSR2 code review for CSA N290.13 [B.24-66] included in [B.24-68]. In this clause, the BDBA term from RD-337 has been replaced by DEC, and AOOs have been added to the qualification requirements. However, all requirements in this clause have been addressed in the other code reviews. The qualification of equipment for DECs is specifically addressed in the PSR2 code review for REGDOC-2.3.2, "Accident Management" [B.24-44]. Pickering is compliant with this requirement.	**Compliant** Addressed in other PSR2 Safety Factor and code reviews.

REDGOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
7.9 Instrumentation and control **7.9.1 General** *The design shall include provision of instrumentation to monitor plant variables and systems over the respective ranges for operational states, DBAs and DECs, in order to ensure that adequate information can be obtained on plant status.* **...** *This shall include instrumentation for measuring variables that can affect the fission process, the integrity of the reactor core, the reactor cooling systems, and containment, as well as instrumentation for obtaining any plant information that is necessary for its reliable and safe operation…*	Sections 6 and 7 of Part 2 of the Pickering Safety Reports [B.24-54] and [B.24-55] provide a high level summary of instrumentation and control related to safety systems and process control. Further specific detail can be found in the USI 60000-series of Design Manuals. As part of PSR2 scope, three separate code reviews relating to instrumentation and control have been performed against the following: • CSA N290.1, "Requirements for Shutdown Systems of Nuclear Power Plants" [B.24-78] • CSA N290.4, "Reactor Control Systems of Nuclear Power Plants" [B.24-79] • CSA N290.6, "Requirements for Monitoring and Display of Nuclear Power Plant Safety Functions in the Event of an Accident" [B.24-80] The qualification of instrumentation for DECs is specifically addressed in the PSR2 code review for REGDOC-2.3.2, "Accident Management" [B.24-44]. There are no gaps for any of the above standards relating to instrumentation and control, and there are no PSR2 gaps.	**Compliant** Addressed in other PSR2 Safety Factor and code reviews.
7.9.2 Use of computer-based systems or equipment *Appropriate standards and practices for the development and testing of computer hardware and software shall be established and implemented throughout the lifetime of the system or equipment, and, in particular, throughout the software development cycle.* …	A separate PSR2 code review of CSA N290.14 (2015) [B.24-81], relating to qualification of hardware and software has been performed. The review found that new applications and changes comply with the standard. However, one gap was identified relating to absence of categorization and qualification for some legacy real-time process computing applications. This gap is also applicable to this clause. Hence, there is a Gap for PSR2.	Addressed in other PSR2 code review. PSR2 CSA N290.14-15 Gap

REGDOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
7.9.3 Accident monitoring instrumentation *Instrumentation and recording equipment shall be such that essential information is available to support plant procedures during and following DBAs and DECs…*	PSR2 review against CSA N290.6-09 [B.24-80] determined that Pickering is compliant with these requirements including BDBA conditions. Guidance under 7.9.3 identifies that instrumentation shall be available for measuring hydrogen concentration. This requirement is also a requirement for new plants in CSA N290.3-11 [B.24-111], Clauses 9.5.5 and 10.2.3. This issue was identified as being acceptably addressed given the implementation of PARs and improvements made to hydrogen concentration estimation in SAMG. Pickering is compliant for PSR2.	**Compliant**
7.10 Safety support system *The safety support systems shall ensure that the fundamental safety functions are available in operational states, DBAs and DECs. …* *2. support continuity of the fundamental safety functions until long-term (normal or backup) service is re-established:* *a. without the need for operator action to connect temporary onsite services for at least 8 hours.* *b. without the need for offsite services and support for at least 72 hours…* *Pre-installed equipment can be credited for accident mitigation after 30 minutes where only control room actions are needed or after 1 hour if field actions are needed. These actions should be limited to operating valves, starting pumps, etc....*	The 72 hours is consistent with the guidance for PSA and the BDBA requirements per [B.24-60] and [B.24-61]. For the DEC reference cases, where the Deaerator Storage Tank inventory remains available, the 8 hour requirement is satisfied [B.24-65]. Additionally, this clause introduces new requirements for times after which main control room and local field action may be credited of 30 minutes and 1 hour respectively. This is a change from Clause 8.10.4 of RD-337 which had the 15 and 30 minute requirements, hence DNGS was compliant. The Pickering Safety Reports [B.24-56] and [B.24-57] – Part 3 – Section 1 Integrated Summary, assess operator credits after 15 minutes for control room and 30 minutes for field actions. The Safety Analysis assumptions have a different perspective than these design requirements for a new plant. The current design requirements are established to provide ample margin to account for actual plant condition and experience. Whereas the action times in safety analysis generally assume that the relevant parameters, conditions and alarms are at a conservative limit. Additionally, for a mature plant,	**PSR2 REGDOC-2.5.2 Gap #6** (Allowable Times for Crediting On-Site Operator Actions) Safety Factor 1 (Plant Design)

REGDOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
	there is significant operating experience (including drills and exercises) that provide assurance that the credited times can be achieved. Notwithstanding the above, the issues relating to on-site operator action times are a PSR2 gap.	
7.11 Guaranteed shutdown state (GSS) This clause identifies the high level requirements of a GSS.	GSS requirements for Pickering 1,4 and Pickering 5-8 are documented in the respective Moderator System Operational Safety Requirements [B.24-82] and [B.24-83] which, along with their associated Compliance Tables and implementing documents, satisfy all requirements. Pickering is compliant for PSR2.	**Compliant**
7.12 Fire safety *The design of the NPP, including that of external buildings and SSCs integral to plant operation, shall include provisions for fire safety.*	For Darlington, this clause was a gap for RD-337 [B.24-21]. The Darlington gaps were reviewed in B.24.2.1 above, and no specific gaps were identified as being applicable to Pickering. However, for PSR2, a separate code review is being performed for fire protection, CSA N293-12 [B.24-34]. Additionally, as specified in [B.24-9], code reviews are being performed for the: - Fire protection aspects of the 2010 version of the "National Building Code of Canada" (NBCC), - the 2010 version of "National Fire Code of Canada" (NFCC), - NFPA-20 (2016), "Standard for the Installation of Stationary Pumps for fire Protection" and, - NFPA-24 (2016), "Standard for the Installation of Private Fire Service Mains and their Appurtenances".	Addressed in other PSR2 code reviews. PSR2 CSA N293-12 Gap
7.13 Seismic qualification and design *The seismic qualification of all SSCs shall meet the requirements of Canadian national or equivalent standards...* **7.13.1 Seismic design and classification**	The guidance section of clause 7.13.1 has been added in REGDOC-2.5.2 and provides specific references to best-practices and industry guidance for seismic qualification. Additionally, the	**<u>PSR2 REGDOC-2.5.2 Gap #7</u>** (Seismic Qualification and Design)

REGDOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
A beyond-design-basis earthquake (BDBE) shall be identified that meets the requirements for identification of DECs as described in section 7.3.4. SSCs credited to function during and after a BDBE shall be demonstrated to be capable of performing their intended function under the expected conditions. Such demonstration shall provide high confidence of low probability of failure (HCLPF) under BDBE conditions for these SSCs. This demonstration need not be seismic qualification by testing. … *Seismic input motion, derived from the DBE, should be based on seismicity and geologic conditions at the site and expressed in such a manner that it can be applied for the qualification of SSCs. The DBE is defined by multiplying the mean site specific uniform hazard spectrum with a probability of occurrence of 10^{-4}/yr by a design factor, defined in the standard ASCE 43-05, Seismic Design Criteria for Structures, Systems and Components in Nuclear Facilities. The probability of occurrence of the defined DBE is therefore equivalent to the probability of DBAs.* *…..* ***Guidance*** *Any evaluation for BDBE should utilize the methodology in the Electrical Power Research Institute, (EPRI) TR-103959, Methodology for Developing Seismic Fragilities to determine if a HCLPF goal is met.* *…* *Beyond-design-basis margin should be such that seismically induced SSC failure probabilities do not contribute to the total core damage frequency and small and large release frequency to the extent that they do not meet the safety goals. To support meeting the safety goals, the acceptance criterion for BDBE should demonstrate that the plant HCLPF is at least 1.67 times the DBE.*	concept of qualification for DECs has been introduced. Per [B.24-9], the CSA N289 series of standards have been reviewed as part of PRS2 scope and the reviews are documented in [B.24-68]. These reviews did not identify any gaps relating to the requirements in this clause. Pickering 5-8 was originally designed to be seismically qualified as detailed in Section 2.2 of [B.24-55]. Pickering 1,4 was assessed for seismic qualification using Seismic Margin Assessment (SMA) methodology as documented in Part 2, Section 2.3 of [B.24-54] and in [B.24-84]. Subsequently Pickering 5-8 was also assessed using SMA, as part of the seismic PSA [B.24-85]. A seismic PSA has also been performed for Pickering 1,4 [B.24-86]. The SMAs address the DEC seismic event as detailed in [B.24-64]. The clause requires the DBE be defined by multiplying the mean Uniform Hazard Spectrum (UHS) with an occurrence of 10^{-4} occ/yr by a design factor. The SMAs both use a representative event, the Review Level Earthquake (RLE) which has a frequency commensurate with that for the DEC. For both stations, the RLE is defined as the 84th percentile Unified Hazard Response Spectrum (UHRS) with a return period of 10,000 years (i.e., 10^{-4} occ/yr), having a peak ground acceleration of 0.23g. Pickering 1,4 plant-limiting HCLPF is 0.23g as per [B.24-86]. The Pickering 5-8 plant-limiting HCLPF is 0.18g as per [B.24-85]. The DBE is 0.05g for the site [B.24-86]. This demonstrates that there is greater than 1.67 times margin between the RLE and the existing DBE. However, the margin between the new plant 10^{-4} occ/yr DBE and the Beyond	Safety Factor 1 (Plant Design)

REDGOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
	Design Basis Earthquake referred to has not been confirmed. Hence, this is a PSR2 gap relating to Beyond Design Basis Earthquake qualification requirement margin.	
7.14 In-service testing, maintenance, repair, inspection and monitoring The clause details high level requirements for facilitation of the above to be considered and addressed in the plant design.	This is a new clause that was not in RD-337. Surveillance and repair of SSCs is addressed by the following OPG programs: • N-PROG-MA-0026, "Equipment Reliability" • N-PROG-MA-0017, "Component and Equipment Surveillance" • N-PROG-MA-0025, "Major Components" • N-PROG-MP-0008, "Integrated Aging Management" • N-PROG-MP-0014, "Reactor Safety Program" • N-STD-RA-0033, "Reliability Monitoring And Reporting Of Systems Important To Safety" In addition, the clause refers to the following codes that are included in the scope of the PSR2 code reviews per [B.24-9]: • REGDOC-2.6.3, "Aging Management" • CSA N287.7, "In-service Examination and Testing Requirements for Concrete Containment" • CSA N285.4, "Periodic Inspection of CANDU Nuclear Power Plant Components" • CSA N285.5, "Periodic Inspection of CANDU Nuclear Power Plant Components" • CSA N291, "Requirements for Safety-Related Structures for CANDU Nuclear Power" Therefore, at a high level Pickering has addressed the requirements of	**Compliant** Also addressed in other PSR2 code reviews.

REDGOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
	this clause with having the capability to perform adequate surveillance and maintenance and is compliant for PSR2.	
7.15 Civil structure This clause is subdivided into three sub-sections: • Structure Design • Surveillance of Structures • Handling of large loads	Heading	n/a
7.15.1 Structural Design *The NPP design shall specify the required performance for the safety functions of the civil structures in operational states, DBAs and DECs.* *Civil structures important to safety shall be designed and located so as to minimize the probabilities and effects of internal hazards such as fire, explosion, smoke, flooding, missile generation, pipe whip, jet impact, or release of fluid due to pipe breaks.* *External hazards such as earthquakes, floods, high winds, tornadoes, tsunamis, and extreme meteorological conditions shall be considered in the design of civil structures.* *….* *The design should identify all DEC loads considered in the structure design and provide the assessment methodology and acceptance criteria.*	Clause 7.15.1 in RD-337 was previously generic to all safety-related structures and did not differentiate between containment and other structures. Pickering structural design has been assessed to meet and facilitate all DBA safety credits. The requirement relating to DECs is a new requirement and is applicable to all structures. Pickering structures were not specifically designed for some external DECs. However, assessments for all hazards have been completed. Internal and external hazard assessments have been done for Pickering as part of screening for the PSA per [B.24-72] and [B.24-73]. The screening assesses the impact of the hazards on structures. Those hazards that cannot be screened out based on low frequency and/or consequences are subjected to further assessment. All internal hazards have been explicitly analyzed or screened out. Only external hazards relating to high wind and seismic were subject of additional assessment in terms of structural capability. The Pickering units' nuclear safety success paths for high wind DECs have been addressed in the PSAs [B.24-87] and [B.24-88]. Although some structural vulnerabilities were identified, there were alternate	**Compliant**

REGDOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
	success paths available or the overall risk from failures was acceptably low. Assessments for Pickering Units 1,4 and 5-8, have been addressed in the seismic PSAs [B.24-85] and [B.24-86]. These have demonstrated that the credited safety-related structures are sufficiently robust that they will support the required safety functions for DEC seismic events. Based on the above, it is concluded that the Pickering plant structural design is sufficiently robust to provide assurance that the required nuclear safety functions will be available for DEC conditions. Therefore, Pickering is compliant with this clause.	
Containment Structures *The design should specify the safety requirements for the containment building or system, including, for example, its structural strength, leak tightness, and resistance to steady-state and transient loads (such as those arising from pressure, temperature, radiation, and mechanical impact) that could be caused by postulated internal and external hazards…*	Safety requirements for the Pickering 1,4 and 5-8 Containment structures are addressed in the Operational Safety Requirements [B.24-89] and [B.24-90]. The clause also specifies requirements for the CSA N287 series of standards for concrete containment structures: • N287.1, General Requirements • N287.2, Material Requirements for Concrete Containment Structures • N287.3, Design Requirements • N287.4, Construction, Fabrication and Installation Requirements • N287.5, Examination and Testing Requirements • N287.6, Pre-operational Proof and Leakage Rate Testing Requirements The N287 reviews are being addressed in separate code reviews as part of PSR2 as defined in the PSR2 basis document [B.24-9].	Addressed in other code reviews. PSR2 CSA N287 Gap

REGDOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
Safety-related structures *The safety-related structures other than the containment should be designed and constructed in accordance with CSA N291, "Requirements for safety-related structures for CANDU nuclear power plants."*	The introduction of CSA N291 [B.24-91] [B.24-89] is a new requirement that was not previously included in RD-337. However, a separate PSR2 review is being performed for this code.	Addressed in other PSR2 code review. PSR2 CSA N291-15 Gap
7.15.2 Surveillance *The design shall enable implementation of periodic inspection programs for structures important to safety in order to verify that the as-constructed structures meet their functional and performance requirements.* *The design shall also facilitate in-service monitoring for degradations that may compromise the intended design function of the structures. In particular, the design shall permit monitoring of foundation settling.* *Pressure and leak testing shall be conducted on applicable structures to demonstrate that the respective design parameters comply with requirements.* *The design shall facilitate routine inspection of sea, lake, and river flood defences and demonstrate fitness for service.*	Clause 7.15.2 was a gap for the Darlington RD-337 [B.24-21] review relating to Periodic Inspection Programs. The periodic inspection requirements for containment are addressed under 7.15.1 above. The surveillance requirements in REGDOC-2.5.2 also apply to safety-related structures. Pickering has established inspection specifications for safety related structures [B.24-92]. Pickering has established a surveillance (Preventative Maintenance ID) for the periodic inspection and is in the process of implementing the Inspection and Test Plan. Any issues identified during the inspection will be addressed via the corrective action program, hence, Pickering is compliant with this requirement for PSR2.	**Compliant**
7.15.3 Lifting and Handling of Large Loads *The lifting and handling of large and heavy loads, particularly those containing radioactive material, shall be considered in the NPP design…* *The drop of large loads lifted and handled in areas where there are systems and components that are important to safety shall be taken into account in the design. The potential load due to the large load drop shall be taken into account in the analysis of DBAs.*	As part of the Hazard Screening reviews for Pickering [B.24-72] and [B.24-73], dropping of heavy loads during craning has been screened out based on low frequency of exposure. For Pickering, craning in the Reactor Building Boiler Rooms is not possible at power and it is a prerequisite that the reactor be shutdown and in a Guaranteed Shutdown State. Specific assessments have been performed for flasking over and around the Reactivity Mechanism Decks at Pickering and the risk of load drops and consequential damage has been evaluated and determined to be acceptable.	**Compliant**

REDGOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
	In general, hoisting over safety-related equipment/systems (e.g., Class III power, Service Water piping) is evaluated in lift planning and is governed by N-STD-MA-0018 [B.24-93] "Hoisting And Rigging", which identifies the process and management controls for hoisting operations, and the inspection and maintenance of lifting devices and rigging equipment. Pickering is compliant for PSR2.	
7.16 Construction and Commissioning This clause identifies high level requirements relating to ensuring that the new plant design addresses constructability and facilitates design confirmation including functional and performance testing.	Pickering is a mature plant, however, all modifications must adhere to the Engineering Change Control (ECC) process [B.24-75]. The ECC process requires that Constructability, Operability, Maintainability and Safety (COMS) be addressed [B.24-94]. The Modification Process [B.24-95] identifies the method of testing the modification and specifications for the testing are prepared in accordance with [B.24-96]. Pickering is compliant with this requirement.	**Compliant**
7.17 Aging and wear *The design shall take due account of the effects of aging and wear on SSCs. For SSCs important to safety, this shall include: …*	A reference to RD-334 "Aging Management" has been introduced and is a new requirement. However, it has already been superseded by REGDOC-2.6.3. An incremental review of REGDOC-2.6.3 has been performed as part of PSR2 and is documented in [B.24-68]. Two gaps have been identified against Safety Factor 4. This is a PSR2 gap.	Addressed in other PSR2 Safety Factor and code review. PSR2 REGDOC-2.6.3 Gap
7.18 Control of Foreign Material *The design provides for exclusion and removal of all foreign material and corrosion products that may have an impact on safety.*	This clause specifically relates to foreign material resulting from process system operation as opposed to that generated during maintenance activities. Systems (e.g., Heat Transport Pressure and Inventory Control, Moderator, Liquid Zone Control, Fuel Handling) and that are exposed to radiation levels where foreign	**Compliant**

REDGOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
	material exposure to radiation could result in activation products are designed with concentrators (strainers, filters, ion exchange) to remove foreign material. This is performed for radiological safety and maintainability reasons. Pickering is compliant with this requirement.	
7.19 Transport and packaging of fuel and radioactive waste The clause identifies the high level requirements that handling of new and used fuel must be addressed in the design.	The handling of new and used fuel is addressed at a high level in the Pickering Safety Report Part 2, Section 10 of [B.24-54] and [B.24-55]. Additionally, radioactive waste management is addressed in Section 13 of [B.24-54] and [B.24-55]. Additional details can be found in the applicable system Design Manuals and operating procedures. Pickering is compliant with this requirement.	**Compliant**
7.20 Escape routes and means of communication *The design shall provide a sufficient number of safe escape routes that will be available in operational states, DBAs and DECs, including seismic events. These routes shall be identified with clear and durable signage, emergency lighting, ventilation and other building services essential to their safe use.*	Fire protection, communication and emergency lighting are addressed the Pickering Safety Report Part 2, Section 11.5 of [B.24-54] and [B.24-55]. Additionally, code reviews are being performed for 2010 versions of the "National Building Code of Canada" (NBCC) and "National Fire Code of Canada" (NFCC), as specified in [B.24-9]. These have specific requirements relating to emergency egress. There is also a requirement that egress from containment be available regardless of containment pressure. This is addressed in Containment Airlock designs as detailed in Section 3.2.3 of [B.24-54] for Pickering 1, 4 and Section 3.2.4 of [B.24-55] Pickering 5-8. Pickering has clearly delineated seismic pathways and areas for safe operator access, occupancy and escape from threatened areas as documented in [B.24-97] and [B.24-98]. Pickering is compliant with this requirement.	**Compliant** Also addressed in other PSR2 Safety Factor and code reviews.

REGDOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
7.21 Human factors *The design shall include a human factors engineering program plan. Relevant and proven systematic analysis techniques shall be used to address human factors issues within the design process….*	This clause was a gap for RD-337 [B.24-21] for Human Factors and has been addressed in Darlington ISR Issue D260 in Section B.24.2.1 above. This clause has been revised from the previous clause in RD-337 to include significantly more background and guidance. Pickering 5-8 and more so, Pickering 1,4 design predates the formalized standards for application of human factors engineering (HFE) to nuclear power. Rather, the Pickering plant designs were based on the 'best practice' of the day and reflected operating experience from earlier nuclear plants including the Nuclear Power Demonstration plant, Douglas Point, and in the case of Pickering 5-8, Bruce A and Pickering A. Furthermore, OPG (formally Ontario Hydro) had extensive experience with control center and plant design based on experience from both fossil fueled and hydro power plants. Since the original plant designs, significant operating experience has been acquired and improvements incorporated relating to integration of human factors into the existing plant. Such areas include; improved annunciation prioritization, and display systems; nuisance alarm reduction; improved procedures including annunciation response manuals, etc. Training and use of the full scope simulators allows operators to simulate and practice procedures and response to normal evolutions and upset plant response. Procedure validation, pre-job briefings, table tops, field walkthroughs, and procedure adherence, facilitate safe and effective human interface with the plant. As demonstrated in the review of Human Factors as part of the PSR2 code reviews and the Plant Design Safety Factor, Pickering for many	**<u>PSR2 REGDOC-2.5.2 Gap #8</u>** (Human Factors in Design) Safety Factor 1 (Plant Design)

REGDOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
	years has had a fully compliant human factors program applicable to plant modifications. Although the original plant design predates the more structured application of HFE to the design, many of the important elements of HFE have been retroactively integrated into plant operations. However, the absence of systematic application of HFE principles to the original plant design is a PSR2 REGDOC-2.5.2 Gap #8 (Human Factors in Design).	
7.22 Robustness against malevolent acts	Not in the PSR2 scope.	n/a
7.23 Safeguards	Not in the PSR2 scope.	n/a
7.24 Decommissioning *Future plant decommissioning and dismantling activities shall be taken into account, such that:* *…*	Decommissioning is not within the scope of PSR2.	n/a
8. System-Specific Requirements	Heading	n/a
8.1 Reactor core This clause deals with a broad range of reactor core related issues, including: • Mechanical/structural design • Aging • Reactor physics (e.g., criticality, power density, reactivity, stability) • Core management In addition, there are two subsections: 8.1.1 Fuel design 8.1.2 Control system *Reactor core parameters and their limits shall be specified. The design shall consider all foreseeable reactor core configurations for normal operation.* *The reactor core, including the fuel elements, reactivity control mechanisms, reflectors, fuel channel and structural parts, shall be designed so that the reactor can be shutdown, cooled and held subcritical with an adequate margin in operational states, DBAs and DECs. …*	The high level reactor core design is described in Part 2 of the Safety Report Section 4 of [B.24-54] and [B.24-55]. Structural design codes for the reactor core are addressed in PSR2 under the CSA N285 series of code reviews per [B.24-9]. Aging is addressed in [B.24-68] as part of PSR2. The original reactor core physics design is addressed in [B.24-99] and [B.24-100], for Pickering 1,4 and Pickering 5-8 respectively. Fuel Design Manuals for both Pickering are included under the 37000 USI. The Operational Safety Requirements for Fuel and Reactor Physics [B.24-101] and [B.24-102] address the safety analysis assumptions and credits relating to the core design. Stability and reactivity management were addressed in the PSR2 CSA N290.4 [B.24-79] code review. There was one Gap identified	**Compliant** Also, addressed in other PSR2 Safety Factor and code reviews.

REDGOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
The design limits for power densities (and thus for peaking factors) during normal operation should be such that acceptable fuel design limits are not exceeded during AOOs and that other limits are not exceeded during DBAs and DECs…	relating to Reactor Regulating System credits for AOOs. This does not represent a gap against this clause. The DEC requirement relating to sub-criticality is new and did not appear in RD-337. However, maintaining sub-criticality is addressed in BDBA and SAMG, and hence this is not a gap (See clause 7.3 above). Also, the DEC fuel design requirements are new. The reference DEC case for Emergency Mitigating Equipment [B.24-103] credits the boilers as the long term heat sink. This does not result in conditions any more severe than a Seismic Event, which is already addressed in DBA analysis. Sensitivity DEC cases with HTS leakage, or where the HTS void becomes sufficiently large that the boilers cannot be maintained as a heatsink, can result in elevated fuel temperatures until HTS make-up can be established. DEC with moderator as the ultimate Heat Sink can result in a significant fission product release to the HTS and Containment, even though fuel channel integrity and coolable fuel geometry is maintained. Hence, although no explicit design limits for fuel have been established for DECs, there are qualitative objectives for CDFs and BDBA response that address this issue. Therefore, Pickering is compliant with this clause.	
8.1.1 Fuel elements, assemblies and design *Fuel assembly design shall include all components in the assembly, such as the fuel matrix, cladding, spacers, support plates, movable rods inside the assembly etc. The fuel assembly design shall also identify all interfacing systems…*	The design basis relating to fuel design is provided in the clause immediately above. In addition, performance of fuel for DBAs is evaluated extensively in the Safety Report Part 3 – Accident Analysis [B.24-56] and [B.24-57], for	**Compliant** Also, addressed in other PSR2 code reviews.

REGDOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
	Pickering 1,4 and Pickering 5-8 respectively. As part of PSR2 a review of CNSC REGDOC-2.4.1 [B.24-14] is being performed for Deterministic Safety Analysis. No gaps have been identified relating to fuel design or performance. Pickering is compliant with this clause.	
8.1.2 Control systems *The design shall provide the means for detecting levels and distributions of neutron flux. This shall apply to neutron flux in all regions of the core during normal operation (including after shutdown and during and after refuelling states), and during AOOs.* *The reactor core control system shall detect and intercept deviations from normal operation with the goal of preventing AOOs from escalating to accident conditions.*	The details in this clause are addressed in the PSR2 CSA N290.4 [B.24-79] code review. There was one Gap identified relating to Reactor Regulating System credits for AOOs. This has been identified as a PSR2 gap relating to CSA N290.4.	Addressed in other PSR2 code review PSR2 CSA N290.4-11 Gap
8.2 Reactor coolant system *The design shall provide the reactor coolant system (RCS) and its associated components and auxiliary systems with sufficient margin to ensure that the appropriate design limits of the reactor coolant pressure boundary are not exceeded in operational states or DBAs.* *The design shall ensure that the operation of pressure relief devices will not lead to significant radioactive releases from the plant, even in DBAs. The RCS shall be fitted with isolation devices to limit any loss of radioactive coolant outside containment.* *The material used in the fabrication of the component parts shall be selected so as to minimize corrosion and activation of the material.* *Operating conditions in which components of the pressure boundary could exhibit brittle behaviour shall be avoided.* *The design shall take into account all conditions of the boundary material in normal operation (including maintenance and testing), AOOs, DBAs and DECs, as well as expected end-of-life properties affected by aging mechanisms, the rate of*	The high level description of the Heat Transport System (HTS) design is provide in Part 2 of the Safety Report Sections 4 and 5 [B.24-54], [B.24-55], for Pickering 1,4 and Pickering 5-8 respectively. Additional detail can be found in the HTS Design Manuals under USI 33100. The requirement for aging combined with DEC conditions is a new requirement. No explicit treatment of DECs has been included in HTS aging. Although a gap has been identified against the PSR2 Aging Safety Factor review relating to operation beyond 2020, this does not specifically deal with DEC requirements. Assessments of DECs have not identified any vulnerabilities with respect to safely shutting down the reactor(s). This is because SDS trips are assumed to occur due to loss of AC power (e.g., loss of Class IV/III), and resultant HTS, Feedwater or manual trips. Also, the Shutdown	**Compliant**

REGDOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
deterioration, and the initial state of the components.	and Regulating systems are designed to fail safe on loss of control power to logic and reactivity mechanisms (Shutoff Rods, Liquid Injection Shutdown System valves, Control Absorbers and Dump Valves). The DEC assessment objective is to minimize the risk of fission product release and to maintain HTS integrity, by maintaining the containment envelope and establishing a long term heat sink, and, hence is achieved. The reference cases for DECs at Pickering 5-8 have the boilers maintained as the long-term heat sink. The limiting case DEC for HTS integrity is a seismic event rather than high wind (Tornado), flooding or fire. All events are assumed to result in a loss of AC power. Per the Seismic PSA [B.24-85], the HTS has been shown to have large margins for DECs. HTS over pressure protection was addressed as part of the Fukushima FAI-1.1.1 [B.24-53]. If the long term boiler heat sink fails or is unavailable for a DEC, the HTS pressure boundary is assumed to eventually fail due to a loss of inventory. Once HTS failure occurs either a HTS make-up heat sink or moderator heatsink is credited. Hence, maintaining HTS integrity is not a requirement for these conditions. For Pickering 1,4, the Shutdown Cooling System is used as the HTS heat sink for the seismic DECs. The Seismic Margin Assessment (SMA) [B.24-84] has shown that HTS and support systems are suitably qualified for the seismic RLC. Similar to Pickering 5-8, if the SDC or Boiler Heat sink fail, either a HTS or moderator make-up heat sink is credited. Hence, maintaining HTS integrity is not a requirement for these conditions. Given that a DEC assessment should be based on 'best estimate' and	

REGDOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
	'reasonable confidence', whereas HTS fitness for service and stress limits are based on bounding deterministic assumptions, HTS pressure boundary integrity in order to support a long-term heat sink would not be challenged by aging. Rather, requirements for DBA mitigation are more limiting than those for DECs. Sensitivity cases for DECs are evaluated with consequential HTS leakage. For DECs where the HTS is open prior (e.g., outage) or where a random DEC LOCA occurs, the HTS pressure boundary is already failed, hence, aging is not an issue. Therefore, although Pickering does not have any explicit assessments of DEC conditions with an aged HTS, based on analysis and assessments completed, this is not a safety issue. Therefore, Pickering is compliant with this clause.	
8.2.1 In-Service pressure boundary inspections *The components of the reactor coolant pressure boundary shall be designed, manufactured, and arranged in a manner that permits adequate inspections and tests of the boundary, support structures and components throughout the lifetime of the plant.* *The design shall also facilitate surveillance in order to determine the metallurgical conditions of materials for which metallurgical changes are anticipated.*	Surveillance of the HTS is addressed by the following OPG programs: • N-PROG-MA-0025, "Major Components" • N-PROG-MP-0008, "Integrated Aging Management" In addition, the clause refers to the following codes that are included in the scope of the PSR2 code reviews: • REGDOC-2.6.3, "Aging Management" • CSA N285.4, "Periodic Inspection of CANDU Nuclear Power Plant Components" Pickering is compliant with this clause.	**Compliant**

REGDOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
8.2.2 Reactor coolant system inventory *Taking volumetric changes and leakage into account, the design shall provide control of coolant inventory and pressure so as to ensure that specified design limits are not exceeded in operational states. This requirement shall extend to the provision of adequate capacity (flow rate and storage volumes) in the systems performing this function.* *The inventory in the RCS and its associated systems shall be sufficient to support cool down from hot operating conditions to zero-power cold conditions without the need for transfer from any other systems.*	There was a gap for Darlington for this clause 8.2.2 [B.24-21] and Issue D227 [B.24-35], relating to the amount of D_2O inventory in a unit. This is not a PSR2 gap as there is sufficient inventory available in each Pickering unit to accommodate HTS cooldown of all units (see assessment in Section B.24.2.1, above).	**Compliant**
8.2.3 Reactor coolant system cleanup *The design shall provide for adequate monitoring and removal of impurities and radioactive substances from the reactor coolant, including activated corrosion products and fission products leaking from the fuel. The safety limit for activity in the reactor coolant shall be defined.*	The Pickering HTS systems have connected purification circuits as detailed in Part 2 of the Safety Report Section 5 [B.24-54], [B.24-55] for Pickering 1,4 and 5-8 respectively. Limits for radioactive content of the HTS are provided in the HTS Operational Safety Requirements [B.24-36] and [B.24-37]. Therefore, Pickering is compliant with this clause.	**Compliant**
8.2.4 Removal of residual heat from reactor core *The design shall provide a means (i.e., backup) of removing residual heat from the reactor for all conditions of the RCS. The backup shall be independent of the configuration in use.*	Pickering units have several residual heat removal mechanisms as detailed in the Safety Reports–Part 2 [B.24-54], [B.24-55]. These are: • SRVs with Boilers supplied by inventory from normal or auxiliary feed water or Pickering 1,4 EBWS or Pickering 5-8 EWS. • Shutdown Cooling Heat Exchangers supplied by normal or emergency high pressure service water. • Emergency heat removal with High Pressure Emergency Coolant Injection (HPECI) and ECI recovery, with Reactor Building Air Coolers. • Additionally, Pickering 5-8 has emergency heat removal with EWS to the HTS. Pickering 1,4	**Compliant**

REDOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
	has a Firewater supply to either the HTS or Moderator. Therefore, Pickering is compliant with this clause.	
8.3.1 Steam lines *The steam piping up to and including the turbine generator governor valves and, where applicable, the steam generators shall allow sufficient margin to ensure that the appropriate design limits of the pressure boundary are not exceeded in operational states and DBAs. This provision shall take into account the operation of control and safety systems.* *The main steam isolation valves (MSIVs) shall be installed in each of the steam lines leading to the turbine, and located as close as practicable to the containment structure.*	Pickering satisfies all requirements in this clause except that the units do not have Main Steam Isolation Valves. For both Pickering 1,4 and Pickering 5-8, Safety Analysis for Main Steam Line Breaks has been performed with the assumption that the break is not isolated, Appendix 7 of [B.24-56] and [B.24-57] for Pickering 1,4 and Pickering 5-8 respectively. This analysis assumption maximizes the harsh environment (temperature, pressure and humidity) conditions in the Powerhouse [B.24-104] and [B.24-105]. The environmental qualification and qualified success path is established based on these bounding conditions. The reactor safety consequences have been demonstrated to be acceptable in the absence of break isolation. Therefore, this issue is not considered to be safety significant. Additionally, Pickering has performed Containment analysis with steam line breaks inside containment without credit for isolation and demonstrated acceptable consequences. While MSIVs could also be used to establish a secondary containment if the primary boundary fails (e.g., boiler tube(s), steam line), boiler tube failures can be isolated by existing boiler isolating valves while the closed main steam supply loop remains intact as a boundary for pipe breaks inside containment. Finally, as noted in REGDOC-2.5.2, remote manual action cannot be credited before 30 minutes, therefore manual closure of MSIVs would have minimal benefit in terms of reducing consequences. Automatic closure of MSIVs would need to have sufficient reliability both for closure and avoidance of spurious closure.	**Compliant**

REGDOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
	Based on the above, the provision of MSIVs is not significantly beneficial from a reactor safety perspective and the absence of MSIVs has acceptable safety consequences. Therefore, Pickering is compliant with this clause.	
8.3.2 Steam and feedwater system piping and vessels *All piping and vessels shall be typically separated from electrical and control systems, to the extent practicable…*	Pickering 5-8 credits the Group 2 systems for Steam and Feedline failures. Pickering 1,4 has hardened and/or protected systems credited for Steam and Feedline failures. Critical control (including personnel) and electrical equipment for the main control rooms has been protected by providing a steam protection barrier between the Reactor Auxiliary Bay and the Turbine Auxiliary Bay. Therefore, Pickering is compliant with this clause.	**Compliant**
8.3.3 Turbine generators *The design shall provide over-speed protection systems for the turbine generators to minimize the probability of turbine disk failure leading to generation of missiles.* *The design shall be such as to minimize the potential for any missiles from a turbine break-up striking the containment, or striking other SSCs important to safety.*	This clause 8.3.3 was a gap for Darlington [B.24-21] (Issue D278) [B.24-38]. This issue was closed because failure resulting in missiles was explicitly analyzed and demonstrated acceptable. This was previously assessed in C-6 R1 reviews for Pickering Units 1,4 and Units 5-8 and is addressed in the PSR2 Hazard Analysis Safety Factor report and the Hazard Analysis screening [B.24-72], [B.24-73] for Pickering 1,4 and Pickering 5-8 respectively. These assessments are consistent with the 'minimize probability' requirement, hence, this is not a PSR2 gap.	**Compliant**
8.4 Means of shutdown *The design shall provide means of reactor shutdown capable of reducing reactor power to a low value, and maintaining that power for the required duration,…* *The design shall include two separate, independent, and diverse means of shutting down the reactor.*	Pickering 5-8 has two fully independent and diverse shutdown systems and is compliant with this requirement. Pickering 1,4 has two means of reactor shut down with independent parameters and logic using SDSA and SDSE. The shutoff rods and	**Compliant** Addressed in other PSR2 code review.

REDGOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
At least one means of shutdown shall be independently capable of quickly rendering the nuclear reactor subcritical from normal operation in AOOs and DBAs, by an adequate margin, on the assumption of a single failure. For this means of shutdown, a transient recriticality may be permitted in exceptional circumstances if the specified fuel and component limits are not exceeded.	dump valves are initiated from either of the two logic trains. This configuration was an acceptable deviation for PSR1 and was assessed for PSR2 as part of the CSA 290.1 [B.24-78] standard review. Pickering is compliant with this clause for PSR2.	
8.4 Means of shutdown (Con't) *…* *Means shall be provided to ensure that there is a capability to shut down the reactor in DECs, and to maintain the reactor subcritical even for the most limiting conditions of the reactor core, including severe degradation of the reactor core….*	Light water addition is available for HTS or Moderator addition at both Pickering 1,4 and Pickering 5-8 from installed systems or EME water. This was addressed in SAMG via the SAG-2 guideline, hence, this is not a PSR2 gap.	**Compliant**
8.4.1 Reactor trip parameters *The design authority shall specify derived acceptance criteria for reactor trip parameter effectiveness for all AOOs and DBAs, and shall perform a safety analysis to demonstrate the effectiveness of the means of shutdown.* *For each credited means of shutdown, the design shall specify a direct trip parameter to initiate reactor shutdown for all AOOs and DBAs in time to meet the respective derived acceptance criteria. Where a direct trip parameter does not exist for a given credited means, there shall be two diverse trip parameters specified for that means.* *For all AOOs and DBAs, there shall be at least two diverse trip parameters unless it can be shown that failure to trip will not lead to unacceptable consequences.*	AOO Safety Analysis has not been completed for Pickering and this has been identified as a PSR2 gap. However, the issue of AOO analysis is identified under the Deterministic Safety Analysis Safety Factor review and the code review for CNSC REGDOC-2.4.1. Therefore, this is not repeated as a PSR2 gap against REGDOC-2.5.2.	Addressed in other PSR2 code reviews. PSR2 REGDOC-2.4.1 Gap
8.4.2 Reliability *The design shall permit ongoing demonstration that each means of shutdown is being operated and maintained in a manner that ensures continued adherence to reliability and effectiveness requirements…*	Compliance with this requirement is evidenced by the shutdown system reliability target, actual past unavailability and predicted future unavailability as documented in the Annual Reliability report [B.24-76] relating to shutdown systems. Therefore, Pickering is compliant with this clause.	**Compliant**
8.4.3 Monitoring and operator action *Once automatic shutdown is initiated, it shall be impossible for an operator to prevent its actuation.*	This is the same requirement that is in Clause 4.3.5.4 of CSA N290.1, "Requirements for the Shutdown	**Compliant**

REDGOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
	Systems" [B.24-78], which was reviewed as part of PSR2 and was compliant. Once a 2 of 3 channel trip seals-in, a trip cannot be reset. The trip seal-in timer is provided to avoid spurious trips, however, its seal-in time is so short (<350 msec) [B.24-106], [B.24-107], that it is essentially impossible for an operator to intervene. Additionally, operator training and expectations dictate that they never interfere with automatic shutdown system action. Pickering is compliant with this clause.	
8.5 Emergency core cooling system (ECCS) *All ECCS components that may contain radioactive material shall be located inside containment or in an extension of containment. ECCS piping in an extension of containment that may contain radioactivity from the reactor core shall be subject to the following requirements:* *1. As a piping extension to containment, it meets the requirements for metal penetrations of containment.* *2. All piping and components of the ECCS recovery flow path piping that are open to the containment atmosphere are designed for a pressure greater than the containment design pressure.* *3. All ECCS recovery flow paths are housed in a confinement structure which prevents leakage of radioactivity to the environment and to adjacent structures.* *4. This housing includes detection capability for leakage of radioactivity, and the capability to either return the radioactivity to the flow path, or to collect the radioactivity and store (or process it) in a system designed for this purpose.* *Intermediate or secondary cooling piping loops shall have leak detection, whether the ECCS recovery system is inside or outside of containment, with the leak detection being such that upon detection of radioactivity from the ECCS recovery flow, the loops can be isolated as per the requirements for containment isolation.*	The ECI systems have been assessed as part of PSR2 in a review of CSA N290.2, "Requirements for Emergency Core Cooling Systems of Nuclear Power Plants" [B.24-110], and a gap was identified relating to recovery strainer instrumentation. All of ECI in Pickering 1,4 is inside containment including the Heat Exchangers (HXs). Because the HXs are common to the Moderator system, leak detection is available. Therefore, Pickering 1,4 is compliant. Pickering 5-8 has the ECI recovery piping, pumps and HXs outside of Containment. Components penetrating and outside containment are all DBE qualified and nuclear Class 2 [B.24-108], in accordance with the Design Guide [B.24-109]. The system has leakage collection, recovery and radiation monitoring in the vicinity. However, there is no direct leak detection for an ECI recovery HX tube leak. This clause is therefore a PSR2 Gap.	**PSR2 REGDOC-2.5.2 Gap #9** (Detection/Isolation of ECI HX Tube Leak) Safety Factor 1 (Plant Design)

REGDOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
8.6 Containment There are twelve sub-clauses relating to Containment. These are: 8.6.1 General 8.6.2 Strengthen (sic) of the containment structure 8.6.3 Capability for pressure test 8.6.4 Leakage 8.6.5 Penetrations 8.6.6 Isolations 8.6.7 Airlocks 8.6.8 Internal Structures 8.6.9 Containment Pressure and Energy Management 8.6.10 Control and cleanup of containment atmosphere 8.6.11 Coverings, coatings and materials 8.6.12 Design extension conditions	A review of CSA N290.3, "Requirements for the Containment System" [B.24-111], has been performed as part of PSR2. This review also includes assessing the requirements that were only applicable to a new plant. Only one gap was identified relating to the Energy Management System (EMS) and Phase 2 EME not being fully implemented. The approach to the assessment of the sub-section requirements is to only evaluate incremental requirements that were not assessed in the CSA 290.3 review. These are addressed in the subsections below.	**Compliant**
8.6.1 General *Leakage rate limits* …. **Guidance** *A modern containment should be able to achieve a leakage rate less than 0.5% containment air mass per day at the maximum containment pressure from any DBA. For example, modern designs achieve a maximum leakage rate of 0.1% to 0.5% containment air mass per day at design pressure.*	Pickering demonstrates compliance with the licence and safety leakage limits in the Operating Policies & Principles (OP&Ps) and the Containment Operational Safety Requirements [B.24-89], [B.24-90]. These are 2% Vol/hr for Pickering 5-8 and 2.7% Vol/hr for Pickering 1,4 at design pressure of 41.4 kPa(g). All units use an operational leakage target of <1% Vol/hr. However, Pickering has a Negative Pressure Containment System (NPCS). Following a DBA which requires the NPCS to activate, pressure in the accident Reactor Building is quickly reduced and controlled at a pressure dictated by the Instrumented Pressure Relief Valves (e.g., at a Reactor Building pressure <~4.5 kPa(g)) [B.24-89], [B.24-90]. The guidance provided in this clause is applicable to a new single unit plant with a lined (typically steel lined) containment. Such a unit may be predicted to have significant prolonged high pressure in containment for DBAs. Hence, the leakage rates referred to are not	**Compliant**

REGDOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
	applicable to a multi-unit design with NPCS. Therefore, this issue is not a PSR2 gap.	
8.6.5 Containment penetrations *The number of penetrations through the containment shall be kept to a minimum.* *All containment penetrations shall be subject to the same design requirements as the containment structure itself, and shall be protected from reaction forces stemming from pipe movement or accidental loads, such as those due to missiles generated by external or internal events, jet impact, and pipe whip.* *All penetrations shall be designed to allow for periodic inspection and testing.* *If resilient seals such as elastomeric seals, electrical cable penetrations, or expansion bellows are used with penetrations, they shall have the capacity for leak testing at the containment design pressure. To demonstrate continued integrity over the lifetime of the plant, this capacity shall support testing that is independent of determining the leak rate of the containment as a whole.*	Darlington was compliant for Clause 8.6.5 of RD-337 [B.24-21]. Darlington was designed and built with testable penetrations but it does not rely on them for in service leak detection. Per the RD-337 Compliance Discussion for Clause 8.6.4 [B.24-21], Darlington relies on the operational leak rate testing to confirm containment is sufficiently leak tight. With very few exceptions, Pickering does not have testable penetrations. Similar to Darlington, Pickering relies on operational leak rate testing and PIP testing to confirm the leak tightness of containment. Operational pressure testing of the containment envelope is conducted via several, sometimes overlapping, tests identified in the Appendices of [B.24-76]. As part of CSA N287.7 compliance, full Reactor Building Pressure tests are required every 6 years and a Negative Pressure Containment pressure test is required every 10 years [B.24-2]. A review of Pickering 5-8 against CSA N285.3-88 is included in Reference [B.24-10]. For CSA N285.3 Clause 7.8.4 relating to penetration testing, Pickering 5-8 was compliant. Given the above, although Pickering does not have individually testable electrical penetrations, this is considered to be acceptable based on the suitable alternate surveillance/testing being performed to assure the safety function is maintained. Therefore, this issue is not a PSR2 gap.	**Compliant**

REGDOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
8.6.8 Internal structures of the containment *The design shall provide for ample flow routes between separate compartments inside the containment. The openings between compartments shall be large enough to prevent significant pressure differentials which may cause damage to load-bearing and safety systems during AOOs, DBAs and DECs.* *The design of internal structures shall consider the hydrogen control strategy, and assist in the effectiveness of that strategy.…*	Clause 8.6.8 was a gap for RD-337 for Darlington [B.24-21], relating to treatment of hydrogen. Treatment of hydrogen in containment has been addressed for Pickering per the FAI [B.24-12] closure which included the installation of Passive Autocatalytic Recombiners and improvements to SAMG hydrogen determination. Hydrogen produced for DBA is addressed by hydrogen igniters, PARs and by assessment of standing flame consequences. Relief paths between compartments inside the Pickering Reactor Buildings, the Pressure Relief Duct and Vacuum Building were addressed in the design for DBAs and limit the pressure differentials across credited structures and equipment. The credited relief and pressure equalization features are referenced in the Containment OSRs (References [B.24-89] and [B.24-90]) and in the 21000 USI series of Design Manuals. DECs do not introduce incremental requirements relating to flowpaths and pressure equalization. Limiting static differential pressure due to water level is addressed in BDBA and SAM guidelines. Therefore, this is not a PSR2 gap.	**Compliant**
8.6.9 Containment pressure and energy management *The design shall enable heat removal and pressure reduction in the reactor containment in operational states, DBAs and DECs. Systems designed for this purpose shall be treated as part of the containment system, and are capable of:* *1. minimizing the pressure-assisted release of fission products to the environment* *2. preserving containment integrity* *3. preserving required leak tightness*	The introduction of DEC is a new requirement and the energy management system (EMS) has been addressed in other PSR2 code reviews. There was a gap in the CSA N290.3 "Requirements for the Containment System" [B.24-111] and the REGDOC-2.3.2 "Accident Management" [B.24-44], because Phase 2 EME has not yet been fully implemented to support the EMS. This is a PSR2 gap.	Addressed in other PSR2 code reviews. PSR2 CSA N290.3-11 Gap PSR2 REGDOC-2.3.2 Gap

REGDOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
8.6.10 Control and cleanup of the containment atmosphere *The design shall provide systems to control the release of fission products, hydrogen, oxygen, and* *other substances into the reactor containment, as necessary, to:* *1. reduce the amount of fission products that might be released to the environment during an accident* *2. prevent deflagration or detonation that could jeopardize the integrity or leak tightness of the containment …*	Clause 8.6.10 was a gap for RD-337 for Darlington [B.24-21] relating to there not being hydrogen mitigation for DECs. PARs have been installed at Pickering, hence this is not a PSR2 gap.	**Compliant**
8.6.11 Coverings, coatings and materials *The coverings and coatings for components and structures within the containment shall be carefully selected, and their methods of application shall be specified to ensure fulfillment of their safety functions. …*	This issue was assessed for Pickering B ISR and classified as an AD. The conclusions are applicable to Pickering Units 1,4 (See discussion in Section B.24.2.1). This is not a PSR2 gap.	**Compliant**
8.6.12 Design extension conditions *Following onset of core damage, the containment boundary shall be capable of contributing to the reduction of radioactivity releases to allow sufficient time for the implementation of offsite emergency procedures. …* *The ability of the containment system to withstand loads associated with design extension conditions (DECs) shall be demonstrated in design documentation, and shall include the following considerations:* *1. various heat sources, including residual heat, metal-water reactions, combustion of gases, and standing flames* *2. pressure control* *3. control of combustible gases* *4. sources of non-condensable gases* *5. control of radioactive material leakage* *6. effectiveness of isolation devices* *7. functionality and leak tightness of airlocks and containment penetrations* *8. effects of the accident on the integrity and functionality of internal structures* *The design authority shall demonstrate that complementary design features have been incorporated that will:* *1. prevent a containment melt-through or failure due to the thermal impact of the core debris* *2. facilitate cooling of the core debris*	The SAMG related requirements have been closed for Pickering per Section B.24.2.1, above. The requirement that containment should not exceed the design leakage for DECs for a period (at least 24 hours per Guidance) is a new requirement. This imposes a requirement that containment venting should not be required for 24 hours. While this requirement can likely be achieved for DECs, where the Boilers remain an effective heatsink, it is likely unachievable if there is core damage with limited Containment energy management capability as identified in Clause 8.6.9 above. The Level 2 PSA addresses this issue from a risk perspective, however, it may not address the deterministic design requirement. Therefore, this Clause is a PSR2 gap relating to DEC containment leakage limits to support implementation of offsite emergency measures.	**PSR2 REGDOC-2.5.2 Gap #3** (Containment Leak Tightness for DECs) Safety Factor 1 (Plant Design)

REDGOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
3. minimize generation of non-condensable gases and radioactive products *4. preclude unfiltered and uncontrolled release from containment* ***Guidance*** *The containment leakage rate in DECs with core damage should not exceed the design leakage rate for a sufficient period to allow for the implementation of offsite emergency measures. This period should be demonstrated, with reasonable confidence, to be at least 24 hours…*		
8.8 Emergency heat removal system *The design shall include an emergency heat removal system (EHRS) which provides for removal of residual heat in order to meet fuel design limits and reactor coolant boundary condition limits…. If the design of the plant is such that the EHRS is required to mitigate the consequences of a DBA, then the EHRS shall be designed as a safety system. There shall be reasonable confidence that the EHRS will function during DECs, if required…*	Clause 8.8 was a gap for RD-337 for Darlington [B.24-21]. The Darlington gap related to a fire coincident with LOCA relying on the common Emergency Service Water System. This gap is not an issue for Pickering where fire water is supplied by diesel pumps (Pickering Units 1,4) [B.24-54] or High Pressure Service Water (Pickering Units 5-8) [B.24-55]. For Pickering 5-8, there is reasonable confidence that Emergency Power Supply (EPS) and EWS will be available as an EHRS for DECs. For Pickering 1,4, the EHRS relies on firewater or EME water supply to maintain fuel cooling for DECs. Therefore, Pickering is compliant with this clause.	**Compliant**
8.9 Electrical power systems *The design shall specify the required functions and performance characteristics of each electrical power system that provides normal, standby, emergency and alternate power supplies to ensure: …* Guidance… *The normal AC electrical power systems should have the capacity and capability to supply all plant electrical loads during operational states, DBAs and DECs.* *Normal AC power supplies should be designed to:* • *prevent deviations from normal operation*	A PSR2 code review has been performed for CSA N290.5 "Requirements for Electrical Power…" [B.24-112]. The only gap identified in the review related to treatment of AOOs, hence, this gap is not repeated in this review. DEC assessments have been performed for some EME power (i.e., EME generator connection and corresponding distribution). Provision has been established using existing distribution systems. The requirement for the 'normal AC electrical power' to have capability	Addressed in another PSR2 code review. PSR2 CSA N290.5-06 Gap

REDGOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
• *prevent single failures from impacting more than one redundant division of electrical power supply* *avoid preventable challenges to standby and emergency systems as a result of an electrical system disturbance, transient, or upset condition (e.g., turbine-generator trip)…*	for DBAs and DECs is new and seems to impose qualification for DEC conditions. (Note, the term 'normal' appears to be used to distinguish the existing plant systems from the 'Alternate AC power supply' in clause 8.9.3). The extent of qualification for the 'Normal AC power' is addressed in the clause below.	
8.9.1 Standby and emergency power systems *The standby and emergency power systems shall have sufficient capacity and reliability, for a specified mission time, and in the presence of a single failure to provide the necessary power to:* *1. maintain the plant in a safe shutdown state and ensure nuclear safety in DBAs and DECs* *2. support severe accident management actions* *Dedicated onsite fuel storage facilities shall have a sufficient quantity of fuel to operate standby and emergency power sources while supplying connected loads…*	The clause imposes DEC qualification on EPS (Pickering 5-8) and Class III (Pickering 1,4). Pickering 5-8 EPS and Emergency Power Generators (EPGs) have been assessed to have external DEC capability [B.24-113]. Pickering 1,4 Class III distribution and Standby Generators have been assessed for external DECs for a seismic event [B.24-84] and high wind [B.24-87]. However, vulnerabilities exist for high wind at DECs. Therefore, a probabilistic assessment has been performed to demonstrate acceptable consequences [B.24-87] . This approach acceptably addresses the DEC qualification for Pickering 1,4 power, and hence, this is not a PSR2 gap. Pickering is compliant with this clause.	**Compliant**
8.9.2 DC and uninterruptible power systems *The design of the direct current (DC) power systems and uninterruptible AC power systems (if applicable) shall specify operating mission times when performing the intended safety functions of the connected loads and meet the capacity requirements of section 7.10.…*	This is a new clause. Pickering is compliant and a PSR2 review for CSA N290.5 [B.24-112] has not identified any gaps relating to the Class I and II (AC and DC) power systems. Pickering is compliant with this clause.	**Compliant** Addressed in other PSR2 code review.
8.9.3 Alternate AC power supply *The electrical power system design shall include provisions for mitigating the complete loss of onsite and offsite AC power. This is accomplished by the use of onsite portable, transportable or fixed power sources or offsite portable or transportable power sources, or a combination of these…*	This clause imposes requirements on the 'Alternate' as opposed to 'Normal' power supply. 'Alternate' power is not addressed in CSA N290.5. The purpose of this system is to address "Station Blackout" (SBO). At Pickering, this is a BDBA involving total loss of all AC power. This imposes requirements on stored	**Compliant**

REDOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
	water systems for heatsinks with make-up from Emergency Mitigating Equipment (EME) water pumps supplying Boilers. The alternate power required for the SBO is for monitoring and limited control. This power is provided by EME power supplies involving portable power supplies and AC power from diesel generators. These power supplies are detailed in EME guidelines and in [B.24-60] and [B.24-61]. Pickering is compliant with this clause.	
8.10 Control facilities **8.10.1 Main control room** *The design shall provide for a main control room (MCR) from which the plant can be safely operated, and from which measures can be taken to maintain the plant in a safe state or to bring it back into such a state after the onset of AOOs, DBAs or DECs…* *The design of the MCR shall take ergonomic factors into account to provide both physical and visual accessibility to controls and displays, without adverse impact on health and comfort. This includes hardwired display panels as well as computerized displays, with the aim of making these displays as user-friendly as possible.*	Clause 8.10.1 had a RD-337 gap for Darlington [B.24-21] relating to Human Factors and cabling between the MCR and Common Secondary Control Area. Pickering 5-8 and 1,4 comply with all requirements in this clause relating to cabling. Pickering 5-8 has Unit Secondary Control Areas that are Group 2, remote and isolated from the MCR. Pickering 1,4 has remote monitoring and reactor trip capability in the SDSE Instrument Rooms. Similar to Pickering 5-8 the instrumentation and equipment in these rooms is remote from and physically isolated from the MCR. As identified for Clause 7.21 above, there is a gap relating to the systematic application of Human Factors Engineering principles into the original Pickering plant design.	**PSR2 REGDOC-2.5.2 Gap #8** (Human Factors in Design) Safety Factor 1 (Plant Design)
8.10.1.1 Safety parameter display system *The MCR shall contain a safety parameter display system (SPDS) that presents sufficient information on safety-critical parameters for the diagnosis and mitigation of DBAs and DECs.* *The SPDS shall have the following capabilities:* *1. display safety-critical parameters within the full range expected in operational states, DBAs and DECs …*	Both Pickering 5-8 and 1,4 have the equivalent to SPDSs. However, a new requirement relating to DECs qualification has been introduced. The safety parameter display systems (SPDSs) or critical safety parameter monitoring (CSPM) display are not qualified for DECs where there is a total loss of AC power. Further, the CSPM displays have not been qualified for events such as seismic and fire.	**PSR2 REGDOC-2.5.2 Gap #10** (Safety Parameter Display System Qualification for DECs) Safety Factor 1 (Plant Design)

REGDOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
The SPDS shall be designed and installed such that the same information is made available in a secure manner to the emergency response facility.	As part of the Fukushima follow-up, instrumentation to support critical parameters required to function for DECs has been evaluated for survivability in [B.24-114]. The instrument loops associated with these parameters have been identified for use in CSPM and BDBA accident procedures. The indications from these loops are not in one central location and, in some cases, require field action (e.g., power) to obtain data. Although this does not fully satisfy the requirements to have these parameters available from a SPDS in the MCR and Secondary Control Area (SCA) (clause below), it is considered to be an acceptable alternative. However, this is a PSR2 gap relating to the new plant requirement to have SPDS that is DEC qualified and with parameters available in the in the MCR and SCA. The CSPM displays are available in the emergency Site Management Center. Therefore, this is a PSR2 gap.	
8.10.2 Secondary control room *The design shall provide an SCR that is physically and electrically separate from the MCR, and from which the plant can be placed and kept in a safe shutdown state when the ability to perform essential safety functions from the MCR is lost…* *The SCR shall be equipped with an SPDS similar to that in the MCR. As a minimum, this display system shall provide the information required to facilitate placing and keeping the plant in a safe shutdown state when the MCR is uninhabitable…*	Clause 8.10.2 had a RD-337 gap for Darlington [B.24-21] relating to fire code compliance and cabling issue. This has been assessed for Pickering and is not a PSR2 gap. Pickering 5-8 has Unit Emergency Control Centers (UECCs) for each unit which are credited to address Common Mode and Harsh Environment events. These rooms are part of the Group 2 safety functions. A high level description of the functions of and controls in the UECCs is included in Section 7.5 of [B.24-55]. There are no SPDS displays in the UECCs. Pickering 1,4 MCRs have been hardened to provide protection for Seismic and Harsh Environment events. Therefore, functional credits associated with these events are addressed from the MCR and via	**Compliant**

REGDOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
	field actions. The Shutdown System Enhancement Instrument Rooms (SDSE IRs) provide diverse and separate shutdown logic and monitoring capability, independent of the MCR. For events where the MCR is uninhabitable, the SDSE IR, in conjunction with remote field actions (e.g., Steam Reject Valve control), provide equivalent capability to that required for a SCA. In addition, there is a SPDS display in the SDSE IR. Therefore, this clause does not represent a PSR2 gap for Pickering 1,4. Pickering Units 5-8 do have SCAs but there is no SPDS system in them. This issue is included in REGDOC-2.5.2 Clause 8.10.1.1 immediately above.	
8.10.3 Emergency support facilities *The design shall provide for onsite emergency support facilities that are separate from the plant control rooms for use by the technical support staff and emergency support staff in the event of an emergency…* *The design shall ensure that the emergency support facilities:* *1. includes provisions to protect occupants over protracted periods from the hazards resulting from DBAs and DECs* *2. is equipped with adequate facilities to allow extended operating periods…*	The Consolidated Nuclear Emergency Plan [B.24-62], establishes and governs the emergency facilities. The EITER program assures that EP provisions are available [B.24-67]. The BDBA standard [B.24-59] provides assurance that procedures and staff are capable of addressing BDBAs. The Emergency Operations Centre (EOC) is in the plant at a location remote from the MCR. There is a Site Management Center (SMC) outside the station boundary in close proximity to the plant. Back-up emergency support facilities are available at Darlington or the CEOF in Whitby. Hence, Pickering is PSR2 compliant.	**Compliant**
8.10.4 Credit for operator action *If operator action is required for actuation of any safety system or safety support system equipment, all of the following requirements shall apply:* *1. there are clear, well-defined, validated, and readily available operating procedures that identify the necessary actions* *2. there is instrumentation in the control rooms to provide clear and unambiguous indication of the necessity for operator action*	Credited operator actions are identified in Section 1 of Part 3 of the Safety Reports [B.24-56], [B.24-57] for Pickering 1,4 and Pickering 5-8 respectively. DBA event based response is per the Abnormal Incidents Manuals (AIM). For less severe events, operators may respond via Annunciation Response Manuals (ARMs) and/or Section 5 of Operating Manuals. All Emergency	**<u>PSR2 REGDOC-2.5.2 Gap #6</u>** (Allowable Times for Crediting On-Site Operator Actions) Safety Factor 1 (Plant Design)

REDGOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
3. following indication of the necessity for operator action inside the control rooms, there are at least 30 minutes available before the operator action is required *4. following indication of the necessity for operator action outside the control rooms, there is a minimum of 1 hour available before the operator action is required …* **Guidance** *If operator action is required for actuation of any safety function, other than meeting the requirements of this regulatory document, the analysis should also demonstrate that:* • *there is sufficient time available for the operator to perform the required manual action* *the operator can perform the actions correctly and reliably in the time available*	Operating Procedures in the AIMs are validated. As noted in Clause 7.10 above, the action timing requirements have changed from 15 and 30 min to 30 min and 1 hr. Assessments in the Safety Reports are based on 15 and 30 minutes. Therefore, Pickering is not compliant with this requirement. This gap has previously been identified against Clause 7.10 above. However, it should be noted that the guidance provides the option to perform an analysis to support an earlier credit. Given that the operator actions have all been validated using a combination of simulator exercises, tabletop walkthroughs, drills and testing, the existing operator action time credits are well supported. Therefore, it may be possible to reclassify aspects of this gap as an acceptable deviation.	
8.11 Waste treatment and control *The design shall include provisions to treat liquid and gaseous effluents in a manner that will keep the quantities and concentrations of discharged contaminants within prescribed limits, and that will support application of the ALARA principle.*	A high level description of Pickering waste treatment and control is provided in Section 13 of Part 2 of the Safety Report [B.24-54], [B.24-55] for Pickering 1,4 and Pickering 5-8 respectively. Reference [B.24-115] establishes the program that manages all aspects of radiological impact on the environment. A separate PSR2 Safety Factor review is performed for Radiological Impact on the Environment, Safety Factor 14 per [B.24-9]. Additionally, PSR2 reviews of the following standards relating to radioactive emission control and ALARA have been performed: • CNSC G-129 (2004), "Keeping Radiation Exposures and Doses 'As Low As Reasonably Achievable (ALARA)'" • CSA N288.1-14, "Guidelines for Calculating Derived Release Limits for Radioactive Material Airborne	**Compliant** Addressed in PSR2 Safety Factor and other PSR2 code reviews

REDGOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
	and Liquid Effluents for Normal Operation of Nuclear Facilities" • CSA N288.4-10, "Environmental Monitoring Program at Class I Nuclear Facilities and Uranium Mines and Mills" • CSA N288.5-11, "Effluent Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills" • CSA N288.6-12, "Environment Risk Assessments at Class I Nuclear Facilities and Uranium Mines and Mills" • CSA N288.3.4-13, "Performance Testing of Nuclear Air-Cleaning Systems at Nuclear Facilities" Pickering is compliant with this requirement.	
8.11.1 Control of liquid releases to the environment *The design shall include provisions to treat liquid and gaseous effluents in a manner that will keep the quantities and concentrations of discharged contaminants within prescribed limits, and that will support application of the ALARA principle.*	Addressed in clause immediately above.	**Compliant** Addressed in PSR2 Safety Factor and other PSR2 code reviews
8.11.2 Control of airborne material within the plant The clause details requirements for controlling airborne radioactive contamination within the plant. Additionally it details requirements for establishing radiological zoning.	Clause 8.11.2 had a RD-337 gap for Darlington [B.24-21] relating to an Off-gas management system. However, as addressed in Section B.24.2.1 above, this is not a gap for Pickering. A description of the ventilation and filtration systems is provided in Section 11 of Part 2 of the Safety Report [B.24-54], [B.24-55] for Pickering 1,4 and Pickering 5-8 respectively. The overriding program for control of radiation within the plant is the "Radiation Protection Requirements" [B.24-116]. The control of radioactive contamination in the plant is fully	**Compliant** Addressed in PSR2 Safety Factor and other PSR2 code reviews

REGDOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
	addressed in PSR2 Safety Factor 15 Radiation Protection [B.24-9].	
8.11.3 Control of gaseous release to the environment The clause details requirements for control of radioactive gas emissions and for removing radioactivity from release paths to atmosphere.	A description of the ventilation and filtration systems is provided in Section 11 of Part 2 of the Safety Report [B.24-54], [B.24-55] for Pickering 1,4 and Pickering 5-8 respectively. All elements of this clause are addressed in the clause 8.11 review above.	**Compliant** Addressed in PSR2 Safety Factor and other PSR2 code reviews
8.12 Fuel handling and storage *There shall be barriers to prevent the insertion of incorrect, defective or damaged fuel into the reactor.* *There shall be provisions to prevent contamination of the fuel and the reactor.* *The design shall meet the requirements found in CNSC RD-327, Nuclear Criticality Safety.*	A high level description of the Pickering fuel handling process and systems is included in Section 10 of Part 2 of the Safety Report [B.24-54], [B.24-55] for Pickering 1,4 and Pickering 5-8 respectively. The introduction of RD-327, "Nuclear Criticality Safety", is a new requirement. However, given that Pickering uses natural uranium fuel, criticality events while handling new and used fuel outside the reactor are not credible. Pickering is compliant with these requirements.	**Compliant**
8.12.1 Handling and storage of non-irradiated fuel	Addressed in Clause 8.12 above.	**Compliant**
8.12.2 Handling and storage of irradiated fuel *The design of the handling and storage systems for irradiated fuel shall:* *1. ensure nuclear criticality safety* *2. permit adequate heat removal in operational states, DBAs and DECs* *3. permit inspection of irradiated fuel…* *The design of irradiated fuel storage pools shall include means for preventing the uncovering of fuel in the pool in operational states, DBAs and DECs.* *The design for a water pool used for fuel storage shall include provisions for DECs by:* *1. ensuring that boiling in the pool does not result in structural damage* *2. providing temporary connections to enable the refill of the pool using temporary supplies*	The handling of used fuel is addressed in Clause 8.12 above. DEC is a new addition. However EME and SAMG response addresses loss of cooling and failures in the irradiated fuel bays per [B.24-12] and [B.24-53]. This addresses all DEC requirements. Failure of fuel in fueling machines or transfer mechanisms for DECs is bounded by DBA analysis already in the Safety Reports, Appendix 1 of [B.24-56] and [B.24-57], for Pickering 1,4 and Pickering 5-8 respectively.	**Compliant**

REGDOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
3. providing temporary connections to heat removal systems for power and cooling water *4. providing hydrogen mitigation in the spent fuel pool area* *5. ensuring that severe accident management actions related to the spent fuel pool can be carried out*	Pickering is compliant with these requirements.	
8.12.3 Detection of failed fuel *The design shall provide a means for allowing reliable detection of fuel defects in the reactor, and the subsequent removal of failed fuel, if action levels are exceeded.*	The presence of failed fuel is usually detected by chemistry sampling. Once elevated Iodine samples are detected, a review of fueling activities and transients is initiated. The "Reactor Physics Fuel Management and Core Surveillance" manual [B.24-117] details the systematic process for locating and removing failed/defective fuel. Pickering is compliant with these requirements.	**Compliant**
8.13 Radiation Protection *The design and layout of the plant shall make suitable provision to minimize exposure and contamination from all sources. This shall include the adequate design of SSCs to:* *1. control access to the plant* *2. minimize exposure during maintenance and inspection* *3. provide shielding from direct and scattered radiation* *4. provide ventilation and filtering to control airborne radioactive materials* *5. limit the activation of corrosion products by proper specification of materials* *6. minimize the spread of active material* *7. monitor radiation levels* *8. provide suitable decontamination facilities…*	A description of the Radiation Protection provisions is provided in Section 12 of Part 2 of the Safety Report [B.24-54], [B.24-55] for Pickering 1,4 and Pickering 5-8 respectively. The overriding program for control of radiation within the plant is the "Radiation Protection Requirements" [B.24-116]. The control of radioactive contamination in the plant is fully addressed in PSR2 Safety Factor 15 Radiation Protection. This addresses the heading and all sub-clauses except for 8.13.3 which is addressed below.	**Compliant** Addressed in PSR2 Safety Factor and other PSR2 code reviews
8.13.3 Radiation monitoring *Equipment shall be provided to ensure that there is adequate radiation monitoring in operational states, DBAs and DECs.* *Stationary alarming dose rate meters shall be provided:* *1. for monitoring the local radiation dose rate at places routinely occupied by operating personnel*	Pickering has alarming radiation monitoring equipment distributed within and outside containment in areas where there is a risk of elevated radiation. The locations of detectors, detection ranges and alarm levels are documented in [B.24-118] and [B.24-119]. The DECs requirement is a new addition that was not previously included in RD-337.	**Compliant**

REDGOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
2. where the changes in radiation levels may be such that access may be limited for periods of time *3. to indicate, automatically and in real-time, the general radiation level at appropriate locations in operational states, DBAs and DECs* *4. to give sufficient information in the control room or at the appropriate control location for operational states, DBAs and DECs, to enable plant personnel to initiate corrective actions when necessary.*	Pickering has installed an Automated Source Term Gamma Monitoring System (ASTGMS) for emergency and source term radiation monitoring. Its availability is managed via the EITER program [B.24-67]. ASTGMS is not qualified for all DECs in the event that power is lost. In the absence of ASTGMS availability, the Critical Safety Parameter Monitoring AIM provides a list of alternate indications, including manual radiation measurements. Given that there are alternate provisions available, this issue has low safety significance and is not considered to be a PSR2 gap.	
9. Safety Analysis **9.1 General** *A safety analysis of the plant design shall include hazard analysis, deterministic safety analysis, and probabilistic safety assessment (PSA) techniques. The safety analysis shall demonstrate achievement of all levels of defence in depth, and confirm that the design is capable of meeting the applicable expectations, dose acceptance criteria and safety goals…* *Radioactive sources other than the reactor core, such as the spent fuel pool and fuel handling systems, shall be considered. Impacts for multiple units at a site if applicable, shall be included.* *The first step of the safety analysis shall be to identify PIEs using a systematic methodology, such as failure modes and effects analysis. Both direct and indirect events shall be considered in PIE identification. Requirements and guidance for identification of PIEs is given in section 7.4 of this document.*	This clause identifies the high level requirements for hazard analysis, deterministic safety analysis, and probabilistic safety assessment, with which Pickering is in alignment. The second paragraph is new and introduces irradiated fuel bay (IFB) issues. PIEs have been identified consistent with REGDOCs-2.4.1 and 2.4.2, and identified in both the PSA initiating event tables and in the DSAs for Pickering [B.24-56] and [B.24-57]. IFB failures are addressed in the DSA and in BDBA assessments and response per [B.24-53]. Pickering is compliant with these requirements.	**Compliant**
9.2 Analysis objectives *The final safety analysis shall: …* *2. account for postulated aging effects on SSCs important to safety…* *8. demonstrate that the design incorporates sufficient safety margins*	This clause states high level objectives. Darlington was compliant with these requirements for RD-337 [B.24-21]. Given the generic application of safety analysis methodology for both Darlington and Pickering, the Darlington compliance assessment is applicable to PSR2. Items 2 and 8 are new additions. These are both addressed in the	Addressed in PSR2 Safety Factor and other PSR2 code reviews. PSR2 REGDOC-2.4.1 Gap.

REDGOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
	assessment of PSR2 for REGDOC-2.4.1 [B.24-14], and in the PSR2 Safety Factor 5 review for Deterministic Safety Analysis.	
Guidance The *Class I Nuclear Facilities Regulations* requires a preliminary safety analysis report demonstrating the adequacy of the NPP design to be submitted in support of an application for a licence to construct a Class I nuclear facility. A final safety analysis report demonstrating the adequacy of the design is required for an application for a licence to operate a Class I nuclear facility.	This is a new requirement. Final Safety Analysis Reports (FSARs) are not used for existing plants. The applicable constituents of the FSAR are referenced in the Licence [B.24-2] and the LCH [B.24-3], and include the OP&Ps, SOE, DSA, PSA, Radiation Protection, etc. Therefore, Pickering is compliant with the intent of this requirement.	**Compliant**
9.3 Hazard analysis **Guidance** The objective of the hazard analysis is to determine the adequacy of protection of the NPP against internal and external hazards, while taking into account the plant design and site characteristics. To ensure the availability of required safety functions and operator actions, all the SSCs important to safety (including the main control room, secondary control room and emergency support facilities) should be adequately protected against relevant internal and external hazards…	This is a new requirement. Hazard Analysis is addressed in the PSR2 Safety Factor 7 report. This clause required there to be a specific Hazard Analysis report for the plant. The required contents of the report are addressed in various other reports for Pickering such as the Safety Report, Risk Assessments, Fire Safe Shutdown Analysis, Seismic Margin Assessment, etc. It is concluded that Pickering is compliant with the intent of this clause.	Addressed in PSR2 Safety Factor review.
9.4 Deterministic safety analysis The deterministic safety analysis shall be conducted in accordance with the requirements specified in CNSC regulatory document REGDOC-2.4.1, *Deterministic Safety Analysis*.	Reference to REGDOC-2.4.1 [B.24-14] is a new addition. A specific code review is being performed for this standard as part of PSR2.	Addressed in PSR2 code reviews.
9.5 Probabilistic safety assessment The probabilistic safety assessment shall be conducted in accordance with the requirements specified in CNSC REGDOC-2.4.2, *Probabilistic Safety Assessment (PSA) for Nuclear Power Plants*.	Reference to REGDOC-2.4.2 [B.24-15] is a new addition. A specific code review is being performed for this standard as part of PSR2.	Addressed in PSR2 code review.
10.1 Design for environmental protection **Guidance** The design should incorporate the "best available technology and techniques economically achievable" (BATEA) principle for aspects of the design related to environmental protection.	This is a new requirement in REGDOC-2.5.2. However, environmental requirements with no potential radiological implications are not part of PSR2 scope. Per the PSR Basis Document [B.24-9], Safety Factor 14 only deals with radiological impact on the environment.	n/a

REDGOC-2.5.2 Clauses (text in italics is verbatim from the code)	**PSR2 Review**	**Gap or Compliance?**
10.2 Release of nuclear and hazardous substance Pollution prevention principles shall be applied when considering the technological design options for cooling water systems, in order to minimize adverse environmental impact.	See clause immediately above. For radiological impact on the environment refer to PSR2 Safety Factor 14 review and assessment under clause 8.11 above.	See Clause 10.1 above
11. Alternative Approaches The requirements in this regulatory document are intended to be technology neutral for watercooled reactor designs. It is recognized that specific technologies may use alternative approaches…	For information, only.	n/a
Appendix A: Structural Analysis of Containment Structures	New appendix Appendix A was added to outline acceptance criteria when performing structural analysis of containment structures for robustness against malevolent acts. Treatment of malevolent acts is not within the PSR2 review scope.	n/a

B.24.3 Compliance Summary for Pickering PSR2

There are ten PSR2 CNSC REGDOC-2.5.2 (2014) gaps. The gaps and their associated Safety Factor are identified below:

1. Safety Factor 5 (Deterministic Safety Analysis) – Clauses 4.2.1, 6.4 and 7.3 of REGDOC-2.5.2 introduce new requirements and limits for Anticipated Operational Occurrences (AOOs), Design Basis Accidents (DBAs) and Beyond Design Basis Accidents (BDBAs) and include specific dose limits for AOOs and DBAs. Current Pickering Safety Report analyses do not identify and classify events into these categories. Dose limits currently used in Pickering are aligned with the single failure / dual failure limits in accordance with the Pickering Licence Conditions Handbook. This issue has therefore been identified as a PSR2 gap. It is being addressed as part of REGDOC-2.4.1 implementation.

2. Safety Factor 6 (Probabilistic Safety Assessment) – Clause 4.2.2 of REGDOC-2.5.2 introduces new requirements and limits for probabilistic analysis risk limits, such as a core damage frequency limit of $<10^{-5}$ yrs/yr. It has not been demonstrated that these requirements can be achieved. Therefore, this has been identified as a PSR2 gap.

3. Safety Factor 1 (Plant Design) – Containment Leak Tightness for Design Extension Conditions (DECs): Clauses 7.3 and 8.6.12 of REGDOC-2.5.2 require containment to provide a leak tight barrier following DECs with severe core damage for a period sufficient to implement off-site emergency measures. REGDOC-2.5.2 guidance suggests this period be at least 24 hours. Such a requirement does not exist in Beyond Design Basis Accident (BDBA)/Severe Accident (SA) mitigation, so this represents a PSR2 gap.

4. Safety Factor 1 (Plant Design) – On-Demand Reliability of Safety Systems: Clause 7.6 of REGDOC-2.5.2 requires all Structures, Systems, and Components (SSCs) important to safety (SIS) to meet an on-demand failure rate of $<10^{-3}$ yrs/yr. This requirement is not met for several systems including Pickering 1,4 Emergency Coolant Injection (ECI) and is therefore identified as a PSR2 gap.

5. Safety Factor 1 (Plant Design) – Sharing of Safety Systems and Turbine Hall: Clause 7.6.5 of REGDOC-2.5.2 has a new requirement that sharing of safety systems and the turbine generator building not be permitted. Pickering Units share Emergency Coolant Injection (ECI) and Negative Pressure Containment (NPC), as well as the turbine hall; therefore, this has been identified as a PSR2 gap.

6. Safety Factor 1 (Plant Design) – Allowable Times for Crediting On-Site Operator Actions: Clauses 7.10 and 8.10.4 of REGDOC-2.5.2 establish new time limits for crediting operator actions, i.e., 30 minutes for Main Control Room (MCR) actions and 1 hour for field actions. Pickering NGS has not demonstrated that deterministic safety analysis consequences are acceptable if MCR and field action are not credited for these times respectively. Therefore, this has been identified as a PSR2 gap.

7. Safety Factor 1 (Plant Design) – Seismic Qualification and Design: Clause 7.13.1 of REGDOC-2.5.2 requires that Beyond Design Basis (BDB) Earthquake seismic margin be a factor of 1.67 beyond that required for the new plant Design Basis Earthquake (DBE). Fragility evaluations were completed for seismic mitigating Structures, Systems and Components (SSCs), however, based on available information it could not be confirmed that the new plant BDB Earthquake margin of 1.67 would be achieved. Therefore, this has been identified as a PSR2 gap.

8. Safety Factor 1 (Plant Design) – Human Factors in Design: Clauses 7.21 and 8.10.1 of REGDOC-2.5.2 introduce new requirements for the systematic application of Human Factors Engineering (HFE) principles to plant design. Many years of safe and reliable operating experience indicate that the design and processes for integration of human interactions with the plant were and remain robust. However, Pickering plant design predates the current requirements for incorporating HFE into the design and the existing plant has not been systematically demonstrated to meet the requirements for a new plant. Therefore, this has been identified as a PSR2 gap.

9. Safety Factor 1 (Plant Design) – Detection/Isolation of Emergency Coolant Injection (ECI) Heat Exchanger Tube Leak: Clause 8.5 of REGDOC-2.5.2 requires ECI recovery heat exchanger tube leak detection capability. Pickering Units 5-8 ECI recovery heat exchangers do not have leak detection capability on the cooling water side. Therefore, this has been identified as a PSR2 gap.

10. Safety Factor 1 (Plant Design) – Safety Parameter Display System Qualification for Design Extension Conditions (DECs): Clause 8.10.1.1 of REGDOC-2.5.2 requires the Main Control Room (MCR) to contain a Safety Parameter Display System (SPDS) that presents sufficient information on safety-critical parameters for the diagnosis and mitigation of Design Basis Accidents (DBAs) and DECs. The SPDSs are to be qualified for DEC and have parameters available in both the MCR and Secondary Control Areas (SCA), per Clause 8.10.2. Pickering SPDSs are not Review Level Condition (RLC) qualified or available in all locations. As part of the Fukushima follow-up, instrumentation to support critical parameters required to function for DECs has been evaluated for survivability. The instrument loops associated with these parameters have been identified for use in Critical Safety Parameter Monitoring (CSPM) and Beyond Design Basis Accident (BDBA) procedures. However, the indications from these loops are not in one central location and, in some cases, require field action (e.g., power) to obtain data. This does not fully satisfy the requirements to have these parameters available from a SPDS in the MCR and SCA. Therefore, this has been identified as a PSR2 gap relating to the new plant requirement to have SPDS that is DEC qualified and with parameters available in the MCR and SCA.

B.24.4 References

[B.24-1] CNSC Regulatory Document REGDOC-2.5.2, *Design of Reactor Facilities: Nuclear Power Plants*, May 2014.

[B.24-2] CNSC, PROL-48.02/2018 Licence, *Nuclear Power Reactor Operating Licence: Pickering Nuclear Generating Station*, December, 2015.

[B.24-3] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.24-4] CNSC Regulatory Document RD-337, *Design of New Nuclear Power Plants*, November 2008.

[B.24-5] CNSC Regulatory Document RD-337 version 2, *Design of New Nuclear Power Plants*, July 2012 DRAFT.

[B.24-6] CNSC Guidance Document GD-337, *Guidance for the Design of New Nuclear Power Plants,* September 2012 DRAFT.

[B.24-7] CSNC website, *Archived - CNSC publishes REGDOC-2.5.2, Design of Reactor Facilities: Nuclear Power Plants*, http://news.gc.ca/web/article-en.do?nid=851769, May 28, 2014.

[B.24-8] CNSC INFO-0824, *CNSC Fukushima Task Force Report*, October 2011.

[B.24-9] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.24-10] OPG Report, NK30-REP-03680-00001 R000, *Pickering NGS-B Integrated Safety Review – Plant Design Safety Factor*, August 2007.

[B.24-11] OPG Report, NK30-REP-03680-00015 R000, *Pickering NGS-B Integrated Safety Review (ISR) – Final ISR Report*, August 2009.

[B.24-12] OPG Report, NK30-REP-03680-00005, *Pickering NGS B – Integrated Safety Review – Safety Analysis Review*, June 2007.

[B.24-13] OPG Letter, Woods to Santini & Rinfret, N-CORR-00531-06906, *OPG Progress Report No. 7 on CNSC Action Plan – Fukushima Action Items,* November 30, 2015.

[B.24-14] CNSC Regulatory Document REGDOC-2.4.1, *Deterministic Safety Analysis*, May 2014.

[B.24-15] CNSC Regulatory Document, REGDOC-2.4.2, *Probabilistic Safety Assessment (PSA) for Nuclear Power Plants*, May 2014.

[B.24-16] OPG Plan, NK30-PLAN-00531-00001 R000, *Pickering B Continued Operations Plan,* September 23, 2010.

[B.24-17] CNSC G-323, *Ensuring the Presence of Sufficient Qualified Staff at Class I Nuclear Facilities - Minimum Shift Complement*, July 2007*.*

[B.24-18] OPG Instruction, P-INS-09100-00003 R009, *Pickering Minimum Shift Complement*, December 2014.

[B.24-19] CNSC Letter, E-Docs # 3752906/4.01.02, OPG File No. P-CORR-00531-03640 R000, M. Santini to G. Jager, *OPG Request for Concurrence with Minimum Shift Complement documents, P-INS-09100-00003 and P-INS-09260-0008, Action Items 2004-4-09, 2004-8-10 and 2006-4-01*, August 9, 2011.

[B.24-20] CSA Standard, N290.0-11, *General Requirements for Safety Systems of Nuclear Power Plants*, October 2011.

[B.24-21] OPG Report, NK38-REP-03680-10109 R000, *Review of RD-337 Design of New Nuclear Power Plants for Darlington Integrated Safety Review*, September 2011.

[B.24-22] OPG Report, NK38-REP-03680-10024 R000, *Review of IAEA NS-R-1 (October 2000) Safety of Nuclear Power Plants: Design for Darlington Integrated Safety Review*, September 2011.

[B.24-23] OPG Report, NK38-REP-03680-10209 R000, *Code Refresh Review of IAEA SSR-2/1 (2012) Safety of Nuclear Power Plants: Design*, February 2014.

[B.24-24] OPG Report, NK38-REP-03680-10104 R000, *Darlington NGS Integrated Safety Review (ISR) – Final ISR Report*, October 2011.

[B.24-25] OPG Report, NK38-REP-00770-0417591 R001, *DARA Scope and Completion - Issue # D025,* February 8, 2013.

[B.24-26] OPG Report, NK38-REP-00770-0421330 R001, *Nuclear Refurbishment Issue Resolution Form – Darlington – Issue #D040 – Human Factors Design*, November 2012.

[B.24-27] OPG Report, NK38-REP-00770-0425837 R001, *Nuclear Refurbishment Issue Resolution Form – Darlington – Issue #D260, Human Factors Annunciation Improvements,* January, 2013.

[B.24-28] OPG Report, NK38-REP-00770-0425541 R002, *Nuclear Refurbishment Issue Resolution Form – Darlington – Issue #D143 – Severe Accident and Beyond Design Basis Accident (BDBA) Program/ SAMG,* November 2013.

[B.24-29] OPG Report, NK38-REP-00770-0426309 R003, *Nuclear Refurbishment Issue Resolution Form – Darlington – Issue #D279 – Cabling Between Main Control Room and Secondary Control Area*, July 2013.

[B.24-30] OPG Report, NK38-REP-00770-0462013 R003, *Nuclear Refurbishment Issue Resolution Form – Darlington – Issue #D430 – CNSC Acceptance of FHA and FSSA,* April 15, 2015.

[B.24-31] OPG Guide, NK30-DG-30-68000-00005 R000, *Engineering Design Guide – Pickering G. S. B Location and Separation Requirements for Special Safety Systems*, September 1976.

[B.24-32] OPG Report, NK30-REP-71400-10002 R002, *Fire Hazard Assessment – Pickering B Nuclear Generating Station*, November 2011.

[B.24-33] OPG Report, NK30-REP-71400-00001 R002, *Fire Safe Shutdown Analysis – Pickering B Nuclear Generating Station,* October 2011.

[B.24-34] CSA N293-12, *Fire Protection for Nuclear Power Plants,* October 2012.

[B.24-35] OPG Report, NK38-REP-00770-0426307 R000, *Nuclear Refurbishment Issue Resolution Form – Darlington – Issue #D277 – Primary Heat Transport Inventory for Multi Unit Shutdown*, May 2011.

[B.24-36] OPG Manual, NA44-OSR-08131.02-00013-R02, *Pickering NGS A Operational Safety Requirements: Heat Transport System*, October 2009.

[B.24-37] OPG Manual, NK30-OSR-08131.02-00013-R02, *Pickering NGS B Operational Safety Requirements: Heat Transport System*, February 2011.

[B.24-38] OPG Report, NK38-REP-00770-04216308 R001, *Nuclear Refurbishment Issue Resolution Form – Darlington – Issue #D278 – Turbine Orientation*, April 2011.

[B.24-39] OPG Report, NK38-REP-00770-0425807 R001, *Nuclear Refurbishment Issue Resolution Form – Darlington – Issue #D225 – Fire Protection Water Supply,* July 2013.

[B.24-40] OPG Report, NK38-REP-00770-0426310 R000, *Nuclear Refurbishment Issue Resolution Form – Darlington – Issue #D280 – Off-Gas Management System,* April 2011.

[B.24-41] OPG Letter, R. Strickert and W. Robinson to B. Parsons, P-CORR-00531-00693, *Offgas Management System, Pickering A & B – Abandon System In-place,* July 5, 2000.

[B.24-42] OPG Memorandum (E-Mail communication) to CNSC, R. MacEacheron to M. Dallaire, N-CORR-00531-06294, *OPG Comments on Draft REGDOC-2.5.2, Design of Reactor Facilities: Nuclear Power Plants*, October 21, 2013.

[B.24-43] OPG Program, N-PROG-RA-0016 R009, *Risk and Reliability Program*, June 2016.

[B.24-44] CNSC Regulatory Document REGDOC-2.3.2, *Accident Management*, Version 2, September 2015.

[B.24-45] OPG Nuclear Charter, N-CHAR-AS-0002 R019, *Nuclear Management System*, August 2016.

[B.24-46] CSA Standard N290.15-10, *Requirements for the Safe Operating Envelope of Nuclear Power Plants*, August 2010.

[B.24-47] CNSC Regulatory Document REGDOC-2.10.1, *Nuclear Emergency Preparedness and Response, Version 2*, February 2016.

[B.24-48] OPG Nuclear Program, N-PROG-MP-0007 R012, *Conduct of Engineering*, October 26, 2012.

[B.24-49] OPG Standard, N-STD-MP-0024 R001, *Engineering and Design Authority*, June 6, 2016.

[B.24-50] CSA Standard, N286-05 Update No. 1, *Management System Requirements for Nuclear Power Plants*, November 2007.

[B.24-51] OPG Policy, NA44-OPP-03600 R035, *Pickering NGS-A Operating Policies and Principles,* April 27, 2015.

[B.24-52] OPG Policy, NK30-OPP-03600 R037, *Pickering NGS-B Operating Policies and Principles,* September 6, 2016.

[B.24-53] OPG Report, N-REP-03600-10003 R007, *Fukushima Action Item Report*, November 27, 2015.

[B.24-54] OPG Report, NA44-SR-01320-00001 R015, *Pickering A Safety Report – Part 2,* July 24, 2012.

[B.24-55] OPG Report, NK30-SR-01320-00002 R004, *Pickering B Safety Report – Part 2,* October 10, 2012.

[B.24-56] OPG Report, NA44-SR-01320-00002 R004, *Pickering Nuclear 1-4 Safety Report: Part 3 – Accident Analysis,* October 31, 2013.

[B.24-57] OPG Report, NK30-SR-01320-00003 R004, *Pickering Nuclear 5-8 Safety Report: Part 3 – Accident Analysis,* October 30, 2014.

[B.24-58] OPG List, P-LIST-06937-00001 R00, *Pickering A and B List of Safety Related Systems*, February 18, 2011.

[B.24-59] OPG Standard, N-STD-MP-0019 R006, *Beyond Design Basis Accident Management*, August 19, 2016.

[B.24-60] OPG Guideline, NA44-GUID-03600-00001 R000, *Pickering 1-4 Beyond Design Basis Functional Safety Requirements,* October 2014.

[B.24-61] OPG Guideline, NK30-GUID-03600-00001 R000, *Pickering 5-8 Beyond Design Basis Functional Safety Requirements,* October 2014.

[B.24-62] OPG Program, N-PROG-RA-0001 R014, *Consolidated Nuclear Emergency Plan*, June 2015.

[B.24-63] OPG Standard, N-STD-MP-0016 R002, *Safe Operating Envelope,* June 2012.

[B.24-64] OPG Guideline, N-GUID-01130-10000 R001, *Modifications for Beyond Design Basis Accidents,* March 2015.

[B.24-65] OPG Manual, N-BDB-03600-00002 R000, *OPG Emergency Mitigating Equipment for Beyond Design Basis Accidents: Technical Basis Document*, October 2015.

[B.24-66] CSA Standard N290.13-05, *Environmental Qualification of Equipment for CANDU Nuclear Power Plants*, 2005; Update No. 1: October 2009.

[B.24-67] OPG Procedure, N-PROC-RA-0133 R000, *Management of Equipment Important to Emergency Response*, December 2014.

[B.24-68] OPG Report, P-REP-03680-00004 R000, *Pickering NGS Periodic Safety Review 2: Code and Standard Reviews for Safety Factors 2 (Actual Condition of SSCs), 3 (Equipment Qualification) and 4 (Aging)*, July 2016.

[B.24-69] OPG Report, N-GUID-03611-10001, Volume 1 R004, *OPG Probabilistic Risk Assessment (PRA) Guide - Level 1 (At-Power)*, November, 2014.

[B.24-70] OPG Report, NA44-REP-03611-00012 R000, *Pickering NGS A Level 1 At-Power Internal Events Risk Assessment (PARA-L1P)*, September 2013.

[B.24-71] OPG Report, NK30-REP-03611-00006 R001, *Pickering NGS B Level 1 At-power Internal Events Risk Assessment*, November 2012.

[B.24-72] OPG Report, NA44-REP-03611-00011 R000, *Hazards Screening Analysis – Pickering A*, January 2012.

[B.24-73] OPG Report, NK30-REP-03611-00008 R000, *Hazards Screening Analysis – Pickering B*, May 2012.

[B.24-74] OPG Program, N-PROG-MP-0009 R011, *Design Management*, January 13, 2015.

[B.24-75] OPG Program, N-PROG-MP-0001 R014, *Engineering Change Control,* January 7, 2015.

[B.24-76] OPG Report, P-REP-09051.1-00015 R000, *Pickering NGS – 2015 Annual Risk and Reliability Report*, March 16, 2016.

[B.24-77] OPG Manual, NK30-OM-5-33000-08 R009, *Pickering Units 5-8 Heat Transport System, Auxiliary Service Failures*, May 28, 2009.

[B.24-78] CSA Standard, N290.1-13, *Requirements for the Shutdown Systems of Nuclear Power Plants*, December 2013.

[B.24-79] CSA Standard, N290.4-11, *Requirements for Reactor Control Systems of Nuclear Power Plants*, October 2011.

[B.24-80] CSA Standard N290.6-09, *Requirements for Monitoring and Display of Nuclear Power Plant Safety Functions in the Event of an Accident*, March 2009.

[B.24-81] CSA Standard, N290.14-15, *Qualification of Digital Hardware and Software for Use in Instrumentation and Control Applications for Nuclear Power Plants*, November 2015.

[B.24-82] OPG Manual, NA44-OSR-08131.02-00012 R002, *Pickering NGS-A Operational Safety Requirements: Moderator System*, November 10, 2010.

[B.24-83] OPG Manual, NK30-OSR-08131.02-00010 R002, *Pickering NGS-B Operational Safety Requirements: Moderator System*, February 4, 2011.

[B.24-84] OPG Report, NA44-REP-02004-0073 Volume 2 R000, *Seismic Assessment of Pickering A NGS Summary Report*, February 25, 1998.

[B.24-85] OPG Report, NK30-REP-03611-00013 R001, *Pickering NGS B (PNGS-B) PRA Based Seismic Margin Assessment (SMA),* March 2014.

[B.24-86] OPG Report, NA44-REP-03611-00034 R000, *Pickering NGS 'A' PRA-Based Seismic Margin Assessment (PARA-Seismic) Fukushima Action Item (FAI) Update,* February 2014.

[B.24-87] OPG Report, NA44-REP-03611-00033 R000, *Pickering NGS A High Wind Probabilistic Risk Assessment - FAI Update*, February 2014.

[B.24-88] OPG Report, NK30-REP-03611-00027 R000, *Pickering NGS B Level-1 High Wind Probabilistic Risk Assessment – FAI Update*, February 2014.

[B.24-89] OPG Manual, NA44-OSR-08131.02-00002 R03, *Pickering NGS 1-4 Operational Safety Requirements: Negative Pressure Containment System*, March 2015.

[B.24-90] OPG Manual, NK30-OSR-08131.02-00003 R004, *Pickering NGS 5-8 Operational Safety Requirements: Negative Pressure Containment System*, March 2015.

[B.24-91] CSA Standard N291-15, *Requirements for Safety-Related Structures for Nuclear Power Plants*, November 2015.

[B.24-92] OPG Memorandum, P-CORR-20000-0608706, *Pickering NGS, Inspection Criteria for Non-Containment Buildings and Structures (including safety-related structures and components),* August 17, 2016.

[B.24-93] OPG Standard, N-STD-MA-0018 R009, *Hoisting and Rigging,* September 22, 2015.

[B.24-94] OPG Procedure, N-PROC-MP-0083 R009, *Constructability, Operability, Maintainability & Safety*, April 2016.

[B.24-95] OPG Procedure, N-PROC-MP-0090 R013, *Modification Process*, July 2016.

[B.24-96] OPG Instruction, N-INS-00960-10000 R004, *Detailed Commissioning Specifications and Commissioning Reports*, April 2014.

[B.24-97] OPG Design Manual, NK30-DM-21002-10001, *Seismic Route*, April 1999.

[B.24-98] OPG Drawing, NA44-DRAW-20000-10001-0001 R00, *Reactor Auxiliary Bay Floor El. 254' -0" & 274'-0" Interior Seismic Route General Layout*, December 2015.

[B.24-99] OPG Design Manual, NA44-01100 R000, *Pickering Generating Station, Reactor Physics*, December 01, 1973.

[B.24-100] OPG Design Manual, NK30-03100, R001, *Pickering Generating Station B, Reactor Physics*, July 1987.

[B.24-101] OPG Manual, NA44-OSR-08131.02-00003 R003, *Pickering NGS-A Operational Safety Requirements: Fuel and Reactor Physics*, December 2015.

[B.24-102] OPG Manual, NK30-OSR-08131.02-00002 R003, *Pickering NGS-B Operational Safety Requirements: Fuel and Reactor Physics*, December 2015.

[B.24-103] OPG Manual, N-BDB-03600-00002 R000, *OPG Emergency Mitigating Equipment for Beyond Design Basis Accidents: Technical Basis Document*, October 13, 2015.

[B.24-104] OPG Manual, NA44-MAN-03651-10001 R002, *Environmental Qualification Room Conditions - Pickering A*, October 31, 2014.

[B.24-105] OPG Manual, NK30-MAN-03651-10001 R002, *Environmental Qualification Room Conditions - Pickering B*, November 06, 2014.

[B.24-106] OPG Manual, NA44-OSR-08131.02-00001 R002, *Pickering NGS A Operational Safety Requirements: Shutdown System*, June, 2010.

[B.24-107] OPG Manual, NK30-OSR-08131.02-00004 R003, *Pickering NGS-B Operational Safety Requirements: Shutdown Systems,* February 2016.

[B.24-108] OPG Design Manual, NK30-DM-33350-00002 R002, *Emergency Coolant Injection System Part 1 – General Requirements and Overview*, September 28, 2012.

[B.24-109] OPG Reference, NK30-REF-68000-0379145 R001, Engineering Design Guide, DG-30-68000-6, *Containment Provisions for Extensions of the Containment Envelope for Pickering Generating Station 'B'*, May 1977.

[B.24-110] CSA Standard, N290.2-11, *Requirements for Emergency Core Cooling Systems of Nuclear Power Plants*, October 2011.

[B.24-111] CSA Standard, N290.3-11, *Requirements for the Containment System of Nuclear Power Plants*, October 2011.

[B.24-112] CSA Standard, N290.5-06 (R2011) including Update No. 1, *Requirements for Electrical Power and Instrument Air Systems of CANDU Nuclear Power Plants*, December 2006; Update No. 1: November 2011.

[B.24-113] OPG Report, N-REP-03500-0401509, *Implications of the Fukushima Daiichi Event on OPG Nuclear Power Plants: A Summary Report,* July 2012.

[B.24-114] OPG Report, N-REP-09013-10007 R000, *Ontario Power Generation Severe Accident Management Guidance Instrumentation and Equipment Survivability - Summary Report*, December 2013.

[B.24-115] OPG Program, N-PROG-OP-0006 R018, *Environmental Management*, April 21, 2015.

[B.24-116] OPG Report, N-RPP-03415.1-10001 R007, *Radiation Protection Requirements – Nuclear Facilities*, June 2001.

[B.24-117] OPG Manual, P-MAN-37000-00001 R002, *Reactor Physics Fuel Management and Core Surveillance*, September 2016.

[B.24-118] OPG Design Manual, NK30-DM-67873-00001 R000, *Fixed Area Gamma Radiation Monitoring*, July 2011.

[B.24-119] OPG Design Manual, NA44-DM-67873-00002 R000, *Fixed Area Gamma Monitors*, March 2013.

B.25 CNSC G-144 (2006), "Trip Parameter Acceptance Criteria for the Safety Analysis of CANDU Nuclear Power Plants"

B.25.1 Background

The following paraphrased from the purpose and scope of CNSC G-144 (2006) [B.25-1] provides a brief overview of the purpose of this Regulatory Guide and the requirements expressed therein:

> *The purpose of G-144 is to provide guidance to licensees who operate CANDU nuclear power plants regarding reactor trip parameters that will preclude direct or consequential failures of reactor fuel or reactor pressure tubes.*
>
> *G-144 outlines the criteria that the selected reactor trip parameters for the plant are expected to meet, under all postulated design basis accidents other than the following:*
>
> 1. *Large loss of coolant accidents (LLOCA);*
> 2. *Very slow loss of reactivity control (VSLORC) accidents;*
> 3. *Fast loss of reactivity control (FLORC) accidents;*
> 4. *Fuelling machine accidents;*
> 5. *Single channel accidents; and*
> 6. *Accidents in the spent fuel bay.*

G-144 is relevant to Safety Factor 5 (Deterministic Safety Analysis). The latest version is G-144 (2006) [B.25-1]. G-144 is not identified in Appendix C or E of the R04 Licence Conditions Handbook [B.25-2], meaning it is not in Pickering PROL 48.02/2018 or cited by the CNSC as "Guidance and Criteria".

The CNSC recently communicated in Reference [B.25-3] plans to retire the Code on the basis that the acceptance criteria as outlined in G-144 are no longer applicable and have been replaced by a new set of acceptance criteria documented in COG Report COG-13-9035 R00, "Derived Acceptance Criteria For Deterministic Safety Analysis" [B.25-4].

The results of PSR1 G-144 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)) have been assessed for applicability to PSR2 in Section B.25.2. As identified in Reference [B.25-5], the Pickering PSR2 review of CNSC G-144 (2006) is an Incremental Review. PSR2 Incremental Review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.

- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.25.2 Compliance Assessment for Pickering PSR2

B.25.2.1 Application of PSR1 Reviews

The versions of G-144 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

The Pickering B ISR was performed using the latest version of the guide, i.e., G-144 (2006) [B.25-1] and was documented in OPG Report NK30-REP-03680-00005 R000 [B.25-6]. Reference [B.25-6] documented a clause-by-clause review of G-144 (2006) for Safety Factor 5. The review identified five cases where compliance was deemed to be an Acceptable Deviation. The Acceptable Deviations were related to dual trip parameter coverage, no dry-out criterion for the first trip, allowable duration of post dry-out operation for the second trip and interpretation of the acceptance criteria relating to acceptable duration of post dry-out operation.

Reference [B.25-6] identified that Pickering B compared favorably against the newest CANDU stations in terms of trip effectiveness. It was noted that the G-144 criteria were surrogates to the real criteria (fuel and channel integrity) and are conservative with respect to the actual safety criteria. Further, it was noted that in many instances of deviations there are either:

- A large number of redundant instruments that, while considered single parameter coverage, have in fact a large amount of redundancy (e.g., the large number of NOP detectors available in each logic channel during a slow Loss of Regulation event); or

- Trips are available, but they come in either slightly after dryout, or slightly beyond the 600°C/60 second criteria specified in G-144.

Based on the above findings, which are not impacted by operation beyond 2020, there are no gaps for PSR2 resulting from the Pickering B ISR reviews of G-144.

Pickering Units 1,4

G-144 had not yet been issued at the time of Pickering A Return to Service and as such a code review was not performed. However, the compliance assessment performed for the Pickering B ISR is not specific to any design differences which exist between Pickering Units 5-8 and Pickering Units 1,4, and as such, the findings are applicable to Pickering Units 1,4.

Darlington NGS

The Darlington ISR was performed using the latest version of the guide, i.e., G-144 (2006) [B.25-1] and was documented in OPG Report NK38-REP-03680-10015 R000 [B.25-7]. Reference [B.25-7] documented a high-level intent review of G-144 (2006) for Safety Factor 5. One gap in compliance was identified with Clause 5.0, which provides the specifics of the trip parameter effectiveness acceptance criteria. This gap was addressed through the industry initiative discussed in Section B.25.2.2.

B.25.2.2 Application of Post-PSR1 Reviews

Subsequent to completion of the Darlington ISR, the Industry, with CNSC input, convened work through COG associated with Safety Report analysis acceptance criteria. As part of that work, an Independent Technical Panel on trip effectiveness acceptance criteria was established to review the experimental database and provide recommendations on fuel and fuel channel integrity acceptance criteria for application to accidents associated with regulatory guide G-144 (2006). COG-10-2010, "ITP Recommendations for the CANDU Trip Effectiveness Acceptance Criteria" [B.25-8] was issued with recommendations for acceptance criteria. COG-13-9035 [B.25-4] documents fuel and fuel channel derived acceptance criteria and technical bases for slow events that are intended to replace the requirements of G-144 (2006). The CNSC staff completed their review of the technical basis for the new set of acceptance criteria documented in COG-13-9035 and concluded that the new criteria were developed in accordance with CNSC Document REGDOC-2.4.1, "Deterministic Safety Analysis" [B.25-9] and on that basis identified that G-144 is no longer applicable and will be retired.

COG-13-9035 [B.25-4] outlines the implementation processes for the new criteria as agreed to by the Industry and the CNSC. The derived acceptance criteria outlined in COG-13-9035 for slow events are less restrictive than those outlined in the current Pickering 1,4 and Pickering 5-8 Safety Reports ([B.25-10], [B.25-11]), providing confidence that the derived acceptance criteria documented in COG-13-9035 will not adversely impact on existing margins as demonstrated in the Safety Reports.

In Reference [B.25-12], OPG communicated to the CNSC that going forward, OPG will no longer provide analysis results for comparison against G-144 screening criteria, and instead will use the derived acceptance criteria outlined in COG-13-9035 [B.25-4]. On the basis of the Industry work undertaken to replace the requirements of G-144 (2006) with the derived acceptance criteria outlined in COG-13-9035, per Reference [B.25-3], the CNSC plans to retire G-144 and will update the Compliance Verification Criteria in the Pickering Licence Conditions Handbook to include COG-13-9035 under Licence Condition 5.1, Safety Analysis.

B.25.3 Compliance Summary for Pickering PSR2

There are no PSR2 gaps for CNSC G-144 (2006) [B.25-1]. Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with G-144 (2006).

B.25.4 References

[B.25-1] CNSC Regulatory Guide G-144, *Trip Parameter Acceptance Criteria for the Safety Analysis of CANDU Nuclear Power Plants*, May 2006.

[B.25-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.25-3] CNSC Correspondence, e-Doc 4981431, OPG File No. N-CORR-00531-18045 R000, *Darlington and Pickering NGS: Derived Acceptance Criteria for Deterministic Safety Analysis of Slow Events and Re-Categorization of the CANDU Safety Issue PF18 Fuel Bundle/Element Behaviour under Post Dryout*, April 21, 2016.

[B.25-4] COG Report, COG-13-9035 R00, *Derived Acceptance Criteria For Deterministic Safety Analysis*, November 2014.

[B.25-5] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.25-6] OPG Report, NK30-REP-03680-00005 R000, *Pickering NGS B – Integrated Safety Review – Safety Analysis Review*, June 2007.

[B.25-7] OPG Report, NK38-REP-03680-10015 R000, *Review of CNSC G-144 (May 2006) Trip Parameter Acceptance Criteria for the Safety Analysis of CAND Nuclear Power Plants for Darlington Integrated Safety Review,* August 2011.

[B.25-8] COG Report, COG-10-2010, *ITP Recommendations for the CANDU Trip Effectiveness Acceptance Criteria,* November 2011.

[B.25-9] CNSC Regulatory Document REGDOC-2.4.1, *Deterministic Safety Analysis,* May 2014.

[B.25-10] OPG Report, NA44-SR-01320-00002-R004, *Pickering Nuclear 1-4 Safety Report: Part 3, Accident Analysis,* October 2013.

[B.25-11] OPG Report, NK30-SR-01320-00003-R004, *Pickering Nuclear 5-8 Safety Report: Part 3 – Accident Analysis,* October 2014.

[B.25-12] OPG Correspondence, N-CORR-00531-17997 R000, *Implementation of Derived Acceptance Criteria for Design Basis Accidents in Deterministic Safety Analyses and Request for Re-Categorization of CSI PF18,* April 6, 2016.

B.26 CNSC G-149 (2000), "Computer Programs Used In Design and Safety Analyses of Nuclear Power Plants and Research Reactors"

B.26.1 Background

The following paraphrased from the purpose and scope of CNSC G-149 (2000) [B.26-1] provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *The purpose of G-149 is to provide guidance to licensees involved in the development, maintenance and use of computer programs used in the design and safety analysis of nuclear power plants and research reactors so that a high degree of confidence may be placed in both the programs and the results of their application.*
>
> *G-149 applies to licensees whose computer programs are used in:*
>
> - *designing or supporting the design of a nuclear power plant or research reactor*
> - *analyzing operational transients, incidents or accidents.*
>
> *G-149 does not apply to operational control systems software. For computer programs developed before the effective date of this regulatory guide, the degree of applicability is specified in section 8 of this guide.*

G-149 is relevant to Safety Factors 1 (Plant Design), 5 (Deterministic Safety Analysis), 6 (Probabilistic Safety Assessment) and 7 (Hazard Analysis). G-149 is not identified in Appendix C or E of the R04 Licence Conditions Handbook [B.26-2], meaning it is not in Pickering PROL 48.02/2018 or cited by the CNSC as "Guidance and Criteria". CNSC G-149 (2000) is the first edition of this Regulatory Guide.

The results of PSR1 G-149 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Review (ISRs)) have been assessed for applicability to PSR2 in Section B.26.2. As identified in Reference [B.26-3], the Pickering PSR2 review of CNSC G-149 (2000) is an Incremental Review. PSR2 Incremental Review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- <u>Compliance:</u> Compliance indicates that the change in the safety requirement, per the topical review, is met.
- <u>Gap</u>: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.26.2 Compliance Review History

B.26.2.1 Application of PSR1 Reviews

The versions of G-149 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

<u>Pickering Units 5-8</u>

A clause-by-clause review of CNSC G-149 (2000) was performed at a high level as part of the Pickering B ISR [B.26-4] for the Deterministic, Probabilistic Safety Analysis and Hazard Analysis Safety Factors (Safety Factors 5, 6, and 7). The review of the guide performed as part of the Pickering B ISR was a high level review which confirmed that OPG's governance and processes adequately addressed the intent of the guide.

OPG first implemented its software quality assurance programs for Nuclear Safety Analysis based on the requirements of CSA Standard N286.7-94, well in advance of G-149 being issued in 2000. Reference [B.26-4] identifies that, in general, the content and definitions of G-149 are very similar to CSA N286.7-99, i.e., the version of CSA N286.7 reviewed as part of the Pickering B ISR. As such, demonstrating Direct Compliance with CSA N286.7-99, which was the case, was sufficient to meet the intent of G-149. Reference [B.26-4] identified one notable exception between G-149 and CSA N286.7-99 which related to Clause 8.0 of G-149. Clause 8.0 of G-149 pertains to *Existing Computer Codes*. Reference [B.26-4] identifies that Generic Action Item (GAI) 98G02 *Validation of Computer Programs used in Safety Analysis of Power Reactors* and its closure were considered to have met the intent of Clause 8.0 as it pertains to legacy codes. On that basis, the Pickering B ISR found that OPG software governance was in Indirect Compliance with G-149.

The review performed as part of the Pickering B ISR did not identify any gaps against G-149. There are therefore no PSR2 gaps which result from the Pickering B ISR.

<u>Pickering Units 1,4</u>

CNSC G-149 was not included in the list of codes and standards reviewed as part of Pickering A Return to Service. The subsequent clause-by-clause review of CNSC G-149 against OPG software governance that was performed as part of the Pickering B ISR [B.26-4] identified no gaps. Since the review was performed against relevant OPG governance, the conclusions of Reference [B.26-4] are considered equally applicable to Pickering Units 1,4 and Units 5-8. Further, as discussed below, findings from the Darlington ISR are programmatic and applicable to Pickering NGs.

Darlington NGS

OPG Report NK38-REP-03680-10016 R000 [B.26-5] documents a high level review of OPG software governance that was performed against the requirements of Regulatory Guide G-149 (2000) as part of the Darlington ISR. The review was conducted by comparing the requirements of G-149 against OPG software quality assurance governance. The OPG software governance was found to be largely compliant with G-149 (2000), as discussed below.

Reference [B.26-5] notes that early versions of CSA N286.7 contained a number of clauses that required interpretation or expansion to be implemented for which complete guidance was not provided, and which the CSA N286.0 series allowed for a graded interpretation. Consequently, COG Guideline 153-507230-COG-001 R0, "Guideline for the Application of CSA N286.7" [B.26-6] was created by a collaborative industry effort to provide guidance to assist owner organizations and participants in preparing and implementing software quality assurance processes that comply with CSA N286.7. Reference [B.26-5] notes that the COG guideline forms the main basis for OPG Standard N-STD-MP-0008, "Development, Qualification and Use of Scientific, Engineering and Safety Analysis Software" [B.26-7], and as such, its concepts are reflected throughout the OPG software governance. G-149 predates the COG Guideline (Reference [B.26-6]) and reflects the CNSC interpretation of CSA N286.7 at the time of issuance of G-149.

The Darlington ISR found that the OPG Software Quality Assurance is largely compliant with the Regulatory Guide G-149. However, gaps were found to exist in the level of detail of specification for certain activities:

- The relationships between development phases (input and outputs) are not explicitly specified in the OPG software governance. However, these relationships are logical and intrinsic to the development of computer codes, and are likely understood by code developers and testers.

- The OPG software governance does not contain an explicit requirement to place coding produced during the development phase under configuration management. This is not a significant gap as test output with the exception of that arising from regression analysis is ephemeral. Regression test results are necessarily placed under configuration management as they must be available for comparison with the results obtained with a new code version.

- The OPG software governance does not contain an explicit requirement to perform a design review.

The requirements from G-149 that give rise to these gaps are not reflected in CSA N286.7-99 (i.e., the version of CSA N286.7 reviewed as part of the Darlington ISR). Furthermore, Reference [B.26-5] concluded that the gaps arise from an over-specification of detail for certain processes that have an inherent logic that would necessarily be executed through compliance with the OPG software governance. Since OPG software governance was found to be fully compliant with the version of N286.7 reviewed as part of the Darlington ISR, these gaps were not considered to be significant. As noted in Reference [B.26-8], these gaps were identified as Acceptable Deviations under Issue D146 of the Darlington ISR. CNSC acceptance of these gaps as Acceptable Deviations was documented in Reference [B.26-9]. The results of the Darlington ISR from the perspective of G-149 are considered applicable to Pickering Units 1,4 and 5-8, given that the assessment was performed against OPG governance applicable to all of the OPG Nuclear fleet. Hence, there are no PSR2 gaps which result from the Darlington ISR.

B.26.2.2 Application of Post-PSR1 Reviews

G-149 has not been revised since the PSR1 G-149 reviews (Pickering B and Darlington ISRs) were completed and there have been no additional compliance assessments completed since the PSR1 G-149 reviews outlined in Section B.26.2.1 were undertaken. The results of the PSR1 G-149 reviews identified no gaps as the requirements of G-149 are built into existing OPG software governance. Hence, no PSR2 gaps are identified against G-149.

B.26.3 Compliance Summary for PSR2

There are no PSR2 gaps for CNSC G-149 (2000) [B.26-1]. Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with G-149 (2000).

B.26.4 References

[B.26-1] CNSC Regulatory Guide G-149, *Computer Programs Used in Design and Safety Analysis of Nuclear Power Plants and Research Reactors*, October 2000.

[B.26-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.26-3] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.26-4] OPG Report, NK30-REP-03680-00005 R000, *Pickering NGS B – Integrated Safety Review – Safety Analysis Review*, June 2007.

[B.26-5] OPG Report, NK38-REP-03680-10016 R000, *Review of Regulatory Guide G-149 (October 2000), Computer Programs Used in Design and Safety Analyses of Nuclear Power Plants and Research Reactors for Darlington Integrated Safety Review*, August 22, 2011.

[B.26-6] COG Guideline, 153-507230-COG-001 R0, *Guideline for the Application of CSA N286.7*, July 2007.

[B.26-7] OPG Standard, N-STD-MP-0008 R004, *Development, Qualification and Use of Scientific, Engineering and Safety Analysis Software,* October 2013.

[B.26-8] OPG Report, NK38-REP-03680-10104 R000, *Darlington NGS Integrated Safety Review (ISR) – Final ISR Report*, October 2011.

[B.26-9] OPG Report, NK38-REP-03680-10016-ADD-001 R000, *Addendum to the CNSC G-149 Code Review Report for Darlington ISR*, January 21, 2014.

B.27 CNSC R-77 (1987), "Overpressure Protection Requirements for Primary Heat Transport Systems in CANDU Power Reactors Fitted With Two Shutdown Systems"

B.27.1 Background

The following paraphrased from the introduction of CNSC Regulatory Document R-77 (1987) [B.27-1] provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *The overpressure protection requirements of Article NB 7000 of Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) are incorporated in the National Standard of Canada N285.1. These requirements do not refer to a particular nuclear system design. This is recognized in paragraphs NCA-2141 and NB-7I20 of the ASME Code which make reference to the requirements of the appropriate regulatory authority for guidance.*
>
> *For CANDU power reactors fitted with two shutdown systems, some guidance is given in the Atomic Energy Control Board (AECB) Regulatory Document R-10, but this does not address overpressure protection as a specific topic and further clarification is required. This document seeks to provide such clarification.*

CNSC R-77 (1987) [B.27-1] is applicable to Safety Factor 1 (Plant Design).

Compliance with CNSC R-77 is not currently a licence requirement for Pickering NGS (in accordance with PROL 48.02/2018) per the R04 Pickering Licence Conditions Handbook [B.27-2].

This regulatory document is no longer available. The content of R-77 [B.27-1] has been incorporated into Section 7.6 of N285.0-12.

The results of PSR1 CNSC R-77 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.27.2. As identified in Reference [B.27-3], the Pickering PSR2 review of CNSC R-77 (1987) is an Incremental Review. PSR2 Incremental Review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.
- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.27.2 Compliance Assessment for Pickering PSR2

B.27.2.1 Application of PSR1 Reviews

The versions of R-77 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

A clause-by-clause review against R-77 was performed and documented in report NK30-REP-03680-00001 R000 [B.27-4] as part of the Pickering B ISR Plant Design Safety Factor. The objective of this review was to complete a clause-by-clause review of relevant sections of CNSC R-77, to assess whether the design requirements in CNSC R-77 are covered by design requirements in CSA Standard N285.0 (the PROL requires compliance with N285.0). This approach was taken since CNSC R-77 contains upper tier design requirements for the design of the Primary Heat Transport and Safety Systems and therefore it is sufficient to demonstrate that the design requirements contained in R-77 are covered by the design requirements in CSA N285.0. Compliance with CSA N285.0 was demonstrated in a separate section of Reference [B.27-4].

The Pickering B ISR assessment concluded that all of the design requirements contained in R-77, except one clause, are covered by the design requirements in CSA N285.0. The one exception on Clause 3.6 states:

> *In a case where only one trip parameter is installed in the second shutdown system, this trip parameter may be credited.*

The report states that although Pickering B is in compliance with this clause, it is recommended that CSA N285.0 be modified to include this clause. This has since been addressed in Clause 7.6.2.3 (c) of N285.0 [B.27-5]. There is no PSR2 gap associated with this issue.

The report also documented one Acceptable Deviation, on Clause 3.1 addressing Allowable Service Conditions. It specifies events to analyze based on their probability of occurrence. The report describes that since R-77 was not in place at the time of the issuance of the first Pickering B Operating Licence, the Safety Report used the Siting Guide rules, which are consistent with the intent of those specified in R-77. This rationale remains valid and is not affected by Pickering operation beyond 2020. The Pickering Units 1,4 and 5-8 Safety Reports [B.27-6], [B.27-7] demonstrate that for Design Basis Accidents resulting in Heat Transport System (HTS) overpressure, e.g. a loss of Class IV power, peak stresses are below those allowed for the assigned ASME service level consistent with R-77. Therefore, there is no PSR2 gap associated with this issue.

Pickering Units 1,4

As part of Pickering Units 1,4 Return to Service (PARTS), code reviews were conducted. The main submission for PARTS [B.27-8] committed to perform a review of R-77 [B.27-1]. This review was completed in AECL Report, "Review of Pickering A Design Against Current Codes and Standards" [B.27-9].

The review demonstrated that Pickering Units 1,4 comply with the requirements of R-77. The report also reviewed Safety Report events resulting in HTS overpressure. It concluded that for each of these events, peak stresses are below those allowed for the ASME service level assigned to the event according to R-77 guidelines. These conclusions remain valid per the Safety Report as described above.

Darlington NGS

The Darlington ISR review of R-77 was documented in OPG Report NK38-REP-03680-10019 R000 [B.27-10]. This clause-by-clause review demonstrated that Darlington NGS was compliant with R-77.

B.27.2.2 Application of Post PSR1 Reviews

CNSC Regulatory Document R-77 has not been revised since the initial version issued in 1987. Therefore the conclusions of the Regulatory Document R-77 reviews described in Section B.27.2.1 for Pickering still apply and do not pose any PSR2 gaps.

In addition, per the CSA Impact Statement for the N285.0-12, General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants [B.27-11], the requirements in R-77 were migrated into N285.0-12 (Clause 7.6.2). There has been no change made to these requirements in the latest version, i.e., N285.0-12 including Updates No. 1 and No. 2, per the associated CSA Impact Statement [B.27-12]. Also, the PSR2 assessment of CSA N285.0-12 did not identify any gaps associated with Clause 7.6.2.

Therefore, there are no PSR2 gaps associated with Regulatory Document R-77.

B.27.3 Compliance Summary for Pickering PSR2

There are no PSR2 gaps for CNSC R-77 (1987) [B.27-1]. Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with CNSC R-77 (1987).

B.27.4 References

[B.27-1] CNSC Regulatory Document R-77, *Overpressure Protection Requirements for Primary Heat Transport Systems in CANDU Power Reactors Fitted with Two Shutdown Systems*, October 1987.

[B.27-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.27-3] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.27-4] OPG Report, NK30-REP-03680-00001 R000, *Pickering NGS-B Integrated Safety Review – Plant Design Safety Factor*, August 2007.

[B.27-5] CSA Standard N285.0-12/N285.6 Series-12 including Update No. 1, *General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants/Material Standards for Reactor Components for CANDU Nuclear Power Plants, 2012*; Update No. 1: September 2013.

[B.27-6] OPG Report, NA44-SR-01320-00002 R004, *Pickering Nuclear 1-4 Safety Report: Part 3 – Accident Analysis*, September 2013.

[B.27-7] OPG Report, NK30-SR-01320-00003 R004, *Pickering Nuclear 5-8 Safety Report: Part 3 – Accident Analysis*, October 2014.

[B.27-8] OPG Letter, R.J. Strickert to J.S.C. Tong, NA44-CORR-00531-00381, *Pickering A Updated Basis for Return to Service Document*, April 20, 2001.

[B.27-9] AECL Assessment Document, 44RS-00531-ASD-001 Rev. 04, *Review of Pickering A Design Against Current Codes and Standards*, November 2000.

[B.27-10] OPG Report, NK38-REP-03680-10019 R000, *Review Of CNSC R-77 (October 1987), Overpressure Protection Requirements for Primary Heat Transport Systems in CANDU Power Reactors Fitted with Two Shutdown Systems For Darlington Integrated Safety Review*, June 2011.

[B.27-11] CSA Impact Statement, *Notification of CSA N285.0-12 General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants/Material Standards for Reactor Components for CANDU Nuclear Power Plants*, Date not provided.

[B.27-12] CSA Impact Statement for Publication, *Notification of CSA N285.0-12 Amendment No. 2, General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants/Material Standards for Reactor Components for CANDU Nuclear Power Plants*, Date not provided.

B.28 CSA N288.2-14, "Guidelines for Calculating Radiological Consequences to the Public from a Release of Airborne Radioactive Material for Nuclear Reactor Accidents"

B.28.1 Background

The following paraphrased from the introduction of CSA N288.2-14 [B.28-1] provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *CSA N288.2 provides guidelines and a methodology for calculating effective doses and thyroid doses to people (either individually or collectively) in the path of airborne radioactive material released from a nuclear facility following a hypothetical accident.*
>
> *The specific radionuclides considered in CSA N288.2 are those associated with substances having the greatest potential for becoming airborne in reactor accidents [e.g., tritium (HTO), noble gases and their daughters (Kr-Rb, Xe-Cs), and radioiodines (I)]; as well as certain radioactive particulates (e.g., Cs, Ru, Sr, Te) that may become airborne under exceptional circumstances.*
>
> *CSA N288.2 focuses on the calculation of radiation doses for:*
>
> > *(a) External exposures from radioactive material in the cloud;*
> >
> > *(b) Internal exposures from inhalation of radioactive material in the cloud and also skin penetration of tritium; and*
> >
> > *(c) External exposures from radionuclides deposited on the ground, during and after passing of the cloud.*

CSA N288.2 is relevant to Safety Factor 5 (Deterministic Safety Analysis). CSA N288.2 is not discussed in the R04 Pickering Licence Conditions Handbook [B.28-2].

CSA N288.2-14 is the second edition of this standard, and supersedes the previous version published in 1991 under the title *Guidelines for Calculating Radiation Doses to the Public from a Release of Airborne Radioactive Material under Hypothetical Accident Conditions in Nuclear Reactors* [B.28-3]. According to the preface of the 2014 edition, it has been updated to reflect current industry practice and new research and analysis methods. Major changes in the 2014 edition are discussed in Section B.28.2.2 below.

The results of PSR1 CSA N288.2 reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.28.2. As identified in Reference [B.28-4], the Pickering PSR2 review of CSA N288.2-14 is an Incremental Review. PSR2 Incremental Review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.
- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.28.2 Compliance Assessment for Pickering PSR2

B.28.2.1 Application of PSR1 Reviews

The versions of N288.2 (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

For the Pickering B ISR, a high level review of CSA N288.2-M91 [B.28-3] was conducted in 2007 and included in Appendix E of the Safety Analysis Safety Factor Review Report [B.28-5]. The review concluded that:

> *The OPG methodology is in compliance with the intent of CSA N288.2 (Indirect Compliance).*

There were no ISR gaps identified and no CNSC comments raised regarding the code compliance assessment against N288.2. However, the same report also noted that there was an open Safety Report Update Issue #115 (defined as the need to update the existing dosimetric information in safety analysis) that would be used to track revision of the OPG methodology for calculating radiological doses from airborne releases. Relevant open Safety Report Analysis Issues were taken into account in the development of the Pickering RD-310 Implementation Plan [B.28-6]. As discussed in the PSR2 review of REGDOC-2.4.1, the prioritization of gaps to be addressed as part of the RD-310 Implementation Plan was based in part on the projected end of commercial operations by 2020. These prioritization criteria were carried forward into the REGDOC-2.4.1 Implementation Plan [B.28-7] which superseded the RD-310 Implementation Plan and reconsideration in the context of extended operations beyond 2020 is already addressed by PSR2 REGDOC-2.4.1 Gap #2.

Pickering Units 1,4

CSA N288.2-M91 was not reviewed as part of Pickering A Return-to-Service ISR because it was concluded that the standard "*Pertains mostly to design support analysis*" [B.28-8]. CSA N288.2 is not mentioned in the R04 Pickering Licence Conditions Handbook [B.28-2] or Pickering PROL Renewal Application [B.28-9]. However, as discussed above, the Pickering B ISR reviewed CSA N288.2-M91 and the conclusions of that work are applicable to Pickering Units 1,4 as the same methodology for dose assessment was applied in both Pickering A and B Safety Reports.

Darlington NGS

Compliance with CSA N288.2-M91 (R2008) [B.28-10] was assessed as part of the Darlington ISR in 2011 [B.28-11]. This was a high-level intent review and Darlington was considered compliant with the exception of a gap relating to Clause 6.2.3. Clause 6.2.3 discusses the assessment of the effect associated with the buildup of radionuclides from decay of parents. The review was unable to find documented evidence that analysis had been carried out to explicitly address buildup of progeny from the point of release under accidental conditions.

This Darlington gap was identified as Gap #400 and was addressed under Issue D026 [B.28-12] and summarized in Reference [B.28-13]. The resolution of the gap resulted in it being reclassified as an Acceptable Deviation and concluded it to have low safety significance with no further action required. Given the similarity of dose calculations between Pickering and Darlington, the same conclusion is applicable to Pickering NGS and there is therefore no PSR2 gap.

B.28.2.2 Application of Post-PSR1 Reviews

As discussed in Section B.28.1, CSA N288.2-14 is the second edition of this standard, and supersedes the previous version published in 1991 under the title *Guidelines for Calculating Radiation Doses to the Public from a Release of Airborne Radioactive Material under Hypothetical Accident Conditions in Nuclear Reactors.* According to the preface of the 2014 edition, major changes include [B.28-1]:

> *a) Updating definitions and terminology in accordance with current usage;*
>
> *b) Incorporation of new guidance from relevant national and international publications that address doses from accidental releases (e.g., new ICRP [International Commission on Radiological Protection] guidance on dose coefficients);*
>
> *c) Broadening the applicability to include assessments that are conducted for licensing, emergency planning, or environmental assessment purposes;*
>
> *d) Provision of guidance on consequence assessments for emergency response during a real event;*
>
> *e) Inclusion of all radionuclides that could be released to the atmosphere in a postulated or real accident;*
>
> *f) Allowance for a stochastic treatment of meteorological data in which doses are calculated for many records in the meteorological archives at a given site;*
>
> *g) Discussion of the uncertainty in the dose estimates;*
>
> *h) Provision of guidance on how to obtain the meteorological information required by the models (e.g., stability class) and performance requirements for data measurement;*
>
> *i) Inclusion of health risks resulting from the predicted doses (including organ doses for deterministic effects);*

j) Consideration of approaches to account for time-dependent releases to the environment;

k) Provision of guidance on the location and age of the representative person for whom doses are calculated;

l) Advanced methods for treating the release of tritium;

m) Provision of guidance on how to determine individual doses from stochastic results consistent with regulatory expectations for conservative analysis; and

n) Provision of guidance on the attributes that atmospheric dispersion computer codes should consider for use in the Canadian regulatory context.

As discussed in the Standard, the guidance in the 2014 edition was updated to reflect current industry practice and new research and analysis methods. Further, the Standard states [B.28-1]:

The new edition does not mandate a single approach or code, or provide detailed equations to construct a code. It describes acceptable methods that can be used to calculate the consequences of a nuclear accident. The new edition also identifies acceptable data sources and acceptable methodologies to account for specific effects, and recommends standardized end points for the calculations.

Darlington and Pickering NGS have not completed compliance reviews against the 2014 version of the standard. However, per the COG Research and Development Annual Reporting [B.28-14], Work Package WP 50109 - *Assessment of Impact of Proposed Revision*, the Industry Standard Toolset code ADDAM (Atmospheric Dispersion and Dose Analysis Method) is being evaluated against the 2014 version of the standard. This assessment will determine the extent of code compliance and recommend required changes to the ADDAM code to comply with the N288.2-14 standard.

Updates to deterministic safety analysis are scoped in accordance with REGDOC-2.4.1 [B.28-15]. No plans to systematically update the existing Pickering safety analysis due to changes in N288.2-14 have been identified in the REGDOC-2.4.1 Implementation Plan [B.28-7]. Currently, the only update identified for Pickering is the development of the Common Mode Events Appendices. According to the Technical Basis Document for these new appendices [B.28-16], the ADDAM code discussed above will be used for the atmospheric dispersion and dose calculations.

As captured in PSR2 REGDOC-2.4.1 Gap #2, the scope of the REGDOC-2.4.1 Implementation Plan will be reconsidered to align with Pickering operation beyond 2020. The REGDOC-2.4.1 Implementation Plan update will consider the incremental implications of Pickering operation beyond 2020, including any considerations of N288.2 revisions. This is identified as a PSR2 gap (**PSR2 CSA N288.2-14 Gap #1**). It is being addressed as part of REGDOC-2.4.1 implementation.

B.28.3 Compliance Summary for Pickering PSR2

There is one PSR2 CSA N288.2-14 gap which relates to Safety Factor 5 (Deterministic Safety Analysis):

1. Safety Report upgrades currently underway for Pickering as part of REGDOC-2.4.1 implementation for the period of 2017-2021 will utilize methods consistent with N288.2-14. The REGDOC-2.4.1 Implementation Plan update will consider the incremental implications of Pickering operation beyond 2020, including any considerations of N288.2 revisions. This issue has therefore been identified as a PSR2 gap. It is being addressed as part of REGDOC-2.4.1 implementation.

B.28.4 References

[B.28-1] CSA Standard N288.2-14, *Guidelines for Calculating Radiological Consequences to the public of a Release of Airborne Radioactive Material for Nuclear Reactor Accidents*, December 2014.

[B.28-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.28-3] CSA Standard N288.2-M91, *Guidelines for Calculating Radiation Doses to the Public from a Release of Airborne Radioactive Material under Hypothetical Accident Conditions in Nuclear Reactors*, April 1991.

[B.28-4] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.28-5] OPG Report, NK30-REP-03680-00005 R000, *Pickering NGS B Integrated Safety Review – Safety Analysis Review*, June 2007.

[B.28-6] OPG Plan, N-PLAN-03500-0439621 R001, *Pickering A & Pickering B RD-310 Implementation Plan*, March 2013.

[B.28-7] OPG Plan, N-PLAN-03500-0500515 R003, *REGDOC-2.4.1 Implementation Plan*, May 2015.

[B.28-8] OPG Letter, R. Strickert to J. Tong, NA44-CORR-00531-00381 R000, *Pickering A - Updated Basis for Return to Service Document*, April 20, 2001.

[B.28-9] OPG Letter, P-CORR-00531-03719 R000, G. Jager to M. A. Leblanc, *Application for Renewal of Pickering Nuclear Generating Station Power Reactor Operating Licence*, July 4, 2012.

[B.28-10] CSA Standard N288.2-M91 (R2008), *Guidelines for Calculating Radiation Doses to the Public from a Release of Airborne Radioactive Material under Hypothetical Accident Conditions in Nuclear Reactors*, April 1991.

[B.28-11] OPG Report, NK38-REP-03680-10037-R000*, Review of CAN/CSA-N288.2-M91(R2008) (January 1991), Guidelines for Calculating Radiation Doses to the Public From a Release of Airborne Radioacive Material under Hypothetical accident Conditions in Nuclear Reactors for Darlington Integrated Safety Review*, August 2011.

[B.28-12] OPG Report, NK38-REP-00770-0417594, *Decay of Parent Nucleides in Hypothetical Accidents*, May 2011.

[B.28-13] OPG Report, NK38-REP-03680-10104 R000, *Darlington NGS Integrated Safety Review (ISR) – Final ISR Report*, October 2011.

[B.28-14] OPG Letter, R. Manley to M. Santini and F. Rinfret, N-CORR-00531-06905 R000, *REGDOC 3.1.1 Research and Development Annual Reporting*, June 16, 2015.

[B.28-15] CNSC Regulatory Document REGDOC-2.4.1, *Deterministic Safety Analysis*, May 2014.

[B.28-16] OPG Report, P-REP-03500-00004 R000, *Pickering A and B Safety Report Common Mode Events Appendices Technical Basis Document*, August 2016.

B.29 CSA N290.7-14, "Cyber-Security for Nuclear Power Plants and Small Reactor Facilities"

B.29.1 Background

The following paraphrased from the preface and scope of CSA N290.7-14 (including Errata, January 2015) [B.29-1] provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *CSA N290.7 pertains to the securing of essential computer systems and components against cyber-attacks resulting in loss of availability, degradation or loss of ability to perform their intended function, compromise of their integrity, and loss of confidentiality of their information. CSA N290.7 does not apply to business systems (e.g., work management), and offline engineering systems (e.g., analytical, scientific, and design computer programs as per CSA N286.7).*
>
> *CSA N290.7 addresses cyber security at nuclear power plants and small reactor facilities for the following computer systems and components:*
>
> *a) Systems important to nuclear safety;*
>
> *b) Nuclear security;*
>
> *c) Emergency preparedness;*
>
> *d) Production reliability;*
>
> *e) Safeguards; and*
>
> *f) Auxiliary assets or systems which, if compromised, exploited, or failed, could adversely impact Item (a), (b), (c), (d) or (e).*

CSA N290.7-14 is applicable to Safety Factor 1 (Plant Design).

Compliance with CSA N290.7 is not currently a licence requirement for Pickering NGS (in accordance with PROL 48.02/2018) per the R04 Pickering Licence Conditions Handbook [B.29-2].

CSA N290.7-14 is the first edition of this standard. The Impact statement for public review [B.29-3] identifies the following significant features of this standard:

1. *This new standard establishes:*
    a. *The requirements for cyber security for nuclear power plants and small reactor facilities.*
    b. *The graded assessment of computer systems to determine the applicability of cyber security controls.*

c. *Cyber security throughout a computer system's lifecycle, from conceptual design through installation, commissioning and decommissioning.*

d. *The need to interface with other organizations.*

2. *This standard identifies the purpose and expected outcomes but does not attempt to describe detailed methodology.*

As identified in Reference [B.29-4], the Pickering PSR2 review of CSA N290.7-14 is a High Level review. For a PSR2 High Level review, the degree of conformance with clauses or groups of clauses in the Law, Regulation, Code or Standard is demonstrated by supporting evidence stating whether the intent of the requirements stipulated in the requirement document is met. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the intent of the safety requirement is met.
- Gap: A Gap indicates that the intent of the safety requirement is not met.

B.29.2 Compliance Assessment for Pickering PSR2

B.29.2.1 Application of PSR1 Reviews

CSA N290.7 was not reviewed as part of PSR1 as the document did not exist at the time that the previous Darlington and Pickering B Integrated Safety Reviews and Pickering A Return to Service assessments were performed.

B.29.2.2 Application of Post PSR1 Reviews

As discussed above, N290.7 was not reviewed as part of PSR1. However, N-REP-69000-10003 R000, "Gap Analysis Between CSA N290.7-14 Cyber Security Requirements for Nuclear Power Plants and Small Reactor Facilities" [B.29-5], has been completed by OPG and satisfies the intent of this PSR2 High Level Review. The gap analysis and implementation plan for N290.7-14 was accepted by the CNSC, and the related Action Item 2015-OPG-7041 was closed [B.29-6]. For reasons of security and confidentiality, the findings of N-REP-69000-10003 R000 will not be discussed in PSR2.

B.29.3 Compliance Summary for Pickering PSR2

As discussed in Section B.29.2.2, for reasons of security and confidentiality, the findings of the gap analysis for N290.7-14 will not be discussed in PSR2.

B.29.4 References

[B.29-1] CSA Standard N290.7-14, *Cyber Security for Nuclear Power Plants and Small Reactor Facilities*, December 2014; Errata, February 2015.

[B.29-2] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.29-3] CSA Impact Statement for Public Review, *Product: New Standard; Product Designation: CSA N290.7-14*, Date not provided.

[B.29-4] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.29-5] OPG Report, N-REP-69000-10003 R000, *Gap Analysis Between CSA N290.7-14 "Cyber Security Requirements for Nuclear Power Plants and Small Reactor Facilities"*, March 2016.

[B.29-6] CNSC Letter, e-Doc # 5057238, OPG File No. N-CORR-00531-18195 R000, H. Khouaja and M. Santini to B. McGee and B. Duncan, *Darlington and Pickering NGS: Implementation of CSA N290.7-14 Cyber Security – Closure of Action Item 2015-OPG-7041*, August 16, 2016.

B.30 NBCC (2010), "National Building Code of Canada"

B.30.1 Background

The following, paraphrased from the National Building Code of Canada (NBCC) 2010 [B.30-1] provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *The National Building Code of Canada sets out technical provisions for the design and construction of new buildings. It also applies to the alteration, change of use and demolition of existing buildings. The NBCC details the minimum provisions acceptable to maintain the safety of buildings, with specific regard to public health, fire protection, accessibility and structural sufficiency.*

Appendix A of NBCC 2010 [B.30-1] states:

> *Application to Existing Buildings: This Code is most often applied to existing or relocated buildings when an owner wishes to rehabilitate a building, change its use, or build an addition, or when an enforcement authority decrees that a building or class of buildings be altered for reasons of public safety. It is not intended that the NBCC be used to enforce the retrospective application of new requirements to existing buildings or existing portions of relocated buildings, unless specifically required by local regulations or bylaws.*

The NBCC is relevant to Safety Factor 1 (Plant Design).

Compliance with NBCC is not a licence requirement for Pickering NGS (PROL 48.02/2018), although it is referred to in the R04 Pickering Licence Conditions Handbook [B.30-2] under Section 9.1 as a requirement for Structures, Systems and Components (SSCs) in the protected area for which OPG governance has specified that CSA N293-07 [B.30-3] is not applied. Also, it may be considered an indirect licence requirement as N293 is a licence requirement and refers to the NBCC as follows:

> *5.1.3 Where specific design or operational requirements are not addressed in this Standard, the NBCC, or the NFCC [National Fire Code of Canada], good engineering practice shall apply and, where appropriate, recognized Standards (such as those of the National Fire Protection Association [NFPA]) shall be used.*
>
> *5.5.2.2 Except as otherwise indicated in this Standard, plants shall be designed, modified, and constructed in accordance with all applicable requirements of the NBCC.*

The 2010 edition of the NBCC is the thirteenth edition of this standard. It includes revisions and errata released on December 21, 2012 and October 31, 2013. This edition incorporates a number of technical changes from the 2005 edition as outlined in Section B.30.2.2.

The results of PSR1 NBCC reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.30.2. As identified in Reference [B.30-4], the Pickering PSR2 review of the NBCC 2010 is an Incremental Review. PSR2

Incremental Review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- Compliance: Compliance indicates that the change in the safety requirement, per the topical review, is met.
- Gap: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.30.2 Compliance Assessment for Pickering PSR2

B.30.2.1 Application of PSR1 Reviews

The versions of the NBCC (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

Pickering Units 5-8

OPG Report NK30-REP-03680-00001 R000, "Pickering NGS-B Integrated Safety Review - Plant Design Safety Factor" [B.30-5] performed a clause-by-clause evaluation of Safe Operating Envelope (SOE) systems and Systems Important to Safety (SIS) against Part 4, "Structural Design Requirements" of the 2005 version of the NBCC [B.30-6]. The sections of the NBCC Volume 2 listed below are not applicable to the safety functions of SOE or SIS systems and were therefore not reviewed:

- Part 1, "General";
- Part 2, "Reserved";
- Part 5, "Environmental Separation";
- Part 6, "Heating, Ventilating and Air Conditioning";
- Part 7, "Plumbing Services";
- Part 8, "Safety Measures at Construction and Demolition Sites"; and
- Part 9, "Housing and Small Buildings".

NBCC Part 3, "Fire Protection, Occupant Safety and Accessibility" was reviewed as part of OPG Report NK30-REP-71400-10001 R001, "Fire Protection Code Compliance Review Pickering Nuclear Generating Station B" [B.30-7] and will be discussed separately below.

The buildings within the scope of the review performed in NK30-REP-03680-00001 R000 were the Reactor Auxiliary Building, Turbine Hall, Turbine Auxiliary Bay, Emergency Water System, Emergency Power System Building, Powerhouse, Standby Generator Building and Unit Emergency Control Centres. The assessment did not include the Pickering B Concrete Containment Structures (CCSs) since they were assessed separately in the Pickering B ISR against the CSA N287 Series Standards (i.e., N287.1, N287.2 and N287.3, which exceed the requirements in the NBCC as stated in Section 1.1 of CSA N287.1 [B.30-8]). It is noted that CSA N287.1, N287.2, N287.3 and N287.5 have also been assessed separately as part of Pickering PSR2.

NK30-REP-03680-00001 R000 [B.30-5] found a number of gaps which were all classified as Acceptable Deviations as outlined below (text in italics taken verbatim):

- Sections 4.1.5.15, 4.1.5.16 and 4.1.5.17 - Loads on Guardrails: *The NBC*[18] *1970 specifies a live load of 2.2 kN/m whereas the NBC 2005 has addressed the requirement of 3.0 kN/m on grandstands. However, this does not impact the design of the SOE & SIS systems… The NBC 1970 specifies a concentrated load of 0.56 kN at any point of access ways to equipment platforms whereas the NBC 2005 has addressed the requirement of 1.0 kN on similar locations. However, this does not impact the design of the SOE & SIS systems.*

- Section 4.1.6 - Loads Due to Snow and Rain: *As per the evaluation conducted for NBC 2005 load due to rain or snow and associated rain, the conclusion are as follows: i. Loads due to rain need not be considered associated with snow. ii. The loads due to snow is higher than that of rain, therefore rain will not have major impact on design and is not considered. iii. The maximum permissible loads due to snow on the structures as per NBC 2005 are less than the snow load considered for the design of structures of Pickering B as per their Design Manuals.*

- Section 4.2.3.6 - Protection Against Chemical Attack: *As stated in Reference 9, Part 2, Section 2A2 (Technical Specification for Pickering B concrete) [Specification NK30-LH-20541-01, Part 2], the foundations were designed to meet A23.1-1973. Although the recent version of CSA A23.1 may introduce new requirements, the intent of this issue is considered to be met.*

- 4.3.2.1 Design Basis for Plain & Reinforced Masonry: *CSA S304.1 is the general standard for masonry design and was used in the Pickering B design. Although the recent version of CSA S304.1 may introduce new requirements, the intent of this issue is considered to be met.*

- 4.3.3.1 Design Basis for Plain, Reinforced & Pre-Stressed Concrete: *The requirements of the Article 4.3.3.1 are in compliance with Article 4.5.2.1 of NBC 1970. Although the*

[18] Note that NK30-REP-03680-00001 R000 (as well as other OPG documents cited in this review) use the acronym NBC instead of NBCC for the National Building Code of Canada.

recent version of CSA A23.3 may introduce new requirements, the intent of this issue is considered to be met.

- 4.3.4.1 Design Basis for Structural Steel: *Reference 5 (NBC 1970 Design Supplement), Section 1.3 specifies that structural is to conform to the requirements of S16-1965. Although the recent version of CSA S16 may introduce new requirements, the intent of this issue is considered to be met.*
- 4.3.4.2 Design Basis for Cold-Formed Steel: *The requirements of the Article 4.3.4.2 are in compliance with Article 4.6.2.2 of NBC 1970. Although the recent version of CSA S136 may introduce new requirements, the intent of this issue is considered to be met.*

All Acceptable Deviations (1-051 to 1-061) were documented and dispositioned in OPG Report, NK30-REP-03680-00015 R000, "Pickering NGS-B Integrated Safety Review (ISR) - Final ISR Report" [B.30-9]. The rationale for these findings being classified as Acceptable Deviations is not impacted by Pickering NGS operation past 2020.

NK30-REP-03680-00001 R000 [B.30-5] concluded:

> *The structures of the buildings have consistently met their performance requirements throughout 25 years of station operation and there are ongoing inspection and test programs to assess fitness for service similar to the Periodic Inspection Plan which exists for Vacuum Building, Pressure Relief Duct & Reactor Building for Concrete Components and other in-service inspection and leak testing of structures. These test programs… can be credited with the ability to detect and monitor any safety significant ageing mechanism and to provide assurance of continued fitness for the service of those structures.*

As discussed under the PSR2 reviews of CSA N287.1, N287.3 and N287.5, the CCSs at Pickering B were built and tested to meet 1970 NBCC requirements [B.30-10], supplemented by specific loading requirements and the requirements of Design Manuals (e.g., see [B.30-11], [B.30-12], [B.30-13], [B.30-14]) and Design Guides (e.g., see [B.30-15], [B.30-16], [B.30-17]). The original Pickering concrete specifications (L-715-80 [B.30-18] and NK30-LH-20541-01 [B.30-19]) included requirements for quality control and compliance with CSA A23.1, A23.2 and A23.3 (which address concrete materials, methods of concrete construction and test methods and standard practices for concrete). Ongoing confirmation that the Pickering NGS CCSs remain fit for service is demonstrated via periodic inspections and in-service testing (i.e., CSA N287.7, "In Service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants" and N285.5, "Periodic inspection of CANDU Nuclear Power Plant Containment Components"). This testing, together with the aging management program, are credited with the ability to detect and monitor any safety significant degradation mechanisms and thus to provide assurance of continued fitness for service of the Pickering NGS CCSs.

As discussed earlier, Part 3 of the NBCC 2010, "Fire Protection, Occupant Safety and Accessibility" was reviewed as part of OPG Report NK30-REP-71400-10001 R001 [B.30-7]. As part of the Code Compliance Review (CCR), changes between the 1995 and 2005 editions of the NBCC were identified. The CCR concluded that the changes between the current and previous editions of the NBCC and other standards referenced therein have no safety impact on the existing physical features of the station, as they are not intended to be applied retroactively. As such, they would be applicable only to new construction including modifications undertaken at the station subsequent to adoption of the new codes and standards under the site operating license. Section 2.1 of [B.30-7] states:

> *The 1970 edition of the NBCC and the 1963 edition of the NFCC were in effect at the time that PNGS B was issued a construction permit (dated July 19, 1974). Therefore, fire safety requirements were enforced by application of the 1970 NBCC and the 1963 NFCC to PNGS B as the Codes of Record and are documented as such in the station operating license.*
>
> *The 1970 edition of the NBCC and the 1963 NFCC were used for the 2010 CCR as those portions of the 2005 NFCC that relate to the design and installation of structures, systems, and components are not retroactive. This concept was applied to any fixed system or component such as fire barriers, tanks, and piping systems for the transfer of combustible liquids. For these features, the requirements of the 1963 Edition of the NFCC were considered applicable…*
>
> *Systems and/or structures installed or constructed since May 2000 are required to be installed in accordance with the applicable NBCC and NFCC that were in force as referenced in the Power Reactor Operating License (PROL) for PNGS B at the time of design. Those installed or constructed between May 2000 and June 2008 are required to conform to the 1995 edition of the Codes. Those installed subsequent to June 2008 are required to conform to the 2005 edition.*

The updated 2010 CCR identified a number of new Deviations against the 1970 and 1975 editions of the NBCC which were minor in nature, as identified in Appendix C of NK30-REP-71400-10001 R001 [B.30-7]. These Deviations were subsequently addressed, per OPG Letter NA44-CORR-00531-06935 R000, "Pickering NGS 'A' - Request for CNSC Acceptance of the "Fire Safe Shutdown Analysis" (FSSA) and "Fire Hazard Assessment" (FHA) Reports and Status Update on CCR/ITM Deviations" [B.30-20]. The closure criteria for these findings is not impacted by Pickering NGS operation past 2020.

It is noted that CSA N293 addresses specific nuclear fire protection design, operational and performance requirements, and a review against the latest version of N293-12 (including of design-related aspects) is addressed as part of PSR2. In addition, Nuclear Oversight conducts an annual audit to assess Fire Protection provisions in accordance with CSA N293 Appendix E, "Fire Inspections and Audits" to provide assurance that the inspection requirements of the NFCC, including Inspection, Testing, and Maintenance (ITM) activities, are being met pursuant to the Nuclear Power Reactor Operating License [B.30-21].

Based on the above, there are no PSR2 gaps associated with the Pickering B ISR review (or subsequent CCR) which addressed compliance against the 2005 version of the NBCC. Compliance against the 2010 version of the NBCC is addressed under Section B.30.2.2 below.

Pickering Units 1,4

OPG Letter, NA44-CORR-00531-00381 R000, "Pickering A - Updated Basis for Return to Service Document" [B.30-22] included a CCR against the 1995 version of the NBCC, taking into account differences from the 1965 NBCC requirements [B.30-23] to which the CCSs at Pickering A were built and tested to meet. No significant deficiencies were identified. OPG undertook Regulatory Commitments to address the identified deviations prior to the return to service of Units 1 and 4, and these commitments were reflected in Section VI, Page VI-58 of [B.30-22]. The past dispositions of these deviations are not impacted by Pickering NGS operation beyond 2020.

OPG Report NA44-REP-71400-10001 R001, "Pickering Nuclear Generating Stations "A" Fire Protection Code Compliance Review" [B.30-24] provided an update to the original March 2000 CCR [B.30-25]. Changes between the 1995 and 2005 editions of the NBCC were identified. With respect to design-related aspects of the NBCC, the Pickering A CCR made similar arguments as in the Pickering B review with respect to retroactive application, i.e., that it is generally not practicable to make structural changes to existing buildings without rebuilding them. The updated 2010 CCR identified a number of new Deviations against the 1965 and 1970 editions of the NBCC which were minor in nature, per Appendix C of NA44-REP-71400-10001 R001 [B.30-24]. The findings were subsequently addressed to the satisfaction of the CNSC, per OPG Letter NA44-CORR-00531-06837 R000, "Pickering NGS A – CNSC Acceptance of Fire Protection Code Compliance Review and Third Party Review, Fixed Fire Protection Systems Inspection Testing and Maintenance" [B.30-26]. The closure criteria for these findings is not impacted by Pickering NGS operation past 2020.

Based on the above, there are no PSR2 gaps associated with the Pickering A Return to Service review (or the subsequent CCR) which addressed compliance against the 2005 version of the NBCC. As discussed earlier, compliance against the 2010 version of the NBCC is addressed under Section B.30.2.2 below.

Darlington NGS

OPG Report NK38-REP-03680-10048 R001, "Review of National Building Code of Canada (2005), National Building Code of Canada for Darlington Integrated Safety Review" [B.30-27] identified gaps at Darlington NGS related to requirements for the seismic design of buildings, guardrails and firewall design for lateral loads. OPG Report NK38-REP-03680-10104 R000, "Darlington NGS Integrated Safety Review (ISR) - Final ISR Report" [B.30-28] combined the gaps into Issues D077/D078, placed the fire protection related gaps under N293 Issues D044/D222, and classified all as Acceptable Deviations with very low safety significance and no further action required.

OPG Report NK38-REP-03680-10125 R002, "Gap Analysis to National Building Code of Canada 2005, Part 3, for Darlington Integrated Safety Review" [B.30-29] reviewed Part 3, "Fire Protection, Occupant Safety and Accessibility" of the NBCC as it had not been included in the NK38-REP-03680-10048 R001 review. This review found a number of gaps, but concluded:

> *Based on the small number of gaps identified relative to the number of clauses, and the generally minor nature of such gaps, the design and construction of the buildings*

within the scope of this review is deemed to be substantially in accordance with the requirements of the NBCC 2005.

Given the reviews completed for Pickering Units 1,4 and 5-8 against the 2005 version of the NBCC, the above reviews are not assessed further for applicability to PSR2. However, it is noted that OPG Reports NK38-REP-03680-10173 R000, "Code Refresh Review of National Building Code of Canada (2010)" [B.30-30] and NK38-REP-03680-10190 R001, "Code Review Refresh of the National Building Code of Canada: 2010 Edition" [B.30-31] compared the 2010 NBCC to the 2005 NBCC to identify if there were additional gaps beyond those identified in OPG Report NK38-REP-03680-10048 R001 [B.30-27]. NK38-REP-03680-10173 R000 [B.30-30] confirmed that changes in the 2010 NBCC do not affect the existing ISR, and the contents of that review, together with the reviews of NK38-REP-03680-10190 R001 [B.30-31], NK38-REP-03680-10048 R001 [B.30-27] and NK38-REP-03680-10125 R002 [B.30-29], have been utilized in Section B.30.2.2 below to assess Pickering NGS compliance against the 2005 and 2010 versions of the NBCC.

B.30.2.2 Application of Post PSR1 Reviews

Per the National Research Council of Canada, the major changes in the NBCC from 1995 to 2005 are summarized as follows [B.30-32]:

- Changes to Part 5 (Environmental Separation) and Part 6 (Heating, Ventilating and Air Conditioning (HVAC)), including new requirements for controlling air leakage and vapour diffusion, and revised wording to clarify the types of spaces in buildings that do not require mechanical ventilation (such as closets, storage rooms or other such spaces). These changes are not safety significant for PSR2.

- Changes to Part 9 (Housing and Small Buildings) which are not applicable to Pickering NGS.

- Changes to Part 3 (Fire Protection, Occupant Safety and Accessibility):

  - Materials of limited combustibility that pose a low fire risk are now allowed based on specific new test criteria. This is a relaxation of requirements and is therefore not safety significant for PSR2.

  - There has been a change from a prescriptive requirement to a more performance-based requirement to allow for materials other than masonry or concrete for the construction of firewalls requiring a rating of up to two hours (which would allow firewalls to be constructed of gypsum board provided certain conditions are met). This change in prescriptiveness is not safety significant for PSR2.

  - A number of changes have been made regarding mezzanines including: allowing the enclosure of the space below open mezzanines, redefining the point of reference to calculate the area of open mezzanines, imposing an area limit of 10% of the suite in which the mezzanine is located, allowing an enclosed space on open mezzanines, and clarifying the provisions for means of egress from mezzanines. This was assessed to be an Acceptable Deviation for Darlington in NK38-REP-03680-10125 R002 [B.30-29], since the implications of having to upgrade the building to meet the current code requirements (due to the code change in the determination of building height) are substantial and it is cost prohibitive to do so. Furthermore, Article 3.2.2.2 of NBCC 2005, which applies to the Powerhouse, provides for special considerations for structures of unusual proportion, special occupancy hazards and use. These considerations and conclusions also apply to Pickering NGS.

  - Larger non-metallic conduits within a fire compartment (without penetrating a fire separation) are permitted. This is a relaxation of requirements and is therefore not safety significant for PSR2.

- Changes to Part 4 (Structural Design), including:

  - Separation of snow and rain loads from "live load". Darlington was assessed to meet the intent of these changes in NK38-REP-03680-10048 R001 [B.30-27], given the use of conservative load factors and additional load combinations being considered in the design of various plant safety-related structures. These considerations also apply to Pickering NGS.

  - Earthquake data is more often described in the form of spectral acceleration values (related to motion in the ground) and is more geographically specific than the previous zonal values. Dynamic analysis was also established as the default method for analyzing earthquake design. Darlington was assessed to meet the intent of these changes in NK38-REP-03680-10048 R001 [B.30-27], since the seismic design of structures which contain safety-related systems is based primarily on the CSA N289 series of standards and not the NBCC. Since NBCC seismic requirements are only used on parts of the nuclear power plant that do not impact nuclear safety, it follows that changes to the NBCC requirements do not have an impact on nuclear safety. These considerations and conclusions also apply to Pickering NGS. It is noted that the CSA N289 series of standards are assessed separately as part of PSR2.

  - To establish a harmonized approach for calculating the environmental design loads for different categories of buildings, a table of "Importance Categories" (i.e., "low," "normal," "high" and "post-disaster" categories delineating hazard in the event of failure or required functionality in the event of a disaster) was created based on their use and occupancy. Darlington was assessed to meet the intent of these changes in NK38-REP-03680-10048 R001 [B.30-27], since the intent of the clauses was deemed to be unchanged from previous versions of the NBCC. Given that the intent has not changed, this is not safety significant for PSR2.

Per the National Research Council of Canada, the major changes in the NBCC from 2005 to 2010 are summarized as follows [B.30-33]:

- Changes to the organization of the Code (including clarifications). These do not affect the requirements.

- Changes to requirements for housing and small buildings, theatres and arenas/stadia (NBCC Part 9). These are not applicable to Pickering NGS.

- Crane and vehicle loads are more explicitly defined (NBCC Part 4). These changes are not safety significant for PSR2.

- Changes to the requirements for non-structural features such as window openings, stairs, handrails, radon levels and HVAC/ventilation (including particulate, ozone and carbon monoxide levels) (NBCC Parts 5 and 9). These changes are not safety significant for PSR2.

- A new occupancy classification for residential care facilities has been created (Group B3 occupancy) that relaxes requirements for smaller care occupancies having a limited number of occupants (NBCC Part 3, NFCC Part 2). These items are shared with the National Fire Code of Canada, and are not applicable to Pickering NGS.

- Revisions were made to seismic requirements related to site properties, irregularities, steel structures, static and dynamic procedures, and diaphragms. As outlined in [B.30-30], NBCC 2005 has similar requirements and the intent of these revisions has not changed (NBCC Part 4). Therefore, the conclusions of past Pickering NBCC 2005 code reviews remain valid.

- There are new wind load requirements for buildings with very long periods of vibration stating that they must now be designed by experimental methods and not dynamic calculations (NBCC Part 4). As outlined in [B.30-30], NBCC 2005 has similar requirements and the intent of these additions has not changed. Therefore, the conclusions of past Pickering NBCC 2005 code reviews remain valid.

- Seismic effects are now taken into account only for post-disaster buildings (i.e., buildings essential to the continued provision of services in the event of a disaster) (NBCC Part 5). As outlined in [B.30-30], NBCC 2005 has similar requirements and the intent of these additions has not changed. Therefore, the conclusions of past Pickering NBCC 2005 code reviews remain valid.

- To draw a clear line between the roles of the NBCC and the NFCC, building design requirements presently in the NFCC were moved to the NBCC (except for spill control measures). Appropriate cross-referencing between the two codes was added. This does not affect the requirements.

- New construction, sprinkler, emergency power and fire alarm requirements were added that are shared with the NFCC (NBCC Part 3, NFCC Part 2). In addition, Part 3 "Fire Protection, Occupant Safety and Accessibility" was updated to reflect the following changes:

- Additional fire protection requirements were introduced relating to the construction of all buildings in proximity to one another or to the property line;
- New requirements and clarifications were introduced for smoke alarm placement, commissioning of life safety and fire safety systems, and when fire alarm components must be installed;
- Definitions for "fire stops" and "fire blocks" have been added, as were several changes addressing penetrations through fire separations;
- Requirements addressing green pictograms conforming to ISO standards and photoluminescent exit signs were introduced;
- New protection requirements are provided for electrical conductors located in high buildings, fire pumps, refuge areas, and contained use areas; and
- New requirements were introduced for electrical supervision of water supply system valves.

The above changes were addressed in OPG Report NK38-REP-03680-10190 R001 [B.30-31], and gaps were identified that were subsequently grouped into ISR Issues D519 and D524 to D530. These were all later classified as Acceptable Deviations with no safety impact in References [B.30-34] to [B.30-41]. The rationale for these gaps being classified as Acceptable Deviations at Darlington NGS is also generally applicable to Pickering NGS, and include the following arguments:

- Fire Hazard Analysis (FHA) and the Fire Safety Shutdown Analysis (FSSA) demonstrate that the safety objectives of the Station can be met under postulated fire scenarios. An FHA and FSSA have also been completed for Pickering Units 1,4 and Units 5-8.
- In the event of a fire at the Station, personnel, as per training and as instructed via the public address system, are to avoid the incident unit and area as well as to refrain from using the elevators. This also applies to Pickering NGS.
- Access to the Station is limited to trained personnel who are familiar with the hazards and safety features of site buildings. Multiple means of egress are available and existing signage is clearly recognizable and understood by the personnel accessing the Station buildings. This also applies to Pickering NGS.
- Although there is a lack of smoke detection near the entrance to certain areas of the Station, it unlikely that these fires would go undetected and unreported given the plant is served by multiple alternate means of egress. There are also multiple alternate means of egress at Pickering NGS.
- With respect to electrical conductors, there are no high-rise buildings, areas of refuge, or contained use areas located at Darlington NGS. This also applies to Pickering NGS. The Darlington ISR originally identified that there was no evidence that the electrical cables serving Darlington fire pumps are protected in

according with this requirement; however, Pickering NGS does not have electrically driven fire pumps so this finding is not applicable.

- Although the absence of electronic supervision for fire protection water supply valves could result in a valve being closed without sending an alert to central alarm monitoring, water supply valves without electronic supervision are locked in an 'open' position at Darlington which would prevent unintended closure. As per Section 11.3 of Appendix B of [B.30-42] for Pickering Units 1,4 and Section 4.2 of [B.30-43] for Pickering Units 5-8, this is the expectation for Pickering NGS as well. Nevertheless, as discussed under the PSR2 review for NFPA 24, a number of yard post indicator valves at Pickering NGS without electronic supervision do not have locking mechanisms. Resolution of this finding is currently in progress with locks now installed on the majority of the affected valves. This was identified as a PSR2 gap under the NFPA 24 review (PSR2 NFPA 24 Gap #1) and is not duplicated as part of this NBCC review.

Based on the above, there are no Pickering NGS safety significant findings (PSR2 gaps) against the 2005 or 2010 versions of the NBCC. The changes to the 2005 and 2010 versions of the NBCC are largely incremental in nature due to minor improvements in knowledge and the addition of clarifications, and do not impact the overall adequacy of the previous Pickering Units 1,4 and Units 5-8 reviews against the NBCC (which are not impacted by Pickering NGS operation past 2020).

B.30.3 Compliance Summary for Pickering PSR2

There are no PSR2 gaps for NBCC (2010) [B.30-1]. Per the definition of Compliance for an Incremental Review, Pickering has a PSR2 Compliance associated with NBCC (2010).

B.30.4 References

[B.30-1] National Research Council Canada: Canadian Commission on Building and Fire Codes, *National Building Code of Canada 2010,* 2010; Includes Revision and Errata released on December 2012 and October 2013.

[B.30-2] CNSC Report, LCH-PNGS R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.30-3] CSA Standard, N293-07, *Fire Protection for CANDU Nuclear Power Plants*, February 2012.

[B.30-4] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.30-5] OPG Report, NK30-REP-03680-00001 R000, *Pickering NGS-B Integrated Safety Review – Plant Design Safety Factor*, August 2007.

[B.30-6] National Research Council Canada: Canadian Commission on Building and Fire Codes, *National Building Code of Canada 2005,* 2005; Includes Revision and Errata released on December 2007 and June 2008.

[B.30-7] OPG Report, NK30-REP-71400-10001 R001, *Fire Protection Code Compliance Review Pickering Nuclear Generating Station B*, November 23, 2010.

[B.30-8] CSA Standard N287.1-14, *General Requirements for Concrete Containment Structures for Nuclear Power Plants*, February 2014.

[B.30-9] OPG Report, NK30-REP-03680-00015 R000, *Pickering NGS-B Integrated Safety Review (ISR) – Final ISR Report,* August 2009.

[B.30-10] National Research Council of Canada, *National Building Code of Canada, 1970,* 1971.

[B.30-11] Ontario Hydro Pickering Generating Station B Design Manual, NK30-21000 R002, *Reactor Building - General*, issued February 1980, revised September 1988 and February 1989.

[B.30-12] Ontario Hydro Pickering Generating Station B Design Manual, NK30-21040 R001, *Reactor Building Floor Loadings*, issued August 1979, revised September 1988.

[B.30-13] Ontario Hydro Pickering Generating Station B Design Manual, NK30-21140 R002, *Reactor Building Foundation and Perimeter Wall*, issued November 1979, revised August 1982 and November 1988.

[B.30-14] Ontario Hydro Pickering Generating Station B Design Manual, NK30-21149 R002, *Reactor Building Dome*, issued January 1979, revised August 1982 and December 1988.

[B.30-15] AECL Engineering Design Guide, NK30-REF-68000-0379145 (DG-30-68000-6), *Containment Provisions for Extensions of the Containment Envelope for Pickering NGS B*, May 1977.

[B.30-16] AECL Engineering Design Guide, DG-00-01040-1 Rev. 02, *Earthquake Design Requirements for CANDU Nuclear Power Plants*, November 1974.

[B.30-17] OPG Design Basis Document, NK30-DBD-34200-00001 R000, *Containment System DBD*, February 2000.

[B.30-18] Specification L-715-80, *Specification for Concrete Placing and Workmanship.*

[B.30-19] Tendering and Contract Document, NK30-REF-20541-{47603}, *Supply of Pre-Mix Concrete in Ready Mix Trucks - Units 5-8*, NK30-LH-20541-01, April 1974.

[B.30-20] OPG Letter, NA44-CORR-00531-06935 R000, *Pickering NGS 'A' - Request for CNSC Acceptance of the "Fire Safe Shutdown Analysis" (FSSA) and "Fire Hazard Assessment" (FHA) Reports and Status Update on CCR/ITM Deviations*, June 28, 2012.

[B.30-21] OPG Plan, P-PLAN-09100-00001 R003, *Pickering Fire Safety Plan*, November 2015.

[B.30-22] OPG Letter, NA44-CORR-00531-00381, R.J. Strickert to J.S.C Tong, *Pickering A – Updated Basis for Return to Service Document*, April 20, 2001.

[B.30-23] National Research Council of Canada, *National Building Code of Canada, 1965*, 1966.

[B.30-24] OPG Report, NA44-REP-71400-10001 R001, *Pickering Nuclear Generating Station "A" Fire Protection Code Compliance Review*, March 2011.

[B.30-25] OPG Report, NA44-REP-71400-10001 R000, *Pickering Nuclear Generating Station "A" Fire Protection Code Compliance Review*, April 2000.

[B.30-26] OPG Letter, NA44-CORR-00531-06837 R000, *Pickering NGS A – CNSC Acceptance of Fire Protection Code Compliance Review and Third Party Review, Fixed Fire Protection Systems Inspection Testing and Maintenance,* December 9, 2011.

[B.30-27] OPG Report, NK38-REP-03680-10048 R001, *Review of National Building Code of Canada (2005), National Building Code of Canada for Darlington Integrated Safety Review*, October 2011.

[B.30-28] OPG Report, NK38-REP-03680-10104 R000, *Darlington NGS Integrated Safety Review (ISR) – Final ISR Report*, October 2011.

[B.30-29] OPG Report, NK38-REP-03680-10125 R002, *Gap Analysis to National Building Code of Canada 2005, Part 3, for Darlington Integrated Safety Review*, June 2013.

[B.30-30] OPG Report, NK38-REP-03680-10173 R000, *Code Refresh Review of National Building Code of Canada (2010)*, January 2014.

[B.30-31] OPG Report, NK38-REP-03680-10190 R001, *Code Review Refresh of the National Building Code of Canada*, *2010 Edition*, February 2014.

[B.30-32] National Research Council Canada website, *Archived – Significant Technical Changes in the 2005 NBC, NFC and NCP*, accessed October 25, 2016, from http://www.nrc-cnrc.gc.ca/ci-ic/article/v10n3-2.

[B.30-33] National Research Council Canada website, *Significant Technical Changes in the 2010 National Model Construction Codes*, accessed October 25, 2016, from https://www.nrc-cnrc.gc.ca/ci-ic/article/v15n4-2.

[B.30-34] NK38-REP-00770-0489261 R000, *Nuclear Refurbishment Issue Resolution Form – Darlington Issue #D523 Piping System Requirements*, January 2014.

[B.30-35] NK38-REP-00770-0489262 R000, *Nuclear Refurbishment Issue Resolution Form – Darlington Issue #D524 Fire and Smoke Characteristics of Electrical Wires and Cables*, January 2014.

[B.30-36] NK38-REP-00770-0489263 R000, *Nuclear Refurbishment Issue Resolution Form – Darlington Issue #D525 Elevator Requirements*, January 2014.

[B.30-37] NK38-REP-00770-0489264 R000, *Nuclear Refurbishment Issue Resolution Form – Darlington Issue #D526 Electronic Supervision of Water Supply Valves*, January 2014.

[B.30-38] NK38-REP-00770-0489265 R000, *Nuclear Refurbishment Issue Resolution Form – Darlington Issue #D527 Entrance Walkway Smoke Detector*, January 2014.

[B.30-39] NK38-REP-00770-0489266 R000, *Nuclear Refurbishment Issue Resolution Form – Darlington Issue #D528 Voice Communication Systems*, January 2014.

[B.30-40] NK38-REP-00770-0489267 R000, *Nuclear Refurbishment Issue Resolution Form – Darlington Issue #D529 Electrical Conductors for Fire Protection Systems*, January 2014.

[B.30-41] NK38-REP-00770-0489268 R000, *Nuclear Refurbishment Issue Resolution Form – Darlington Issue #D530 Exit Signage*, January 2014.

[B.30-42] OPG Design Manual, NA44-DM-71400-00002 R000, *Fire Protection Systems (Water),* February 2014.

[B.30-43] OPG Design Manual, NK30-DM-71400-00001 R006, *Fire Protection System*, January 2016.

B.31 NFCC (2010), "National Fire Code of Canada"

B.31.1 Background

The following, paraphrased from the 2010 National Fire Code of Canada (NFCC) [B.31-1], provides a brief overview of the purpose of this standard and the requirements expressed therein:

> *The National Fire Code of Canada purpose is to limit the probability that, as a result of specific circumstances related to the building or facility, a person in or adjacent to the building or facility will be exposed to an unacceptable risk of injury.*

As identified in Appendix A of [B.31-1]:

> *A.2.1.3.1.(.1) The National Building Code of Canada [NBCC] is most often applied to existing buildings when an owner wishes to rehabilitate a building, change its use, or build an addition; or when an enforcement authority decrees that a building, or a class of buildings, be altered for reasons of public safety. It is not intended that either the NBC [National Building Code] or the NFC be used to enforce the retrospective application of new requirements in the NBC to existing buildings…*
>
> *It is usually difficult to change structural features of an existing building when undertaking alterations or additions, but the installation of "active" fire protection systems, such as alarms, sprinklers and standpipes, in existing buildings may be possible. These systems may be considered as contributing to an adequate degree of life safety in cases where the structural features of a building do not conform to the NBC.*
>
> [Clause A.2.1.3.1.(.1)] *is intended to address the installation of fire alarm, sprinkler and standpipe systems in existing buildings presently not so equipped, and in existing buildings that do not provide an acceptable level of safety to meet the current installation standards specified in the NBC. It is not intended that existing fire protection systems that provide an acceptable level of life safety be upgraded with each new edition of the NBC or in conjunction with the inclusion of new requirements not in force at the time that a building was constructed.*

The NFCC is relevant to Safety Factor 1 (Plant Design).

Compliance with the NFCC is not a licence requirement for Pickering NGS (PROL 48.02/2018), although it is referred to in Section 9.1 of the R04 Pickering Licence Conditions Handbook [B.31-2] which states that OPG governance has identified specific Structures, Systems and Components (SSCs) in the protected area for which the requirements of CSA N293 [B.31-3] are not applied, and the requirements of the NFCC apply in those cases. Also, it may be considered an indirect licence requirement as N293 is a licence requirement and refers to the NFCC as follows:

> *5.1.3 Where specific design or operational requirements are not addressed in this Standard, the NBCC, or the NFCC, good engineering practice shall apply and, where*

appropriate, recognized Standards (such as those of the National Fire Protection Association [NFPA]) shall be used.

5.5.2.3 Except as otherwise indicated in this Standard, plants shall comply with all applicable requirements of the NFCC.

The 2010 edition of the NFCC is the ninth edition of this standard, and includes errata released in December 2012 and revision released in November 2013. This edition incorporates a number of technical changes from the 2005 edition as outlined in Section B.31.2.2.

The results of PSR1 NFCC reviews (Pickering A Return to Service assessments, and Pickering B and Darlington Integrated Safety Reviews (ISRs)), as well as reviews performed since PSR1, have been assessed for applicability to PSR2 in Section B.31.2. As identified in Reference [B.31-4], the Pickering PSR2 review of the NFCC (2010) is an Incremental Review. PSR2 Incremental Review includes an assessment of the intent of recent changes to the Law, Regulation, Code or Standard on a topic or subject-matter basis where there is potential to impact nuclear safety. The review identifies Compliances and Gaps, where required, as defined below:

- <u>Compliance:</u> Compliance indicates that the change in the safety requirement, per the topical review, is met.
- <u>Gap</u>: A Gap indicates that the change in the safety requirement, per the topical review, is not met.

B.31.2 Compliance Assessment for Pickering PSR2

B.31.2.1 Application of PSR1 Reviews

The versions of the NFCC (or its predecessors) subject to previous PSR1 reviews conducted for Pickering and Darlington, as well as their applicability to Pickering PSR2, are identified and discussed below.

Pickering NGS

<u>Pickering Units 5-8</u>

OPG Report NK30-REP-03680-00001 R000, "Pickering NGS-B Integrated Safety Review - Plant Design Safety Factor" [B.31-5] performed a clause-by-clause evaluation of Safe Operating Envelope (SOE) systems and Systems Important to Safety (SIS) against the 2005 version of the NFCC. The review found Pickering B to be in compliance, with 2 Acceptable Deviations (1-063/1-079) and 1 Discrepancy (1-469).

With respect to Discrepancy 1-469, NK30-REP-03680-00001 R000 stated:

According to Pickering B Design Manual, EPS [Emergency Power Supply] Generators Fuel Oil System, NK30-54860, Rev. 2, Jun 1982 and the Pickering B Design Manual, Standby Generators Fuel Oil System, NK30-54660, Rev. 2, Jul 1982, the Standby Generator and Emergency Power Generator tanks are allowed to be separated by 1/6 of the sum of the *two tank diameters. This is inconsistent with Section 4.3.2.2 (1) of the National Fire Code (NFC).*[19] *This is a documentation discrepancy only; the actual configurations of the storage tanks meet NFC requirements.*

Action Request # 28134694 (Assignment -06, relating to revision to Fuel Oil System Design Manuals to resolve the documentation discrepancies) was subsequently completed and Discrepancy 1-469 closed, per NK30-PLAN-00531-00001 R005, "Pickering 5-8 Continued Operations Plan" [B.31-6]. This resolution is not impacted by Pickering NGS operation past 2020.

Issue 1-079 was related to Clause 4.5.6.7 of NFCC 2005 regarding piping for flammable or combustible liquids at the entrance to buildings being above grade, and which was later reclassified from an Acceptable Deviation to a Discrepancy in OPG Letter NK30-CORR-00531-04739 R000, "Pickering NGS-B Integrated Safety Review - Discrepancy Resolution" [B.31-7]. OPG Report NK30-REP-03680-00016 R000, "OPG Response to CNSC Comments on Pickering NGS-B Integrated Safety Review - Plant Design, Safety Analysis, Safety Performance, Ageing and Equipment Qualification Safety Factors and Discrepancy Resolutions" [B.31-8] acknowledged CNSC agreement with OPG's disposition for Issue 1-079. In CNSC Letter NK30-CORR-00531-06324 R000 [B.31-9], CNSC staff requested additional information regarding Issue 1-079, and indicated it would be tracked via new Action Item I03-1. NK30-PLAN-00531-00001 R005 [B.31-6] addressed this and stated:

OPG has determined that the Pickering B Standby Generator and EPG [Emergency Power Generator] Fuel Oil storage tank systems comply with the NFCC. A review provided in Enclosure 5 NK30-REP-54600-00053, "Preliminary Tank Assessment Report Review of Pickering B Standby Generator and Emergency Power Generator Fuel Oil System" confirms OPG's interpretation of NFC Part IV Section 4.1.1.1 and subsequent implementation of CSA B139-00 and concludes that the Standby Generator and EPG Fuel Oil storage tank systems as designed meet the requirements of CSA B139-00. Furthermore, no changes to fuel oil piping requirements were identified from a review of CSA B139-00 to CSA B139-04. Hence OPG is in compliance with CSA B139-04 regarding piping at building entrances. Gap 1-079 is closed and this action is complete.

Based on the above, there is no Pickering Units 5-8 gap associated with Issue 1-079 (Action Item I03-1). However, a similar PSR2 gap for Pickering Units 1,4 has been identified in OPG Report P-REP-03680-00024 R000, "Pickering 5-8 Continued Operations Plan Review in Support of PNGS Periodic Safety Review 2" [B.31-10]. ID # 44, IIP Code I03-1, from Section 4.0 of P-REP-03680-00024 R000 states:

An assessment that shows that Standby Generator fuel tanks supporting Units 1,4 comply with NFCC could not be found.

The rationale for classification of Issue 1-063 as an Acceptable Deviation (which was related to NBCC 2005 Clauses 2.1.6.10 and 4.3.7.2 regarding the installation of impermeable dyke liners

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[19] Note that NK30-REP-03680-00001 R000 (as well as other OPG documents cited in this review) use the acronym NFC instead of NFCC for the National Fire Code of Canada.

for Standby Generator and the Emergency Power Generator tanks) is not impacted by operation past 2020.

OPG Report NK30-REP-03680-00001 R000 [B.31-5] did not assess the Pickering B Fire Protection System as part of the review, as it was not classified as an SOE or SIS system. However, a Code Compliance Review (CCR) was prepared in 2000 in OPG Report, NK30-REP-71400-10001 R000, "Fire Protection Code Compliance Review Pickering Nuclear Generating Station B" [B.31-11] to document compliance of Pickering B with the requirements of the NBCC, NFCC and applicable NFPA standards. Since completion of the original 2000 CCR, a new edition of CSA N293 was added to the Pickering PROL [B.31-2]. The updated standard (N293-07) requires that the original CCR be updated to reflect current station conditions. This was completed in OPG Report NK30-REP-71400-10001 R001, "Fire Protection Code Compliance Review Pickering Nuclear Generating Station B" [B.31-12]. A gap analysis was performed and documented prior to initiating the 2010 CCR update [B.31-13]. As part of the gap analysis, changes between the 1995 and 2007 editions of CSA N293 deemed to impact the CCR were identified as well as those between the 2005 and 1995 editions of the NBCC/NFCC.

Three main evaluations were performed for the 2010 CCR [B.31-12]:

- Update the status of code deviations identified in the original 2000 CCR [B.31-11];
- Evaluate new system/construction; and
- Evaluate buildings not covered by the original CCR (including Filtered Air Discharge (FAD) Tower[20], FAD Building, and Emergency Coolant Injection Shield Tower).

The 2010 CCR NK30-REP-71400-10001 R001 [B.31-12] states:

> *The approach used in the original CCR was to first analyze the structure in relation to the specific design requirements of the NBCC and NFCC. During the initial plant review and evaluation, experienced fire protection personnel identified areas of potential deviation from code criteria. The identification of deviations was based on the following considerations: a) The field inspection teams visually inspected existing fire protection features, including walls and doors used to enclose exit stairs, b) The field inspection teams reviewed existing fire protection feature documentation, inclusive of inspection and testing procedures, for identification of operational deficiencies, and c) Design deficiencies were identified by a review of design documentation, supplemented by walk downs as appropriate. Each deficiency was subjected to an engineering evaluation… The purpose of the engineering evaluation was to identify if the current plant construction and/or administrative control met the safety intent of the code through an alternate means… Where an alternate method of compliance for a potential design deficiency could not be substantiated, recommendations for corrections were provided. The recommendations, which consist of changes to physical plant or administrative*

[20] The FAD Tower houses the FAD system suction line.

controls, appropriately consider compliance to current versions of the code. The 2010 CCR followed the same general analysis approach used in the original CCR.

All past deviations from the 2000 CCR were resolved per Appendix E of NK30-REP-71400-10001 R001 [B.31-12]. The following new deviations against NFCC 2005 were identified in Appendix C of [B.31-12] (text taken verbatim is in italics):

- Clause 6.1.1.2 (Deviations No. 2010-0800 and 2010-0801): *Two fire hose cabinets were identified as being out of service. These fire hose cabinets included: Service Wing Extension, elevation 324 - #71410-FHC44, and RAB, elevation 317 - #71410-FHC83. Maintenance should be performed and the fire hose cabinets placed back in service in order to maintain the intended level of fire and life safety in the station. In addition, two fire hose cabinets were identified as out of service in the Service Wing Extension which include: FHC44, elevation 324 ft, Unit 018, and FHC 37, elevation 274, Unit 018. Signage provided on this FHC identified an out of service date of Feb 17, however, a year was not identified. Fire protection installations shall be maintained in operating condition (2005 NFCC, 6.1.1.2.). The fire hose cabinets should be repaired and placed back in service.*

These Deviations were subsequently addressed, as identified in OPG Letter NA44-CORR-00531-06935 R000, "Pickering NGS 'A' - Request for CNSC Acceptance of the "Fire Safe Shutdown Analysis" (FSSA) and "Fire Hazard Assessment" (FHA) Reports and Status Update on CCR/ITM Deviations" [B.31-14].

NK30-REP-71400-10001 R001 [B.31-12] states:

> *With respect to the NFCC, the gap analysis concluded that that the changes between the current and previous* [1995 and 2005] *edition of Code that impact the CCR are limited to inspection, testing and maintenance requirements for fire protection systems serving the station… The original CCR included a detailed analysis of the inspection, testing and maintenance requirements of fire protection equipment and features. This updated CCR excludes such discussion. The inspection, testing and maintenance requirements are addressed in separate evaluation reports addressing compliance of the fire protection program at the station; specifically the inspection, testing and maintenance of automatic and manual fire protection systems.*

The Inspection, Testing, and Maintenance (ITM) requirements that applied to the fire alarm life safety systems at Pickering B were addressed under OPG Report NK30-REP-71400-00027 R000, "Third Party Review: Fixed Fire Protection Systems Inspection, Testing and Maintenance Report" [B.31-15]. The objective of the Third Party Review was to assess OPG's evaluation of the ITM program as it applies to fixed fire protection systems at Pickering B for compliance with the requirements of applicable codes and standards and other relevant documents referenced therein, which included the 2005 edition of the NFCC. The review of OPG's compliance evaluation resulted in the identification of nine deviations specifically related to NFCC 2005 as discussed below (text taken verbatim is in italics). In each case, the Third Party vendor concurred with OPG's disposition.

- Clause 6.5.1.6 (1) (Deviation No. 64): *Description - A visual inspection of self-contained emergency lighting units must be conducted monthly. OPG Disposition - PMIDs will be created for direct compliance.*

- 6.5.1.7 (Deviation No. 66): *Description - Emergency lights must be inspected annually to ensure that they are functional. OPG Disposition - Emergency lights operating on Class II power do not require annual inspection to confirm functionality. These lights are in continuous operation, demonstrating their functionality. During a power outage Class II power switches to a battery supply, this supply is tested as per E1, E2, E5, and ES. Any light that fails during normal operation is repaired in accordance with N-PROC-MA-0008. Light failures would be detected by personnel conducting normal daily activities, or during shifty operator rounds, P-INS-09100-00004.*

- 6.5.1 (2)(a) (Deviation No. 67): *Description - Self-contained emergency lighting units must be tested monthly for operation upon failure of primary power supply.* (The Vendor reviewed Emergency Response Maintainer Procedure P-ERP-71400-00010, "Monthly Exit Inspections", and agreed this document addresses the issue.)

- 6.5.1.6 (2)(b) (Deviation No. 69): *Description - Test self-contained emergency lighting units to ensure that unit provides emergency lighting for duration equal to the design criterion under simulated power failure conditions. OPG Disposition - OPG will update PMIDs to test the emergency lights for the design period.*

- 6.5.1.6 (3) (Deviation No. 70): *Description - The charging conditions for voltage, current, and the recovery prior of self-contained emergency lighting units must be tested annually. OPG Disposition - PMIDs to be created for direct compliance.*

- 2.7.2.1 (1) (Deviation No. 76): *Description - All doors forming part of a means of egress shall be tested at intervals not greater than one month to ensure that they are operable. OPG Disposition - Meet the intent of this requirement as fire doors are checked monthly, exterior doors are checked by security daily and there are no public corridors where the passage of smoke would be a concern. Also, the buildings are large with high ceilings and staff are advised via PA [Public Address] if an incident occurs. The buildings are not open to the public, and OPG staff are required to initiate repairs to the plant, including doors, as per N-PROC-MA-0008 (work initiation, approval and prioritization).*

- 2.7.2.1 (2) (Deviation No. 77): *Description - The safety features of revolving doors shall be tested at intervals not greater than 12 months. OPG Disposition - Not applicable since PNGS B does not utilize revolving doors.*

- 2.7.2.1 (3) (Deviation No. 78): *Description - Sliding doors that are required to swing on their vertical axis in the direction of egress when pressure is applied shall be tested at intervals not greater than 12 months. OPG Disposition - Not applicable since PNGS B does not utilize sliding doors.*

- 2.7.2.1 (4) (Deviation No. 79): *Description - When doors are equipped with electrometric locks, these locks shall be tested at intervals not greater than 12 months. OPG Disposition - Not applicable since PNGS B does not utilize electromagnetic doors.*

OPG Report NK30-REP-71400-00027 R000 [B.31-15] states that "a satisfactory disposition has been reached to resolve all noted deviations."

Based on the above, there are no PSR2 gaps associated with the Pickering B ISR review (or subsequent CCR) which addressed compliance against the 2005 version of the NFCC. Compliance against the 2010 version of the NFCC is addressed under Section B.31.2.2 below.

Pickering Units 1,4

OPG Letter, NA44-CORR-00531-00381 R000, "Pickering A - Updated Basis for Return to Service Document" [B.31-16] included a CCR against the 1995 version of the NFCC, taking into account differences from the 1963 NFCC requirements [B.31-17] which Pickering A was built and tested to meet. No significant deficiencies were identified. OPG undertook Regulatory Commitments to address the identified deviations prior to the return to service of Units 1 and 4, and these commitments were reflected in Section VI, Page VI-58 of [B.31-16]. The past dispositions are not impacted by Pickering NGS operation past 2020.

As was done for Pickering B, a CCR was prepared in 2000 in OPG Report NA44-REP-71400-10001 R000, "Pickering Nuclear Generating Station "A" Fire Protection Code Compliance Review" [B.31-18] to document compliance of Pickering A with the requirements of the NBCC, NFCC and applicable NFPA standards. The original 2000 CCR was updated in 2010 to reflect current station conditions, as outlined in OPG Report NA44-REP-71400-10001 R001, "Pickering NGS A Fire Protection Code Compliance Review (CCR)" [B.31-19]. A gap analysis was also performed and documented prior to initiating the 2010 CCR update [B.31-20]. As part of the gap analysis, changes between the 2005 and 1995 editions of the NFCC were assessed. The 2010 CCR found nine deviations from the NFCC (including one remaining deviation from the 2000 CCR). All deviations were minor in nature with no impact on safety (e.g., missing labelling, burnt out light bulbs). The findings were addressed to the satisfaction of the CNSC per OPG Letter NA44-CORR-00531-06837 R000 [B.31-21]. The rationale for these findings being classified as non-safety significant is not impacted by Pickering NGS operation past 2020.

In addition to the above work, the ITM requirements that applied to the fire alarm life safety systems at Pickering Units 1,4 were addressed under OPG Report NA44-REP-71400-00022 R000, "Third Party Review: Fixed Fire Protection Systems Inspection, Testing and Maintenance Report" [B.31-22]. The objective of the Third Party Review was to assess OPG's evaluation of the ITM program as it applies to fixed fire protection systems at Pickering A for compliance with the requirements of applicable codes and standards and other relevant documents referenced, which included the 2005 edition of the NFCC. The review of OPG's compliance evaluation resulted in the identification of 10 deviations applicable to the 2005 NFCC. Similar to Pickering B, these deviations were not safety significant and have subsequently been addressed to the satisfaction of the CNSC, as identified in [B.31-14] and [B.31-21].

Based on the above, there are no PSR2 gaps associated with the Pickering A Return to Service review (or subsequent CCRs) which addressed compliance against the 1995 and 2005 versions of the NFCC. As discussed earlier, compliance against the 2010 version of the NFCC is addressed under Section B.31.2.2 below.

Darlington NGS

OPG Reports NK38-REP-03680-10049 R000, "Review of National Fire Code of Canada (2005) for Darlington Integrated Safety Review" [B.31-23] and NK38-REP-03680-10130 R002, "Gap Analysis of the National Fire Code of Canada 2005 Edition for the Darlington Integrated Safety Review" [B.31-24] found gaps against a number of clauses of NFCC 2005, largely because solid supporting evidence of compliance could not be retrieved. NK38-REP-03680-10130 R002 concluded:

> *Based on the small number of gaps identified relative to the number of clauses, and the generally minor nature of many of the gaps, the Station is deemed to be substantially in accordance with the requirements of the NFCC.*

Given the CCR reviews completed for Pickering Units 1,4 and 5-8 against the 2005 version of the NFCC, the above Darlington ISR reviews were not assessed further for applicability to PSR2. However, it is noted that OPG Report NK38-REP-03680-10188 R001, "Code Review Refresh of the National Fire Code of Canada, 2010 Edition" [B.31-25] reviewed the 2005 edition of the NFCC against the 2010 edition. Twelve gaps were identified and all were classified as having low safety significance. The content of NK38-REP-03680-10188 R001 has been utilized in Section B.31.2.2 below to assist in assessing Pickering NGS compliance against the 2010 version of the NFCC.

B.31.2.2 Application of Post PSR1 Reviews

Per the National Research Council of Canada, the major changes in the NFCC from 2005 to 2010 are summarized as follows [B.31-26]:

- Adjacent buildings or facilities must now be protected from fires originating from demolition or construction sites. Requirements for fire safety plans and fire department access to sites were improved. Specific requirements on the commissioning and decommissioning of standpipe systems, as well as restrictions on rooftop bitumen kettle placement, have been added.

  These changes are either not applicable to Pickering NGS, or are addressed via OPG Plan P-PLAN-09100-00001 R003, "Pickering Fire Safety Plan" [B.31-27]. Per [B.31-27]: "The Fire Safety Plan meets the requirements documented in N-PROG-RA-0012, "Fire Protection", in CAN/CSA N293-07, "Fire Protection for CANDU Nuclear Power Plants" (Fire Protection Plan, A6 Operation), and in the National Fire Code of Canada 2010, Section 2.8.2 Fire Safety Plan".

- A new occupancy classification for residential care facilities has been created (Group B3 occupancy) that relaxes requirements for smaller care occupancies having a limited number of occupants. These items are shared with the National Building Code of Canada, and are not applicable to Pickering NGS.

- New construction, sprinkler, emergency power and fire alarm requirements were added. These items are shared with the National Building Code of Canada, and have been addressed separately for PSR2 as part of that review.

- To draw a clear line between the roles of the NBCC and the NFCC, building design requirements presently in the NFCC were moved to the NBCC (except for spill control measures). Appropriate cross-referencing between the two codes was added. This does not affect the requirements.

- Limits to quantities of flammable and combustible liquids stored within buildings have been updated, including addition of new passive and active fire protective measures. Changes dealing with leak detection and monitoring, as well as handling of certain dangerous goods, have been introduced. Existing requirements relating to the detection and monitoring of storage tanks, sumps, and piping systems containing flammable and combustible liquids were also revised and new ones added.

  As outlined in NK38-REP-03680-10188 R001 [B.31-25], gaps identified for the Darlington ISR on these items related to referencing newer standards/guides, specifying different valve/pipe identification requirements, and minor changes to requirements in the 2010 NFCC versus the 2005 version. These items are not safety significant.

Based on the above, there are no Pickering NGS safety significant findings (PSR2 gaps) against the 2010 version of the NFCC. The changes to NFCC 2010 are largely incremental in nature due to minor improvements in knowledge and the addition of clarifications, and do not impact the overall adequacy of the previous Pickering Units 1,4 and Units 5-8 reviews against the 2005 version of the NFCC (which are not impacted by Pickering NGS operation past 2020).

It is noted that if a fundamental change in understanding occurs that could have a negative impact on safety, this is addressed in a timely fashion through Industry Operating Experience (OPEX). CSA N293 addresses specific nuclear fire protection design, operational and performance requirements, and a review against the latest version of N293-12 (including design-related aspects) is addressed as part of PSR2. In addition, Nuclear Oversight conducts an annual audit to assess Fire Protection provisions in accordance with CSA N293 Appendix E, "Fire Inspections and Audits" to provide assurance that the inspection requirements of the NFCC, including ITM activities, are being met pursuant to the Nuclear Power Reactor Operating License [B.31-27].

Further, OPG has a major fire protection program assessing and addressing any fire protection findings at Pickering NGS. A sampling of recent work includes:

- OPG Report NK30-REP-71400-00018 R000, "Fixed Fire Protection Systems Inspection, Testing and Maintenance" [B.31-28].

- OPG Report NA44-REP-71400-00021 R000, "Pickering A Fixed Fire Protection Systems Inspection, Testing, and Maintenance Code Compliance Report" [B.31-29].

- OPG Report, NA44-REP-71400-00027 R000, "Pickering NGS 014 Compliance with CSA N293-07, Fire Protection for CANDU Nuclear Power Plants" [B.31-30].

- OPG Report, NA44-REP-71400-00023 R000, "Fire Safe Shutdown Analysis - Pickering A Nuclear Generating Station" [B.31-31].

- OPG Report, NA44-REP-71400-10003 R001, "Fire Hazard Assessment - Pickering A Nuclear Generating Station" [B.31-32].

- OPG Report, NK30-REP-71400-00033 R000, "Pickering NGS 058 Compliance with CSA N293-07, Fire Protection for CANDU Nuclear Power Plants" [B.31-33].

- OPG Report, NK30-REP-71400-00001 R002, "Fire Safe Shutdown Analysis - Pickering B Nuclear Generating Station" [B.31-34].

- OPG Report, NK30-REP-71400-10002 R002, "Fire Hazard Assessment - Pickering B Nuclear Generating Station" [B.31-35].

- OPG Report, NA44-REP-71400-00034 R000, "Pickering A Buried Piping Third Party Review" [B.31-36].

- OPG Letter, NA44-CORR-00531-07592 R000, "Submission of Fire Protection Independent Third Party Code Compliance Review for Pickering A Firewater Pipe Replacement Project 13-80069" [B.31-37].

- OPG Letter, NK30-CORR-00531-06032 R000, "Pickering B - Response to CNSC Action Item 20118-2289 - Type II Inspection of Fire Protection Water Supply Systems" [B.31-38].

- OPG Letter, NA44-CORR-00531-06269 R000, "Pickering "A" - Installation of Diesel Engine Driven Fire Pumps (MEC 91665)" [B.31-39].

The above demonstrates that Pickering NGS has an extensive fire management program that is continually assessed for compliance against modern standards and any issues are addressed as part of the program.

B.31.3 Compliance Summary for Pickering PSR2

There is one PSR2 gap for NFCC (2010) [B.31-1], related to piping for flammable or combustible liquids at building entrances. The gap is related to Safety Factor 1 (Plant Design). As discussed in Section B.31.2.1, this issue is identified as a PSR2 gap in OPG Report P-REP-03680-00024 R000, "Pickering 5-8 Continued Operations Plan Review in Support of PNGS Periodic Safety Review 2". Therefore, a duplicate gap under NFCC (2010) has not been created.

B.31.4 References

[B.31-1] Canadian Commission on Building and Fire Codes, National Research Council Canada, *2010 National Fire Code of Canada*, 2010; Includes Revision and Errata released on December 2012 and November 2013.

[B.31-2] CNSC Report, LCH-PNGS R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.31-3] CSA Standard N293-12, *Fire Protection for Nuclear Power Plants*, October 2012.

[B.31-4] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.31-5] OPG Report, NK30-REP-03680-00001 R000, *Pickering NGS-B Integrated Safety Review – Plant Design Safety Factor*, August 2007.

[B.31-6] OPG Plan, NK30-PLAN-00531-00001 R005, *Pickering 5-8 Continued Operations Plan*, December 2015.

[B.31-7] OPG Letter, NK30-CORR-00531-04739, D.P. McNeill and P.F. Tremblay to T.E. Schaubel, *Pickering NGS-B Integrated Safety Review – Discrepancy Resolution*, April 17, 2008.

[B.31-8] OPG Report, NK30-REP-03680-00016 R000, *OPG Response to CNSC Comments on Pickering NGS-B Integrated Safety Review – Plant Design, Safety Analysis, Safety Performance, Ageing and Equipment Qualification Safety Factors and Discrepancy Resolutions*, September 2009.

[B.31-9] CNSC Letter, NK30-CORR-00531-06324 R000, *Pickering NGS-B: CNSC Staff Assessment of OPG's 2011 Continued Operations Plan (Action Item 2010-8-05 (2461)) and Path Forward*, e-Doc 3947907, June 19, 2012.

[B.31-10] OPG Report, P-REP-03680-00024 R000, *Pickering 5-8 Continued Operations Plan Review in Support of PNGS Periodic Safety Review 2 (PSR2),* January, 2017.

[B.31-11] OPG Report, NK30-REP-71400-10001 R000, *Fire Protection Code Compliance Review Pickering Nuclear Generating Station B*, May 2000.

[B.31-12] OPG Report, NK30-REP-71400-10001 R001, *Fire Protection Code Compliance Review Pickering Nuclear Generating Station B*, November 23, 2010.

[B.31-13] OPG Report, NK30-REP-71400-0299410 R000, *Fire Safety Assessment: Definition of Scope of Work for CSA N293-07 Compliance at PNGS-B*, August 2009.

[B.31-14] OPG Letter, NA44-CORR-00531-06935 R000, *Pickering NGS 'A' - Request for CNSC Acceptance of the "Fire Safe Shutdown Analysis" (FSSA) and "Fire Hazard Assessment" (FHA) Reports and Status Update on CCR/ITM Deviations*, June 28, 2012.

[B.31-15] OPG Report, NK30-REP-71400-00027 R000, *Third Party Review: Fixed Fire Protection Systems Inspection, Testing and Maintenance Report*, February 23, 2011.

[B.31-16] OPG Letter, NA44-CORR-00531-00381, R.J. Strickert to J.S.C Tong, *Pickering A – Updated Basis for Return to Service Document*, April 20, 2001.

[B.31-17] National Research Council Canada: Associate Committee on National Fire Codes, *1963 National Fire Code of Canada*, 1963.

[B.31-18] OPG Report, NA44-REP-71400-10001 R000, *Pickering Nuclear Generating Station "A" Fire Protection Code Compliance Review*, April 2000.

[B.31-19] OPG Report, NA44-REP-71400-10001 R001, *Pickering Nuclear Generating Station "A" Fire Protection Code Compliance Review*, March 2011.

[B.31-20] OPG Report, NA44-REP-71400-0300967 R000, *Fire Safety Assessment: Definition of Scope of Work for CSA N293-07 Compliance at PNGS-A*, August 2009.

[B.31-21] OPG Letter, NA44-CORR-00531-06837 R000, *Pickering NGS A – CNSC Acceptance of Fire Protection Code Compliance Review and Third Party Review, Fixed Fire Protection Systems Inspection Testing and Maintenance,* December 9, 2011.

[B.31-22] OPG Report, NA44-REP-71400-00022 R000, *Third Party Review: Fixed Fire Protection Systems Inspection, Testing and Maintenance Report,* March 15, 2011.

[B.31-23] OPG Report, NK38-REP-03680-10049 R000, *Review of National Fire Code of Canada (2005) for Darlington Integrated Safety Review*, August 2011.

[B.31-24] OPG Report, NK38-REP-03680-10130 R002, *Gap Analysis of the National Fire Code of Canada 2005 Edition for the Darlington Integrated Safety Review*, June 2013.

[B.31-25] OPG Report, NK38-REP-03680-10188 R001, *Code Review Refresh of the National Fire Code of Canada, 2010 Edition*, February 2014

[B.31-26] National Research Council Canada website, *Significant Technical Changes in the 2010 National Model Construction Codes*, accessed October 25, 2016, from https://www.nrc-cnrc.gc.ca/ci-ic/article/v15n4-2.

[B.31-27] OPG Plan, P-PLAN-09100-00001 R003, *Pickering Fire Safety Plan*, November 2015.

[B.31-28] OPG Report, NK30-REP-71400-00018 R000, *Fixed Fire Protection Systems Inspection, Testing and Maintenance*, November 2010.

[B.31-29] OPG Report, NA44-REP-71400-00021 R000, *Pickering A Fixed Fire Protection Systems Inspection, Testing, and Maintenance Code Compliance Report*, January 2011.

[B.31-30] OPG Report, NA44-REP-71400-00027 R000, *Pickering NGS 014 Compliance with CSA N293-07, Fire Protection for CANDU Nuclear Power Plants*, September 23, 2015.

[B.31-31] OPG Report, NA44-REP-71400-00023 R000, *Fire Safe Shutdown Analysis - Pickering A Nuclear Generating Station*, April 5, 2012.

[B.31-32] OPG Report, NA44-REP-71400-10003 R001, *Fire Hazard Assessment - Pickering A Nuclear Generating Station*, April 30, 2012.

[B.31-33] OPG Report, NK30-REP-71400-00033 R000, *Pickering NGS 058 Compliance with CSA N293-07, Fire Protection for CANDU Nuclear Power Plants*, September 23, 2015.

[B.31-34] OPG Report, NK30-REP-71400-00001 R002, *Fire Safe Shutdown Analysis - Pickering B Nuclear Generating Station*, October 5, 2011.

[B.31-35] OPG Report, NK30-REP-71400-10002 R002, *Fire Hazard Assessment - Pickering B Nuclear Generating Station*, November 23, 2011.

[B.31-36] OPG Report, NA44-REP-71400-00034 R000, *Pickering A Buried Piping Third Party Review*, January 8, 2016.

[B.31-37] OPG Letter, NA44-CORR-00531-07592 R000, *Submission of Fire Protection Independent Third Party Code Compliance Review for Pickering A Firewater Pipe Replacement Project 13-80069*, March 16, 2016.

[B.31-38] OPG Letter, NK30-CORR-00531-06032 R000, *Pickering B – Response to CNSC Action Item 20118-2289 – Type II Inspection of Fire Protection Water Supply Systems*, September 23, 2011.

[B.31-39] OPG Letter, NA44-CORR-00531-06269 R000, *Pickering "A" – Installation of Diesel Engine Driven Fire Pumps (MEC 91665)*, February 23, 2010.

B.32 CSA N290.8-15, "Technical Specification Requirements for Nuclear Power Plant Components"

B.32.1 Background

The following text from the Preface and Scope of CSA N290.8-15 [B.32-1] provides a brief overview of the purpose of this Standard and the requirements expressed therein:

> *This Standard is intended to ensure that technical specifications used to procure components are concise, consistent, and complete (i.e., identify all of the technical requirements and acceptance criteria). This Standard is not intended to add new requirements, codes, and standards, or interpretations to the component's design basis.*
>
> *This Standard has been written as a general standard for specifying components that will be installed in nuclear power plants. It establishes the requirements for design, procurement, installation, commissioning/testing, operation, maintenance, packaging, shipping that are to appear in the technical specifications for components.*
>
> *This Standard is one of a series of standards on reactor control systems, safety systems, and instrumentation for nuclear power plants.*
>
> *The CSA N-Series of Standards provides an interlinked set of requirements for the management of nuclear facilities and activities. CSA N286 provides overall direction to management to develop and implement sound management practices and controls, while the other CSA nuclear Standards provide technical requirements and guidance that support the management system. This Standard works in harmony with CSA N286 and does not duplicate the generic requirements of CSA N286; however, it may provide more specific direction for those requirements…*
>
> *The Standard:*
>
> a) *provides a consistent approach across the industry to produce specifications for components used in normal operation, anticipated operational occurrences (AOOs), and design basis accidents (DBAs)...*
>
> b) *provides a common set of attributes to be used when specifying component requirements related to plant design requirements;*
>
> c) *utilizes the product specific knowledge of the industry to provide common criteria for specifying technical requirements;*
>
> d) *provides a common set of attributes for use when specifying requirements for analysis, testing, design, manufacturing, and associated documentation to demonstrate components can meet their design requirements; and*
>
> e) *is not intended to require update of existing specifications.*

CSA N290.8-15 [B.32-1] is relevant to Safety Factor 1 (Plant Design).

CSA N290.8-15 is the first edition of this standard [B.32-1]. According to the N290.8-15 CSA Impact Statement and Public Review Notice [B.32-2]:

> *This Standard provides requirements and guidance on the content of technical specifications. The value of this standard is dependent on its implementation by operators…*
>
> *The Standard assumes that organizations preparing technical specifications has and maintains a quality assurance program that complies with the requirements of CSA N286-12.*
>
> *It is not necessary or expected that the requirements of this Standard would be applied retroactively to existing specifications that have been previously used to procure safety related components for use in a nuclear power plant (NPP).*
>
> *The Standard does not apply to the preparation of technical specifications for catalogue items that can be purchased without a detailed technical specification prepared by or on behalf of a NPP. Neither does the Standard apply to the preparation of technical specifications for raw material (plate steel, consumable materials, conduit etc.), services, and civil structures.*
>
> *The Standard does not provide requirements for specifications that are "strictly of a commercial nature" (i.e., Liability, warranty, escalation, insurance, liquidated damages, quantity, cancellation, etc.).*

Compliance with CSA N290.8-15 [B.32-1] is not currently a licence requirement for Pickering NGS (in accordance with PROL 48.02/2018) per the R04 Pickering Licence Conditions Handbook [B.32-3].

As identified in Pickering NGS PSR2 Basis Document [B.32-4], the Pickering PSR2 review of CSA N290.8-15 is a High Level review. For a PSR2 High Level review, the degree of conformance with clauses or groups of clauses in the Law, Regulation, Code or Standard (L/R/C/S) is demonstrated by supporting evidence stating whether the intent of the requirements stipulated in the requirement document is met. The review identifies Compliances and Gaps, where required, as defined below:

- <u>Compliance:</u> Compliance indicates that the intent of the safety requirement is met.
- <u>Gap</u>: A Gap indicates that the intent of the safety requirement is not met.

B.32.2 Compliance Assessment for Pickering PSR2

B.32.2.1 Application of PSR1 Reviews

CSA N290.8 was not reviewed as part of PSR1 as the document did not exist at the time that the previous Darlington and Pickering B Integrated Safety Reviews, and the Pickering A Return to Service assessments, were performed. A high level review of CSA N290.8-15 [B.32-1] is provided in Section B.32.2.2 below.

B.32.2.2 Application of Post PSR1 Reviews

According to the N290.8-15 CSA Impact Statement and Public Review Notice [B.32-2], the following is a "Summary of Significant Features" of N290.8:

- *Feature 1: This new Canadian Standard provides requirements for specifications used to procure components (or assemblies) for use in nuclear safety related applications within a Nuclear Power Plant (NPP).*
- *Feature 2: This Standard provides guidance to promote that technical specifications for components have been vetted with available industry representatives (suppliers) thus ensuring that suppliers are able to supply the component specified.*
- *Feature 3: This Standard promotes a consistent approach for use across the industry.*
- *Feature 4: This Standard identifies critical attributes necessary to promote that components perform their intended function(s) are listed within the technical specification(s) used to procure those components.*
- *Feature 5: This Standard recommends the preparer to exclude references which are not relevant to procurement, to avoid burdening the vendor with unnecessary documentation. This enables the vendors to be more efficient and ultimately avoid unnecessary cost.*
- *Feature 6: The Standard promotes that technical specifications include requirements which define the interface between the component and other components within NPP system.*
- *Feature 7: The Standard promotes that appropriate Codes and Standards are referenced within the technical specification.*
- *Feature 8: The Standard promotes that all appropriate documentation defining the pedigree of the component will be supplied.*
- *Feature 9: Annex A of the Standard provides informative guidance for use in preparing technical specification data sheets. Annex A also provides sample tech spec data sheets for a variety of components.*
- *Feature 10: Annex B of Standard provides informative guidance on the selection and inclusion of graded quality assurance requirements that may be included within the technical specifications as an alternative to the now withdrawn CSA Z299 series.*

A compliance review against CSA N290.8-15 [B.32-1] was not undertaken as part of previous PSR1 reviews and the following High Level assessment has been completed. In the review below, the degree of conformance with clauses or groups of clauses in the L/R/C/S is assessed for Pickering NGS by reference to supporting evidence stating whether the intent of the requirements stipulated in the L/R/C/S is met.

Due to the nature of N290.8-15, it is important to clearly state the intent of this standard within the context of PSR2.

1) CSA N290.8-15 is not intended to be applied retroactively

   The Impact Statement and Public Review Notice for N290.8-15 [B.32-2] includes explicit statements that the standard is not intended to be applied retroactively to previously issued specifications.

   For the purposes of PSR2, demonstrating whether the intent of the standard is met is determined by reviewing existing procedures that would be used to prepare specifications going forward.

2) CSA N290.8-15 is not a source of component-specific technical requirements

   As stated in the Preface section of N290.8-15 [B.32-1], the intent of this standard is to ensure that technical specifications used to procure components are concise, consistent, and complete. Technical requirements for a given component will vary on a case-by-case basis and be specified in accordance with the design basis for the corresponding system (e.g., per the corresponding system Design Manual). N290.8-15 provides direction on the type of requirements that need to be identified in a technical specification, but is not a source of specific technical requirements.

   For the purposes of PSR2, compliance is assessed by determining whether existing procedures governing the preparation of specifications provide direction for staff to include the information specified by N290.8-15.

Governing procedures for the preparation of technical specifications are N-PROC-MP-0059, "Preparation, Review, and Approval of Engineering Specifications" [B.32-5] and N-PROC-MP-0089, "Design Specifications, Design Reports and Overpressure Protection Reports" [B.32-6]. The majority of technical specifications will be prepared in accordance with N-PROC-MP-0059 [B.32-5] as this procedure is written such that it is not discipline-specific (e.g., procedure can be applied to procurement of mechanical, civil, electrical, instrumentation and control components, or software components). Only a small sub-set of mechanical components that perform a pressure boundary function require a technical specification to be prepared in accordance with N-PROC-MP-0089 [B.32-6].

Procedures and templates used to prepare specifications are written at a high level such that they can be readily used for components covering a range of technical disciplines (i.e., N-PROC-MP-0059 [B.32-5]) and/or a range of components within an individual technical discipline (i.e., N-PROC-MP-0059 [B.32-5] and N-PROC-MP-0089 [B.32-6]). Thus, the determination of whether the intent of a particular section of N290.8-15 is satisfied is primarily made by confirming that OPG governing procedures (and associated document templates) prompt the preparer of the specification to include the type of information specified in N290.8-15. Where appropriate, activities performed by Supply Chain and/or Procurement Engineering staff are considered as part of the PSR2 Review.

CSA N290.8-15 Clause	**PSR2 Review**	**Compliant or Gap**
0 Introduction	There are no requirements specified. The introduction describes the purpose of CSA N290.8-15.	N/A
1 Scope	There are no requirements specified. Sets context.	N/A
2 Reference Publications	There are no requirements specified. Describes the publications that CSA N290.8-15 refers to.	N/A
3 Definitions and Abbreviations	There are no requirements specified. Defines various words or phrases, or acronyms, used in CSA N290.8-15.	N/A
4 Preparation of Technical Specifications	There are no requirements specified. Heading only.	Compliant
4.1 Procurement *The technical specification should identify any unique procurement requirements that are not addressed by standard commercial or procurement provisions.*	The procurement of new items is governed by OPG-PROC-0060, "Requisitioning Items and Services" [B.32-7]. This procedure directs Supply Chain and Procurement Engineering to complete the following key activities: 1) Requisitioner identifies a suitable CAT ID in accordance with N-PROC-MM-0008, "Catalogue Information: Create, Maintain, and Replenish" [B.32-8]. 2) As part of setting a CAT ID to "Ready" status, a pre-purchase technical review is performed by Supply Chain staff and where appropriate, Procurement Engineering staff. Reviews by Procurement Engineering staff are performed in accordance with N-PROC-MP-0098, "Procurement Engineering Activities" [B.32-9]. 3) Once a CAT ID has been set to "Ready" and associated Material Request has been approved, OPG-PROC-0060 [B.32-7] directs Supply Chain staff to procure the component in accordance with OPG-PROC-0058, "Procurement Activities" [B.32-10]. The activities described above ensure that standard commercial or procurement provisions are translated into procurement requirements. As part of performing these activities, Supply Chain and Procurement Engineering staff will seek additional information to ensure unique procurement requirements (i.e., component-specific technical requirements) are documented and identified. Depending on the nature of the component that is being procured, these unique requirements are documented in an Engineering Specification in accordance with N-PROC-MP-0059, "Preparation, Review, and Approval of Engineering Specifications" [B.32-5] or a Design Specification in accordance with N-PROC-MP-0089, "Design Specifications,	Compliant

CSA N290.8-15 Clause	PSR2 Review	Compliant or Gap
	Design Reports and Overpressure Protection Reports" [B.32-6].	
4.2 Catalogue Component This covers: • Consideration of whether a technical specification is required if there exists a supplier's catalogue component that meets end-use requirements. • Identification of additional modifications/tests required for a supplier's catalogue components to satisfy end-use requirements.	OPG governance dictates that technical specifications will be prepared even if there exists a supplier's catalogue component that meets end-use requirements. This satisfies the intent of the requirement as the organization is not precluded from using technical specifications in these circumstances. Specifically, components are procured either through a Request for Quotation (RFQ) or a Request for Proposal (RFP) in accordance with OPG-PROC-0058 [B.32-10]. The RFQ/RFP package identifies all requirements needed to procure the component and makes reference to an Engineering Specification or a Design Specification.	Compliant
4.3 References Codes, Standards, and Generic Specifications This covers: • Identification of applicable Codes, Standards, and generic specifications. • Identification of year-date or revision number for Standards referenced in the specification. • Where applicable, identification of acceptable or required deviations from referenced Codes, Standards, or generic specifications.	Requirements related to applicable Codes, Standards, and generic specifications are documented in an Engineering Specification or a Design Specification. Engineering Specifications are prepared in accordance with N-PROC-MP-0059 [B.32-5] using N-TMP-10019, "Engineering Specification" [B.32-11]. Section 4.0 of N-TMP-10019 prompts the preparer to identify applicable Codes and Standards for the component [B.32-11]. In addition, Section 1.2 of N-PROC-MP-0059 requires the preparer to determine if there is a generic specification available which is adequate to define the requirements for the component [B.32-5]. Design Specifications are prepared for components performing a pressure boundary function in accordance with N-PROC-MP-0089 [B.32-6] using N-TMP-10190, "Design Specification" [B.32-12] or N-FORM-11612, "Valve Specification Data Sheet – Nuclear Class 1, 2, or 3 Valves" [B.32-13]. Appendix A of N-TMP-10190 requires the preparer to identify applicable Codes and Standards for the component [B.32-12]. Item 3 in N-FORM-11612 prompts the preparer to identify applicable Codes and Standards [B.32-13].	Compliant

CSA N290.8-15 Clause	PSR2 Review	Compliant or Gap
4.4 Component Standardization This covers: • Providing direction to select catalogue components if such components are determined to meet all technical requirements. • Minimization of inventory and maintenance costs by using organization's component standardization practices (e.g., use of previously purchased components that satisfy requirements).	Components are procured in accordance with OPG-PROC-0060 [B.32-7]. Step 1.1.2 of OPG-PROC-0060 [B.32-7] prompts the requisitioner to ensure technical requirements are identified prior to initiating a requisition. Components that are to be procured are assigned CAT IDs in accordance with N-PROC-MM-0008 [B.32-8]. Section 1.1.2 of N-PROC-MM-0008 [B.32-8] requires the requestor to complete a thorough search of the Master Materials Catalog and/or Standard Materials Catalog to determine if the required item already has a CAT ID. Once a CAT ID has been assigned, Appendix A of OPG-PROC-0060 [B.32-7] prompts the requisitioner to order the component through the OPG Web Catalogue, if possible. In addition, procedures governing the preparation of technical specifications direct staff to minimize the use of custom specifications as noted below: • Section 1.1 of N-PROC-MP-0059 [B.32-5] directs staff to maximize the use of generic Engineering Specifications. • Section 1.5 of N-PROC-MP-0089 [B.32-6] prompts the preparer to determine if the required document has previously been prepared.	Compliant
4.5 Supplier Capability *The preparer should consider the capability of the supplier community when preparing a technical specification so that gaps between the capability of available components and the critical characteristics required by the NPP application are addressed.*	Supplier capabilities are taken into consideration during the preparation of Engineering Specifications and Design Specifications. For Engineering Specifications, Section 1.2.3 of N-PROC-MP-0059 directs the preparer to ensure that there is sufficient information provided to enable vendors to design, manufacture, or supply components that meet the specified requirements [B.32-5]. In addition, Section 1.3 of N-PROC-MP-0059 specifies that Engineering Specifications are reviewed by Strategic Sourcing and Supply Planning and Procurement to review the manufacturing processes that have been specified and ensure alignment with good industry and business practices [B.32-5]. Only suitably qualified vendors on OPG's Approved Suppliers List would be asked to supply and/or procure these components [B.32-10]. Design Specifications, in accordance with N-PROC-MP-0089 [B.32-6] for pressure boundary components, must comply with the ASME Boiler and Pressure Vessel Code (BPVC) [B.32-14]. Only vendors on OPG's Approved Suppliers List qualified as being capable of performing pressure boundary work would be asked to supply and/or procure these components [B.32-10].	Compliant

CSA N290.8-15 Clause	PSR2 Review	Compliant or Gap
4.6 Units *The technical specification should maintain a single system of measurement consistent with that used in the applicable NPP and, when multiple units are used, the sequence of units of measure should be consistent.*	Pickering NGS is designed and operated with a standardized set of units of measurement that are reflected in all plant documentation. The development of technical specifications include extracting relevant information from existing plant documentation. Thus, the process of preparing a technical specification is structured such that requirements are identified in accordance with the units of measurement used by the station.	Compliant
4.7 Supplier Submissions *The technical specification shall require that the supplier list all* *a) Assumptions;* *b) Exceptions; and* *c) Identification and description of all digital items.*	The RFQ/RFP process described in OPG-PROC-0058 [B.32-10] prompts the supplier to identify any conditions to their responses or exceptions to requirements specified by OPG. Any conditions or exceptions noted by the supplier are considered in the bid evaluation process. However, OPG-PROC-0058 does not include explicit direction for the supplier to be required to identify/describe the use of all digital items in their equipment. The situation of interest is where a RFP/RFQ is issued for equipment that is not expected to include any digital items, but the supplier's design includes digital items. If the use of digital items is identified in advance of issuing the RFP/RFQ, applicable requirements would be specified in an Engineering Specification (per N-PROC-MP-0059 [B.32-5]) or a Design Specification (per N-PROC-MP-0089 [B.32-6]). However, if the equipment is not expected to contain any digital items, the use of these procedures would not result in a requirement for the supplier to self-identify whether their product contains any digital items. As requirements for supplier self-identification are not in OPG governance, this is identified as **PSR2 CSA N290.8-15 Gap #1**.	Gap
4.8 Design The technical specification shall include all applicable technical requirements and invoke the applicable referenced Codes and Standards.	Component-specific requirements are identified in accordance with the design requirements for the system that the component is to be installed in. Design requirements (e.g., physical limitations, functional requirements, performance requirements, etc.) are documented in an Engineering Specification or a Design Specification. Engineering Specifications are prepared in accordance with N-PROC-MP-0059 [B.32-5] using N-TMP-10019 [B.32-11]. Sections 4.0 and 5.0 of N-TMP-10019 [B.32-11] prompt the document preparer to include applicable design requirements. Design Specifications are prepared for components that perform a pressure boundary function in accordance with N-PROC-MP-0089 [B.32-6] using N-TMP-10190 [B.32-12] or N-FORM-11612 [B.32-13]. Section 2.1 and Appendix A of N-TMP-10190 [B.32-12] requires the preparer of the document to include applicable design requirements. Similarly, N-FORM-11612 [B.32-13] require the preparer of the document to	Compliant

CSA N290.8-15 Clause	PSR2 Review	Compliant or Gap
	include applicable design requirements. Additional direction is provided in Sections 1.2.5 and 1.2.6 of N-PROC-MP-0089 [B.32-6] for the preparer to ensure that all applicable technical requirements are captured in the Design Specification or Valve Specification Data Sheet.	
4.9 Component Interaction If the component being procured is part of an assembly, the technical specification should identify requirements for the interactions between individual components in the assembly. Where applicable, these requirements include: • Description of interaction between individual components in an assembly. • Identification of interface requirements (physical attributes and functional attributes). • Identification of individual component performance requirements and acceptance criteria.	Component interaction requirements are documented in an Engineering Specification or a Design Specification, depending on the components that are to be procured. Engineering Specifications are prepared in accordance with N-PROC-MP-0059 [B.32-5] using N-TMP-10019 [B.32-11]. Sections 4.0, 5.0, and 6.0 require the preparer to include requirements related to component interaction within an assembly. Design Specifications are prepared for components performing a pressure boundary function in accordance with N-PROC-MP-0089 [B.32-6] using N-TMP-10190 [B.32-12] or N-FORM-11612 [B.32-13]. Both N-TMP-10190 and N-FORM-11612 require the preparer to list all of the subject components and their associated requirements.	Compliant
4.10 Free Issue Material and Services *The technical specification may provide the supplier with information on the control of free issue material and services that will be made available by the operating organization.*	N-PROC-MP-0059 [B.32-5] and N-PROC-MP-0089 [B.32-6] do not prompt the preparer of a technical specification to include information regarding the control of free issue material and services. The control of free issue material is maintained in accordance with OPG-PROC-0060, "Requisitioning Items and Services" [B.32-7]. Existing OPG processes (i.e., OPG-PROC-0060 [B.32-7]) meet the intent of this requirement as N290.8-15 does not require the technical specification to be the governing document for the control of free issue material and services. The use of OPG-PROC-0060 meets the intent of the requirement for the use of free issue material.	Compliant
4.11 Operating Organization Supplied Design *In instances where portions of the design are to be supplied by the operating organization, the*	The intent of this requirement is to ensure that the division of responsibilities between OPG and its supplier for the completion of design activities is clearly documented. For activities being performed under an Engineering, Procurement, and Construction (EPC) agreement, Section 1.2	Compliant

CSA N290.8-15 Clause	PSR2 Review	Compliant or Gap
technical specification shall identify the portions of the design to be supplied and identify the supplier's responsibilities for integrating, and implementing the portions supplied by the operating organization.	of N-STD-MP-009, "Contractor/Owner Engineering Interface and Oversight" [B.32-15] dictates that the interface requirements between OPG and its supplier must be formally documented. N-COI-00120-00001, "Contractor/Owner Interface Requirements for Nuclear" [B.32-16] is used as a basis for preparing interface requirements which may be a standalone document or incorporated into other documentation such as an Engineering Specification. For activities being performed outside of an EPC agreement, arrangements for design services are made in accordance with OPG-PROC-0060 [B.32-7].	
4.12 QA Requirements *The technical specification shall identify the QA requirements applicable to the component and/or assembly to ensure that it meets all of its requirements.*	QA requirements for a component and/or assembly are documented in an Engineering Specification or a Design Specification. Engineering Specifications are prepared in accordance with N-PROC-MP-0059 [B.32-5] using N-TMP-10019 [B.32-11]. Section 3.0 of N-TMP-10019 [B.32-11] requires the document preparer to identify applicable QA requirements. Design Specifications are prepared for components performing a pressure boundary function in accordance with N-PROC-MP-0089 [B.32-6] using N-TMP-10190 [B.32-12] or N-FORM-11612 [B.32-13]. Both N-TMP-10190 and N-FORM-11612 are structured to identify QA requirements applicable to the ASME BPVC. Additionally, Sections 1.2.5 and 1.2.6 of N-PROC-MP-0089 require the preparer of the specification to include any additional QA requirements that apply to the subject component [B.32-6]. Once completed, Engineering Specifications and Design Specifications are included in a RFP/RFQ package. Activities related to the RFP/RFQ process are performed in accordance with OPG-PROC-0058 [B.32-10], which provides direction to ensure that applicable QA requirements are identified in the documentation package provided to potential suppliers.	Compliant
4.13 Documentation Requirements The technical specification shall provide a list of all engineering documents required to be provided by the supplier to design, install, operate, commission and maintain the component. Where applicable, the required documentation may include:	Documentation requirements are included in an Engineering Specification or a Design Specification. Engineering Specifications are prepared in accordance with N-PROC-MP-0059 [B.32-5] using N-TMP-10019 [B.32-11]. Section 10.0 of N-TMP-10019 requires the preparer of the document to identify all of the documents required to be provided by the supplier [B.32-11]. Design Specifications are prepared for components performing a pressure boundary function in accordance with N-PROC-MP-0089 [B.32-6] using N-TMP-10190 [B.32-12] or N-FORM-11612 [B.32-13]. Sections 1.2.5 and 1.2.6 of N-PROC-MP-0089 require the preparer to add sections or appendices to the form/template in order to identify	Compliant

CSA N290.8-15 Clause	PSR2 Review	Compliant or Gap
• Engineering documentation. • Installation manuals. • Commissioning/ operation manuals. • Maintenance manuals. • History files. • Other documentation as required.	documentation requirements specific to the subject component [B.32-6].	
4.14 Reliability and Maintainability Reliability and maintainability requirements shall be identified in a technical specification, including the corresponding operations and maintenance activities that shall be performed following component installation in order to demonstrate compliance with the specified requirements.	Reliability and maintainability requirements are documented in an Engineering Specification or a Design Specification. Engineering Specifications are prepared in accordance with N-PROC-MP-0059 [B.32-5] using N-TMP-10019 [B.32-11]. Sections 5.0 and 6.0 of N-TMP-10019 [B.32-11] require the preparer of the document to identify applicable reliability and maintainability requirements. Design Specifications are prepared in accordance with N-PROC-MP-0089 [B.32-6] using N-TMP-10190 [B.32-12] or N-FORM-11612 [B.32-13]. Sections 1.2.5 and 1.2.6 of N-PROC-MP-0089 [B.32-6] require the preparer of the document to add sections or appendices to the form/template in order to identify applicable reliability and maintainability requirements.	Compliant
4.15 Testing Required testing activities to support the procurement of a component shall be identified in a technical specification. These requirements may include: • Identification of testing requirements for the component that is being procured. • Identification of applicable type tests and/or production tests.	Testing requirements are documented in an Engineering Specification or a Design Specification. Component-specific testing requirements are identified based on safety credits for the corresponding plant system and the location where the component is to be installed (e.g., if a component is going to be installed in a system important to safety in a location that is subjected to harsh environmental conditions, as determined by the Environmental Qualification Room Conditions Manual, appropriate testing requirements for environmental qualification would be included in the specification). Engineering Specifications are prepared in accordance with N-PROC-MP-0059 [B.32-5] using N-TMP-10019 [B.32-11]. Section 6.0 of N-TMP-10019 [B.32-11] requires the preparer of the document to identify applicable testing requirements. Design Specifications are prepared for components performing a pressure boundary function in accordance with N-PROC-MP-0089 [B.32-6] using N-TMP-10190 [B.32-12] or	Compliant

CSA N290.8-15 Clause	PSR2 Review	Compliant or Gap
	N-FORM-11612 [B.32-13]. Both N-TMP-10190 and N-FORM-11612 require the preparer to identify testing requirements in accordance with the applicable sections of the ASME BPVC. Additionally, Sections 1.2.5 and 1.2.6 of N-PROC-MP-0089 [B.32-6] prompt the preparer of the document to add sections or appendices to the form/template in order to identify any additional testing requirements.	
4.16 Marking and Labelling *The technical specification shall identify the marking and labelling requirements consistent with the applicable Codes and Standards for the component being specified.*	Marking and labelling requirements are documented in an Engineering Specification or a Design Specification. Engineering Specifications are prepared in accordance with N-PROC-MP-0059 [B.32-5] using N-TMP-10019 [B.32-11]. Section 7.0 of N-TMP-10019 [B.32-11] requires the preparer of the document to identify applicable marking/labelling requirements. Design Specifications are prepared for components performing a pressure boundary function in accordance with N-PROC-MP-0089 [B.32-6] using N-TMP-10190 [B.32-12] or N-FORM-11612 [B.32-13]. Both N-TMP-10190 and N-FORM-11612 require the preparer to identify marking/labelling requirements in accordance with the applicable sections of the ASME BPVC. Additionally, Sections 1.2.5 and 1.2.6 of N-PROC-MP-0089 [B.32-6] prompt the preparer of the document to add sections or appendices to the form/template in order to identify any additional marking/labelling requirements.	Compliant
4.17 Cleanliness and Foreign Material Exclusion (FME) Requirements Where applicable, the technical specification shall identify requirements related to cleanliness and foreign material exclusion. These requirements may include : • Listing of FME covers, transportation supports, or other temporary items. • Requirements to control substances in contact with components and to prevent banned substances entering the plant where applicable.	Cleanliness and FME requirements are documented in an Engineering Specification or a Design Specification. Engineering Specifications are prepared in accordance with N-PROC-MP-0059 [B.32-5] using N-TMP-10019 [B.32-11]. Sections 3.0, 5.0, and 8.0 of N-TMP-10019 [B.32-11] require the preparer of the document to identify requirements related to cleanliness and FME exclusion. Design Specifications are prepared for components performing a pressure boundary function in accordance with N-PROC-MP-0089 [B.32-6] using N-TMP-10190 [B.32-12] or N-FORM-11612 [B.32-13]. Both N-TMP-10190 and N-FORM-11612 require the preparer of the document to identify requirements for cleanliness and FME in accordance with the applicable sections of the ASME BPVC. Additionally, Sections 1.2.5 and 1.2.6 of N-PROC-MP-0089 prompt the preparer of the document to add sections or appendices to the form/template in order to identify any additional cleanliness or FME requirements.	Compliant

CSA N290.8-15 Clause	**PSR2 Review**	**Compliant or Gap**
4.18 Lifting Points *In instances where the operating organization has special lifting constraints or requirements, the technical specification shall identify these requirements.*	Special lifting constraints or requirements are documented in an Engineering Specification or a Design Specification. These are a specific type of design requirement, as the component would need to be designed and fabricated to facilitate any lifting activities required to support installation or maintenance of the component. Engineering Specifications are prepared in accordance with N-PROC-MP-0059 [B.32-5] using N-TMP-10019 [B.32-11]. Section 5.0 of N-TMP-10019 [B.32-11] requires the preparer of the document to identify applicable design and fabrication requirements. Design Specifications are prepared for components performing a pressure boundary function in accordance with N-PROC-MP-0089 [B.32-6] using N-TMP-10190 [B.32-12] or N-FORM-11612 [B.32-13]. Sections 1.2.5 and 1.2.6 of N-PROC-MP-0089 [B.32-6] require the preparer of the document to add sections or appendices to the form/template in order to identify any special lifting requirements.	Compliant
4.19 Human Factors *The technical specification shall identify any applicable human factors requirements.*	Human factors requirements are a specific type of design requirement that would be documented, if applicable, in an Engineering Specification or a Design Specification. N-PROC-MP-0090, "Modification Process" [B.32-17] provides direction to identify the required design inputs from a human factors perspective. Engineering Specifications are prepared in accordance with N-PROC-MP-0059 [B.32-5] using N-TMP-10019 [B.32-11]. Sections 4.0 and 5.0 of N-TMP-10019 [B.32-11] require the document preparer to include applicable design requirements. Design Specifications are prepared for components performing a pressure boundary function in accordance with N-PROC-MP-0089 [B.32-6] using N-TMP-10190 [B.32-12] or N-FORM-11612 [B.32-13]. Section 2.1 and Appendix A of N-TMP-10190 [B.32-12] requires the preparer to include applicable design requirements. Similarly, N-FORM-11612 [B.32-13] requires the preparer to include applicable design requirements. Additional direction is provided in Sections 1.2.5 and 1.2.6 of N-PROC-MP-0089 [B.32-6] for the preparer of the document to ensure that all applicable technical requirements are captured in the Design Specification or Valve Specification Data Sheet.	Compliant

CSA N290.8-15 Clause	PSR2 Review	Compliant or Gap
4.20 Fire Protection and Hazards *The technical specification shall identify any applicable fire protection and hazard requirements.*	N-PROC-MP-0090, “Modification Process” [B.32-17] provides direction to identify the required design inputs from a fire protection perspective. When N-PROC-MP-0090 [B.32-17] determines fire protection requirements are applicable for a specific component, appropriate requirements would be documented in an Engineering Specification or a Design Specification. Engineering Specifications are prepared in accordance with N-PROC-MP-0059 [B.32-5] using N-TMP-10019 [B.32-11]. Sections 4.0 and 5.0 of N-TMP-10019 [B.32-11] require the document preparer to include applicable design requirements, including applicable Laws, Regulations, Codes and Standards (specific section, step, or clause of the document, including the applicable edition of the code). Therefore, design requirements related to fire protection would be specified in accordance with relevant Codes and Standards, such as the National Fire Code of Canada [B.32-18] and CSA N293-07, “Fire Protection for CANDU Nuclear Power Plants” [B.32-19]. Design Specifications are prepared for components performing a pressure boundary function in accordance with N-PROC-MP-0089 [B.32-6] using N-TMP-10190 [B.32-12] or N-FORM-11612 [B.32-13]. Section 2.1 and Appendix A of N-TMP-10190 [B.32-12] requires the preparer to include design requirements based on applicable Laws, Regulations, Codes and Standards. Similarly, N-FORM-11612 [B.32-13] requires the preparer to include design requirements. Additional direction is provided in Sections 1.2.5 and 1.2.6 of N-PROC-MP-0089 [B.32-6] for the preparer of the document to ensure that all applicable technical requirements are captured in the Design Specification or Valve Specification Data Sheet. Therefore, design requirements related to fire protection would be specified in accordance with relevant Codes and Standards, such as the National Fire Code of Canada [B.32-18] and CSA N293-07 [B.32-19].	Compliant
4.21 Operating Conditions The technical specification shall include the most bounding set of operating conditions anticipated during service for the intended application.	The conditions that a component must be designed to operate under are documented in an Engineering Specification or a Design Specification. However, CSA N290.8-15 [B.32-1] is not intended to be used as the basis for identifying the conditions/scenarios a component must be designed to withstand. This would be determined in accordance with the design basis for the system that the component is to be installed in. Engineering Specifications are prepared in accordance with N-PROC-MP-0059 [B.32-5] using N-TMP-10019 [B.32-11]. Section 5.0 of N-TMP-10019 [B.32-11] requires the document preparer to include the operating conditions that the component must be designed to be capable of withstanding. Design Specifications are prepared for components performing a pressure boundary function in accordance with	Compliant

CSA N290.8-15 Clause	PSR2 Review	Compliant or Gap
	N-PROC-MP-0089 [B.32-6] using N-TMP-10190 [B.32-12] or N-FORM-11612 [B.32-13]. Both N-TMP-10190 and N-FORM-11612 require the preparer of the document to identify the operating conditions that the component must be designed to be capable of withstanding.	
4.22 Component Qualification The technical specification shall identify all requirements needed to provide documented assurance that the component will operate as intended under the conditions dictated by the requirements.	Component qualification requirements, including the completion of required testing/analysis activities and production of corresponding documentation, are documented in an Engineering Specification or a Design Specification. Engineering Specifications are prepared in accordance with N-PROC-MP-0059 [B.32-5] using N-TMP-10019 [B.32-11]. Sections 4.0, 5.0, 6.0, and 10.0 of N-TMP-10019 [B.32-11] require the preparer of the document to include requirements such that the component can be qualified for its intended use. Design Specifications are prepared for components performing a pressure boundary function in accordance with N-PROC-MP-0089 [B.32-6] using N-TMP-10190 [B.32-12] or N-FORM-11612 [B.32-13]. N-FORM-11612 [B.32-13] and Section 3.0 of N-TMP-10190 [B.32-12] require the preparer to include applicable component qualification requirements. Additionally, Sections 1.2.5 and 1.2.6 of N-PROC-MP-0089 require the preparer to add sections or appendices to the form/template in order to identify documentation requirements specific to the subject component [B.32-6].	Compliant
4.23 Digital Items Requirements related to the use of digital items shall be included in a technical specification. Where applicable, the technical specification shall address the following requirements: • Identification of whether the use of a digital item is acceptable. • If it is acceptable to use a digital item, identification of documentation required to enable qualification to CSA N290.14.	Per Section 3 of CSA N290.8-15 [B.32-1], a digital item encompasses pre-developed software, custom software, software engineering tools and digital hardware. However, CSA N290.8-15 is not intended to be used as a source of requirements for the use of digital items; such requirements would be specified in accordance with the design basis for the system that the component is to be installed in. Where applicable, requirements related to digital items would be documented in an Engineering Specification in accordance with N-PROC-MP-0059 [B.32-5] using N-TMP-10019 [B.32-11]. Section 10.0 of N-TMP-10019 requires the preparer of the document to identify all of the documents required to be provided by the supplier [B.32-11]. Additional requirements regarding the use of digital items would be identified using the governance listed below, that provide direction to document applicable requirements in an Engineering Specification: • N-PROC-MP-0049, "Procurement of Software and Products Containing Software" [B.32-20] • N-PROC-MP-0099, "Development of Real-Time Process Computing System" [B.32-21]	Compliant

CSA N290.8-15 Clause	PSR2 Review	Compliant or Gap
	• N-STI-69000-10013, "Computer System Requirements and Design" [B.32-22] • N-INS-69000-10002, "Computer Spare Parts and Other Electronic Components Acquisition by Computers and Control Design Department" [B.32-23]. A separate review has been performed in support of PSR2 to assess compliance with CSA N290.14, "Qualification of Digital Hardware and Software for Use in Instrumentation and Control Applications for Nuclear Power Plants" [B.32-24].	
4.24 Cyber Security *The technical specification shall include the applicable cyber security requirements from CSA N290.7.*	Where applicable, cyber security requirements are identified in accordance with N-PROC-MP-0049 [B.32-20], which directs staff to prepare an Engineering Specification. Engineering Specifications are prepared in accordance with N-PROC-MP-0059 [B.32-5] using N-TMP-10019 [B.32-11]. Sections 3.0, 4.0, 5.0, 6.0, and 10.0 of N-TMP-10019 [B.32-11] require the preparer of the document to identify applicable design requirements, which includes cyber security requirements. A separate review [B.32-25] of compliance with CSA N290.7 has been performed, which meets the intent of PSR2. Due to security and confidentiality constraints, the findings of this review will not be discussed in PSR2.	Compliant
5 Instrumentation and Control Components This section of CSA N290.8-15 covers the identification of discipline-specific requirements not explicitly stated in Section 4. This includes: • Design requirements. • Functional requirements. • Performance requirements. • Testing requirements.	Per CSA N290.8-15 [B.32-1], Section 5 is intended to be used in conjunction with Section 4 to ensure requirements specific to Instrumentation & Control (I&C) components are identified in technical specifications. CSA N290.8-15 is not intended to be used as a source of technical requirements. Thus, compliance is assessed by determining if applicable OPG procedures and templates direct staff to include the type of information specified in Section 5 of N290.8-15. Specifications for I&C components are prepared in accordance with N-PROC-MP-0059 [B.32-5] using N-TMP-10019 [B.32-11]. Sections 4.0, 5.0, and 6.0 of N-TMP-10019 [B.32-11] require the preparer of the document to include the type of information specified by N290.8-15.	Compliant
6 Mechanical Components This section of N290.8-15 covers the identification of discipline-specific requirements not	Per CSA N290.8-15 [B.32-1], Section 6 is intended to be used in conjunction with Section 4 to ensure requirements specific to mechanical components are identified in technical specifications.	Compliant

CSA N290.8-15 Clause	PSR2 Review	Compliant or Gap
explicitly stated in Section 4. This includes: • Design requirements. • Functional requirements. • Performance requirements. • Testing requirements. • Component qualification requirements. • Valve requirements. • Pump requirements. • Heat exchanger requirements. • Requirements for Heating, Ventilation and Air Conditioning (HVAC) components. • Requirements for other mechanical components not listed above.	CSA N290.8-15 is not intended to be used as a source of technical requirements. Thus, compliance is assessed by determining if applicable OPG procedures and templates direct staff to include the type of information specified in Section 6 of N290.8-15. Requirements for mechanical components are documented in an Engineering Specification or a Design Specification, as appropriate. Engineering Specifications are prepared in accordance with N-PROC-MP-0059 [B.32-5] using N-TMP-10019 [B.32-11]. Sections 4.0, 5.0, and 6.0 of N-TMP-10019 [B.32-11] require the preparer of the document to include the type of information specified by N290.8-15. Design Specifications are prepared for components performing a pressure boundary function in accordance with N-PROC-MP-0089 [B.32-6] using N-TMP-10190 [B.32-12] or N-FORM-11612 [B.32-13]. Section 2.1 and Appendix A of N-TMP-10190 [B.32-12] require the preparer of the document to include the type of information specified by N290.8-15 (i.e., identification of applicable requirements per ASME BPVC). Similarly, N-FORM-11612 [B.32-13] requires the preparer to include the type of information specified in N290.8-15. Additional direction is provided in N-PROC-MP-0089 for the preparer of the document to ensure that all applicable technical requirements are captured in the Design Specification or Valve Specification Data Sheet [B.32-13].	
7 Electrical Components This section of N290.8-15 covers discipline-specific requirements not explicitly stated in Section 4. This includes: • Design requirements. • Functional requirements. • Performance requirements. • Testing requirements. • Maintenance requirements.	Per CSA N290.8-15 [B.32-1], Section 7 is intended to be used in conjunction with Section 4 to ensure requirements specific to electrical components are identified in technical specifications. CSA N290.8-15 is not intended to be used as a source of technical requirements. Thus, compliance is assessed by determining if applicable OPG procedures and templates direct staff to include the type of information specified in Section 7 of N290.8-15. Specifications for electrical components are prepared in accordance with N-PROC-MP-0059 [B.32-5] using N-TMP-10019 [B.32-11]. Sections 4.0, 5.0, and 6.0 of N-TMP-10019 [B.32-11] require the preparer of the document to include the type of information specified by N290.8-15.	Compliant

CSA N290.8-15 Clause	PSR2 Review	Compliant or Gap
Annex A (Informative) – Data Sheet Contents	Provides information and does not establish any requirements.	N/A
Annex B (Informative) – National and International Standards Bodies	Provides information and does not establish any requirements.	N/A
Annex C (Informative) – I&C Safety Categories	Provides information and does not establish any requirements.	N/A
Annex D (Informative) – Commercial Guidance	Provides information and does not establish any requirements.	N/A

There are no references to CSA N290.8-15 in OPG governance. Specifically, [B.32-5] does not refer to the standard. However, because the above high level review identified that OPG's governance aligns with all but one of the requirements in the standard, the absence of a reference to the standard does not have any safety significance. Furthermore, CSA N290.8-15 is not currently listed in the Pickering PROL or Licence Conditions Handbook. Therefore, this is not a PSR2 gap.

B.32.3 Compliance Summary for Pickering PSR2

There is one PSR2 CSA N290.8-15 gap which relates to Safety Factor 1 (Plant Design):

1. Clause 4.7 of CSA N290.8-15 mandates that the technical specification requires the supplier to identify and describe all digital items included in their equipment. In the event that the use of digital items is identified by OPG in advance of issuing a Request for Proposal (RFP) or Request for Quotation (RFQ), existing OPG procedures are adequate for ensuring that requirements related to digital items are documented in the technical specification. However, a requirement for a supplier to self-identify whether their product contains any digital items is not reflected in OPG governing documents. This has therefore been identified as a PSR2 gap.

B.32.4 References

[B.32-1] CSA Standard N290.8-15, *Technical Specification Requirements for Nuclear Power Plant Components*, November 2015.

[B.32-2] CSA Communication, Impact Statement and Public Notice for CSA N290.8, *Technical Specification Requirements for Nuclear Power Plant Components*, December 2015.

[B.32-3] CNSC Report, LCH-PNGS-R004, OPG File No. P-CORR-00531-04633, *Pickering NGS: Licence Conditions Handbook*, December 2015.

[B.32-4] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016.

[B.32-5] OPG Procedure, N-PROC-MP-0059 R009, *Preparation, Review and Approval of Engineering Specifications*, April 2015.

[B.32-6] OPG Procedure, N-PROC-MP-0089 R011, *Design Specifications, Design Reports and Overpressure Protection Reports*, November 2015.

[B.32-7] OPG Procedure, OPG-PROC-0060 R005, *Requisitioning Items and Services,* October 2015.

[B.32-8] OPG Procedure, N-PROC-MM-0008 R014, *Catalogue Information: Create, Maintain and Replenish,* October 2015.

[B.32-9] OPG Procedure, N-PROC-MP-0098 R007, *Procurement Engineering Activities*, June 2014.

[B.32-10] OPG Procedure, OPG-PROC-0058 R009, *Procurement Activities*, May 2015.

[B.32-11] OPG Template, N-TMP-10019 R012, *Engineering Specification*, May 2016.

[B.32-12] OPG Template, N-TMP-10190 R004, *Design Specification*, October 2013.

[B.32-13] OPG Form, N-FORM-11612 R001, *Valve Specification Data Sheet – Nuclear Class 1, 2, or 3 Valves,* January 2016.

[B.32-14] ASME Boiler and Pressure Vessel Code, Section III Division 1, NCA 3000 series.

[B.32-15] OPG Standard, N-STD-MP-0009 R005, *Contractor/Owner Engineering Interface and Oversight*, August 2014.

[B.32-16] OPG Contractor/Owner Interface Agreement, N-COI-00120-00001 R00, *Contractor/Owner Interface Requirements for Nuclear*, July 2013.

[B.32-17] OPG Procedure, N-PROC-MP-0090 R012, *Modification Process,* April 2015.

[B.32-18] Canadian Commission on Building and Fire Codes, National Research Council Canada, *2010 National Fire Code of Canada,* 2010.

[B.32-19] CSA Standard, N293-07, *Fire Protection for CANDU Nuclear Power Plants,* February 2012.

[B.32-20] OPG Procedure, N-PROC-MP-0049 R008, *Procurement of Software and Products Containing Software*, March 2014.

[B.32-21] OPG Procedure, N-PROC-MP-0099 R004, *Development of Real-Time Process Computing Systems*, March 2016.

[B.32-22] OPG Standard, N-STI-69000-10013 R002, *Computer System Requirements and Design*, April 2016.

[B.32-23] OPG Instruction, N-INS-69000-10002 R000, *Computer Spare Parts and Other Electronic Components Acquisition by Computers and Control Design Department*, March 2009.

[B.32-24] CSA Standard, N290.14-15, *Qualification of Digital Hardware and Software for Use in Instrumentation and Control Applications for Nuclear Power Plants,* November 2015.

[B.32-25] OPG Report, N-REP-69000-10003 R000, *Gap Analysis Between CSA N290.7-14 Cyber Security Requirements for Nuclear Power Plants and Small Reactor Facilities,* March 2016.

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ONTARIO POWER GENERATION	
ACCEPTED	✓
ACCEPTED AS NOTED	
REVISE AND RESUBMIT	
Signature	Date: 17 Jan 2017
Name: Mike Ruffolo, Manager	
Dept: Pickering Engineering - Aging Management & Strategic Initiatives	
OPG Proprietary	
Doc No.: P-REP-03680-00024	Rev: 000
This acceptance does not relieve the contractor from responsibility for errors or omissions or from any obligations or liability under this contract.	

Pickering 5-8 Continued Operations Plan Review in Support of PNGS Periodic Safety Review 2 (PSR2)

PS112/RP/021 R01

January 12, 2017

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Revision Summary

Rev	Date	Author	Comments
R00	December 2, 2016	S. B. Harvey	Initial issue for OPG review and comment.
R01	January 12, 2017	S. B. Harvey	Address OPG comments

EXECUTIVE SUMMARY

Ontario Power Generation (OPG) is performing a Periodic Safety Review (PSR) in support of continued operation of Pickering Nuclear Generating Station (NGS) beyond 2020. The PSR (referred to as "PSR2") is a subsequent PSR building on the review basis of earlier OPG Integrated Safety Reviews and other associated assessments.

The Pickering 5-8 Continued Operations Plan (COP) actions with the exception of those related to the Environmental Assessment were reviewed to determine if there were implications for PSR2. PSR2 implications were identified if closed COP actions needed to be re-assessed given the potential to operate Pickering to 2028 rather than 2020, which had been factored into the closure criteria for some COP actions. PSR2 implications were also identified if they were applicable to Pickering Units 1,4. Where there are implications of extended operation, or to Pickering Units 1,4, a PSR2 gap has been identified. These gaps will be considered in the Global Assessment Report.

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1.0 PURPOSE

The Continued Operations Plan (COP) was developed to document the actions that help support continued operation of Pickering 5-8 following a decision not to refurbish the units in 2009. It was developed based on a planned operation to 2020. The COP incorporated actions from the Environmental Assessment (EA) and from the 2009 Integrated Safety Review (ISR) as documented in the Integrated Implementation Plan (IIP) (Reference [1]) and the Global Assessment Report (Reference [2]).

The COP was updated annually from 2010 through to 2015 (References [3] to [8]). A Final Update was provided in December 2015 (Reference [9]).

Per the PSR2 Basis Document [10], the COP actions are being reassessed based on operation to the end of 2028. This approach is consistent with the recommendation from the Canadian Nuclear Safety Commission (CNSC) staff [11].

In Reference [11], the CNSC recommended that OPG utilize the CNSC response to the 2013 version of the COP [12] as the basis for the review, as this version included the most comprehensive COP action listing. The 2013 COP included 95 actions. Seven of the actions were related to the Pickering B Refurbishment EA, and are not within the scope of the Periodic Safety Review 2 (PSR2) (Reference [10]) since they did not originate in the ISR. The remaining 88 actions are addressed in this report.

The purpose of this report is to review the 88 actions from the Pickering 5-8 COP referred to in Reference [12] and as documented in Reference [13] (95 actions minus 7 actions related to the EA) that need to be reassessed for PSR2 based on extended operation to 2028. The 88 COP actions were also reviewed for applicability to Pickering Units 1,4.

2.0 SCOPE

The COP included four sources of actions:

- "E" actions are from the EA (outside the scope of this review);
- "F" actions are from Life Cycle Management Plans or Condition Assessments;
- "G" actions arose from the global assessment; and
- "I" actions originated in the ISR.

This report addresses actions that originated as F, G, and I actions as well as log entries that are related to those actions. The following actions are related to the EA as well as log entries related to the EA and are therefore not included in the scope of PSR2 (Reference [10]):

- E01-01
- E01-01(a)
- E01-02
- E01-2(a)
- E02-01
- E02-01(a)
- E03-01

3.0 METHODOLOGY

In Section 4.0, an assessment is done of each of the 88 Pickering 5-8 COP actions taken directly from Reference [13]. The identification number from the 2015 COP update (Reference [8], [9]) is included. All entries from the 2015 COP update are included in the review. The original wording of the issue is summarized. Impacts of extended operation are identified. Consistent with the PSR2 Basis Document [10], extended operation is assumed to 2028. However, in some cases, systems are needed after shutdown until the units are defueled; where this is the case, the period over which nuclear safety needs to be demonstrated is identified. Implications for Pickering Units 1,4 are also identified.

COP actions that have implications for extended operation or for Pickering Units 1,4 are considered to be PSR2 gaps. The Safety Factors to which these gaps are applicable are identified. Gaps are given a label similar to the labels used in the Safety Factor reports. The labels have the form **Pickering PSR2 Gap COP-x.** These gaps will be assessed in the Global Assessment.

The associated Safety Factors are assigned with numbering consistent with the PSR2 Basis Document [10]:

1. Plant Design
2. Actual Condition of SSCs Important to Safety
3. Equipment Qualification (environmental and seismic)
4. Aging
5. Deterministic Safety Analysis
6. Probabilistic Safety Assessment
7. Hazard Analysis
8. Safety Performance
9. Use of Experience from Other Nuclear Power Plants and Research Findings
10. Organization, Management System and Safety Culture
11. Procedures
12. Human Factors
13. Emergency Planning
14. Radiological Impact on the Environment
15. Radiation Protection.

4.0 ASSESSMENT OF COP ACTIONS FOR APPICABILITY TO PSR2

IIP ID "E" actions are from the Environmental Assessment; "F" actions are from Life Cycle Management Plans (LCMP) or Condition Assessments (CA)[1], "G" actions arose from the Global Assessment, "I" actions originated in the Integrated Safety Review.

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
1	6	E01-01	IIP ID numbers 1-7 from Reference [13] originated as "E" actions from the Environmental Assessment and did not originate from the ISR. These actions are not within the scope of the PSR2 review.				
2		E01-01(a)					
3	7	E01-02					
4		E01-2(a)					
5	8	E02-01					
6		E02-01(a)					
7	1	E03-01					
8	9	F01	Develop a strategy to provide evidence that the Calandria Tube (CT) to Liquid Injection Shutdown System (LISS) nozzle gap will be maintained beyond 247k Equivalent Full Power Hours (EFPH) for all Pickering B units.	Complete	Actions were constrained by the shutdown date of 2020 assumed in the 2011 business plan. The strategy for CT - LISS nozzle gap preservation may apply beyond 2025, but has not been	This gap is addressed in the Safety Factor 4 report (Reference [14]) as **PSR2 Gap SF4-18.** For completeness, the specific wording of this gap is repeated in Section 4.1 of this report.	4

[1] Condition Assessments were referred to as Component Condition Assessments (CCAs) during the ISR.

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
					confirmed. This is therefore identified as a gap for Pickering PSR2. There is no implication for Pickering Units 1,4 since there are no LISS nozzles.		
9	10	F01-1	This is a log entry. OPG was requested to consider the proposed action that follows. Review the prediction model for CT/LISS Nozzle time to contact to identify over-conservatism.	Dispositioned	Entries 9, 10, 11, and 12 are all related to entry 8.	This is a log entry that expands upon F01. Refer to F01.	Not Applicable
10	11	F01-2	This is a log entry. OPG was requested to consider the proposed action that follows: Perform CT-LISS gap measurement for unit 7 during P1471.	Dispositioned	Entries 9, 10, 11, and 12 are all related to entry 8.	This is a log entry that expands upon F01. Refer to F01.	Not Applicable
11	12	F01-3	This is a log entry. OPG was requested to consider the proposed action that follows: Conduct a new assessment to estimate revised CT/LISS time to contact with reduced conservatism for all Pickering B units based on new installation information and latest Pressure Tube (PT) sag data.	Dispositioned	Entries 9, 10, 11, and 12 are all related to entry 8.	This is a log entry that expands upon F01. Refer to F01.	Not Applicable

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
12	13	F01-4	This is a log entry. OPG was asked to consider the proposed action that follows: Confirm whether CT/LISS nozzle contact can be precluded or not for Pickering B units prior to 247k EFPH.	Dispositioned	Entries 9, 10, 11, and 12 are all related to entry 8.	This is a log entry that expands upon F01. Refer to F01.	Not Applicable
13	14	F02	Develop Research and Development (R&D) justification for extending Fuel Channel design life to 247k EFPH and beyond in the areas of hydrogen ingress & fracture toughness, spacer mobility and integrity.	Complete	Actions were constrained by the shutdown date of 2020 assumed in the 2011 business plan. This assessment only addressed to 2025. This is therefore a gap for Pickering PSR2. This issue applies to Pickering Units 1,4 and 5-8. Note that OPG is continuing to provide the CNSC with evidence to support the extended operation of the fuel channels. The work is being conducted in a staged approach. (See, for example, Reference [35].)	This gap is addressed in the Safety Factor 4 report (Reference [14]) as **PSR2 Gap SF4-18.**	4

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
14		F02-1a	Provide written update on: (1) Evaluation of operations procedure changes at Pickering Nuclear Generating Station (PNGS) 5-8 to improve margins on Fracture Protection (FP) and Leak-Before-Break (LBB) assessments and (2) Status of installation of annulus gas system Dew-Point Rate of Rise (DPROR).	Closed to another action or process. (Reference [13])	This action was addressed under the Hold Point that is governed by Licence Condition Handbook (LCH) Section 16.3 (Reference [13]). The action was subsequently closed in Reference [9]. (1) OPG implemented procedure changes to improve margins. (2) The DPROR capability has been installed and commissioned on Pickering NGS Units 5-8. The issue is impacted by extended operation since the probabilistic LBB assessments that have been done must take into consideration the requirements for pressure tube life. This issue is also applicable to Pickering Units 1,4. Note that OPG is continuing to conduct work in this area in a staged	The probabilistic LBB assessments have not yet been fully completed for the entire extended operating period, as work is being performed to demonstrate fitness for service for the pressure tubes in a staged approach. **(<u>Pickering PSR2 Gap COP-1</u>)**	2

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
					approach. (See, for example, References [36] and [37].)		
15		F02-1b	Provide update on: (1) Development of probabilistic LBB and FP methodologies, (2) Schedule for LBB and FP assessments using improved models, methodologies and operating procedures (as applicable) (3) Rolled Joint Deuterium (RJD) ingress model improvements, (4) Rolled Joint (RJ) scrape sampling results.	Closed to another action or process. (Reference [13])	This action was addressed under the Hold Point that is governed by LCH Section 16.3 (Reference [13]). The action was subsequently closed in Reference [9]. Since COP action parts 1, 2, and 3 are complete for Pickering Units 5-8; since the same methodologies are used for Pickering Units 1,4, there is no impact on Pickering Units 1,4. With respect to COP action part 4, this request for specific information for Pickering Units 5-8 was addressed. Going forward, the periodic inspection program requires rolled joint scrapes at defined intervals for Pickering Units 1,4 and 5-8. This action is not impacted by extended operation.	No gap	Not Applicable

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
16		F03	Determine if reconfiguration of P7 & P8 fuel channels is required prior to 238k EFPH and if necessary, schedule reconfiguration for completion prior to 238k EFPH in the Fuel Channel Life Cycle Management Plan.	Closed to another action or process. (Reference [13])	This action was addressed under the "Aging Management" that is governed by LCH Section 7.1. For clarity, the requirements identified in the LCH Section 7.1 – Aging Management are reproduced below: *"The SSC [System, Structures and Components] specific AMPs [Aging Management Plans] or LCMPs [Life Cycle Management Plans] which are submitted in accordance with LC 1.2, are licensing basis documents. As such, any changes to the SSC-specific AMPs or LCMPs will be reviewed by CNSC staff to confirm that they remain within the licensing basis and include all prior OPG commitments with respect to the inspection scope and other relevant commitments related to the continued operation of the Pickering Units. When considering possible changes to activities*	No gap	Not Applicable

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
					identified in the LCMPs, the licensee shall engage CNSC staff early enough to confirm that the changes are within the licensing basis." Any additional required fuel channel reconfiguration as a result of extended operation will be addressed as noted above. This conclusion is also applicable to Pickering Units 1,4. This is not a PSR2 gap.		
17	15	F04	Demonstrate adequate margin to operate the Pickering B Shutdown System 2 (SDS2) LISS to 31 Dec 2025 by performing piping fatigue and aging analysis. If analysis does not demonstrate adequate margin, develop and implement mitigation actions. The analysis timeframe includes 5 years of operation to support stabilization activity planned to commence after the end of commercial operations on 31 Dec 2020. This analysis timeframe may also be expressed as nominal design life (2015) extending 10 years to 2025 to accommodate continued	Complete	Actions were constrained by the shutdown date of 2020 assumed in the 2011 business plan. This assessment only addressed to 2025. This is therefore a gap for Pickering PSR2. There is no implication for Pickering Units 1,4 since there is no LISS on these units. Note: OPG is in the process of updating its	Demonstration of adequate margin to operate the Pickering Units 5-8 SDS2 LISS by performing piping fatigue and aging analysis has not been completed for the extended operations period and for the period until LISS can be demonstrated to no longer be required. **(<u>Pickering PSR2 Gap COP-2)</u>**	2

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
			operations with adequate margin.		condition assessments in support of extended operation.		
18	16	F05	Demonstrate adequate margin to operate the Pickering B Primary Heat Transport (PHT) piping to 31 Dec 2025 by assessing the impact of Flow Accelerated Corrosion (FAC) in PHT piping on fatigue service limits and perform a fatigue analysis if required. If analysis does not demonstrate adequate margin, develop and implement mitigation actions. The analysis timeframe includes 5 years of operation to support stabilization activity planned to commence after the end of commercial operations on 31 Dec 2020. This analysis timeframe may also be expressed as nominal design life (2015) extending 10 years to 2025 to accommodate continued operations with adequate margin.	Complete	Actions were constrained by the shutdown date of 2020 assumed in the 2011 business plan. This assessment only addressed to 2025. This issue applies to Pickering Units 1,4 and 5-8. This is therefore a gap for Pickering PSR2. Note: OPG is in the process of updating its condition assessments in support of extended operation.	Demonstration of adequate margin to operate the Pickering PHT piping has not been completed for the extended operations period and for the period until PHT piping integrity has been demonstrated to no longer be required. **(<u>Pickering PSR2 Gap COP-3</u>)**	2
19	2	F06	Demonstrate adequate margin to operate the Pickering High Pressure Emergency Coolant Injection (HPECI) storage tank foundation piles to 31 Dec 2025 by performing engineering analysis of loss of thickness due	In Reference [11], CNSC staff identified that the issue cannot be closed pending additional	Actions were constrained by the shutdown date of 2020 assumed in the 2011 business plan. This assessment only addressed to 2025. The analysis timeframe includes 5 years	Work has not been completed to demonstrate adequate margin to operate the Pickering HPECI storage tank foundation piles by performing engineering	2

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
			to corrosion. Develop mitigation plan if analysis does not indicate adequate margin. The analysis timeframe includes 5 years of operation to support stabilization activity planned to commence after the end of commercial operations on 31 Dec 2020. This analysis timeframe may also be expressed as nominal design life (2015) extending 10 years to 2025 to accommodate continued operations with adequate margin. It is expected that the demonstration of adequate margin would be documented in formal correspondence and/or in an appropriate engineering document.	information. In Reference [29] the CNSC identified this item as complete and closed.	of HPECIS operation to support stabilization activity planned to commence after the end of commercial operations. Pickering Units 1,4 share the HPECI storage tank with Pickering Units 5-8. OPG has had continued dialogue with the CNSC on this issue. CNSC have subsequently closed COP Action F06 [29]. This issue must be reconsidered in the context of extended operation. This issue is also applicable to Pickering Units 1,4. This is therefore a gap for Pickering PSR2.	analysis of loss of thickness due to corrosion, for the extended operations period and for the period until the ECIS is no longer required. **(Pickering PSR2 Gap COP-4)**	
20	17	F07	Demonstrate adequate margin to operate the Pickering B Fuelling Machines to 31 Dec 2025 by performing engineering analysis of fatigue and aging of Fuelling Machine components.	Complete	OPG performed a fatigue analysis of the Unit 5-8 Fuelling Machine components to demonstrate adequate margin to operate the	The fatigue analysis of the Pickering Fuelling Machine components has not been completed to demonstrate adequate margin to operate the Fuelling	2

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
			Develop mitigation plans if analysis does not indicate adequate margin. The analysis time frame includes 5 years of operation to support stabilization activity planned to commence after the end of commercial operations on 31 Dec 2020. This analysis timeframe may also be expressed as nominal design life (2015) extending 10 years to 2025 to accommodate continued operations with adequate margin. It is expected that the demonstration of adequate margin would be documented in formal correspondence and/or in an appropriate engineering document.		Fuelling Machines to December 2025, documented in report NK30-CALC-35310-00003, and correspondence NK30-CORR-35300-0476979. The analysis concluded all Unit 5-8 Fuelling Machine non-replaceable components satisfy the fatigue requirements for operation to December 2025. The remaining pressure boundary components will be replaced through prescribed maintenance strategies (Reference [9]). The foregoing analysis does not address extended operations and the additional period required for reactor defueling as part of the stabilization activities planned to commence after the end of commercial operations. This issue applies to Pickering Units 1,4 and 5-8. This is a gap for Pickering PSR2.	Machines for the extended operations period and for the period beyond where required for defueling activities. **(Pickering PSR2 Gap COP-5)**	

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
					Note: OPG is in the process of updating its condition assessments in support of extended operation.		
21	3	F08	Demonstrate adequate margin to operate the Pickering B Fuelling Machine Bridge ball screws to 31 Dec 2025 by performing engineering analysis of fatigue and aging of bridge ball screws. Develop mitigation plans if analysis does not indicate adequate margin. The analysis time frame includes 5 years of operation to support stabilization activity planned to commence after the end of commercial operations on 31 Dec 2020. This analysis timeframe may also be expressed as nominal design life (2015) extending 10 years to 2025 to accommodate continued operations with adequate margin. It is expected that the demonstration of adequate margin would be documented in formal correspondence and/or in an appropriate engineering document.	Complete.	In Reference [11], CNSC staff identify that the issue will be monitored up to the new target date of 2028 for end of commercial operations. Actions were constrained by the shutdown date of 2020 assumed in the 2011 business plan. The foregoing analysis does not address extended operations and the additional period required for reactor defueling as part of the stabilization activities planned to commence after the end of commercial operations. This issue applies to Pickering Units 1,4 and 5-8. This is a gap for Pickering PSR2. Note: OPG is in the process of updating its	Demonstration of adequate margin to operate the Pickering Fuelling Machine Bridge ball screws for the extended operations period and for the period beyond as required for defueling activities by performing engineering analysis of fatigue and aging of bridge ball screws, has not been completed. **(Pickering PSR2 Gap COP-6)**	2

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
					condition assessments in support of extended operation.		
22	18	F09	Demonstrate adequate margin to operate the Pickering B Reactor Regulating System (RRS) Control Absorbers to 31 Dec 2025 by performing engineering analysis of RRS control absorber cadmium to demonstrate adequate reactivity-worth. Develop mitigation plans if analysis does not indicate adequate margin. The analysis time frame includes 5 years of operation to support stabilization activity planned to commence after the end of commercial operations on 31 Dec 2020. This analysis timeframe may also be expressed as nominal design life (2015) extending 10 years to 2025 to accommodate continued operations with adequate margin.	Complete	Actions were constrained by the shutdown date of 2020 assumed in the 2011 business plan. This assessment only addressed to 2025. There is no implication for Pickering Units 1,4 since there are no Control Absorbers. This is a gap for Pickering PSR2. Note: OPG is in the process of updating its condition assessments in support of extended operation.	Demonstration of adequate margin to operate the Pickering 5-8 RRS Control Absorbers through the period of extended operations by performing engineering analysis of RRS control absorber cadmium to demonstrate adequate reactivity-worth, has not been completed. **(Pickering PSR2 Gap COP-7)**	2
23		F10	Provide CNSC with a progress report on the OPG Heat Transport System (HTS) Aging Safety Analysis, including a report documenting the comparison of the results from the coupled toolset against a	Closed to another action or process. (Reference [13])	F10-F13: Relate to Heat Transport System aging of the Pickering NGS Units 5 to 8 Actions were constrained by the shutdown date of	Addressed under IIP ID G01-04 (ID number 40). No additional action is required.	Not Applicable

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
			station transient. This action is related to F11, F12, and F13.		2020 assumed in the 2011 business plan. This assessment only addressed to 2025. This is a PSR2 gap that is addressed under IIP ID G01-04.		
24		F11	Update the Pickering B HTS aging model with respect to reactor safety analysis.	Closed to another action or process. (Reference [13])	Actions were constrained by the shutdown date of 2020 assumed in the 2011 business plan. An update to the HTS aging model with respect to safety analysis to assess impact of extended operations has not been done. This issue applies to Pickering Units 1,4 and 5-8. Therefore, this is a gap for Pickering PSR2.	This gap is addressed in the Safety Factor 4 report (Reference [14]) as **PSR2 Gap SF4-18.**	2, 4
25		F12	Develop the Pickering B HTS Aging Management Strategy	Closed to another action or process. (Reference [13])	Actions were constrained by the shutdown date of 2020 assumed in the 2011 business plan. This assessment only addressed to 2025. This issue applies to Pickering Units 1,4 and 5-	This gap is addressed in the Safety Factor 4 report (Reference [14]) as **PSR2 Gap SF4-18.**	2, 4

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
					8. Therefore, this is a gap for Pickering PSR2.		
26	19	F13	Update the Neutron Over Power (NOP) analysis for Pickering B. Analysis should incorporate HTS aging affects and impact, if any, on trip set points.	Complete	As part of the NOP compliance framework specific plant parameters relating to HTS aging are monitored to ensure that the installed NOP trip setpoints remain valid. However, the compliance framework adequacy needs to be confirmed for operation past 2020 for Pickering Units 1,4 and Units 5-8. Therefore, this is a gap for Pickering PSR2.	This gap is addressed in the Safety Factor 4 report (Reference [14]) as **PSR2 Gap SF4-18.**	4, 5
27	20	F14	Provide a roadmap to relate the key aging assessment information similar to the content of Component Condition Assessments (CCAs) for the content of major component LCMPs and Technical Basis Documents (TBDs).	Complete	This action is related to packaging of information not the content of CCAs. There is no impact on extended operation or on operation of Pickering Units 1,4.	No gap	Not Applicable
28	21	F14-1a	With respect to Feeder LCMP, clarify the impact of fuel channel axial elongation during the operation beyond the fuel channel assumed design life of 210k EFPH on feeder stress	Complete	An interim LCMP update for major components is documented in P-CORR-01060-0587604 R000 (Reference [15]), which describes life cycle	This gap is addressed in the Safety Factor 4 report (Reference [14]) as **PSR2 Gap SF4-18.**	4

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
			analysis and acceptable feeder thickness.		management strategies for major components to achieve extended operations to 2024. This assessment must be reconsidered in the context of extended operation. This issue applies to Pickering Units 1,4 and 5-8. This is a gap for Pickering PSR2. Note: OPG is in the process of updating its LCMPs in support of extended operation using a staged approach.		
29	22	F14-1b	Submit the Pickering B feeder replacement schedule/plan including their IDs and the supporting rationale, including life limits, for specific feeder replacement.	Complete	The current feeder LCMP only provides the number of replacement feeders up to 2020. This issue applies to Pickering Units 1,4 and 5-8. Therefore, this is a gap for Pickering PSR2. Note: OPG is in the process of updating its LCMPs in support of	The Pickering feeder replacement schedule/plan (including feeder IDs and the supporting rationale, including life limits, for specific feeder replacement) has not been updated to support extended operation for Pickering Units 1,4 and 5-8. **(<u>Pickering PSR2 Gap</u>**	2

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
					extended operation using a staged approach.	**COP-8)**	
30	23	F14-2a	Demonstrate that sufficient margin remains in the operating life of the steam generators (residual life based on the Time Limiting Ageing Analysis (TLAA)), steam generator tubes and tube supports, shell, attachment welds and other internals considering the original design requirements and the Operating Experience (OPEX) on in-service degradation with respect to: - thermal cyclic fatigue -mechanical fatigue - corrosion allowances - other types of degradation mechanisms.	Complete	An interim LCMP update for major components is documented in P-CORR-01060-0587604 R000 (Reference [15]), which describes life cycle management strategies for major components to achieve extended operations to 2024. This assessment does not address the full extended operations period. This issue applies to Pickering Units 1,4 and 5-8. This is a gap for Pickering PSR2. Note: OPG is in the process of updating its condition assessments in support of extended operation.	Demonstration has not been completed for extended operation to confirm that sufficient margin remains in the operating life of the Pickering steam generators (residual life based on the TLAA), steam generator tubes and tube supports, shell, attachment welds and other internals considering the original design requirements and the OPEX on in-service degradation with respect to: - thermal cyclic fatigue; - mechanical fatigue; - corrosion allowances; and - other types of degradation mechanisms. **(Pickering PSR2 Gap COP-9)**	2

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
31	24	F14-2b	Clarify whether OPG considers the deference of inspections due to radiological concerns as unique situations or repeated occurrences and how OPG expects to deal with them.	Complete	There is no impact on extended operation or on operation of Pickering Units 1,4.	No gap	Not Applicable
32	25	F14-2c	Clarify how OPG derived that there is a 30% probability that the steam generators may not be able to operate until 240k EFPH, and describe any perceived scenarios which could lead to safety concerns related to inoperability of the steam generators before 240k EFPH.	Complete	This action was in response to a specific CNSC inquiry. This action is complete. There is no impact on extended operation or on operation of Pickering Units 1,4. Note: OPG is in the process of updating its condition assessments in support of extended operation.	No gap	Not Applicable
33	26	F14-3a	Provide plan for how Fitness for Service (FFS) will be demonstrated for the Pickering B calandrias.	Complete	This action was in response to a specific CNSC inquiry. This action is complete for Pickering Units 5-8. Fitness for Service demonstration of the Pickering Units 1,4 and 5-8 calandrias does not address extended	Fitness for Service demonstration of the Pickering Units 1,4 and 5-8 calandrias to address the full period of extended operation has not been completed. **(Pickering PSR2 Gap COP-10)**	2

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
					operation. This is a gap for Pickering PSR2. Note that fitness for service evaluations and Life Cycle Management Plans are being updated in support of the extended operating period using a staged approach.		
34	27	F14-3b	Provide a Report on the Calandria Tube Life Assessment for Pickering A, Pickering B and Darlington units beyond 30 years design.	Complete	An interim LCMP update for major components is documented in P-CORR-01060-0587604 R000 (Reference [15]), which describes life cycle management strategies for major components to achieve extended operations to 2024. This issue is not addressed in the context of extended operation. This issue applies to Pickering Units 1,4 and 5-8. This is a gap for Pickering PSR2. Note that Life Cycle	The Pickering Calandria Tube life assessment has not been updated for the full period of extended operations. **(Pickering PSR2 gap COP-11)**	2

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
					Management Plans and Condition Assessments are being updated in support of the extended operating period.		
35	28	F14-3c	Provide the CNSC with updates on inspection and monitoring of reactor components to support the technical basis for continued operation.	Complete	An interim LCMP update for major components is documented in P-CORR-01060-0587604 R000 (Reference [15]), which describes life cycle management strategies for major components to achieve extended operations to 2024. This issue is not addressed in the context of extended operation. This issue applies to Pickering Units 1,4 and 5-8. This is a gap for Pickering PSR2. Note that Life Cycle Management Plans and Condition Assessments are being updated in support of the extended operating period.	OPG requirements and plans for inspection and monitoring of reactor components have not been updated to address the full period of extended operations of Pickering Units 1,4 and 5-8. **(Pickering PSR2 gap COP-12)**	2

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
36	29	F14-3d	Submit calandria and internal structures Technical Basis Document to CNSC, which should include OPEX, and FFS rationale to support calandria and internal components will remain fit for service to end of commercial operation.	Complete	An interim LCMP update for major components is documented in P-CORR-01060-0587604 R000 (Reference [15]), which describes life cycle management strategies for major components to achieve extended operations to 2024. This issue has not been addressed in the context of extended operation. This issue applies to Pickering Units 1,4 and 5-8. This is a gap for Pickering PSR2. Note that Condition Assessments and Life Cycle Management Plans are being updated in support of the extended operating period.	The calandria and internal structures Technical Basis Document for Pickering Units 1,4 and 5-8, which includes OPEX, and FFS rationale to support that calandria and internal components will remain fit has not been updated for the full period of extended operations. **(Pickering PSR2 gap COP-13)**	2
37	30	F14-4.1	Include the periodic inspection programs and LCMPs for the secondary side pressure retaining components and submit them for CNSC review.	Complete	This action was to provide the CNSC with specific information in response to an inquiry. This action is complete. This issue has not been addressed in the context of	This gap is addressed in the Safety Factor 4 report (Reference [14]) as **PSR2 Gap SF4-18.**	4

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
					extended operation. This issue applies to Pickering Units 1,4 and 5-8. This is a gap for Pickering PSR2.		
38	31	F14-4.2	Include the periodic inspection programs and LCMPs for the Safety-significant civil structures that are under the scope of Canadian Standards Association (CSA) N291-08, but not covered by the N287.7 standard.	Complete	This action was to provide the CNSC with specific information in response to an inquiry. This action is complete. This issue has not been addressed in the context of extended operation. This issue applies to Pickering Units 1,4 and 5-8. This is a gap for Pickering PSR2.	Actions F14-4.2, F14-5, and G04-02 from the Pickering Units 5-8 Continued Operations Plan are related to CSA N287.7 and CSA N291, and although complete, have not been updated for implications of the full extended operation for Pickering Units 1,4 and 5-8. This gap is addressed in the Safety Factor 4 report (Reference [14]) as **PSR2 Gap SF4-13.**	4
39	32	F14-5	Submit the Ageing Management Plan for N287.7 Concrete Containment Structures to the CNSC review and provide fitness for service to end of mission time.	Complete	This action was to provide the CNSC with specific information in response to an inquiry. This action is complete. This issue has not been addressed in the context of	This gap is already identified under IIP ID F14-4.2 (ID Number 38) and addressed in the Safety Factor 4 report (Reference [14]) as part of **PSR2 Gap SF4-13.**	4

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
					extended operation. This issue applies to Pickering Units 1,4 and 5-8. This is a gap for Pickering PSR2.		
40	4	G01-04	Demonstrate adequate safety margins to operate Pickering B units from an HTS ageing perspective to Jan 31, 2021.	Complete. In Reference [13], CNSC staff identify that the issue will be monitored up to the new target date of 2028 for end of commercial operations.	As discussed in COP (Reference [9]) Section 2.4: *In 2015, COP improvement action G01-04 was completed with the issuance of the 2015 strategy update to CNSC staff, which provided a progress report on Heat Transport System (HTS) Aging Safety Analysis and related activities, and an updated revision of the HTS Aging Management Strategy (HTS-AMS) for the period 2015-2020. The main focus of the 2015 revision of the HTS-AMS was the progression of the Safety Analyses to demonstrate continued safe operation of Pickering Units 5-8 until currently planned end-of-commercial operation. Updates on several experimental and research & development activities which are in*	This gap is addressed in the Safety Factor 4 report (Reference [14]) as **PSR2 Gap SF4-18.**	4, 5

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
					progress with the intention of demonstrating improved safety margins were also included. This assessment only addressed to 2020. This issue applies to Pickering Units 1,4 and 5-8. This is a gap for Pickering PSR2.		
41		G01-05	Develop an action plan to address the remaining generic deficiencies in the Two Unequal Fluids (TUF) code validation.	Closed to another action or process (Reference [13])	CNSC staff agreed that this action should be tracked within the Pickering regulatory program under Action Item (AI) # 2011 OPG-01, and Regulatory Management (RegM) Action Request (AR) 28137598-03. Therefore, this action is closed in the COP. Based on continued work and communications, the CNSC have subsequently closed Action Item 2011OPG-01 [30]. There is no impact on extended operation or on operation of Pickering Units 1,4.	No gap	Not Applicable

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
					There is no PSR2 gap.		
42		G01-06	Address regulatory concerns raised with the 28-element Boiling Length Average (BLA) Critical Heat Flux (CHF) Correlation analysis plan.	Closed to another action or process (Reference [13])	CNSC staff agreed that this action should be tracked within the Pickering regulatory program under AI 201113-2297 (Reference [13]). The CNSC have closed the associated action item AI 2012-OPG-3464 (N-CORR-00531-06063 [32]) but AI 201113-2297 related to Pickering 28 element fuel remains open (N-CORR-00531-05900 [33]]. The methodology and correlations are applicable to 28-element fuel so Pickering Units 1,4 and 5-8 are addressed. The analysis (Reference [16]) bounds both un-crept and fully crept pressure tubes. Therefore, the methodology and correlations are not affected by extended operation. However, AI 201113-2297 has not been closed by the	There are remaining issues from AI 201113-2297 follow-up related to providing a detailed assessment report of the uncertainty in the implementation and use of BLA CHF correlation for 28 element fuel. **(Pickering PSR2 Gap COP-14)**	5

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
					CNSC for Pickering NGS 28 element fuel. This is a gap for Pickering PSR2.		
43		G02-01	Track to completion Generic Action Item (GAI) 01G01 "Fuel Management and Surveillance Software Upgrade".	Closed to another action or process (Reference [13])	CNSC staff agreed that this action can be tracked within the Sustainable Operations Plan (SOP), under SCA04-08. CNSC provided closure of GAI 01G01, "Fuel Management and Surveillance Software Upgrade" and opened AI 2012-OPG-3465 to track associated actions. Per N-CORR-00531-07323 (Reference [17]), OPG addressed CNSC staff requests for both Pickering Units 1,4 and 5-8 and requested closure of AI 2012-OPG-3465. The issue has been closed (Reference [18]), however follow-up action has been identified per AI 2016-OPG-8250. OPG has provided a response to this Action Item (N-CORR-00531-18204 [31])	There are remaining issues from Generic Action Item (GAI) 01G01 "Fuel Management and Surveillance Software Upgrade" (AI 2016-OPG-8250). **(Pickering PSR2 Gap COP-15)**	5

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
					indicating additional information is planned by May 2017. This issue applies to Pickering Units 1,4 and Units 5-8. There is a gap for Pickering PSR2.		
44		G02-02	Track to completion GAI 95G04 "Positive Void Reactivity Uncertainty - Treatment in Large Loss of Coolant Accident (LOCA) Analysis".	Closed to another action or process (Reference [13])	CNSC staff agreed that this action be tracked within the SOP, under SCA04-09. CNSC staff subsequently closed SCA04-09 indicating the issue was dealt with appropriately under SOP SCA04-07 [13]. SOP SCA-04-07 derives from COP item I09 (ID number 68) which is also considered in this reassessment. There is no impact on extended operation or on operation of Pickering Units 1,4. There is no PSR2 gap.	No gap	Not Applicable
45		G02-03	Track to completion GAI 99G02 "Replacement of Reactor Physics Computer Codes used in Safety	Closed to another action or process	CNSC staff agreed that this action be tracked within	No gap	Not Applicable

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
			Analysis of CANDU Reactors".	(Reference [13])	the SOP, under SCA04-10. CNSC staff subsequently closed SCA04-10 indicating the issue was dealt with appropriately under SOP SCA04-07 [13]. SOP SCA-04-07 derives from COP item I09 (ID number 68) which is also considered in this reassessment. There is no impact on extended operation or on operation of Pickering Units 1,4. There is no PSR2 gap.		
46		G02-04	Track to completion GAI 00G01 "Channel Voiding during a LOCA".	Closed to another action or process (Reference [13])	CNSC staff agreed that this action be tracked within the SOP, under SCA04-11. This issue was tracked as Action Item 2012OPG-3317. CNSC have closed this action (N-CORR-00531-06689 [34]). There is no impact on extended operation or on operation of Pickering Units 1,4.	No gap	Not Applicable

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
					There is no PSR2 gap.		
47		G04-01	Develop and issue a high-level generic leakage rate test requirements document to address the clauses of CSA N287.7-08 relevant to leakage rate testing for the Pickering B Reactor Buildings, entitled Pickering B Reactor Building In-Service Leakage Rate Test Requirements in Accordance with CSA N287.7-08.	Closed to another action or process (Reference [13])	This action was closed to Action G04-02 in the COP (ID Number 48) (Reference [13]) which is also considered in this reassessment. There is no impact on extended operation or on operation of Pickering Units 1,4. There is no PSR2 gap.	No gap	Not Applicable
48	33	G04-02	Revise NK30-PIP-03643.2-00001, Reactor Building Periodic Inspection Program, to comply with CSA Standard N287.7-08 and include the Embedded Parts List. Submit NK30-PIP-03643.2-00001 R003 to CNSC for acceptance. Revise N-PROC-MA-0066 to reference the Leakage Rate Test document.	Complete	Revise NK30-PIP-03643.2-00001, Reactor Building Periodic Inspection Program, to comply with CSA Standard N287.7-08 to address extended operation. This issue has not been addressed in the context of extended operation. This issue applies to Pickering Units 1,4 and 5-8. This is a gap for Pickering PSR2.	This gap is already identified under IIP ID F14-4.2 (ID Number 38) and is addressed in the Safety Factor 4 report (Reference [14]) as part of **PSR2 Gap SF4-13.**	4

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
49		G04-03	Since a Vacuum Building Outage (VBO) will not be completed in 2020, document the licensing/technical basis for maintaining a fully serviceable containment boundary for the 2-3 years during which defueling and dewatering activities will be underway on the Pickering Units.	Closed to another action or process (Reference [13])	Action G04-03 in the COP was transferred to SOP action SCA04-04. (This was formerly identified as SOP action SCA06-07). (Reference [13]). OPG proposed to close this to the Stabilization Activity Plan (SAP) (Reference [13]). The issue must be reconsidered in the context of extended operation. Pickering Units 1,4 share the Vacuum Building with Pickering Units 5-8. This is a gap for Pickering PSR2.	Reassessment of the VBO schedule and basis for maintaining a fully serviceable containment boundary has not been completed for extended operations and until negative pressure containment can be demonstrated to no longer be required. **(Pickering PSR2 gap COP-16)**	5
50	34	G05-01	Issue the Pickering B, Priority 2 Safe Operating Envelope (SOE) systems documentation.	Complete	This action was to issue Priority 2 SOE systems documentation. This action is complete [13]. There is no impact on extended operation. This work has also been completed for Pickering	No gap	Not Applicable

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
					Units 1,4 (Reference [40]). There is no PSR2 gap.		
51	35	G05-02	2009 Global Assessment: The Global Assessment team recommends that OPG management review the results of the deterministic safety analysis to confirm sufficient defence in depth, production reliability and incorporation of conservative decision making and operating experience have been appropriately incorporated into the project scope. Disposition: The OPG Benefit Cost Analysis (BCA) process had been previously reviewed and accepted by the CNSC (Reference CD# NK30-CORR-00531-04417). If OPG had decided to refurbish, OPG would have deterministically reviewed the project scope to confirm sufficient Defence in Depth, operating experience, production reliability, and economic viability had been appropriately integrated into the project. However, given the decision was not to refurbish the Pickering B units, there is no benefit to reviewing this issue	Complete	There is no impact on extended operation or on operation of Pickering Units 1,4. There is no PSR2 gap. Note: Global Assessment under PSR2 is currently in progress.	No gap	Not Applicable

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
			further.				
52	36	G06-01	Demonstrate fire code compliance with CSA N293- 07 and Licence Condition 6.3.	Complete	This issue is not impacted by extended operation. Code Compliance review (Reference [19]) was also completed for Pickering A. There is no PSR2 gap. Note: The modern version of this CSA Standard (CSA N293-12) is in the current PSR2 assessment basis.	No gap	Not Applicable
53	37	G08-01	Complete the installation of the Passive Autocatalytic Recombiners (PARs) in all Pickering B units.	Complete	PARS installation is complete on all Pickering Units. This issue is not impacted by extended operation. There is no PSR2 gap.	No gap	Not Applicable
54	38	G09-01	Complete the installation and place in-service the upgraded seismic monitoring system.	Complete	The new seismic monitoring system was installed, commissioned, and placed in service as documented in P-REP-61150-00002 (Reference [20]). The seismic monitoring system upgrade addressed the entire	No gap	Not Applicable

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
					Pickering station so there is no gap for Pickering Units 1,4. This issue is not impacted by extended operation. There is no PSR2 gap.		
55	39	G10-01	Complete Level 1 Pickering B Probabilistic Risk Assessment (PRA) to demonstrate S-294 licence compliance.	Complete	Level 1 PRA is complete for Pickering Units 5-8 (Reference [38]) and Units 1,4 (Reference [39]). This issue is not impacted by extended operation. There is no PSR2 gap. Note: CNSC S-294 has been superseded by CNSC REGDOC-2.4.2. REGDOC-2.4.2 is in the current PSR2 assessment basis.	No gap	Not Applicable
56	40	G11-01	Perform Level 2 Pickering B PRA in Accordance with S-294 and Pickering B Pickering Reactor Operating Licence (PROL) Condition 3.12.	Complete	Level 2 PRA is complete for Pickering Units 5-8 (Reference [38]) and Units 1,4 (Reference [39]). This issue is not impacted by extended operation.	No gap	Not Applicable

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
					There is no PSR2 gap. Note: CNSC S-294 has been superseded by CNSC REGDOC-2.4.2. REGDOC-2.4.2 is in the current PSR2 assessment basis.		
57	41	I01	Document the level of compliance of Pickering plant structures supporting the operation of Pickering B reactors with CSA N289.3-M81 "Design Procedures for Seismic Qualification of CANDU Nuclear Power Plants", Clause 4.7.1 to 4.7.3. Perform a review of the relevant seismic analysis reports regarding seismic overturning and sliding stability and if seismic reports are not available, perform required calculations to demonstrate compliance. Document the level of compliance in formal correspondence and/or in engineering documents.	Complete	This issue is not impacted by extended operation. Reference [21] was prepared to document overturning and sliding safety factors for the PNGS-B Reactor Building and other structures. The report determined that the seismic overturning and sliding safety factors for these structures exceeded 1.25, thereby exceeding the requirements of CSA N289.3- M81, Clauses 4.7.1 - 4.7.3. The calculation of the safety factor of 1.25 against overturning is not required for Pickering since the structures are built on embedded piles anchored in solid rock. Therefore, overturning or toppling due	No gap	Not Applicable

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
					to earthquake and other loading is not a concern. This conclusion is applicable to all Pickering units given the similarity of the designs. There is no PSR2 gap.		
58	42	I02	Document the level of compliance of Pickering plant structures supporting the operation of Pickering B reactors with CSA N289.3-M81, "Design Procedures for Seismic Qualification of CANDU Nuclear Power Plants", Clause 5.13.2. Review the relevant seismic analysis reports regarding the minimum number of cycles used for seismic fatigue analysis and document the level of compliance by revising the OPG Design Guide, DG-30-68000-2, "Pickering G.S. 'B' Seismic Qualification of Safety Related Systems", (Vendor R02, OPG R00), December 1979, Section 6.4.	Complete	This IIP action was closed based on the number of cycles used in the seismic analysis reports. This issue must be reconsidered in the context of extended operation. This issue also applies to Pickering Units 1,4. This is a gap for Pickering PSR2.	A review that considers the minimum number of cycles used for seismic fatigue analysis has not been completed for the level of compliance of Pickering NGS plant structures supporting the operation of Pickering reactors with CSA N289.3-M81, "Design Procedures for Seismic Qualification of CANDU Nuclear Power Plants", Clause 5.13.2 in the context of extended operations. **(Pickering PSR2 gap COP-17)**	1

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
59	43	I03	Complete the Pickering B PRA based seismic margin assessment, as per Section 3.12 of the PNGS B Reactor Power Operating Licence and document compliance with regulatory document S-294, "Probabilistic Safety Assessments for NPP".	Complete	This issue is not impacted by extended operation. The Pickering NGS-A PRA-Based Seismic Margin Assessment is complete (Reference [22]). There is no PSR2 gap.	No gap	Not Applicable
60	44	I03-1	Provide evidence that the existing Pickering B Standby Generator and Emergency Power Generator (EPG) Fuel Oil storage tank systems comply with the National Fire Code of Canada (NFCC).	Complete	This issue is not impacted by extended operation. This action has been completed for Pickering Units 5-8. Evidence that a similar demonstration has been done for Pickering Units 1,4 with respect to the Standby Generator fuel tanks was not found. This is a gap for Pickering PSR2. Note: The modern version of the NFCC (NFCC-2010) is included in the current PSR2 assessment basis.	An assessment that shows that Standby Generator fuel tanks supporting units 1,4 comply with NFCC could not be found. **(Pickering PSR2 Gap COP-18)**	1
61	45	I04	Review standards ANSI/ASME N509-1980 and N510-1980 for Filtered Air Discharge System (FADS) and Non-SOE air	Complete	This issue is not impacted by extended operation.	A review of Pickering Units 1,4 Non-SOE air cleaning systems against ANSI/ASME N509-1980 and	1

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
			cleaning systems and document the level of compliance and any requirements for further action.		FADS is applicable to Pickering Units 1,4 and 5-8 (Reference [41]) so the basis of closure for Pickering Units 5-8 is also applicable to Pickering Units 1,4. For the Non-SOE air cleaning systems, a review of ANSI/ASME N509-1980 and N510-1980 for Pickering Units 1,4 has not been completed. This is a PSR2 gap.	N510-1980 has not been completed. **(Pickering PSR2 Gap COP-19)**	
62	46	I04-1	Resolution of gap 643 is being managed under project 10-26003, Fire Safety Assessment (FSA) Upgrade Project and correspondence with CNSC on licence amendment, N-CORR-00531-05297. OPG recommends tracking the Fire Protection 2007 Code Update via correspondence following aforementioned letter and associated project. The scope of project 10-26003, FSA Upgrade Project and correspondence with CNSC on the licence amendment, N-CORR-00531-05297; address the discrepancy, effectively closing the gap. This gap will be tracked through correspondences	Complete	This work was completed for Pickering 5-8. This issue is not impacted by extended operation. Fire Hazards Assessment (Reference [23]), Fire Safe Shutdown Assessment (Reference [24]), and Code Compliance review (Reference [19]), were also completed for Pickering A so there is no PSR2 gap.	No gap	Not Applicable

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ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
			associated with N-CORR- 00531-05297 and project 10-26003.				
63		I05	Complete the FSA Upgrade Project in order to comply with CSA N-293-07. This action will be addressed under G06-01.	Closed to another action or process (Reference [13])	This action was closed to Action IIP ID G06-01 (ID Number 52) in the COP (Reference [13]). This issue is not impacted by extended operation. Refer to I04-1 (ID Number 62) for impact on Pickering Units 1,4. There is no PSR2 gap.	No gap	Not Applicable
64	47	I06	Revise PNGS-B Shutdown System SDS1 Design Requirements document, NK30-DR-63720-10001 R0, and SDS2 Design Requirements document, NK30-DR-63730- 10002 R0, (and associated sub-component design requirements documents), as per Document Change Requests (DCRs) 114575, 114576, 114577 in order to document level of compliance with CSA N290.1-80.	Complete	This issue is not impacted by extended operation. A review against CSA N290.1 is done of both Pickering Units 1,4 and Pickering Units 5-8 as part of PSR2. These are no PSR2 gaps.	No gap	Not Applicable
65		I06-1	Provide evidence that ISR Gaps 1-307 to 1-319 and I-358 are closed. CNSC staff disagrees that Gaps 1-307 to 1-319 and 1-	Closed to another action or process (Reference [13])	The action was closed to IIP ID I06 (ID Number 64) which was completed	No gap	Not Applicable

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
			358 be closed. The disposition of Gaps 1-308, 1-310, 1-312, 1-314, 1-319, and 1-358 are to be transferred by OPG to Pickering B COP, and be coded as this new Action I06-1, with a Target Completion Date (TCD) of October 1, 2012; similar to that of Action I06.		(Reference [13]). This issue is not impacted by extended operation. There is no impact on Pickering Units 1,4. There is no PSR2 gap.		
66	48	I07	Revise Emergency Coolant Injection System (ECIS) Design Requirements NK30-DR-33350-10004 and ECIS Design Manual NK30-33350-00002 as per DCR 114573 and Operational Safety Requirements for ECIS NK30-OSR-08131.02-00001 as per DCR 19510, in order to document compliance with Atomic Energy Control Board Regulatory Guide R-9.	Complete	This issue is not impacted by extended operation. CSA N290.2-11 replaces Regulatory Document, R-9, and is largely based on it. A review against CSA N290.2 is done of both Pickering Units 1,4 and Pickering Units 5-8 as part of PSR2. These is no PSR2 gap.	No gap	Not Applicable
67	49	I08	Revise Pickering-B Design Manual for Emergency Power System (EPS) Generators Fuel Oil System NK30-DM-54860-00001, and Pickering-B Design Manual for Standby Generators Fuel Oil System NK30- DM-54660-00001 (as per DCR 114562), in order to reflect compliance with NFCC 2005.	Complete	This issue is not impacted by extended operation. There is no impact on Pickering Units 1,4. There is no PSR2 gap.	No gap	Not Applicable

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
68		I09	Implement the results of COG Joint Project JP-4367 to resolve the PNGSB Legacy analysis codes and methods issues with the analysis for Large Break Loss of Coolant Accident (LBLOCA).	Closed to another action or process (Reference [13])	This action was transferred to the SOP and renamed SCA04-07. This action will continue to be tracked under the Pickering regulatory program (Reference [13]). The licensing basis of existing CANDU reactors for the LBLOCA scenario will continue to be based on conservative safety analysis for which acceptance criteria are established. Since three LBLOCA CANDU Safety Issues (CSIs) applicable to Pickering NGS remain in the high risk category (Category 3) and require further assessment in order to re-classify into a lower risk category and cover operation past 2020, a gap exists for Pickering PSR2. This issue applies to Pickering Units 1,4 and 5-8. This is a gap for Pickering PSR2.	Three LBLOCA CANDU Safety Issues that are applicable to NGS remain in Category 3 and have not been fully reassessed in order to re-classify into a lower risk category and cover operation past 2020. **(Pickering PSR2 Gap COP-20)**	5

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
69		I10	Demonstrate compliance of the PNGS B Safety Report relative to RD-310 "Safety Analysis for NPP" by completing gap assessments of the Safety Report versus RD-310 requirements including code validation as per AI 2010OPG-05. Prepare plans to address identified gaps. Execute the plans as part of the migration to RD-310 compliance.	Closed to another action or process (Reference [13])	This action was closed in the COP, as it was transferred to the SOP and renamed SCA04-13. OPG completed the gap assessments, and CNSC staff accepted the OPG submission (Reference [13]). This action has not been considered in the context of extended operation. This is a gap for Pickering PSR2. Note: CNSC RD-310 has been superseded by CNSC REGDOC-2.4.1. REGDOC-2.4.1 is in the current PSR2 assessment basis.	The updates to the Safety Reports are being conducted in accordance with the REGDOC-2.4.1 Implementation Plan (REGDOC-2.4.1 superseded CNSC RD-310). This plan did not consider operation beyond 2020, for Pickering Units 1,4 and Units 5-8. **(Pickering PSR2 Gap COP-21)**	5
70		I10-01	As part of OPG's plan for RD-310 implementation at Pickering site, OPG plans to complete limited upgrades to the Pickering B Safety Report. These upgrades will include addressing the current gap in covering Common Mode Events. This action will close ISR Gap 5-389 which identified that with respect to	Closed to another action or process (Reference [13])	This action was transferred to the SOP in 2013 and re-named SCA04-14. The criteria for prioritization of the Safety Report upgrades included remaining operating life. This issue has not been addressed in the context of	Implications described under IIP ID I10 (immediately above) are applicable to this action. The same gap applies. **(Pickering PSR2 gap COP-21)**	5

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
			Safety Assessments (Analysis) the set of Postulated Internal Events (PIE) should include events such as fires, explosions, turbine missile impacts and floods of internal origin which could affect the safety of the reactor and cause failure of some of the safety system equipment which provides protection for that initiating event.		extended operation. This issue applies to Pickering Units 1,4 and Units 5-8. This is a gap for Pickering PSR2.		
71		I11	Complete the Pickering B PRA update (Level 2 PRA) in Accordance with S-294 and Pickering B PROL Condition 3.12.	Closed to another action or process (Reference [13])	This action is closed to another COP action G11-01 (ID Number 56) "Perform Level 2 Probabilistic Risk Assessment (PRA) in Accordance with S-294 and Pickering B PROL Condition 3.12". The Pickering Units 5-8 Level 2 PRA is complete. There is no impact of extended operation. The Pickering Units 1,4 Level 2 PRA is complete (Reference [39]). There is no PSR2 gap. Note: CNSC S-294 has	No gap	Not Applicable

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
					been superseded by CNSC REGDOC-2.4.2. REGDOC-2.4.2 is in the current PSR2 assessment basis.		
72	50	I12	Complete Phase III of the Pickering B Severe Accident Management Program in order to align with CNSC G-306, "Severe Accident Management Programs for NPP". References; Gap 393 Severe Accident Management Program to be developed for Pickering B, Gap 398 SAMG [Severe Accident Management Guidance] Capability of containment system components to maintain their function and withstand effects of a severe accident to be considered, Gap 400 SAMG Capability to remove heat from the reactor containment in the event of a severe accident to be considered, Gap 408 SAMG Program is to be implemented based on guidelines developed through the Joint Industry Initiative (COG).	Complete	The basis of closure is that the remaining actions were being tracked under Fukushima actions. As outlined in Reference [25], SAMG has been implemented for all Pickering Units. Fukushima Action Item 3.1.1 is closed. There is no impact of extended operation. There is no PSR2 gap. Note: CNSC G-306 has been superseded by CNSC REGDOC-2.3.2. REGDOC-2.3.2 is in the current PSR assessment basis.	No gap	Not Applicable
73	51	I13-1	Complete analysis to demonstrate that the Post Accident Radiological Monitoring System (PARMS) iodine	Complete	This action is not impacted by extended operation.	No gap	Not Applicable

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
			sampling meets the design accuracy requirements for concentration measurement.		The closure correspondence (References [26] and [27]) is specific to Pickering Units 5-8. However, Pickering Units 1,4 use the same stack, and sampling system (Reference [41]). Therefore, the basis of closure is applicable to Pickering Units 1,4. There is no PSR2 gap.		
74		I14-a	Provide evidence of the effectiveness of how OPG deals with human performance and organizational issues, how procedures are effectively implemented and how they maintain or improve safety performance.	Closed to another action or process (Reference [13])	This action was renamed SOP action SCA02-33 in 2013, as it deals with the Pickering station in the context of the transition to the end of commercial operations, not with providing the technical basis for the incremental life extension of Pickering Units 5-8. There is no impact of extended operation. There is no issue for Pickering Units 1,4 in the context of PSR2. This transitional issue is an	No gap	Not Applicable

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
					element of the Pickering Sustainable Operations Plan (SOP) under SCA-02 [11], [13]. There is no PSR2 gap.		
75		I14-b	OPG should demonstrate, by providing evidence, how changes in the organization as it transitions to end of life will maintain or improve safety performance.	Closed to another action or process (Reference [13])	This action was renamed SOP action SCA01-17 in 2013, as it deals with the Pickering station in the context of the transition to the end of commercial operations, not with providing the technical basis for the incremental life extension of Pickering Units 5-8. There is no impact of extended operation. The organization for Pickering Units 5-8 is also applicable to Pickering Units 1,4 so there are no additional Pickering Units 1,4 issues. This transition issue is an element of the Pickering Sustainable Operations Plan (SOP) under SCA-01	No gap	Not Applicable

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
					[11], [13]. There is no PSR2 gap.		
76	52	I15-1a	Perform TLAAs and include such TLAAs in the LCMPs and in the CCAs. OPG to provide commitment that TLAAs necessary to demonstrate actual conditions of components will be completed.	Complete	This issue applies to Pickering Units 1,4 and 5-8. Although the action is complete, the updates to the TLAAs have not been fully completed to cover operation past 2020 for Pickering Units 1,4 and Units 5-8. Therefore, this is a gap for Pickering PSR2. Note: OPG is in the process of updating its Condition Assessments and LCMPs in support of extended operation.	This gap is addressed in the Safety Factor 4 report (Reference [14]) as **PSR2 Gap SF4-18.**	4
77	53	I15-1b	Provide CCAs on low priority SSCs.	Complete	This action was to provide CNSC with specific CCAs[2] requested. This action was completed. There is no impact of extended operation.	No gap	Not Applicable

[2] CCAs are now referred to as Condition Assessments.

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
					There is no PSR2 gap. Note: OPG is in the process of updating its Condition Assessments in support of extended operation.		
78	54	I15-1c	OPG to verify the case where the actual condition of component could not be determined and its safety significance is not low.	Complete	This action was to provide the CNSC with specific information to address their inquiry. This action was completed. This issue is not impacted by extended operation. This issue does not apply to Pickering Units 1,4. There is therefore no PSR2 gap. Note: OPG is in the process of updating its Condition Assessments in support of extended operation.	No gap	Not Applicable
79	55	I15-1d	Submit the final report of the Service Limits Assessment for Class 1 components for CNSC staff review.	Complete	Service limits were only addressed to 2020 for Pickering 5-8. Further, Pickering Units 1,4 service limits need to be addressed to take into	The final report of the Service Limits Assessment for Class 1 components has not been updated taking the full period of extended operation into account for Pickering Units 1,4 and 5-	1

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
					consideration the extended period of operation. This is a gap for Pickering PSR2.	8. **(<u>Pickering PSR2 gap COP-22</u>)**	
80	56	I15-1e	Provide inspection results and maintenance strategy for ECI components to address concerns raised by CNSC staff. This includes providing; status of HPECI pumps overhauls, the ECI recovery heat exchanger inspection and an assessment of the current HPECI pump motor maintenance strategy.	Complete	This action was to provide the CNSC with information in response to a specific request. This action is complete. Ongoing inspection results and maintenance strategies for current and extended operation are addressed through the Integrated Aging Management Program, N-PROG-MP-0008 (Reference [28]). This issue is not impacted by extended operation. The Aging Management Program applies equally to Pickering Units 1,4. There is no issue for Pickering Units 1,4. Note: OPG is in the process of updating its Condition Assessments in support of extended	No gap	Not Applicable

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
					operation.		
81	57	I15-1f	Submit procedures used to track actions identified in the plant condition assessment reports.	Complete	There is no impact of extended operation. There is no issue for Pickering Units 1,4. There is no PSR2 gap.	No gap	Not Applicable
82	58	I15-1g	This is a log entry. OPG was requested to consider the proposed action that follows. Formalize the integration of TLAA into the Aging Management Program.	Dispositioned	OPG's response to the CNSC on this action included identification of the required elements of TLAA in its aging management governance. OPG continues to address TLAAs within its Integrated Aging Management Program, N-PROG-MP-0008 (Reference [28]). The Integrated Aging Management Program is not impacted by extended operation. The Integrated Aging Management Program also applies to Pickering Units 1,4. There is no PSR2 gap.	No gap	Not Applicable

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
83	59	I15-1h	This is a log entry. OPG was requested to consider the proposed action that follows. CNSC staff requested that OPG identify the cases where components that do not have a low safety significance and the actual condition of a component could not be determined through inspection. For each of these cases, describe OPG's response to such a situation.	Dispositioned	This action was to provide specific information to address a CNSC inquiry. This action is complete. OPG illustrated that the CCAs prepared in support of continued operation include identifying safety significance and method for determining condition of component. If the actual component condition cannot be assessed directly, OPG uses other processes such as review of OPEX, research, inspection of similar components, modelling, etc. OPG ascertains that the maintenance practices and method for assessing conditions were reviewed and in most cases the SSCs were subject to multiple practices with various combinations of Surveillance, Functional Testing, Preventive Maintenance (PM), Predictive Maintenance (PdM), and Inspection (Reference [9]).	No gap	Not Applicable

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
					There is no impact of extended operation. There is no issue for Pickering Units 1,4. There is no PSR2 gap.		
84	60	I15-2	Identify and provide a list of CCAs with components that were to be repaired or replaced during Refurb but now will be maintained.	Action was closed by CNSC as all required CCAs were updated by OPG to reflect continued operation, and the updates were reviewed by CNSC staff [13].	Ongoing Condition Assessments are addressed through the Integrated Aging Management Program, N-PROG-MP-0008 (Reference [28]). This issue is not impacted by extended operation. This issue does not apply to Pickering Units 1,4. There is no PSR2 gap. Note: OPG is in the process of updating its Condition Assessments in support of extended operation.	No gap	Not Applicable

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
85	61	I15-3	Confirm that actions identified in the Station Condition Records (SCRs) recorded in the CCAs that were prepared to support refurbishment were completed.	Complete	There is no impact by extended operation. There is no issue for Pickering Units 1,4. There is no PSR2 gap.	No gap	Not Applicable
86	62	I15-4	Identify any missing CCAs [from the 2010 CCA set] and revise CCAs. Some CCAs are missing in the updated 2010 CCA set. For example, for turbine and generators. These include the following CCAs: a) the Condition Assessments for the list of SSCs that were identified in the response; CCA 236 Vacuum Building (VB) - Upper Chamber Vacuum Pumps, CCA 237 VB - Upper Chamber Pump Seal Water Tank, CCA 217 VB - Main Volume Pump, CCA 238 Emergency Storage Water Pump 501/502, CCA 239 Emergency Storage Water Heat Exchanger, CCA 240 Emergency Storage Water Pump 503, CCA 241 Pressure Relief Panel Bypass Valves A. b) the Condition Assessments for the CCS of each individual reactor bldg (i.e., Units 5-8) and	Complete	There is no impact by extended operation. There is no issue for Pickering Units 1,4. There is no PSR2 gap. Note: OPG is in the process of updating its Condition Assessments in support of extended operation.	No gap	Not Applicable

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
			their associated sub-systems, and c) the condition assessment performed for Pickering A Common unit 0 Containment Structures. d) Turbine Generator set				
87	63	I15-5a	Respond to CNSC staff comment on CCA 32 (33410 Heat Transport Shutdown Cooling HXs, "What is the condition of other pressure boundary welds, runner-bars, baffle-plates, tie rods and seismic supports?)"	Complete	This action was to provide specific information in response to a CNSC inquiry. This action is complete. This issue is not impacted by extended operation. There is no issue for Pickering Units 1,4. There is no PSR2 gap. Note: OPG is in the process of updating its condition assessments in support of extended operation.	No gap	Not Applicable
88	64	I15-5b	Provide overall maintenance strategy for CCA 096 (the 71380 Emergency Water System (EWS) Recovery Pump Motors) and SCR P-2008- 13621 mentioned	Complete	This action was to provide specific information in response to a CNSC inquiry. This action is	No gap	Not Applicable

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
			in OPG 2009 response on CCA 096.		complete. This issue is not impacted by extended operation. There is no issue for Pickering Units 1,4 since there is no EWS. There is no PSR2 gap. Note: OPG is in the process of updating its condition assessments in support of extended operation.		
89	65	I15-5c	Provide a response to the CNSC request that OPG perform the following actions: "With respect to CCA 102 (73110 RH Cooling – Moderator Room Air Conditioning Units (ACUs)), complete the following actions: 1. Initiate the PMs to proactively replace all criticality code 1 and 2 coils. 2. Provide the rationale for replacing (or not) the solid base plate of the fan motors. This response should support the technical basis for continued	Complete	This action was to initiate PMs and provide specific information in response to a CNSC inquiry. This action is complete. This issue is not impacted by extended operation. There is no issue for Pickering Units 1,4. There is no PSR2 gap. Note: OPG is in the process of updating its condition assessments in	No gap	Not Applicable

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
			operation of Pickering B."		support of extended operation.		
90	66	I15-5d	Provide the detailed analysis and other reviews conducted to confirm that corrosion fatigue will not affect the welds in the deaerator and the storage tank.	Complete	This action was to provide a response to a specific CNSC inquiry. This action was completed. This issue has not been addressed in the context of extended operation. The issue has not been assessed in the context of extended operation and for Pickering Units 1,4. Therefore, this is a gap for Pickering PSR2. Note: OPG is in the process of updating its condition assessments in support of extended operation.	A review for the full period of extended operation has not been performed to confirm that corrosion fatigue will not affect the welds in the deaerator and the deaerator storage tank for Pickering Units 1,4 and 5-8. **(<u>Pickering PSR2 gap COP-23</u>)**	2
91	67	I15-5e	Submit the information on the inspection results of areas susceptible to wall thinning, high stress areas of the PHT auxiliary piping system, and a one-time inspection of the PHT pump discharge and boiler inlet and outlet valves.	Complete	This action was to provide a response to a specific CNSC inquiry. This action was completed. This issue has not been considered in the context of extended operation.	A review for extended operation has not been performed of the inspection results for areas susceptible to wall thinning, high stress areas of the PHT auxiliary piping system, and PHT pump discharge and boiler inlet	2

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
					This issue has not been assessed in the context of extended operation and for Pickering Units 1,4. Therefore, this is a gap for Pickering PSR2. Note: OPG is in the process of updating its condition assessments in support of extended operation.	and outlet valves for Pickering Units 1,4 and 5-8. **(<u>Pickering PSR2 gap COP-24</u>)**	
92	68	I15-6a	Provide a clear, complete and adequate rationale supporting OPG conclusions with the "satisfactory" category assigned for the components of containment system.	Complete	This action was to provide a response to a specific CNSC inquiry. This action was completed. This issue is not impacted by extended operation. There is no issue for Pickering Units 1,4. There is no PSR2 gap. Note: OPG is in the process of updating its condition assessments in support of extended operation.	No gap	Not Applicable

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
93	5	I15-6b	Demonstrate adequate margin to operate the Pickering B Reactor Building (RB) foundations to 31 Dec 2025. Demonstrating adequate margin may include inspections, testing or analysis to confirm the integrity of RB concrete footings and foundations as well as steel foundations and H-piles. Develop a mitigation plan if analysis does not indicate adequate margin. The analysis timeframe includes 5 years of operation to support stabilization activity planned to commence after the end of commercial operations on 31 Dec 2020. This analysis timeframe may also be expressed as nominal design life (2015) extending 10 years to 2025 to accommodate continued operations with adequate margin.	In Reference [11], CNSC staff identified that the issue cannot be closed pending additional information. OPG has had continued dialogue with the CNSC on this issue. In Reference [29] the CNSC identified this item as complete and closed.	This assessment only addressed operation to 2025. This issue must be reconsidered in the context of extended operation. This issue is also applicable to Pickering Units 1,4. This is therefore a gap for Pickering PSR2. Note: OPG is in the process of updating its condition assessments in support of extended operation.	An assessment of margin to operate all the Pickering Reactor Building foundations has not been completed for the period of extended operations and until Reactor Building integrity can be demonstrated to no longer be required. This issue applies also to the Vacuum Building and Pressure Relief Duct for the extended operations period and for the period until the negative pressure containment system integrity can be demonstrated to no longer be required. **(Pickering PSR2 gap COP-25)**	2
94	69	I15-7a	Include relevant information from COG JP 4271 calandria and internals Fitness for life Extension Guidelines (FFLEG) in N-PLAN-01060-10003" Reactor Components and Structures LCMP and submit the LCMP to	Complete	An interim LCMP update for major components is documented in P-CORR-01060-0587604 R000 (Reference [15]), which describes life cycle management strategies for	This gap is addressed in the Safety Factor 4 report (Reference [14]) as **PSR2 Gap SF4-18.**	4

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
			the CNSC in accordance with Pickering B PROL 08.20/2013 Licence Condition 1.2.		major components to achieve extended operations to 2024. This issue has not been addressed in the context of extended operation. The issue has not been considered for Pickering Units 1,4. Therefore, this is a gap for Pickering PSR2.		
95	70	I15-7b	With respect to the calandria and internal structures, provide the results of the COG Joint Project #4220 regarding the guide tube springs. The results should provide evidence that guide tube springs will remain fit for service for the mission life.	Complete	An interim LCMP update for major components is documented in P-CORR-01060-0587604 R000 (Reference [15]), which describes life cycle management strategies for major components to achieve extended operations to 2024. This issue has not been addressed in the context of extended operation. This issue applies to Pickering Units 1,4 and 5-8. Therefore, this is a gap for Pickering PSR2.	A review for the full period of extended operation has not been completed of evidence that calandria and internal structure guide tube springs for Pickering Units 1,4 and 5-8 will remain fit for service. **(Pickering PSR2 gap COP-26)**	2

COP Action #		IIP ID or Code	Summary of Item	Status	Impacted by Operation Past 2020 to 2028 or Implications for Pickering Units 1,4?	Implications for PSR2	Safety Factor Applicability
ID number from Ref. [13]	ID number from App A of COP (Ref. [8])			Based on App A of COP (Ref. [8]) unless noted otherwise)			
					Note: OPG is in the process of updating its LCMPs and condition assessments in support of extended operation.		

4.1 PICKERING NGS PSR2 SAFETY FACTOR 4

In Pickering NGS PSR2 Safety Factor 4 report [14], COP actions were evaluated for applicability for PSR2. Two gaps were identified. For completeness, the text of the related PSR2 gaps is repeated below.

Gap SF4-13

The PSR2 gap is as follows:

Actions #31, #32, and #33 from the Pickering Units 5-8 Continued Operations Plan are related to N287.7 and although complete, need to be re-assessed for Pickering operation past 2020 **(Pickering PSR2 Gap SF4-13)**. (IIP Action #31 involved submission of Periodic Inspection Plans and Life Cycle Management Plans for a number of safety-significant civil structures. IIP Action #32 involved submission of Aging Management Plans for concrete containment structures to the CNSC for acceptance. IIP Action #33 involved revising the Reactor Building Periodic Inspection Plan and submitting to the CNSC for acceptance).

Gap SF4-18

The PSR2 gap is as follows:

Review of the Pickering Units 5-8 Continued Operations Plan [8] identified the following closed gaps from the Pickering B ISR that will need to be revisited in the context of continued operation past 2020 for PSR2 Safety Factor 4 (**Pickering PSR2 Gap SF4-18**):

ID number from App A of COP (Ref. [8])	IIP ID [1] or Code	Summary of Item *
4	G01-04	Demonstrate adequate safety margins to operate Pickering B units from a Heat Transport System aging perspective to Jan 31, 2021. The 2015 strategy update to CNSC staff provided a progress report on HTS Aging Safety Analysis and related activities, and an updated revision of the HTS Aging Management Strategy for the period 2015-2020. This needs to be expanded to cover operation past 2020 for Pickering Units 1,4 and Units 5-8. Therefore, this is a gap for Pickering PSR2.
9, 10, 11, 12, 13	F01 (including F01-1, F01-2, F01-3, F01-4)	Develop a strategy to provide evidence that the Calandria Tube (CT) - Liquid Injection Shutdown System (LISS) nozzle gap will be maintained beyond 240,000 Effective Full Power Hours (EFPH) for all Pickering B units. The strategy for CT - LISS nozzle gap preservation may apply beyond 2025, but this needs to be confirmed. Therefore, this is a gap for Pickering PSR2.
14	F02	Develop R&D justification for extending Fuel Channel design life beyond 240,000 EFPH in the areas of hydrogen ingress, fracture toughness, spacer mobility and integrity. Actions were constrained by the shutdown date of 2020 assumed in the 2011 business plan. This needs to be expanded to cover operation past 2020 and is therefore a gap for Pickering PSR2.

ID number from App A of COP (Ref. [8])	IIP ID [1] or Code	Summary of Item *
		Note: An interim LCMP update for major components is documented in P-CORR-01060-0587604 R000 [15], which describes life cycle management strategies for major components to achieve extended operations to 2024. Strategies in this document may apply beyond 2024, but this needs to be confirmed as part of the resolution of this COP-related PSR2 gap.
19	F13	Update the NOP analysis for Pickering B. Actions were constrained by the shutdown date of 2020 assumed in the 2011 business plan. This is primarily relevant to Safety Factor 5 but is also of relevance to Safety Factor 4. This needs to be expanded to cover operation past 2020 for Pickering Units 1,4 and Units 5-8. Therefore, this is a gap for Pickering PSR2.
21	F14-1a	For the Feeder LCMPs, clarify the impact of fuel channel axial elongation during operation beyond the fuel channel assumed design life of 210,000 EFPH on feeder stress analysis and acceptable feeder thickness. This was only addressed to 2025. This needs to be expanded to cover operation to 2028. Therefore, this is a gap for Pickering PSR2. Note: An interim LCMP update for major components is documented in P-CORR-01060-0587604 R000 [15], which describes life cycle management strategies for major components to achieve extended operations to 2024. Strategies in this document may apply beyond 2024, but this needs to be confirmed as part of the resolution of this COP-related PSR2 gap.
30	F14-4.1	Include the periodic inspection programs and LCMPs for the secondary side pressure retaining components and submit them for CNSC review. Although the action to submit PIPs and LCMPs for the secondary side pressure retaining components is complete, these documents will need to be extended to cover operation past 2020 for Pickering Units 1,4 and Units 5-8. Therefore, this is a gap for Pickering PSR2.
52	I15-1a	Perform Time Limiting Aging Analysis (TLAAs) and include such TLAAs in the LCMPs and in the CAs. OPG to provide commitment that TLAAs necessary to determine the actual conditions of components will be completed. Although the action is complete, this will need to be updated to cover operation past 2020 for Pickering Units 1,4 and Units 5-8. Therefore, this is a gap for Pickering PSR2.
69	I15-7a	Include relevant information from COG JP 4271 Calandria and internals. Fitness for Life Extension Guidelines in N-PLAN-01060-10003 “Reactor Components and Structures Life Cycle Management Plan (LCMP)” and submit the LCMP to the CNSC in accordance with Pickering B PROL 08.20/2013 LC 1.2. Although the action is complete, this will need to be updated to cover operation past 2020 for Pickering Units 1,4 and Units 5-8. Therefore, this is a gap for Pickering PSR2. Note: An interim LCMP update for major components is documented in P-

ID number from App A of COP (Ref. [8])	IIP ID [1] or Code	Summary of Item *
		CORR-01060-0587604 R000 (Reference [15]) which describes life cycle management strategies for major components to achieve extended operations to 2024. Strategies in this document may apply beyond 2024, but this needs to be confirmed as part of the resolution of this COP-related PSR2 gap.
Appendix C, Item 5	F11	Update the Pickering B HTS aging model. This action is complete but needs to be reviewed to assess impact of operation past 2020 for Pickering Units 1,4 and Units 5-8. Therefore, this is a gap for Pickering PSR2.
Appendix C, Item 6	F12	Update the Pickering B HTS aging management strategy. This action is complete but needs to be reviewed to assess impact of operation past 2020 for Pickering Units 1,4 and Units 5-8. Therefore, this is a gap for Pickering PSR2.

* Closed Pickering Units 5-8 COP actions were reviewed to determine whether they need to be reassessed (PSR2 gaps identified) to address operation past 2020. Where applicable, equivalent Pickering Units 1,4 PSR2 gaps are also identified where reassessment will be required for operation past 2020.

5.0 ACRONYMS

ACU	Air Cooling Units
AECB	Atomic Energy Control Board
AGS	Annulus Gas System
AMP	Ageing Management Plan
AMS	Aging Management Strategy
AI	Action Item
AR	Action Request
BCA	Benefit-Cost Analysis
BLA	Boiling Length Average
CA	Condition Assessment
CANDU	CANadian Deuterium Uranium
CCA	Component Condition Assessment
CHF	Critical Heat Flux
CNSC	Canadian Nuclear Safety Commission
COG	CANDU Owners Group
COP	Continued Operations Plan

CSA	Canadian Standards Association
CSI	CANDU Safety Issue
CT	Calandria Tube
DCR	Document Change Requests
DPROR	Dew Point Rate of Rise
EA	Environmental Assessment
ECI	Emergency Cooling Injection
ECIS	Emergency Coolant Injection System
EFPH	Equivalent Full Power Hours
EPG	Emergency Power Generator
EPS	Emergency Power System
EWS	Emergency Water System
FAC	Flow Accelerated Corrosion
FADS	Filtered Air Discharge System
FFLEG	Fitness for Life Extension Guidelines
FFS	Fitness for Service
FP	Fracture Protection
FSA	Fire Safety Assessment
GAI	Generic Action item
HPECI	Pickering High Pressure Emergency Coolant Injection
HTS	Heat Transport System
IIP	Integrated Implementation Plan
ISR	Integrated Safety Review
LBB	Leak Before Break
LBLOCA	Large Break Loss of Coolant Accident
LCH	Licence Conditions Handbook
LCMP	Life Cycle Management Plan
LISS	Liquid Injection Shutdown System
LOCA	Loss of Coolant Accident
NFCC	National Fire Code of Canada
NGS	Nuclear Generating Station
NPP	Nuclear Power Plant
NOP	Neutron Over Power

OPEX	Operating Experience
OPG	Ontario Power Generation
PARMS	Post Accident Radiological Monitoring System
PAR	Passive Autocatalytic Recombiner
PdM	Predictive Maintenance
PHT	Primary Heat Transport
PIE	Postulated Internal Events
PM	Preventive Maintenance
PNGS	Pickering Nuclear Generating Station
PRA	Probabilistic Risk Assessment
PROL	Power Reactor Operating Licence
PSR2	Periodic Safety Review 2
PT	Pressure Tube
R&D	Research and Development
RB	Reactor Building
RegM	Regulatory Management
RJ	Rolled Joint
RJD	Rolled Joint Deuterium
RRS	Reactor Regulating System
SAMG	Severe Accident Management Guidance
SCR	Station Condition Record
SDS1	Shutdown System 1
SDS2	Shutdown System 2
SDSA	Shutdown System A
SDSE	Shutdown System Enhancement
SOE	Safe Operating Envelope
SOP	Sustainable Operations Plan
SSC	Structures, Systems, and Components
TCD	Target Completion Date
TUF	Two Unequal Fluids
TBD	Technical Basis Documents
TLAA	Time Limiting Aging Assessment
VB	Vacuum Building

VBO	Vacuum Building Outage

6.0 REFERENCES

[1] OPG Plan, NK30-PLAN-03680-00002 R000, *Pickering B - Integrated Implementation Plan*, December 2011.

[2] OPG Report, NK30-REP-03680-0400585 R003, *Pickering B Integrated Safety Review Global Assessment Report (GAR),* September 2009.

[3] OPG Plan, NK30-PLAN-00531-00001 R000, *Pickering B Continued Operations Plan*, September 2010.

[4] OPG Plan, NK30-PLAN-00531-00001 R001, *Pickering B Continued Operations Plan*, December 2011.

[5] OPG Plan, NK30-PLAN-00531-00001 R002, *Pickering B Continued Operations Plan*, December 2012.

[6] OPG Plan, NK30-PLAN-00531-00001 R003, *Pickering 5-8 Continued Operations Plan*, December 2013.

[7] OPG Plan, NK30-PLAN-00531-00001 R004, *Pickering 5-8 Continued Operations Plan*, December 2014.

[8] OPG Plan, NK30-PLAN-00531-00001 R005, *Pickering 5-8 Continued Operations Plan*, November 2015.

[9] OPG Correspondence, P-CORR-00531-04470 R000, B. McGee to M. Santini, *Pickering 5-8, Continued Operations Plan - 2015 Final Update*, December 15, 2015.

[10] OPG Report, P-REP-03680-00001 R002, Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document, June 2016.

[11] CNSC Correspondence, e-Doc 5024526, OPG File No. P-CORR-00531-04786 R000, H. Khouaja to B. McGee, *Pickering NGS: CNSC Staff Assessment of 2015 COP, SOP, SAP and CALs*, July 6, 2016.

[12] CNSC Correspondence, e-Doc 4452163, OPG File No. P-CORR-00531-04272 R000, M. Santini to B. Phillips, *Pickering NGS: CNSC Staff Assessment of 2013 COP, SOP and CAL Closure of AI 2013-48-4185*, June 18, 2014.

[13] CNSC Correspondence, e-Doc 4454610, Enclosure to OPG File No. P-CORR-00531-04272 R000, *CNSC staff Assessment of Pickering NGS 2013 EOL Consolidated Actions Log (CAL) (COP, SOP, SAP and SSP*), June 18, 2014.

[14] OPG Report, P-REP-03680-00007 R000, *Pickering NGS PSR2 Safety Factor 4 Report: Aging,* July 2016.

[15] OPG Memorandum, P-CORR-01060-0587604, *Fitness for Service of Major Components*, March 29, 2016.

[16] OPG Report, N-REP-03500-10011 R000, *Uncertainty Associated with Using 28-Element BLA CHF Correlation Derived Based on Stern Laboratories CHF Tests*, October 2009.

[17] OPG Correspondence, N-CORR-00531-07323 R000, W. M. Elliott to M. Santini, *Fuel Management Surveillance Software Upgrade – Action Item 2012-OPG-3465 (Post GAI 01G01 Closure),* July 28, 2014.

[18] CNSC Correspondence, e-Doc 5047254, OPG File No. N-CORR-00531-18178 R000, H. Khouaja and M. Santini to S. Woods, *Darlington and Pickering NGS: Closure of Action Item 2012-OPG-3465 – Post Closure of GAI 01G01 Fuel Management and Surveillance Software Upgrade, New Action Item 2016-OPG-8250*, July 26, 2016.

[19] OPG Report, NA44-REP-71400-10001 R001, *Pickering Nuclear Generating Station "A", Fire Protection Code Compliance Review*, March 2011.

[20] OPG Report, P-REP-61150-00002 R001, *Available for Service Report PN Seismic Monitoring System Replacement Project (Syscom Instruments) Project 13-49129 Unit 018*, December 2013.

[21] OPG Report, NK30-REP-02004-00001 R000, *Pickering NGS B Seismic Overturning and Sliding Safety Factors in Compliance with CSA N289.3-M81 Clause 4.7.1 - 4.7.3*, September 2013.

[22] OPG Report, NA44-REP-03611-00022 R000, *PRA-Based Seismic Margin Assessment of PNGS-A*, January 2014.

[23] OPG Report, NA44-REP-71400-10003 R001, *Fire Hazard Assessment - Pickering A Nuclear Generating Station*, April 2012.

[24] OPG Report, NA44-REP-71400-00023 R000, *Fire Safe Shutdown Analysis - Pickering A Nuclear Generating Station*, April 2012.

[25] OPG Report, N-REP-03600-10003 R007, *Fukushima Action Item Status Report*, November 2015.

[26] OPG Correspondence, NK30-CORR-00531-05978 R000, G. Jager to M. Santini, *Pickering B – Type II System Inspection – Filtered Air Discharge System – Request to Close CNSC Action item 2006-8-02,* November 23, 2011.

[27] CNSC Correspondence, e-Doc 3986551, OPG File No. NK30-CORR-00531-06381 R000, M. Santini to G. Jager, *Pickering B – Type II System Inspection: Filtered Air Discharge System – Closure of CNSC Action Item 2006-8-02 (RIB #2413),* August 24, 2012.

[28] OPG Program, N-PROG-MP-0008 R006 B, *Integrated Aging Management*, May 2016.

[29] CNSC Correspondence, e-Doc 5129748, OPG File No. P-CORR-00531-04901, A. Viktorov to B. McGee, *Pickering NGS: Closure of Actions F06 and 115-6b of the Continued Operations Plan (COP)*, November 28, 2016.

[30] CNSC Correspondence, e-Doc 4407343, OPG File No. N-CORR-00531-06509, M. Santini and F. Rinfret to W.M. Elliott, *Closure of Action Item 2011OPG-01 – TUF Validation Work*, March 25, 2014.

[31] OPG Correspondence, N-CORR-00531-18204, W. S. Woods to A. Viktorov and M. Santini, *Darlington and Pickering NGS: Fuel Management Surveillance Software Upgrade – New Action Item 2016-OPG-8250,* September 16, 2016.

[32] CNSC Correspondence, e-Doc 4054739, OPG File No. N-CORR-00531-06063, *Request for additional information on PHTS Aging, Fuel Channel and Plant Thermalhydraulics and CCP uncertainty aspects of the new NOP Analysis Methodology - Closure of Action Item 2012·0PG-3464,* February 13, 2013.

[33] OPG Correspondence, N-CORR-00531-05900, W. M. Elliott to P. A. Webster and M. Santini, *Request for Closure of Action Item 2012-OPG-3464: Request for Additional Information on PHTS Aging, Fuel Channel and Plant Thermalhydraulics and CCP Uncertainty Aspects of the New NOP Analysis Methodology*, November 27, 2012.

[34] CNSC Correspondence, e-Doc 492988, OPG File No. N-CORR-00531-06689, M. Santini and F. Rinfret to W. M. Elliott, *Darlington and Pickering NGS: Scaling Assessment for Channel Voiding Closure of Action Item 20120PG·3317*, September 5, 2014.

[35] CNSC Correspondence, e-Doc 492988, OPG File No. NK30-CORR-00531-07215, M. Santini to B. McGee, *Pickering NGS: Assurance of Structural Fuel Channel Fitness-for-Service for the Target Service Life of Pickering Units 5-8,* March 9, 2016.

[36] OPG Report, NK30-REP-31100-10143, *Pickering 5–8 Probabilistic Assessment of Leak-Before-Break*, July 29, 2014.

[37] OPG Report, NK30-REP-31100-10179, *Probabilistic Evaluation of Pressure Tube Leak-Before-Break in Pickering Units 5, 6, and 8 – 2016 Update,* November 16, 2016.

[38] OPG Report, NK30-REP-31100-00021, *Pickering B Risk Assessment Summary Report,* February 14, 2013.

[39] OPG Report, NA44-REP-03611-00036, *Pickering A Risk Assessment Summary Report,* April 25, 2014.

[40] CNSC Report, *Regulatory Oversight Report for Canadian Nuclear Power Plants: 2014,* September 2015.

[41] OPG Design Manual, NK30-67876.2, *Fixed Gaseous Process Radiation Monitoring, Filtered Air Discharge System,* April 1985.

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ONTARIO POWER GENERATION	
ACCEPTED	✓
ACCEPTED AS NOTED	
REVISE AND RESUBMIT	
Signature	Date 06 Feb 2017
Name: Mike Ruffolo, Manager	
Dept: Pickering Engineering - Aging Management & Strategic Initiatives	
OPG Proprietary	
Doc No.: P-REP-03680-00022	Rev: 000
This acceptance does not relieve the contractor from responsibility for errors or omissions or from any obligations or liability under this contract.	

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Fukushima Action Item Review in Support of PNGS Periodic Safety Review 2 (PSR2)

PS112/RP/019 R01

January 20, 2017

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Revision Summary

Rev	Date	Author	Comments
R00	December 21, 2016	Stan B. Harvey	Initial issue
R01	January 20, 2017	Stan B. Harvey	Incorporate OPG comments

EXECUTIVE SUMMARY

Fukushima Action Items (FAIs) are considered for applicability to Pickering Periodic Safety Review (PSR2) in this report. The PSR2 Basis Document, Section 3.2.1 (Reference [1]) states that Fukushima actions will be assessed to determine if there are any impacts associated with operation past 2020. Also, in Reference [2], the Canadian Nuclear Safety Commission (CNSC) recommended that OPG consider the FAIs that were closed on the basis that Pickering NGS units were to shut down in 2020. The basis of FAI closure has been assessed to determine if it is affected by extending the operation beyond 2020.

There were no Pickering PSR2 gaps identified in this report.

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1.0 INTRODUCTION

Fukushima Action Items (FAIs) are considered for applicability to Pickering Periodic Safety Review (PSR2) in this report. The PSR2 Basis Document, Section 3.2.1 (Reference [1]) states that Fukushima actions will be assessed to determine if there are any impacts associated with operation past 2020. This is consistent with the CNSC recommendation in Reference [2] for OPG to consider the FAIs that were closed on the basis that Pickering NGS units were to shut down in 2020. The CNSC recommendation was to determine if the basis of closure is affected by the potential to operate Pickering to 2028 rather than 2020.

The FAIs were originally identified in Reference [3]. The status of all FAIs for OPG facilities was summarized in OPG's Fukushima Action Item Status Report, N-REP-03600-10003 [4]. As noted in Reference [4], OPG has completed, and the CNSC has closed all FAIs assigned to OPG. The assessment of the impact of extended operation on all FAIs assigned to OPG that are applicable to Pickering NGS is provided in Table 1 in Section 4.0.

2.0 SCOPE OF REVIEW

This report considers all FAIs that were listed in Reference [3] and the follow up actions that have been created to monitor progress related to specific issues. The assessment considers Units 1,4 and Units 5-8. Common systems (Unit 0) are included in the Unit assessments as applicable.

3.0 METHODOLOGY

The FAIs listed in Table 1 are taken directly from Reference [3]. The dates listed with each FAI are those identified in Reference [3] when the FAIs were originally developed. The impact of extended operation was considered for Units 1,4 and for Units 5-8. In cases where Units 1,4 and Units 5-8 have the same implications, these discussions are combined.

4.0 ASSESSMENT FINDINGS

The assessment is documented in Table 1.

Table 1: Impact of Extended Operation on FAIs – Assessment Results

#	Fukushima Action Item [3]	Assessment of Impact of Operation Beyond 2020		Results
		Pickering Units 1,4	Pickering Units 5-8	
1	FAI 1.1.1[1] An updated evaluation of the capability of bleed condenser / degasser condenser relief valves providing additional evidence that the valves have sufficient capacity. December 2012.[2]	As outlined in Reference [4], an assessment was performed that demonstrated that the installed Relief Valves on the Bleed Condenser provide sufficient relief capacity such that pressure boundary failure due to overpressure will not occur. FAI 1.1.1 is closed. This conclusion is not impacted by extended operation.		No gap
2	FAI 1.1.2 If required, a plan and schedule either for confirmatory testing of installation or provision for additional relief capacity. December 2012.	No further action is required based on results of FAI 1.1.1. [4] FAI 1.1.2 is closed. This conclusion is not impacted by extended operation.		No gap

[1] The FAI numbering shown here is from Reference [4], which reflects the CNSC FAI numbering that was adopted following the initial issuance of the FAIs in Reference [3]. For communications that followed after Reference [3], the CNSC (and industry) adopted a standard three digit FAI numbering system which affected some FAI numbers (e.g., FAI 1.1 became FAI 1.1.1). Affected FAI numbers (changes from Reference [3] to Reference [4]) are FAI 1.1 (1.1.1), 1.4 (1.4.1), 1.5 (1.5.1), 1.7 (1.7.1), 1.8 (1.8.1), 1.9 (1.9.1), 1.11 (1.11.1), 2.2 (2.2.1), 4.2 (4.2.1), 5.3 (5.3.1) and 5.4 (5.4.1).

[2] Dates shown are the due dates identified in Reference [3] when the FAIs were originally assigned.

#	Fukushima Action Item [3]	Assessment of Impact of Operation Beyond 2020		Results
		Pickering Units 1,4	Pickering Units 5-8	
3	FAI 1.2.1 An assessment of the capability of shield tank/calandria vault relief. December 2013.	N/A [3] This action does not apply to Pickering Units 1,4 because there is no shield tank in the design of these units.	As outlined in Reference [4], a review of the Calandria Vault (CV) structural integrity has been completed which confirms the adequacy of relief capability. FAI 1.2.1 is closed. This conclusion is not impacted by extended operation.	No gap
4	FAI 1.2.2 If relief capacity is inadequate, an assessment of the benefit available from adequate relief capacity and the practicability of providing additional relief. December 2013.	N/A [3] This action does not apply to Pickering Units 1,4 because there is no shield tank in the design of these units.	As outlined in Reference [4], analyses show that there is sufficient release through the CV plug to prevent over-pressurization of the CV. No further action is required. Refer to FAI 1.2.1 for further information. FAI 1.2.2 is closed. This conclusion is not impacted by extended operation.	No gap
5	FAI 1.2.3 If additional relief is beneficial and practicable, a plan and schedule for provision of additional relief. December 2013.	N/A [3] This action does not apply to Pickering Units 1,4 because there is no shield tank in the design of these units.	As outlined in Reference [4], analyses show that there is sufficient release through the CV plug to prevent over-pressurization of the CV. Refer to FAI 1.2.1 for further information. FAI 1.2.3 is closed. This conclusion is not impacted by extended operation.	No gap

#	Fukushima Action Item [3]	Assessment of Impact of Operation Beyond 2020		Results
		Pickering Units 1,4	Pickering Units 5-8	
6	FAI 1.3.1 Assessments of adequacy of the existing means to protect containment integrity and prevent uncontrolled release in beyond design basis accidents including severe accidents. December 2015.	As outlined in Reference [4], the FAI was closed based on the Level 2 Probabilistic Risk Assessment[3] (PRA) results, and considering overall containment performance under Beyond Design Basis Accident (BDBA) conditions. This included an assessment of the effectiveness of the Phase 1 and Phase 2 Emergency Mitigating Equipment (EME) mitigation of multi-unit "Total Loss of Heat Sink" events. The Phase 1 EME is currently available for service, and addresses the majority of postulated BDBAs. Phase 2 EME is a planned enhancement. FAI 1.3.1 is closed. This conclusion is not impacted by extended operation (Reference [5]). See also further discussion under FAI 1.3.2.		No gap

[3] Note that Probabilistic Risk Assessment is now referred to as Probabilistic Safety Assessment.

#	Fukushima Action Item [3]	Assessment of Impact of Operation Beyond 2020		Results
		Pickering Units 1,4	**Pickering Units 5-8**	
7	FAI 1.3.2 Where the existing means to protect containment integrity and prevent uncontrolled releases of radioactive products in beyond design basis accidents including severe accidents are found inadequate, a plan and schedule for design enhancements to control long term radiological releases and, to the extent practicable, unfiltered releases. December 2015.	As outlined in Reference [4], there are common elements to this activity for Pickering Units 1,4 and Pickering Units 5-8, because of the shared containment envelope. As discussed under FAI 1.3.1, and further reviewed in Reference [6], OPG has completed analyses for Pickering which outline the extent to which containment integrity may be challenged under BDBA conditions. OPG has assessed various options which can be implemented to enhance containment performance under extreme conditions. It has been determined that the existing and planned equipment upgrades (to address Fukushima related issues) are sufficient to address containment integrity challenges under BDBA conditions, and no further modifications are warranted. FAI 1.3.2 is closed. The original FAI 1.3.1/1.3.2 assessment for Pickering is summarized in P-REP-09013-00002, "Pickering NGS - Beyond Design Basis Containment Integrity" [6]. This evaluation supported the closure of FAI 1.3.2 for Pickering NGS. FAI 1.3.1/1.3.2 was reassessed in Reference [5] taking into consideration the extended operation of Pickering NGS. The updated assessment of containment integrity in BDBAs at Pickering NGS concluded that the initial response to these FAIs remains valid for the extension of commercial operation. This was based on a number of considerations outlined in References [5] and [6], including the Beyond Design Basis Nuclear Safety Principles adopted by the Chief Nuclear Engineers of the Canadian Nuclear Utilities which focus on prevention, the defence-in-depth capability of the design features of the plant and the additional EME provisions, consideration of safety goals and benefits, and additional risk improvement initiatives already underway at Pickering NGS [9].		No gap

#	Fukushima Action Item [3]	Assessment of Impact of Operation Beyond 2020		Results
		Pickering Units 1,4	Pickering Units 5-8	
8	FAI 1.4.1 A plan and schedule for the installation of Passive Autocatalytic Recombiners (PARs) as quickly as possible. December 2012.	As indicated in Reference [3] this FAI was closed. PARs have been installed [4]. Programs are in place to ensure continued operation of the PARs [10]. This conclusion is not impacted by extended operation.		No gap
9	FAI 1.5.1 An evaluation of the potential for hydrogen generation in the Irradiated Fuel Bay (IFB) area and the need for hydrogen mitigation. December 2013.	As outlined in Reference [4], OPG has completed an assessment for the IFBs at Pickering NGS under FAI 1.6.1. This assessment demonstrates that there is high confidence that the IFBs at Pickering NGS will remain filled under an extended loss of IFB cooling event, crediting existing IFB coolant make-up strategies. Therefore, hydrogen production, aside from the normal production from radiolysis, will not occur and hydrogen mitigation in the IFBs is not required. FAI 1.5.1 is closed. This conclusion is not impacted by extended operation.		No gap
10	FAI 1.6.1 An evaluation of the structural response of the IFB structure to temperatures in excess of the design temperature, including an assessment of the maximum credible leak rate following any predicted structural damage. December 2013.	As outlined in Reference [4], OPG has completed an assessment for Pickering NGS IFBs. This assessment demonstrated that in the event of a loss of IFB cooling, some IFB leakage may occur as temperatures approach boiling. However, the leakage make-up requirements are well within the capability of portable EME pumps. The assessment showed that for an extended loss of all AC power, there is high confidence that the IFBs at Pickering NGS will be maintained adequately filled and the irradiated fuel adequately cooled using existing IFB coolant make-up strategies and augmented by EME pumps. FAI 1.6.1 is closed. This conclusion is not impacted by extended operation. Note that OPG is in the process of updating IFB Condition Assessments in support of extended operation.		No gap

#	Fukushima Action Item [3]	Assessment of Impact of Operation Beyond 2020		Results
		Pickering Units 1,4	Pickering Units 5-8	
11	FAI 1.6.2 A plan and schedule for deployment of any additional mitigating measures shown to be necessary by the evaluation of structural integrity. December 2013.	As outlined in Reference [4], it has been determined that expected leakage is well within the capability of additional water sources, such as EME, to mitigate. FAI 1.6.2 is closed. This conclusion is not impacted by extended operation.		No gap
12	FAI 1.7.1 A plan and schedule for optimizing existing provisions (to provide coolant makeup to Primary Heat Transport System, Steam Generators, moderator, etc.) and putting in place additional coolant make-up provisions, and supporting analyses. December 2013.	As outlined in Reference [4], EME was successfully deployed. FAI 1.7.1 is closed. This conclusion is not impacted by extended operation.		No gap
13	FAI 1.8.1 A detailed plan and schedule for performing assessments of equipment survivability, and a plan and schedule for equipment upgrade where appropriate based on the assessment. December 2013.	An assessment [4] has demonstrated that there is reasonable assurance that sufficient equipment and instrumentation will be available to facilitate operator actions at Pickering under a wide range of BDBA conditions, and hence no further actions are required. FAI 1.8.1 is closed. This conclusion is not impacted by extended operation.		No gap

#	Fukushima Action Item [3]	Assessment of Impact of Operation Beyond 2020		Results
		Pickering Units 1,4	Pickering Units 5-8	
14	FAI 1.9.1 An evaluation of the habitability of control facilities under conditions arising from beyond-design basis and severe accidents. Where applicable, detailed plan and schedule for control facilities upgrades. December 2014.	As outlined in Reference [4], for Pickering 1,4 and 5-8, plant habitability assessments were performed. The Pickering habitability assessment indicates that OPG's installed and planned upgrades and additional lines of defence (see FAI 1.7.1 which is complete) are sufficient to terminate event progression at or before the early In Vessel Retention stage, thereby supporting station habitability and providing reasonable confidence that essential operator actions can be completed in a timely manner. FAI 1.9.1 is closed. This conclusion is not impacted by extended operation.		No gap
15	FAI 1.10.1 An evaluation of the requirements and capabilities for electrical power for key instrumentation and control. The evaluation should identify practicable upgrades that would extend the availability of key I&C, if needed. December 2012.	As outlined in Reference [4], OPG has completed a comprehensive assessment demonstrating that adequate provisions are in place to ensure electric power supply is available to essential instrumentation and equipment. Following an extended loss of all AC power, the plant can rely on station battery back-up power, followed by the use of EME equipment including temporary portable uninterruptible power supplies if required as a bridging strategy, and then portable EME electrical generators. FAI 1.10.1 is closed. This conclusion is not impacted by extended operation.		No gap
16	FAI 1.10.2 A plan and schedule for deployment of identified upgrades. A target of 8 hours without the need for offsite support should be used. December 2012.	As outlined in Reference [4], OPG has completed a comprehensive assessment demonstrating that adequate provisions are in place to ensure electric power supply is available to essential instrumentation and equipment. Following an extended loss of all AC power, the plant can rely on station battery back-up power, then portable uninterruptible power supplies if required as a bridging strategy, and followed by the use of EME equipment including temporary portable EME electrical generators. FAI 1.10.2 is closed. This conclusion is not impacted by extended operation.		No gap

#	Fukushima Action Item [3]	Assessment of Impact of Operation Beyond 2020		Results
		Pickering Units 1,4	Pickering Units 5-8	
17	FAI 1.11.1 A plan and schedule for procurement (of emergency equipment and other resources that could be stored offsite). December 2012.	As outlined in Reference [4], OPG committed to undertake this work for all OPG nuclear stations as part of its Emergency Mitigating Equipment project (see FAI 1.7). Subsequently OPG procured emergency equipment covered by FAI 1.11.1. FAI 1.11.1 is closed. This conclusion is not impacted by extended operation.		No gap
18	FAI 2.1.1 Re-evaluation, using modern calculations and state of the art methods, of the site specific magnitudes of each external event to which the plant may be susceptible. December 2013.	As outlined in Reference [4], the re-evaluation of external events has been completed. For Pickering, the screening analyses were completed in 2012. The assessment of the unscreened events is detailed in FAI 2.1.2. FAI 2.1.1 is closed. This conclusion is not impacted by extended operation.		No gap
19	FAI 2.1.2 Evaluate if the current site specific design protection for each external event assessed in 1 above is sufficient. If gaps are identified a corrective plan should be proposed. December 2013.	As outlined in Reference [4]: As described in FAI 2.1.1, two unscreened hazards were identified at Pickering 1,4 and 5-8: high winds and seismic events. For Pickering 1,4 and 5-8 analysis of high winds and seismic events was completed. For Pickering 1,4, EME was credited to provide mitigation for some severe wind events considered in these analyses. It was determined that mitigating actions to secure EME to prevent failure and/or impairment resulting from high winds would be beneficial and these actions were undertaken.		No gap

#	Fukushima Action Item [3]	Assessment of Impact of Operation Beyond 2020		Results
		Pickering Units 1,4	Pickering Units 5-8	
		For Pickering 5-8, the high wind PRA estimated the Severe Core Damage Frequency (SCDF) for high winds with a recurrence interval of 10,000 years to be within OPG's safety goal target. A PRA-based Seismic Margin Assessment was completed for Pickering 1,4 and 5-8. The results demonstrate satisfactory station performance under seismic events. A review of seismically induced fires and floods was completed for Pickering 1,4 and Pickering 5-8. The assessment did not identify any fire or flood sources that would significantly affect the estimate of SCDF in the seismic PRA. (Reference [4]) FAI 2.1.2 is closed. This conclusion is not impacted by extended operation.		
20	FAI 2.2.1 Site-specific implementation plans for RD-310. December 2013.	As outlined in Reference [4], the Pickering RD-310 implementation plan was developed and was discussed with CNSC staff in February 2013. The Pickering implementation plan was subsequently submitted to the CNSC. CNSC closed FAI 2.2.1 following formal submission of the Pickering RD-310 implementation plans. As outlined in Reference [8], subsequently, the CNSC replaced RD-310 with REGDOC-2.4.1, and OPG developed a REGDOC-2.4.1 implementation plan. In Reference [8], OPG's response to FAI 2.2.1 is reassessed in the context of extended operation. The conclusion of that reassessment was that the 2014-2017 REGDOC-2.4.1 Implementation Plan for Pickering remains valid. Under this plan, new Safety Report Appendices to address Common Mode Events are being developed, consistent with the graded approach. Note that REGDOC-2.4.1 is in the assessment basis for Pickering PSR2, and the results of the review will be considered in the PSR2 Global Assessment.		No gap

#	Fukushima Action Item [3]	Assessment of Impact of Operation Beyond 2020		Results
		Pickering Units 1,4	Pickering Units 5-8	
21	FAI 3.1.1 Where SAMG has not been developed/finalized or fully implemented; provide plans and schedules for completion. December 2013.	As outlined in Reference [4], plans and schedules were developed, and Severe Accident Management Guidance (SAMG) has been implemented for all Pickering units. FAI 3.1.1 is closed. This conclusion is not impacted by extended operation.		No gap
22	FAI 3.1.2 For multi-unit stations, provide plans and schedules for the inclusion of multi-unit events in SAMGs. December 2013.	As outlined in Reference [4], a detailed plan and schedule for the explicit inclusion of multi-unit effects and consideration in SAMG was prepared and submitted to CNSC. SAMGs have been updated for multi-unit events. FAI 3.1.2 is closed. This conclusion is not impacted by extended operation.		No gap
23	FAI 3.1.3 For all stations, plans and schedules for the inclusion of IFB events in station operating documentation where appropriate. December 2013.	As outlined in Reference [4], evaluations of IFB loss of cooling events were prepared for all Pickering IFBs. Procedural revisions to improve the station response to events in the IFBs were completed. Additional portable EME was procured and usage guidelines implemented at Pickering for responding to IFB events following a total loss of AC power. FAI 3.1.3 is closed. This conclusion is not impacted by extended operation.		No gap
24	FAI 3.1.4 Demonstration of effectiveness of SAMGs via table-top exercise and drills. December 2013.	As outlined in Reference [4], Severe Accident Management Guidance table-top exercises and drills have been conducted at Pickering. FAI 3.1.4 is closed. This conclusion is not impacted by extended operation.		No gap

#	Fukushima Action Item [3]	Assessment of Impact of Operation Beyond 2020		Results
		Pickering Units 1,4	Pickering Units 5-8	
25	FAI 3.2.1 An evaluation of the adequacy of existing modeling of severe accidents in multi-unit stations. The evaluation should provide a functional specification of any necessary improved models. December 2012.	As outlined in Reference [4], the assessment of severe accident modelling in a multi-unit station was jointly prepared by OPG and Bruce Power. FAI 3.2.1 is closed. This conclusion is not impacted by extended operation.		No gap
26	FAI 3.2.2 A plan and schedule for the development of improved modeling, including any necessary experimental support. December 2012.	As outlined in Reference [4], OPG has completed an assessment demonstrating that the current modeling techniques used to assess the impact and consequences of multi-unit events at the Pickering station are sufficient. FAI 3.2.2 is closed. A multi-unit MAAP4-CANDU model was developed to confirm that the multi-unit severe accident progression and consequences are adequately simulated. Test cases demonstrated that modeling techniques are acceptable for multi-unit stations [7]. This conclusion is not impacted by extended operation.		No gap
27	FAI 4.1.1 An evaluation of the adequacy of existing emergency plans and programs. December 2012.	As outlined in Reference [4], the self-assessment and related evaluation of the emergency plan response capability completed in 2011, and resultant initiatives to resolve identified gaps met the intent of CNSC FAI 4.1.1. These initiatives are either complete or underway as either stand-alone projects or activities tracked by the OPG Action Tracking system. FAI 4.1.1 is closed. The required follow-up plan and schedule are considered in FAI 4.1.2. This conclusion is not impacted by extended operation.		No gap

#	Fukushima Action Item [3]	Assessment of Impact of Operation Beyond 2020		Results
		Pickering Units 1,4	Pickering Units 5-8	
28	FAI 4.1.2 A plan and schedule to address any gaps identified in the evaluation. December 2012.	As outlined in Reference [4], information on the evaluation of existing plans and programs is presented in the description for FAI 4.1.1. One of the initiatives from the post-Fukushima evaluation was to incorporate Beyond Design Basis Events (BDBE), including BDBA, into emergency plans. A BDBE accident scenario was prepared and used to benchmark the onsite emergency plan and review its earlier assessments for BDBA. The conclusion was that the existing Emergency Response Organization, command structure, staffing and governance are event-independent and adequate for BDBA. Gaps were identified related to: • Emergency Planning Basis to include BDBA (generic gap) • Site access • Public dose projection tool – Emergency Response Projection (ERP) • Dosimetry and radiation protection for large numbers of support staff. These gaps were addressed according to an approved plan and schedule. On this basis, FAI 4.1.2 is closed. This conclusion is not impacted by extended operation.		No gap
29	FAI 4.2.1 A plan and schedule for the development of improved exercise program. December 2012.	As outlined in Reference [4], the plan and schedule for an improved Emergency Preparedness exercise program was provided and accepted. FAI 4.2.1 is closed. This conclusion is not impacted by extended operation.		No gap
30	FAI 5.1.1 An evaluation of the adequacy of backup power for emergency facilities and equipment.	As outlined in Reference [4], providing a source of backup power to emergency facilities and equipment, including telecommunications and redundant emergency telecommunications, was an industry initiative based on initial Fukushima Operating Experience. OPG addressed this with a comprehensive review of its overall capability in 2011. At the CNSC's		No gap

#	Fukushima Action Item [3]	Assessment of Impact of Operation Beyond 2020		Results
		Pickering Units 1,4	Pickering Units 5-8	
	December 2012.	request, additional details related to this assessment were provided. Based on current perspectives and discussion with the regulator, this included an expanded review and revisiting a 2011 backup power resolution at the Pickering Emergency Operations Center with consideration of Human and Organizational Performance. FAI 5.1.1 is closed. This conclusion is not impacted by extended operation.		
31	FAI 5.1.2 A plan and schedule to address any gaps identified. December 2012.	As outlined in Reference [4], detail on the evaluation of the adequacy of backup power for emergency facilities and equipment is presented under FAI 5.1.1 above. Back-up power for Pickering's emergency facilities and equipment was deemed adequate. FAI 5.1.2 is closed. This conclusion is not impacted by extended operation.		No gap
32	FAI 5.2.1 Identify the external support and resources that may be required during an emergency. December 2012.	As outlined in Reference [4], FAIs 5.2.1, 5.2.2 and 5.2.3 were being managed as a common action for OPG. OPG has agreements in effect with relevant external agencies to support emergency response. As a post-Fukushima collaborative enhancement, OPG and four other major Canadian nuclear operators formalized a Mutual Aid Agreement. FAI 5.2.1 is closed. This conclusion is not impacted by extended operation.		No gap
33	FAI 5.2.2 Identify the external support and resource agreements that have been formalized and documented. December 2012.	As outlined in Reference [4], FAIs 5.2.1, 5.2.2 and 5.2.3 were being managed as a common action for OPG and actions have been completed. FAI 5.2.2 is closed. This conclusion is not impacted by extended operation.		No gap

#	Fukushima Action Item [3]	Assessment of Impact of Operation Beyond 2020		Results
		Pickering Units 1,4	Pickering Units 5-8	
34	FAI 5.2.3 Confirm if any undocumented arrangements can be formalized. December 2012.	As outlined in Reference [4], FAIs 5.2.1, 5.2.2 and 5.2.3 were being managed as a common action for OPG. FAI 5.2.3 is closed. This conclusion is not impacted by extended operation.		No gap
35	FAI 5.3.1 Provide a project plan and installation schedule. December 2012.	As outlined in Reference [4], OPG has installed an automated Near Boundary Gamma Monitoring System to coincide with the location of the Thermo-Luminescent Dosimeter (TLD) Near Boundary sites at the site boundary (nominally 1 km from the plant) at Pickering. These monitors are solar powered with an 8-hour battery backup and provide immediate information on dose rates at the site boundary to the plant. FAI 5.3.1 is closed. This conclusion is not impacted by extended operation.		No gap
36	FAI 5.4.1 Develop source term and dose modeling tools specific to each NPP. December 2012.	Not Applicable [3]. OPG has these measures already in place and therefore, this action does not apply to OPG [4]. This conclusion is not impacted by extended operation.		No gap

5.0 RESULTS AND CONCLUSIONS

Based on the assessment in Table 1, OPG's responses to the FAIs assigned to Pickering NGS remain valid in the context of extended operations beyond 2020. There are no additional gaps identified that must be considered in the PSR2 Global Assessment based on this review.

6.0 ACRONYMS

BDBA	Beyond Design Basis Accident
BDBE	Beyond Design Basis Event
CNSC	Canadian Nuclear Safety Commission
CV	Calandria Vault
EME	Emergency Mitigating Equipment
ERP	Emergency Response Projection
FAI	Fukushima Action Item
IFB	Irradiated Fuel Bay
NGS	Nuclear Generating Station
OPG	Ontario Power Generation
PAR	Passive Autocatalytic Recombiners
PRA	Probabilistic Risk Assessment
PSR2	Pickering Periodic Safety Review
SAMG	Severe Accident Management Guidance
SCDF	Severe Core Damage Frequency

7.0 REFERENCES

[1] OPG Report, P-REP-03680-00001 R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 29, 2016.

[2] CNSC letter, P-CORR-00531-04786, H. Khouaja to B. McGee, *Pickering NGS: CNSC Staff Assessment of 2015 COP, SOP, SAP and CALs*, July 6, 2016.

[3] CNSC letter, N-CORR-00531-05607, G. Rzentkowski to W. M. Elliott, *Opening of Fukushima Action Items (FAIs) on Ontario Power Generation*, February 17, 2012.

[4] OPG Report, N-REP-03600-10003 R007, *Fukushima Action Item Status Report*, November 27, 2015.

[5] OPG Memorandum, P-CORR-03680-0586481, *Pickering NGS Extended Operations - Reassessment of Fukushima Action Item 1.3.1 & 1.3.2 - Containment Integrity*, November 30, 2016.

[6] OPG Report, P-REP-09013-00002 R001, *Pickering NGS - Beyond Design Basis Containment Integrity*, January 27, 2014.

[7] OPG Letter, N-CORR-00531-06963 R000, W. S. Woods to M. Santini and F. Rinfret, *Request for Closure of Fukushima-Related Action Item 2013-OPG-4286: Multi-Unit Severe Accident Modelling,* July 21, 2015.

[8] OPG Memorandum, P-CORR-03680-0586482, *Pickering NGS Extended Operations - Reassessment of Fukushima Action Item 2.2.1 - Deterministic Safety Analysis RD-310 Implementation Plan*, December 13, 2016.

[9] OPG Letter, P-CORR-00531-04672, B. McGee to M. Santini, *Pickering NGS: Risk Improvement Plan Update*, February 26, 2016.

[10] OPG Plan, P-SPM-34200-0453336, System Performance Monitoring Plan: PNGS 1-8 Reactor Building Negative Pressure Containment, February 4, 2016.

Directorate of Power Reactor Regulation

e-Doc 5461487
File 4.01.03
RIB 11979

February 19, 2018

Mr. Randy Lockwood
Senior Vice President
Pickering Nuclear
Ontario Power Generation Inc.
1675 Montgomery Park Road, P41, E3
Pickering, ON L1V 2R5

Subject: Pickering NGS: Periodic Safety Review 2 - CNSC Staff Review of OPG Global Assessment Report (GAR), Revision 1

Dear Mr. Lockwood:

Canadian Nuclear Safety Commission (CNSC) staff have reviewed OPG Global Assessment Report (GAR) Revision 1 [1] produced under Pickering Periodic Safety Review 2 (PSR2).

CNSC staff determined that this Revision 1 of the GAR satisfies the regulatory requirements of CNSC REGDOC-2.3.3, reflects the work performed under Pickering PSR2, and includes satisfactory disposition by OPG of CNSC staff comments in [2].

Should you have any queries, please contact Dr. Al Omar at al.omar@canada.ca or at 613-995-0565.

Yours truly,

Alexandre Viktorov, Ph.D.
Regulatory Program Director
Pickering Regulatory Program Division

c.c.: Pickering RPD
P. Herrera, R. MacEacheron (OPG)

References:

[1] OPG letter, R. Lockwood to A. Viktorov, "Pickering NGS - Periodic Safety Review 2 - Submission of the Global Assessment Report Revision 1", February 12, 2018, CD# P-CORR-00531-05292, e-Doc 5460376. (Enclosure 1: Pickering NGS PSR2 Global Assessment report, P-REP-03680-00032-R001.)

[2] CNSC Letter, A. Viktorov to R. Lockwood "Pickering NGS Periodic Safety Review 2- CNSC Staff Review of OPG Global Assessment Report (GAR)", January 29, 2018, CD# P-CORR-00531-05291, e-Doc 5441553.

ONTARIOPOWER
GENERATION

1675 Montgomery Park Road, P.O. Box 160, Pickering, Ontario L1V 2R5

OPG PROPRIETARY

February 12, 2018

CD# P-CORR-00531-05292

DR. A. VIKTOROV
Director
Pickering Regulatory Program Division

Canadian Nuclear Safety Commission
280 Slater Street
Ottawa, Ontario
K1P 5S9

Dear Dr. Viktorov:

Pickering NGS - Periodic Safety Review 2 – Submission of Global Assessment Report Revision 1

The purpose of this letter is to submit the Pickering Periodic Safety Review 2 (PSR2) Global Assessment Report Revision 1, P-REP-03680-00032-R001 (Enclosure 1).

This Global Assessment Report presents the results of the Pickering NGS PSR2 Global Assessment, completed in support of extended operation of Pickering NGS. The Pickering PSR2 has been conducted in accordance with the PSR2 Basis Document (Reference 1) that was accepted by the CNSC (Reference 2) and is consistent with CNSC REGDOC-2.3.3.

This revision to the Global Assessment Report includes minor enhancements to the earlier version (Revision 0) submitted for CNSC review (Reference 3) and incorporates comments received from the CNSC (Reference 4) as indicated in Attachment 1.

The Global Assessment Report consolidated findings identified from the Safety Factor review phase of the PSR2 project into Global Issues which were prioritized, assessed, and proposed Resolution Plans developed. The proposed Resolution Plans were supported by specific actions documented in the Integrated Implementation Plan that was submitted for CNSC staff acceptance in November 2017 (Reference 5).

The Global Assessment concludes that the Pickering NGS design, operation, processes and management system will ensure continued safe operation of Units 1,4 and 5-8, both in the short term, and for extended operation. Ontario Power Generation and the Pickering Station Leadership Team are committed to investing in the plant, and focusing the organization to strive for continued improvement in the plant condition, operation and performance.

If you have any questions, please contact Paulina Herrera, Manager, Pickering Regulatory Affairs at 905-839-1151 extension 3235.

Randy Lockwood
Senior Vice President
Pickering Nuclear

cc: CNSC Site Office – Pickering
CNSC Pickering Regulatory Program Division (copy to each staff)

References:

1. OPG Letter, B. McGee to H. Khouaja, "Submission of Pickering NGS Periodic Safety Review 2 Basis Document Revision 002", July 6, 2016, CD# P-CORR-00531-04780

2. CNSC Letter, H. Khouaja to B. McGee, "Pickering NGS: CNSC Staff Acceptance of Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document", July 8, 2016, e-Doc 5037314, CD# P-CORR-00531-04789

3. OPG Letter, R. Lockwood to A. Viktorov, "Pickering NGS Periodic Safety Review 2 – Submission of Global Assessment Report" October 30, 2017, CD# P-CORR-00531-05084.

4. CNSC Letter, A. Viktorov to R. Lockwood "Pickering NGS Periodic Safety Review 2 – CNSC Staff Review of OPG Global Assessment Report (GAR)", January 29, 2018, e-Doc 5441553, CD# P-CORR-00531-05291.

5. OPG Letter, R. Lockwood to A. Viktorov, "Pickering NGS Periodic Safety Review 2 – Submission of Integrated Implementation Plan", November 30, 2017, CD# P-CORR-00531-05085.

Attachments:

1. "OPG's Responses to CNSC Staff Review of Global Assessment Report P-REP-03680-00032 Revision 0"

Enclosures:

1. "Pickering NGS PSR2 Global Assessment Report"

Attachment 1

OPG's Responses to CNSC Staff Review of Global Assessment Report P-REP-03680-00032 Revision 0

The following table documents OPG's responses to the CNSC staff review of P-REP-03680-00032 R0 received in Reference A-1.

Table A-1: OPG's Responses to CNSC Staff Review of Global Assessment Report Revision 0

	Section # / Paragraph #	State the Issue	CNSC Staff Comment and Recommendation(s)	OPG Response
1.	SF6-1, SF6-2 (GI-27)	OPG links the resolution of GI-37 and GI-40 to the resolution of GI-27.	GI-37 and GI-40 are mainly design issues. GI-27 focuses on the plant risk improvement items given that all other plant design requirements (SF1 issues) would have been met or resolved. Although the final PSA results need to reflect the design and operation of the plant, meeting safety goals are not used to justify if the design requirements for containment systems can be relaxed. CNSC Staff request that OPG de-link the resolution of GI-27 (PSA modeling and identification of risk improvements) to the design requirements elements of GI-37 and GI-40. Even though the safety goals are met in GI-27, OPG remains obligated to meet the design requirements for the containment.	**Complete** The cross reference between GI-27 and GI-37 has been removed and enhanced wording added to GI-37 and GI-40 to better explain the relationship of how GI-37 is addressed by GI-40, and how these relate to the Risk Reduction activities in GI-27.
2.	Page C-18 GI-12: Extending the Env.	Paragraph 2 states "Regarding deterministic considerations, the Safety Significance	PSR2 Basis Document, e-docs# 5037878, page 63, Table E1, first column, third row, states: "A (safety) function is affected by the issue but the effect does not	**Complete** Both criteria are applicable; the second paragraph has been revised as shown below

Attachment 1 (Page 2 of 4) to OPG Letter, R. Lockwood to A. Viktorov, "Pickering NGS Periodic Safety Review 2 – Submission of Global Assessment Report Revision 1", CD# P-CORR-00531-05292.

	Section # / Paragraph #	State the Issue	CNSC Staff Comment and Recommendation(s)	OPG Response
	Qualification of Equipment Paragraph 2	Level determined from Table E1 in the PSR2 Basis Document is 3, since ensuring Environmental Qualification for extended operation will ensure the capability of safety provisions to effectively terminate an initiating event (first column, third row in Table E1 in the PSR2 Basis Document)."	impair the capability of safety provisions to terminate an anticipated serious process failure." OPG is required to explain in the GAR: 1) Why the Safety Significance Level was determined from Table E1 based on criteria impact on protection against process failures (first column) and not against design basis accidents (DBA) (second column), since EQ is applicable only to DBA. 2) How the following two cited statements are compatible: - OPG draft GAR, page C-18 'will ensure the capability of safety provisions to effectively terminate an initiating event' - PSR2 Basis Document, page 63, Table E1, first column, third row 'does not impair the capability of safety provisions to terminate an anticipated serious process failure.' If EQ for extended operation will ensure the capability of safety provisions to effectively terminate an initiating event, does it mean that lack of EQ for extended operation does not impair	to include the two criteria. Regarding the second part of the comment, the GI is related to a confirmatory activity, i.e., to confirm that Environmentally Qualified (EQ) equipment will continue to be EQ'd for the extended operating period. It is not about lack of EQ for any component. If the GI was intended to qualify previously unqualified equipment, then it might have been appropriate to select a higher Safety Significance Level. *"Regarding deterministic considerations, the Safety Significance Level determined from Table E1 in the PSR2 Basis Document is 3. This is because ensuring Environmental Qualification for extended operation will ensure the capability of safety provisions to effectively terminate an initiating event (first column, third row in Table E1 in the PSR2 Basis Document) and the issue does not affect the safety function capability for more than one level of protection (second*

Attachment 1 (Page 3 of 4) to OPG Letter, R. Lockwood to A. Viktorov, “Pickering NGS Periodic Safety Review 2 – Submission of Global Assessment Report Revision 1”, CD# P-CORR-00531-05292.

	Section # / Paragraph #	State the Issue	CNSC Staff Comment and Recommendation(s)	OPG Response
			the capability of safety provisions to terminate an anticipated serious process failure?	*column, third row in Table E1 in the PSR2 Basis Document). Safety Significance Levels (E2) is considered not applicable, since this Global Issue has a direct nuclear safety impact, whereas E2 primarily relates to issues that impact other objectives or are indirectly related to nuclear safety. The overall Safety Significance Level for deterministic considerations is 3.”*
3.	Page D-61, Appendix D, Section D-9	OPG has referenced “Section 5.0 of OPG Report [N-RPP-03415.1-10001-R07, *Radiation Protection Requirements - Nuclear Facilities*, June 25, 2001] in support of D-188 – Rad. Protection in Design. This document is obsolete and has been superseded by N-PROG-RA-0013, *Radiation Protection*. OPG should ensure they reference the correct document.	OPG should ensure they provide in the GAR the correct reference to address the design features captured on page 61, paragraph 2.	**Complete** Wording modified to include the superseding reference.

Attachment 1 (Page 4 of 4) to OPG Letter, R. Lockwood to A. Viktorov, "Pickering NGS Periodic Safety Review 2 – Submission of Global Assessment Report Revision 1", CD# P-CORR-00531-05292.

	Section # / Paragraph #	State the Issue	CNSC Staff Comment and Recommendation(s)	OPG Response
4.	Page D-61 Appendix D, Section D-9	OPG has provided the following information in support of quarterly reports submitted by OPG: *"Worker radiation dose due to events described in S-99 clauses 6.3.1 (7) or 6.3.1 (8) and the collective radiation dose of each group of workers."* This reference is obsolete and should be updated.	OPG should ensure they provide in the GAR the correct reference to address in their discussion of quarterly reports captured on page 61, paragraph 2.	**Complete** Wording modified to include the superseding reference.
5.	Page B-185	GI-50-RS1: "In addition, revise the Fuel Channel inspection and surveillance requirements to align with CSA N285.4-14"	OPG should remove this last sentence from the Resolution Statement since the Fuel Channel subject related to CSA N285.4-14 has its own Resolution Statement in GI-1.	**Complete** Statement related to fuel channels removed from GI-50-RS1 as it is already addressed in GI-1.

Reference A-1: CNSC Letter, A. Viktorov to R. Lockwood, "Pickering NGS Periodic Safety Review 2 – CNSC Staff Review of OPG Global Assessment Report (GAR)", January 29, 2018, e-Doc 5441553, CD# P-CORR-00531-05291.

Enclosure 1 to OPG Letter, R. Lockwood to A. Viktorov, "Pickering NGS Periodic Safety Review 2 – Submission of Global Assessment Report Revision 1", CD# P-CORR-00531-05292.

Enclosure 1

Pickering NGS PSR2 Global Assessment Report

P-REP-03680-00032-R001

(600 pages including this coversheet)

KINECTRICS QUALITY PROGRAM FORM QF-05-10 R1304

candesco
Division of Kinectrics Inc.

Title: Pickering NGS Global Assessment Report

File: K-421417-00035-R04

ONTARIO POWER GENERATION	
ACCEPTED	✓
ACCEPTED AS NOTED	
REVISE AND RESUBMIT	
Signature: [signature] Date: 8 Feb 2018	
Name: Mike Ruffolo, Manager	
Dept: Pickering Engineering - Aging Management & Strategic Initiatives	
OPG Proprietary	
Doc No.: P-REP-03680-00032 Rev: 001	
This acceptance does not relieve the contractor from responsibility for errors or omissions or from any obligations or liability under this contract.	

A Report Submitted to Ontario Power Generation

February 2018

<table>
<tr><td rowspan="2">Issue
R00</td><td colspan="5">Reason for Issue:
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R01</td><td colspan="5">Reason for Issue:
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<table>
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R02</td><td colspan="5">Reason for Issue:
Revised to address OPG comments on R01 and issued for use.</td></tr>
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R03</td><td colspan="5">Reason for Issue:
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Issue	Reason for Issue:				
R04	For use				
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Executive Summary

This Global Assessment Report presents the results of the Pickering Nuclear Generating Station (NGS) Periodic Safety Review 2 (PSR2) Global Assessment, completed in support of extended operation of Pickering NGS. PSR2 has been conducted in accordance with the PSR2 Basis Document [1] and is consistent with Canadian Nuclear Safety Commission (CNSC) REGDOC-2.3.3 [2].

The current planning basis for Pickering NGS is operation of Pickering NGS Units 1,4 and 5-8 until the end of 2024. To align with the anticipated expiry date of the next Power Reactor Operating Licence (PROL), for the purposes of PSR2 only the period of extended operation of Pickering NGS units is considered to the end of 2028.

The Global Assessment concludes that the Pickering NGS design, operation, processes and management system will ensure continued safe operation of Units 1,4 and 5-8, both in the short term, and for extended operation. Ontario Power Generation (OPG) and the Pickering Station Leadership Team are committed to investing in the plant, and focusing the organization to strive for continued improvement in the plant condition, operation and performance. Pickering NGS units will be operated only if fitness for service of the structures, systems and components (SSCs) important to safety is assured.

In the Global Assessment, Gaps identified in the Safety Factor review phase are consolidated into Global Issues. The Global Issues are prioritized and assessed, and Resolution Plans are proposed. The proposed Resolution Plans will be supported by specific actions in the Integrated Implementation Plan.

The Global Assessment demonstrates that Pickering NGS will operate safely during the extended operating period. In addition, activities are in progress and planned that will further enhance safe plant operation. The justification for this conclusion is based on the following:

- The Pickering Station Leadership Team has effectively aligned the organization to significantly improve performance in a number of key focus areas. Station performance improvement has been recognized through industry reviews. The plant is safe, and is operated safely.
- OPG has comprehensive programs in place to ensure the condition of SSCs important to safety at Pickering Units 1,4 and 5-8 is well understood, to assess the level of fitness for service, and to effectively take action to maintain good plant condition. This has led to continuous improvement in the condition of the plant, and plant performance.
- OPG has made significant improvements to the Pickering NGS plant design and processes. The plant design enhancements, together with the process enhancements, closely align the plant with safety-significant requirements of modern codes and standards (which in some cases are beyond current requirements), and enhance defence-in-depth. In particular, enhancements made in response to the 2011 Fukushima accident have reduced, and will further reduce, the risk associated with Beyond Design Basis Accidents (BDBAs).

- Design and operation of the plant meet the current deterministic safety analysis dose limits, and processes are in place to ensure the safety analysis accounts for any additional aging effects associated with extended operation. The Probabilistic Safety Assessment (PSA) shows that the OPG risk-based Safety Goals for Core Damage Frequency and Large Release Frequency are met. Initiatives have been proposed to further enhance the margins to these goals.
- Radiological dose performance and environmental impact performance are significantly better than regulatory limits. Programs in place ensure the ongoing effectiveness of the radiological protection of workers, the public and the environment.
- The Global Assessment identifies 24 Strengths, indicating that Pickering NGS is well aligned with modern codes, standards and practices in key areas.
- The Global Assessment identifies 51 Global Issues. Resolution plans for Global Issues have been developed, and most are in progress to further enhance safety, including enhancements to further reduce the risk associated with BDBAs. Many of the Global Issue Resolution Plan actions reflect existing work programs and plans at the station. In particular, for the Global Issues of highest safety significance (i.e., fitness for service to cover the extended operating period), OPG was already fully aware of these issues and is actively working on addressing them for the extended operating period. None of the Global Issues identify a safety concern that requires additional planned or urgent action to be taken.
- The Global Issues of highest safety significance pertain to fitness for service of SSCs important to safety over the extended operating period. Units will be operated only if fitness for service of SSCs important to safety is assured. OPG has comprehensive programs in place to ensure the condition of SSCs important to safety is well understood, to assess the level of fitness for service, and to effectively take action to maintain good plant condition. The Resolution Plans for these Global Issues will ensure the ongoing fitness for service of SSCs for the operational life of the plant, and these plans are actively being progressed.
- The Global Assessment includes a Resolution Plan that proposes the investigation and implementation of design, operational, and/or analytical options to further enhance margins to risk-based Administrative Safety Goals.
- The assessment of Acceptable Deviations confirms there is no impact on the conclusion of the Global Assessment, either individually or in aggregate.
- The Defence-in-Depth Assessment shows that Pickering Units 1,4 and Pickering Units 5-8 design and operation have adequate and effective barriers in all levels of defence-in-depth.
- OPG’s organizational structure and management system provides the requisite processes, tools, resources and oversight that will ensure continued safe operation of the plant.

As noted, the Global Assessment concludes that the Pickering NGS design, operation, processes and management system will ensure continued safe operation of Units 1,4 and 5-8, both in the short term, and for extended operation.

Acronyms and Abbreviations

AD	Acceptable Deviation
AECL	Atomic Energy of Canada Limited (now Canadian Nuclear Laboratories)
AG	Additional Gap
ALARA	As Low As Reasonably Achievable
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
BDBA	Beyond Design Basis Accident
CANDU	CANada Deuterium Uranium
CCS	Concrete Containment Structure
CNSC	Canadian Nuclear Safety Commission
COG	CANDU Owners Group
COP	Continued Operations Plan
CRN	Canadian Registration Number
CSA	Canadian Standards Association
DBA	Design Basis Accident
ECI	Emergency Coolant Injection
EFPH	Effective Full Power Hours
EME	Emergency Mitigating Equipment
EMS	Energy Management System
EQ	Environmental Qualification
FADS	Filtered Air Discharge System
FAI	Fukushima Action Item
FFS	Fitness for Service
GI	Global Issue
HFE	Human Factors Engineering
HTS	Heat Transport System
IAEA	International Atomic Energy Agency
IAM	Integrated Aging Management
IAMP	Integrated Aging Management Program
IEC	International Electrotechnical Commission
IFB	Irradiated Fuel Bay

IIP	Integrated Implementation Plan
INPO	Institute of Nuclear Power Operations
ISO	International Standards Organization
ISR	Integrated Safety Review
KI	Potassium Iodide
LBLOCA	Large Break Loss of Coolant Accident
LCMP	Life Cycle Management Plan
LOCA	Loss of Coolant Accident
LOF	Loss of Flow
MCR	Main Control Room
NFA	No Further Action
NFPA	National Fire Protection Association
NGS	Nuclear Generating Station
NOP	Neutron Overpower Protection
NPP	Nuclear Power Plant
OPEX	Operating Experience
OPG	Ontario Power Generation
OSART	Operational Safety Review Team
PARTS	Pickering A Return to Service
PHT	Primary Heat Transport
PIP	Periodic Inspection Program
PROL	Power Reactor Operating Licence
PSA	Probabilistic Safety Assessment
PSR	Periodic Safety Review
PSR1	Periodic Safety Review 1
PSR2	Periodic Safety Review 2
RMS	Radionuclide Management System
RS	Resolution Statement
SBLOCA	Small Break Loss of Coolant Accident
SDS1	Shutdown System No. 1
SDS2	Shutdown System No. 2
SDSA	Shutdown System A (Original Shutdown System)
SDSE	Shutdown System Enhancement
SOE	Safe Operating Envelope

SSCs	Structures, Systems and Components
WANO	World Association of Nuclear Operators
XRF	Cross-Reference

Part I: Methodology

1. Purpose

OPG is conducting a Periodic Safety Review (PSR) in support of extended operation of Pickering NGS [1]. This work is a subsequent PSR for Pickering NGS, and is designated as PSR2. The work is being performed consistent with the requirements of CNSC REGDOC-2.3.3 *Periodic Safety Reviews* [2], which incorporates the requirements of International Atomic Energy Agency (IAEA) SSG-25 [3]. PSR2 is being conducted following the processes and schedule identified in the OPG-CNSC Protocol developed in support of the PSR2 execution interface [4].

The objective of the Global Assessment is to provide an overall assessment of the safety of the plant, and to assess the acceptability of Pickering NGS for continued operation over the PSR2 period, including an assessment of the defence-in-depth capability of Pickering NGS.

As noted in the PSR2 Basis Document [1], the current planning basis for Pickering NGS is that Pickering NGS units will operate until the end of 2024. To align with the anticipated expiry date of the next PROL, for the purposes of PSR2 only the period of extended operation of Pickering NGS units is considered to the end of 2028.

2. Background

OPG owns and is licensed to operate the Pickering NGS, which is located on the shore of Lake Ontario in the City of Pickering, in the Regional Municipality of Durham. The station has a total of eight CANada Deuterium Uranium (CANDU) reactors (Units 1 to 8). Currently, Pickering NGS Units 1,4 and 5-8 are operating, while Units 2 and 3 are no longer operational and are in a safe storage and surveillance state.

In accordance with CNSC REGDOC-2.3.3 [2], the elements of PSR2 consist of the following four phases:

(i) Preparation of a PSR2 Basis Document [1]

(ii) Conduct of the Safety Factor reviews and identification of Compliances and Gaps

(iii) Analysis of the Gaps and identification of potential safety enhancements for Pickering NGS in the Global Assessment process

(iv) Preparation of a plan for the implementation of safety enhancements (Integrated Implementation Plan)

The Pickering NGS PSR2 Basis Document [1] describes the scope and methodology for PSR2. PSR2 is a subsequent PSR; an update building on the review basis of earlier OPG PSR work and other associated assessments (termed here "PSR1").

Specifically, PSR1 consists of:

- The Pickering B Integrated Safety Review (ISR), completed in 2009 and performed in support of refurbishment and continued operation (at that time planned for another 30 years) of Pickering NGS Units 5-8.
- Pickering 1,4 integrated safety assessments performed during the Pickering A Return to Service (PARTS) work (circa 2000), in support of approval to restart Units 1 and 4.
- The relevant programmatic aspects of the Darlington ISR completed in 2011 in support of refurbishment and continued operation of the Darlington units (programmatic parts are applicable to Pickering where programs and practices are common for the OPG fleet).

The second phase of the PSR2 process involves completion of Safety Factor reviews. Safety Factors cover all aspects important to the safety of an operating Nuclear Power Plant (NPP). As identified in the PSR2 Basis Document [1], there are 15 Safety Factors used in the PSR2 review. These 15 Safety Factors are shown in Table 1.

Table 1: PSR2 Safety Factors

Subject Area	Safety Factor	
The Plant	SF1	Plant Design
	SF2	Actual Condition of Structures, Systems and Components Important to Safety
	SF3	Equipment Qualification (environmental and seismic)
	SF4	Aging
Safety Analysis	SF5	Deterministic Safety Analysis
	SF6	Probabilistic Safety Assessment
	SF7	Hazard Analysis
Performance and Feedback from Operating Experience	SF8	Safety Performance
	SF9	Use of Experience from other NPPs and Research Findings
Management	SF10	Organization, the Management System and Safety Culture
	SF11	Procedures
	SF12	Human Factors
	SF13	Emergency Planning
Environment	SF14	Radiological Impact on the Environment
Radiation Protection	SF15	Radiation Protection

The results of the Safety Factor reviews are documented in Safety Factor Reports. The Safety Factor Reports address the Review Tasks derived from IAEA SSG-25 for Safety Factors 1 to 14 and from CNSC REGDOC-2.3.3 for Safety Factor 15. The Safety Factor Reports also document the results of the assessments of Pickering NGS with respect to applicable modern Laws, Regulations, Codes and Standards, as listed in Appendix D of the PSR2 Basis Document [1], and OPG Program effectiveness reviews. Complementary Reviews, such as review of the Pickering B ISR Continued Operations Plan (COP) actions [5] and the Fukushima Action Items (FAIs) [6] were also completed in the Safety Factor phase.

The third phase of the PSR2 process, i.e., the Global Assessment, is documented in this Pickering NGS Global Assessment Report. The Global Assessment takes into account the Gaps identified during the Safety Factor reviews, the findings from Complementary Reviews (COP and FAI), and any findings from CNSC staff reviews of the Safety Factor Reports and Complementary Reviews. The Global Assessment includes consideration of the five levels of defence-in-depth, including consideration of Strengths, the enhancements proposed[1] through Global Issue Resolution Plans, and assessment of the aggregate impact of Acceptable Deviations, in order to make a conclusion on the overall acceptability of operation of the plant over the period considered in PSR2.

Preparation of the Integrated Implementation Plan, which is the fourth phase of the PSR2 process, involves transforming the proposed Resolution Plans resulting from the Global Assessment into actions with corresponding schedules for implementation during the next licensing period.

[1] At the Global Assessment stage, enhancements are identified as "proposed" through Global Issue resolutions. Enhancements are finalized at the Integrated Implementation Plan stage of the PSR2 process.

3. Overview of the Global Assessment Process

The major inputs to the Global Assessment are:

- Safety Factor Reports

  Gaps are listed in the Safety Factor Reports, based on the Review Tasks and assessments against modern Laws, Regulations, Codes and Standards.

- Complementary Reviews (FAIs, Pickering B ISR COP)

The above information is used to commence the Global Assessment process, which is based on the process outlined in the PSR2 Basis Document [1], and consists of the following elements:

1. Identification of Gaps from the Safety Factor Reports, Complementary Reviews (FAIs, Pickering B ISR COP), and any additional gaps that may be identified during the Global Assessment. Gaps arising from CNSC feedback on the PSR2 deliverables are also considered.
2. Development of Global Issues by integrating and consolidating the Gaps. Each Global Issue is comprised of Gaps related to a common theme.
3. Assessment of interfaces between the various Safety Factors, aggregate impact of Global Issues.
4. Prioritization of Global Issues.
5. Development of proposed Resolution Plans with consideration of safety benefit, practicability, and the interfaces between the Gaps and Global Issues.
6. Assessment of Defence-in-Depth, including Strengths, and the aggregate impact of Acceptable Deviations. It is noted that the PSR2 Basis Document also refers to residual Global Issues, but none were identified during the Global Assessment process.
7. Ranking of Global Issue Resolution Statements.
8. OPG senior management review of proposed Resolution Statements.
9. Assessment of overall acceptability of operation of the plant over the period considered in PSR2.
10. Preparation of the Global Assessment Report to summarize the assessments and document the Global Assessment.

Each of the above steps is described in more detail in Section 5 of this report. A pictorial representation of the Global Assessment Process is shown in Figure 1.

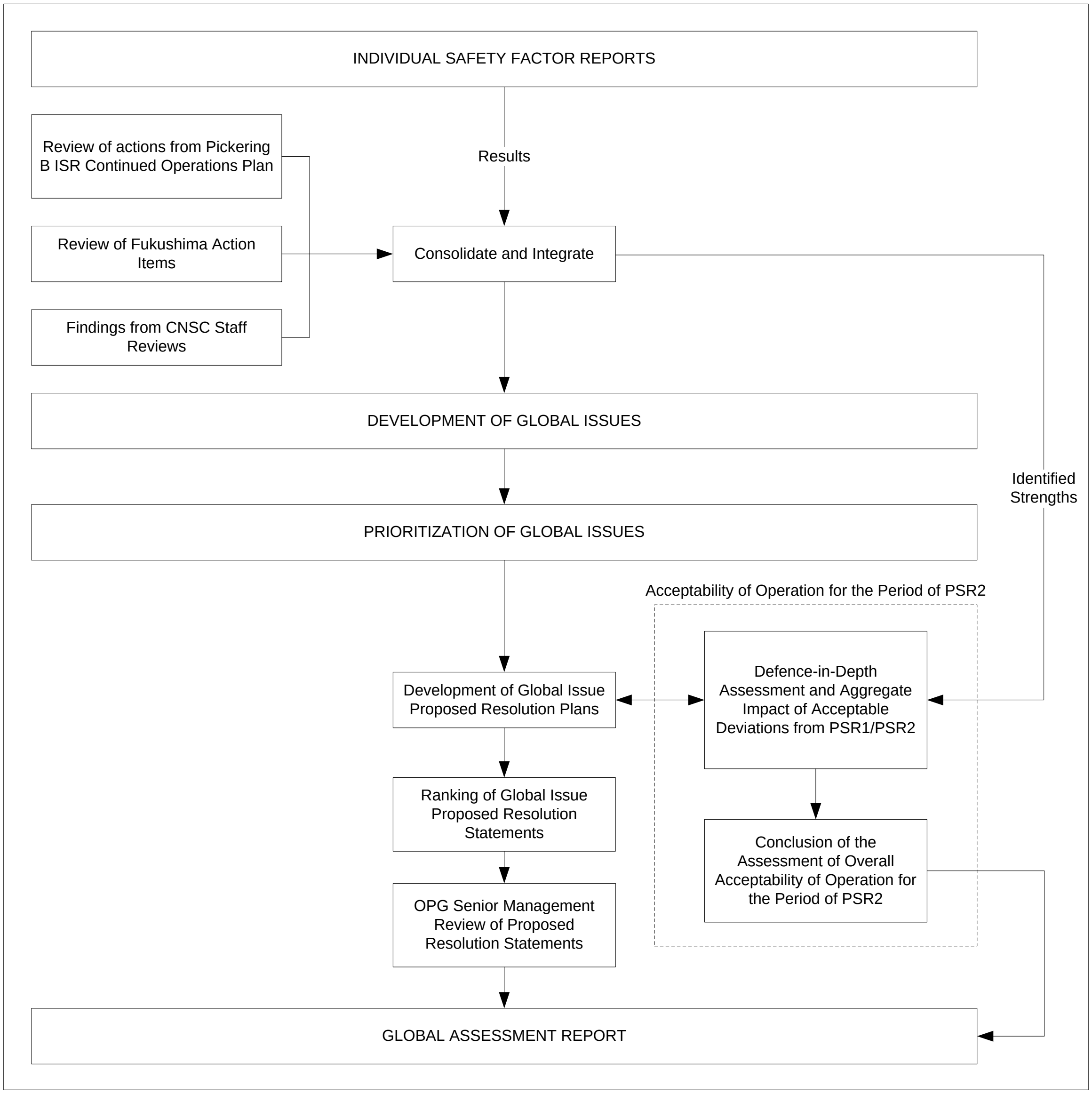

Figure 1: Pickering NGS Global Assessment Process Flowchart

4. Organization of the Report

This report is organized around the work done in each of the steps shown in Figure 1.

Part I: Methodology

Sections 1 and 2 describe the purpose and background of the Global Assessment. Sections 3 and 4 provide an overview of the Global Assessment process and the organization of this report, respectively. Section 5 describes the steps of the overall Global Assessment methodology.

Part II: Safety Factor and Complementary Review Summary

Sections 6 to 9 summarize the Gaps identified in the Safety Factor Reports, as well as the findings from the Complementary Reviews of the Pickering B ISR COP and FAIs, and CNSC feedback on the PSR2 Safety Factor Reports and Complementary Reviews.

Part III: Global Assessment

Section 10 summarizes the numbers of PSR2 Gaps from the various sources described in Part II of the Global Assessment Report. Section 11 groups the PSR2 Gaps into Global Issues, based on their topical similarities, and presents an assessment of the interfaces between the various Safety Factors and the aggregate impact of the Global Issues. Section 12 prioritizes the Global Issues, and Section 13 identifies proposed Resolution Plans for the Global Issues. Section 14 presents the results of the ranking of the proposed Global Issue Resolution Statements that will be considered in the Integrated Implementation Plan. The proposed Resolution Statements are ranked in order of the priority to resolve them based on the magnitude and timeliness of the benefit to be achieved by their resolution. Section 15 summarizes the findings resulting from a third party Expert Panel review of the Global Assessment.

Part IV: Assessment of Overall Acceptability of Operation of the Plant

Pickering NGS Strengths (Section 16) are evaluated for their contribution to the defence-in-depth capability of Pickering NGS (Section 18), which also considers the current Pickering NGS design and performance (Section 18.2) and the proposed Resolution Plans for Global Issues (Section 13). The Defence-in-Depth Assessment also considers the aggregate impact of the PSR1 Acceptable Deviations (Section 17), and PSR2 Acceptable Deviations (Appendix B).

Section 19 presents a justification for the overall acceptability of extended operation of the Pickering NGS.

Part V: References

Section 20 lists the references used in Parts I to IV of the Global Assessment Report.

Appendices

Appendices to the Global Assessment Report are presented at the end of the Global Assessment Report.

5. Global Assessment Methodology

This section describes the methodology for performing the Pickering NGS Global Assessment.

5.1. Identification of Gaps from the Safety Factor Reports and the Complementary Reviews

The first step in the Global Assessment methodology is the review and accumulation of the Gaps identified and described in each Safety Factor Report and other inputs to the Global Assessment (referred to as Complementary Reviews).

The following information for each Gap is also collected:

- Origin of Gap (i.e., Safety Factor Report Number, Complementary Review, other gaps)
- Gap identification number and title
- Associated Safety Factor review task (if applicable)
- Associated modern Laws, Regulations, Codes and Standards (if applicable)
- Relevant governing process or procedure (if applicable)
- Description of the Gap

5.2. Development of Global Issues

The second step in the Global Assessment methodology is the development of Global Issues. There are two aspects to this step, as follows:

- Consolidation of Gaps into Global Issues
- Assessment of the interfaces between the various Safety Factors and the aggregate impact of Global Issues

5.2.1. Consolidation of Gaps into Global Issues

Using the Gap information collected during the process described in Section 5.1, the Gaps are grouped into Global Issues according to their topical similarities, i.e., based on the related discipline, governing process or modern Laws, Regulations, Codes and Standards, with consideration of any interfaces or duplication between the Gaps. The following general steps are followed in consolidating the Gaps into Global Issues:

1. Gaps with clear similarity in themes or topical areas are consolidated into a specific Global Issue, e.g., governance issues where their resolution would require modification to OPG governance documentation. These Gaps are separated from those associated with the implementation or effectiveness of the governance, which are consolidated into a separate Global Issue identified as Governance Implementation/Effectiveness Issues.

The consolidation of Gaps also considers the expected differences between the level of importance of Gaps and their Resolution Plans, e.g., Gaps related to Major Components Fitness for Service (FFS) and Deaerator FFS are identified as separate Global Issues.

2. Any Gap against modern Laws, Regulations, Codes and Standard that is relevant to the theme or subject of an existing Global Issue is appropriately consolidated within the Global Issue.
3. The remaining Gaps that have not been consolidated into Global Issues are further re-examined to ensure their consideration as separate Global Issues is appropriate due to their specific nature.
4. If a Gap is found to be covered by another Gap within an existing Global Issue, it is clearly identified as such. If a Gap is found to be partly covered by an existing Global Issue, the coverage of the remaining part of the Gap is clearly specified in another Global Issue to ensure that the Gap is fully addressed.
5. A confirmatory step is completed to ensure that each Gap is appropriately consolidated into a Global Issue and potential interfaces between the Gaps from various Safety Factor Reports are accounted for.

Any additional Gaps identified during the Global Assessment work are appropriately integrated with the previously identified Gaps and Global Issues where applicable. By the end of this step, all Gaps are mapped to a Global Issue.

5.2.2. Assessment of Interfaces Between Various Safety Factors and Aggregate Impact of Global Issues

The third element of the Global Assessment process, as identified in Section 3.3.2 of the PSR2 Basis Document [1], is an assessment of “interfaces between the various Safety Factors, Aggregate Impact of Global Issues”. This step is carried out by first grouping Global Issues with the same or similar functional topic, independent of their origin from different Safety Factors or another source. For example, there are several Global Issues related to fitness for service, originating from Safety Factors 1, 2, 4 and the COP. These Global Issues are grouped together to form an aggregate group under the topic Fitness for Service. A similar approach is taken for the other Global Issues. In cases where a Global Issue is not related to any others, it is placed into a group with only a single Global Issue.

The second part of this element is completed after the Global Issue proposed Resolution Statements have been developed. Each Global Issue group is reviewed to determine if the interfaces among Global Issues originating from different Safety Factors or other sources warrants adjustment of proposed Resolution Plans for any Global Issue within the group. This review also considers whether the scope and nature of each Global Issue in a group, and the proposed Resolution Plan for each, would interact with or inhibit the successful resolution of any other Global Issues in the group. In either case, the aggregate effect of the Global Issues within the group could be more significant than the sum of the individual Global Issues, and revision of proposed Resolution Plans could be required.

The aggregate assessment considers the number of Global Issues in each aggregate group, the sources of associated Gaps, relevant OPG programs and the Global Issue Resolution Plans developed in Section 13.

5.3. Prioritization of Global Issues

The Global Issues developed in Section 5.2 are prioritized with respect to nuclear safety into one of four categories, based on their Safety Significance Level as described in Section 3.3.3 of the PSR2 Basis Document [1], and shown in Table 2.

Table 2: Safety Significance Level versus Impact on Nuclear Safety

Safety Significance Level	Impact on Nuclear Safety
1	High
2	Medium
3	Low
4	Very Low

The basis for prioritization of the Global Issues is provided in Appendices E and F of the PSR2 Basis Document [1], and includes deterministic and probabilistic considerations. The process is similar to that used in previous OPG ISRs and is used in PSR2 to prioritize Global Issues with respect to their importance to nuclear safety to determine the Safety Significance Level associated with each Global Issue. This supports the resolution evaluation and outcome of the resolution process.

For the purposes of Global Issue prioritization, and to be consistent with the PSR2 Basis Document [1], Table E1 of the PSR2 Basis Document is applied when the Global Issue may have a direct impact on the physical barriers associated with deterministic defence-in-depth considerations, or the other direct factors considered in the table, e.g., initiating events, safety culture. Table E2 is applied when the Global Issue may impact defence-in-depth indirectly or has no impact on defence-in-depth. A broader interpretation of defence-in-depth is applied in the Defence-in-Depth Assessment documented in Section 18 of this report, which considers Global Issues that affect defence-in-depth both directly and indirectly.

The outcome of the Global Issue Safety Significance Level assessment is documented in Section 4 of each Global Issue summary in Appendix B of this report. A Safety Significance Level between 1 and 4, or N/A, is assigned for each of the considerations in Appendix E and Appendix F of the PSR2 Basis Document [1]. N/A means that the Global Issue has no impact on the particular consideration, so the corresponding table in the PSR2 Basis Document is not applicable for the purposes of assessing the Safety Significance Level. The overall Safety

Significance Level for each Global Issue corresponds to the highest impact on nuclear safety (smallest Safety Significance Level number) of the individual considerations.

5.4. Development of Proposed Resolution Plans

Proposed Resolution Plans for the identified Global Issues are formulated with consideration of interfaces between the various Gaps to ensure that the proposed Resolution Plans complement each other. Proposed Resolution Plans are developed for all Global Issues, and consider safety benefit and practicability. Insights from available site Probabilistic Safety Assessments may be used in evaluating the benefit/practicality of potential options, where appropriate.

Proposed Resolution Plans may include proposed Resolution Statements which are actions defined to address a Gap. Proposed Resolution Statements are primarily proposed for Global Issues that have been prioritized with a Safety Significance Level of 1 or 2 (i.e., high or medium impact on nuclear safety), and for Global Issues with Safety Significance Level 3 if a practicable solution is readily evident.

Consistent with Section 3.3.3 of the PSR2 Basis Document [1], Resolution Statements are not proposed for all PSR2 Gaps. Gaps with Safety Significance Level 4 (i.e., very low impact on nuclear safety) are generally assessed as Acceptable Deviations. Gaps with Safety Significance Level 3 (i.e., low impact on nuclear safety) for which a practicable solution is not readily evident are also assessed as Acceptable Deviations. Acceptable Deviations are not tracked beyond the Global Assessment phase of PSR2 [1]. However, the impacts of Acceptable Deviations are considered in the Defence-in-Depth Assessment (refer to Section 5.7) to determine the aggregate impact on the defence-in-depth capability of the plant.

The PSR2 Basis Document [1] also describes the potential for residual Global Issues to be identified in the Global Assessment process. These would comprise aspects of a Global Issue for which a proposed Resolution Statement is not identified, nor is the issue an Acceptable Deviation or categorized as requiring No Further Action. However, no residual Global Issues were identified in the Global Assessment process, hence this topic is not discussed further in this report.

The proposed Global Issue Resolution Plans for each Global Issue are documented in Appendix B – Global Issues and Proposed Resolution Plans. These consist of the following statement types:

- Resolution Statements (RS): An activity is defined to address the Gap(s).
- No Further Action (NFA): Work is already completed or is underway outside of PSR2 to address the related Gap(s), or information has been found to obviate the Gap(s).
- Acceptable Deviation (AD): The Gap(s) have been assessed to have a Very Low Safety Significance Level, or are Low Safety Significance Level items and a practicable resolution is not readily evident.
- Cross Reference (XRF): An action that addresses the Gap(s) is covered by another Resolution Statement.

In conducting the Global Assessment, as Global Issue Resolution Plans are developed, consideration is given as to whether the resolution activities would be different for a scenario with operation to 2024 (the nominal planning basis for the units), or for operation beyond 2024. If the proposed Resolution Plan for a particular Global Issue may be dependent on whether plant operation is assumed to continue beyond 2024, the Global Issue Resolution Plan is flagged accordingly in Section 1 of each Global Issue Table in Appendix B, under a category entitled “Reassessment Beyond 2024”. In such cases where “Reassessment Beyond 2024” is identified as “Y”, detailed resolution activities are only identified to cover the period to 2024. If a decision is subsequently made to operate beyond 2024, the flagged Global Issue will need to be re- evaluated and updated actions will be identified for the proposed Global Issue Resolution Plan if/as appropriate.

To facilitate binning of potential work, proposed Resolution Plans are categorized as one or more of the following types of enhancements:

- Programmatic (changes to governing programs and procedures)
- Engineering (design changes)
- Analytical (engineering analysis, deterministic safety analysis, probabilistic safety assessment or hazard analysis)

The categorization is identified in Section 1 of each Global Issue Table in Appendix B. In some cases, the proposed Resolution Statements entail work in more than one of these categories.

5.5. Ranking of Resolution Statements

The purpose of ranking proposed Resolution Statements is to determine the activities that will be most effective in enhancing safety given the limited extended operating period of the plant. The ranking process recognizes that there are many factors that have to be taken into account in determining the rank of a specific proposed Resolution Statement, but a key consideration is that the ranking takes place within the context of the specific time period related to extended operation. Activities that will take a relatively long time to implement or to take effect may have relatively little practical benefit if the period of extended operation is short.

All Global Issue Resolution Statements with identified actions are ranked from 1 to N in decreasing importance such that 1 is the most important and N, which is the total number of Resolution Statements, is the least important. The ranking is determined through the application of a value-tree method for solving multi-attribute decision problems, as described in Appendix C. Acceptable Deviations and No Further Action statements do not go through the ranking process; only proposed Resolution Statements with identified actions are ranked.

The Value Tree for PSR2 ranking identifies an overall objective for PSR2 as “Enhanced confidence in the continued safe operation of PNGS for the period of PSR2”, with the following fundamental objectives that support this overall objective:

- Enhanced confidence that the design of SSCs supports modern safety practices
- Enhanced confidence in the fitness for service of SSCs

- Enhanced confidence in the Safety Analysis
- Enhanced confidence in safety performance and feedback of experience
- Enhanced confidence in the management processes
- Enhanced confidence in radiation and environmental protection

Using a pair-wise comparison, the above six fundamental objectives are ranked in terms of importance on a scale from 1 to 9.

A relative weight for each fundamental objective is then calculated and accordingly every proposed Resolution Statement is assigned the weight of the fundamental objective that is most relevant to it.

The ranking of each proposed Resolution Statement is based on the weight and a two-variable utility function that accounts for impact and time attributes. The impact attribute is a measure of how directly or strongly the issue impacts the objective while the time attribute accounts for how long it would take to implement and realize the associated objective. Each attribute is rated on a scale of 1 to 5.

The two-variable utility function is used to generate a utility matrix, and the time and impact ratings for each proposed Resolution Statement are used together with the utility matrix to obtain a numerical value that represents the utility score for resolving the proposed Resolution Statement. The Ranking Number of the proposed Resolution Statement is then calculated by multiplying its utility score by its weight.

As a final step, the rankings of the proposed Resolution Statements are confirmed based on engineering judgment applied by the Global Assessment team, and feedback from OPG staff and members of a third party Expert Panel (as described in Section 5.6.1). This considers factors such as the priority previously determined (Safety Significance Level), the contribution to defence-in-depth and the significance of the source (e.g., the type of document that generated the Gap(s) leading to the Global Issue). This also accounts for the extent of impact on multiple Safety Factors.

5.6. Review of Gaps, Global Issues and Proposed Resolution Plans

In addition to initial internal review by the Global Assessment Report preparation team, the Global Issues and proposed Resolution Statements and their rankings undergo a number of subsequent reviews during the Global Assessment process. These subsequent reviews are performed sequentially, as follows:

1. Third party Expert Panel review
2. Review by OPG's PSR2 project staff members and OPG Subject Matter Experts
3. Review and approval of proposed Resolution Statements by OPG's Senior Management Scope Review Board

The functions of the Expert Panel and the Senior Management Scope Review Board are described below.

5.6.1. Pickering PSR2 Expert Panel

The PSR2 Expert Panel is comprised of experienced individuals who are familiar with the design and operation of Pickering NGS (and other nuclear plants) and has been established to provide third party support to the PSR2 process. The Expert Panel has a broad mandate in terms of its activities, including providing third party guidance during the development and preparation of the Global Assessment Report. The framework for the Expert Panel activities is informed by, and is consistent with, the following documents:

- OPG P-REP-03680-00001, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document* [1]
- CNSC REGDOC-2.3.3, *Periodic Safety Reviews* [2]
- IAEA INSAG-10, *Defence in Depth in Nuclear Safety* [7]
- IAEA SRS-46, *Assessment of Defence in Depth for Nuclear Power Plants* [8]
- IAEA SSG-25, *Periodic Safety Review of Nuclear Power Plants* [3]
- IAEA INSAG-12, *Basic Safety Principles for Nuclear Power Plants* [9]

In particular, the Expert Panel activities include third party review of the development of the Global Issues, their prioritization and proposed Resolution Plans, the defence-in-depth review, ranking of proposed Resolution Statements, and the overall assessment of safety.

The objectives of the Expert Panel third party review of Global Issues and proposed Resolution Plans are:

- For each Global Issue, review and agree that the associated Gaps identified in the Global Issue are appropriate.
- For each Global Issue, provide agreement that the Resolution Plan is complete (i.e., resolves the Gaps to the extent practicable).
- For each Global Issue, concur with the Resolution Category (i.e., programmatic, engineering or analytical).
- Agree that Acceptable Deviations are clearly understood and have appropriate rationale.
- Concur that the prioritization of Global Issues is consistent across the Global Assessment.
- Identify additional Gaps, and provide any other pertinent feedback to the Global Assessment team and to OPG staff.

The objectives of the Expert Panel third party review of the Defence-in-Depth Assessment and Global Issue Resolution Statement ranking are:

- Review and comment on the overall defence-in-depth methodology and results.
- Review and comment on the Resolution Statement ranking methodology and results.

In addition to the above activities, the Expert Panel also provides a closing third party review of the Global Assessment Report, including the overall assessment of safety, and states whether the Panel concurs with the conclusions of the Global Assessment.

5.6.2. Senior Management Scope Review Board

The proposed enhancements identified in the PSR2 Global Assessment Report are presented to the OPG Senior Management Scope Review Board for their review and approval. This provides an opportunity for the senior management team to suggest clarifications, confirm the best available options are recommended, or propose changes to enhance safety where such improvements are identified. The Scope Review Board also reviews the PSR2 Acceptable Deviations and No Further Action statements.

This review ensures alignment with the Resolution Plans proposed, their basis and context, and is the means to obtain concurrence that the proposed enhancements are practicable and effective. This also allows the senior management team to consider potential realignment of priorities based on the insights from PSR2. Consistent with OPG Project Management processes, additional approvals are required as the resolution development continues towards full implementation.

The Terms of Reference for the Senior Management Scope Review Board is documented in Reference [10].

5.7. Defence-in-Depth Assessment

As part of the Global Assessment, a Defence-in-Depth Assessment is performed as summarized below and described in more detail in Section 18.

The Defence-in-Depth Assessment supports extended operation of Pickering NGS by demonstrating the extent to which the safety requirements of defence-in-depth are fulfilled.

For each level of defence, the assessment considers the overall plant, as well as the Strengths and the proposed Resolution Plans for the Global Issues identified in PSR2. The impacts of Acceptable Deviations are also assessed to determine the aggregate impact on the defence-in-depth capability of the plant.

Strengths for consideration in the Defence-in-Depth Assessment are identified based on:

- PSR2 Safety Factor reviews and Complementary Reviews.
- Other plant reviews and stakeholder input.
- CNSC Regulatory Oversight Report findings.

The defence-in-depth concept applied to the Global Assessment is consistent with IAEA INSAG-10, *Defence in Depth in Nuclear Safety* [7]. The assessment uses elements of the process described in IAEA SRS-46, *Assessment of Defence in Depth for Nuclear Power Plants* [8], and includes the following:

- Confirmation of the applicable safety principles from IAEA SRS-46 [8] for the defence-in-depth review.
- Establishment of the levels of defence-in-depth impacted for each applicable safety principle (taken from IAEA SRS-46 [8]).
- Mapping of each safety principle to the relevant Safety Factors.
- Assessment of the defence-in-depth aspects of each safety principle in the Pickering NGS design and operation at a high level.
- Consideration of proposed Global Issue Resolution Plans.
- Consideration of Strengths identified during the assessment.
- Consideration of the Acceptable Deviations from PSR1 that are applicable to PSR2.
- Consideration of the Acceptable Deviations identified during the conduct of PSR2.
- For each level of defence, an overall summary integrating the conclusions from the assessment of the related safety principles, Strengths, proposed Resolution Statements and assessment of the aggregate impact of Acceptable Deviations on the defence-in-depth capability of the plant.
- An overall defence-in-depth conclusion summarizing the multiple and overlapping provisions considered in the assessment which support the levels of defence.

5.8. Conclusion of the Assessment of Overall Acceptability of Plant Operation

This step assembles the results of the previous steps in the Global Assessment to assess the overall acceptability of extended operation of Pickering NGS over the designated PSR2 period, on the basis of a balanced view of all of the findings. The assessment includes all aspects of plant operation, and considers the proposed enhancements identified, the Strengths, the assessment of provisions for all levels of defence-in-depth, and the aggregate impact of Acceptable Deviations.

Part II: Safety Factor and Complementary Review Summary

Section	
#	**Title**
6	Review of Safety Factor Reports and Identification of Gaps
7	Review of the Pickering B ISR Continued Operations Plan
8	Review of Fukushima Action Items
9	Findings from CNSC Staff Reviews of Safety Factor Reports and Complementary Reviews

6. Review of Safety Factor Reports and Identification of Gaps

This section summarizes the results of each of the 15 PSR2 Safety Factor Reports and lists the Gaps identified from each Safety Factor Report. The Safety Factor Report summaries are grouped into six subject areas, as follows:

- The Plant
- Safety Analysis
- Performance and Feedback from Operating Experience
- Management
- Environment
- Radiation Protection

6.1. The Plant

6.1.1. SF1: Plant Design

6.1.1.1. Objective

The objective of the review of Safety Factor 1, Plant Design, is to determine the adequacy of the design of the NPP and its documentation by assessment against the current licensing basis and national and international standards, requirements and practices.

6.1.1.2. Scope of the Review

The PSR2 review of Safety Factor 1 contributes to confirming that the design, condition and operation of Pickering Units 1,4 and 5-8 (as well as common systems) will support continued safe operation for the period of PSR2 by:

1. Assessing compliance against the following Review Tasks identified in the PSR2 Basis Document [1], which were derived from IAEA SSG-25, *Periodic Safety Review of Nuclear Power Plants* [3] and CNSC REGDOC-2.3.3 [2]:

   RT1) Confirm that a detailed description of the plant design, documenting the Design Basis, supported by layout, system and equipment drawings exists.

   RT2) Assess the adequacy of design documentation.

   RT3) Identify the SSCs important to safety (Appendix B of the PSR2 Basis Document [1]).

RT4) Review the application of defence-in-depth. This includes an examination of:
- The degree of independence of the levels of defence-in-depth.
- The adequacy of delivery of preventive and mitigatory safety functions.
- Redundancy, separation and diversity of SSCs important to safety.
- Defence-in-depth in the design of structures (e.g., review of integrity of Fuel, cooling circuit and Containment building).

RT5) Confirm that the human-machine interface is considered in the design of the Control Room and other workstations, that analysis of human information requirements and task workload is performed, and that there is linkage to the Probabilistic Safety Assessment, Deterministic Safety Analysis and Hazard Analysis. This review should include a discussion of how guidance such as U.S. NRC NUREG-0700 Revision 2, *Human-System Interface Design Review Guidelines* and NUREG-0711 Revision 2, *Human Factors Engineering Program Review Model* identified in CNSC REGDOC-2.5.2 are relevant to the design of Pickering Units 1,4 and 5-8. (Note: In PSR1, a similar Review Task was addressed in the Human Factors Safety Factor. As it is the only Human Factors activity that deals with plant design it is being assessed as a PSR2 Plant Design Review Task).

RT6) Assess the adequacy of the arrangements for providing radiological protection.

RT7) Where the plant has undergone a significant number of modifications over its lifetime or in the period since PSR1, examine the cumulative effects of all modifications on the design.

RT8) Confirm that the plant SSCs are compliant with the design specifications and consistent with the design documentation.

2. Documenting assessments against applicable modern Laws, Regulations, Codes and Standards defined in the PSR2 Basis Document [1], per Appendix A.

3. Reviewing the effectiveness of the following applicable OPG Nuclear Programs that support the above assessments, through a review of audit and self-assessment results[2].

Document Number	Document Title
N-PROG-MP-0007	Conduct of Engineering
N-PROG-MP-0006	Software
N-PROG-MP-0005	Configuration Management
N-PROG-MP-0009	Design Management

The modern Laws, Regulations, Codes and Standards assessed in the review of this Safety Factor, along with the type of assessment conducted for each, are identified in Appendix A. The

[2] Although there may be content in Nuclear Programs that is applicable to multiple Safety Factors, each N-PROG review is only documented in one Safety Factor Report.

basis for the types of assessments performed is provided in Section 2.2 of the Safety Factor 1 Report [11].

6.1.1.3. Summary and Conclusions

Table 3 lists the thirty-six PSR2 Gaps identified in the Safety Factor 1 report [11], together with the source of each Gap and the associated Global Issue number.

The Safety Factor 1 report states that the review “has confirmed, by assessment against the current licensing basis and applicable standards, requirements and practices, that the design of Pickering NGS and its documentation is adequate” [11].

Table 3: PSR2 Gaps Identified for Safety Factor 1

Gap ID#[3]	Source of Gap			GI #
	Review Task	Modern Laws, Regulations, Codes and Standards	Other	
SF1-1		Canadian Standards Association (CSA) N285.0-12		GI-33
SF1-2		CSA N285.0-12		GI-33
SF1-3		CSA N293-12		GI-48
SF1-4		CSA N293-12		GI-48
SF1-5		CSA N293-12		GI-48
SF1-6		CSA N287.5-11		GI-42
SF1-7		CSA N290.0-11		GI-35
SF1-8		CSA N290.0-11		GI-31
SF1-9		CSA N290.0-11		GI-25

[3] Each Gap ID# entry is hyperlinked to a description of the Gap and to the associated Global Issue in Appendix B.

Gap ID#[3]	Source of Gap			GI #
	Review Task	Modern Laws, Regulations, Codes and Standards	Other	
SF1-10		CSA N290.1-13		GI-34
SF1-11		CSA N290.2-11		GI-36
SF1-12		CSA N290.2-11		GI-36
SF1-13		CSA N290.3-11		GI-37
SF1-14		CSA N290.4-11		GI-31
SF1-15		CSA N290.5-06		GI-31
SF1-16		CSA N290.11-13		GI-31
SF1-17		CSA N290.11-13		GI-38
SF1-18		CSA N290.11-13		GI-31
SF1-19		CSA N290.14-15		GI-39
SF1-20		CSA N291-15		GI-43
SF1-21		CSA N291-15		GI-43
SF1-22		CSA N291-15		GI-43
SF1-23		National Fire Protection Association (NFPA) 24		GI-47
SF1-24		NFPA 24		GI-47
SF1-25		CNSC REGDOC-2.5.2		GI-44
SF1-26		CNSC REGDOC-2.5.2		GI-44

Gap ID#[3]	Source of Gap			GI #
	Review Task	**Modern Laws, Regulations, Codes and Standards**	**Other**	
SF1-27		CNSC REGDOC-2.5.2		GI-44
SF1-28		CNSC REGDOC-2.5.2		GI-44
SF1-29		CNSC REGDOC-2.5.2		GI-44
SF1-30		CNSC REGDOC-2.5.2		GI-44
SF1-31		CNSC REGDOC-2.5.2		GI-44
SF1-32		CNSC REGDOC-2.5.2		GI-44
SF1-33		CNSC REGDOC-2.3.2		GI-40
SF1-34		CSA N290.8-15		GI-15
SF1-35			Additional Review Findings in Section 4.4 of Safety Factor 1 Report	GI-7
SF1-36			Additional Review Findings in Section 4.4 of Safety Factor 1 Report	GI-45

6.1.2. SF2: Actual Condition of Structures, Systems and Components Important to Safety

6.1.2.1. Objective

The objective of the review of Safety Factor 2, Actual Condition of Structures, Systems and Components Important to Safety, is to “determine the actual condition of SSCs important to safety and so to consider whether they are capable and adequate to meet design requirements, throughout the period of PSR2. In addition, the review should verify that the condition of SSCs important to safety is properly documented, as well as reviewing the ongoing maintenance, surveillance and in-service inspection programmes, as applicable” [1].

6.1.2.2. Scope of the Review

The PSR2 review of Safety Factor 2 contributes to confirming that the design, condition and operation of Pickering Units 1,4 and 5-8 (as well as common systems) will support continued safe operation for the period of PSR2 by:

1. Assessing compliance against the following Review Tasks identified in the PSR2 Basis Document [1], which were derived from IAEA SSG-25 [3] and CNSC REGDOC-2.3.3 [2]:

   RT1) Assess and document present conditions of the SSCs important to safety and confirm appropriate measures to address any significant existing or anticipated aging degradation are in place. Any major difference between operating units with respect to aging degradation mechanisms, present condition, or recommended actions shall also be presented.

   RT2) Confirm resources and facilities (on and off site) are available for ongoing plant maintenance.

   RT3) After determining the actual condition of SSCs important to safety, each of these SSCs will be assessed against the current design basis to confirm that design basis assumptions have not been significantly challenged and will remain that way throughout the period of PSR2.

   RT4) Review the condition and operation of spent fuel storage facilities and their effect on the spent fuel storage strategy for Pickering NGS.

   RT5) Assess dependence on obsolescent equipment for which no direct substitute is available.

   RT6) Assess dependence on essential services and/or supplies external to the plant.

2. Documenting assessments against applicable modern Laws, Regulations, Codes and Standards defined in the PSR2 Basis Document [1], per Appendix A.

3. Reviewing the effectiveness of the following applicable OPG Nuclear Programs that support the above assessments, through a review of audit and self-assessment results. Nuclear Program effectiveness reviews associated with Safety Factor 2 are provided in the Safety Factor 4 Report [12], and a synopsis of effectiveness review results is provided in the Safety Factor 4 Report summary.

Document Number	Document Title
OPG-PROG-0009	Items and Services Management
N-PROG-MP-0001	Engineering Change Control
N-PROG-MP-0004	Pressure Boundary

The modern Laws, Regulations, Codes and Standards assessed in the review of this Safety Factor, along with the type of assessment conducted for each, are identified in Appendix A.

6.1.2.3. Summary and Conclusions

Table 4 lists the eighteen PSR2 Gaps identified in the Safety Factor 2 report [13], together with the source of each Gap and the associated Global Issue number.

The Safety Factor 2 report states that the assessment "has not identified any major concerns that the SSCs will continue to operate as per the design basis requirements during life extension" [13]. The report also notes the presence of comprehensive and effective programs in place to ensure the condition of components meets design requirements with margin.

Table 4: PSR2 Gaps Identified for Safety Factor 2

Gap ID#[4]	Source of Gap			GI #
	Review Task	Modern Laws, Regulations, Codes and Standards	Other	
SF2-1	1			GI-1
SF2-2	1			GI-1

[4] Each Gap ID# entry is hyperlinked to a description of the Gap and to the associated Global Issue in Appendix B.

Gap ID#[4]	Source of Gap			GI #
	Review Task	Modern Laws, Regulations, Codes and Standards	Other	
SF2-3	1			GI-1
SF2-4	1			GI-2
SF2-5	1			GI-2
SF2-6	1			GI-3
SF2-7	1			GI-3
SF2-8	1			GI-4
SF2-9	1			GI-4
SF2-10	1			GI-5
SF2-11	1			GI-43
SF2-12	1			GI-8
SF2-13	1			GI-6
SF2-14	1			GI-7
SF2-15	1			GI-8
SF2-16	4			GI-10
SF2-17	4			GI-9
SF2-18	4			GI-9

6.1.3. SF3: Equipment Qualification (Environmental and Seismic)

6.1.3.1. Objective

The objective of the review of Safety Factor 3, Equipment Qualification (Environmental and Seismic), is to determine whether plant equipment important to safety has been properly qualified (including for environmental conditions) and whether this qualification is being maintained through an adequate program of maintenance, inspection and testing that provides confidence in the delivery of safety functions throughout the period of PSR2.

6.1.3.2. Scope of the Review

The PSR2 review of Safety Factor 3 contributes to confirming that the design, condition and operation of Pickering Units 1,4 and 5-8 (as well as common systems) will support continued safe operation for the period of PSR2 by:

1. Assessing compliance against the following Review Tasks identified in the PSR2 Basis Document [1], which were derived from IAEA SSG-25 [3] and CNSC REGDOC-2.3.3 [2]:

    RT1) Confirm there exists a suite of engineering programs or processes to ensure equipment qualification requirements are met and documented.

    RT2) Confirm equipment qualification has been adequately established for all service conditions expected during normal operation, anticipated operational occurrences and accident conditions. These service conditions are subdivided into environmental conditions and operational conditions. Environmental conditions include ambient temperature, pressure, humidity/steam, radiation, water/chemical sprays, fluid submergence, fire and seismic vibration. Operational conditions include process related conditions such as vibration, load cycling, electrical loading parameters, electromagnetic interference, mechanical loads and process fluid condition.

    RT3) Perform an objective confirmation that the installed equipment is qualified to perform its Design Basis function for its operational life and that effective programs exist to monitor for timely maintenance or replacement, as required.

    RT4) Confirm existence of a process for ensuring compliance with equipment qualification programs and of documented previous qualification measures taken to ensure qualification throughout the equipment’s installed life (i.e., prescribed testing, calibration, maintenance, and parts replacement).

    RT5) Confirm existence of a surveillance program and a feedback procedure to ensure aging degradation of qualified equipment remains insignificant.

RT6) Confirm existence of monitoring of actual environmental conditions and identification of ‘hot spots’ of high activity or temperature.

RT7) Confirm existence of an assessment that determines the effects of equipment failures on equipment qualification and appropriate corrective actions and/or safety improvements to maintain equipment qualification.

RT8) Confirm there is protection and adequate separation of qualified equipment from adverse environmental conditions.

RT9) Confirm physical condition and functional capability of qualified equipment is being checked by walkdowns.

RT10) Confirm that changes to equipment classification have occurred, as required, as a result of major design modifications made since PSR1.

2. Documenting assessments against applicable modern Laws, Regulations, Codes and Standards defined in the PSR2 Basis Document [1], per Appendix A.
3. Reviewing the effectiveness of the following applicable OPG Nuclear Programs that support the above assessments, through a review of audit and self-assessment results[5].

Document Number	Document Title
N-PROG-RA-0006	Environmental Qualification
N-PROG-MP-0001	Engineering Change Control
N-PROG-MA-0004	Conduct of Maintenance
N-PROG-MP-0008	Integrated Aging Management
N-PROG-MP-0009	Design Management

The modern Laws, Regulations, Codes and Standards assessed in the review of this Safety Factor, along with the type of assessment conducted for each, are identified in Appendix A. The bases for the types of assessments performed are provided in Section 2.2 of the Safety Factor 3 Report [14].

[5] Although there may be content in Nuclear Programs that is applicable to multiple Safety Factors, each N-PROG review is only documented in one Safety Factor Report.

6.1.3.3. Summary and Conclusions

Table 5 lists the six PSR2 Gaps identified in the Safety Factor 3 report [14], together with the source of each Gap and the associated Global Issue number.

The Safety Factor 3 report states the review "has confirmed that the Pickering NGS equipment important to safety has been properly qualified and that this qualification is being maintained through an adequate program of maintenance, inspection and testing" [14].

Table 5: PSR2 Gaps Identified for Safety Factor 3

Gap ID#[6]	Source of Gap			GI #
	Review Task	Modern Laws, Regulations, Codes and Standards	Other	
SF3-1	3			GI-12
SF3-2		CSA N289.3-10		GI-13
SF3-3		CSA N289.4-12		GI-13
SF3-4		CSA N289.5-12		GI-13
SF3-5			Audit and Self-Assessment Reviews	GI-14
SF3-6			Audit and Self-Assessment Reviews	GI-14

6.1.4. SF4: Aging

6.1.4.1. Objective

The objective of the review of Safety Factor 4, Aging, is to determine whether aging aspects affecting SSCs important to safety are being effectively managed and whether an effective

[6] Each Gap ID# entry is hyperlinked to a description of the Gap and to the associated Global Issue in Appendix B.

aging management program is in place so that all required safety functions will be delivered throughout the period of PSR2.

6.1.4.2. Scope of the Review

The PSR2 review of Safety Factor 4 contributes to confirming that the design, condition and operation of Pickering Units 1,4 and 5-8 (as well as common systems) will support continued safe operation for the period of PSR2 by:

1. Assessing compliance against the following Review Tasks identified in the PSR2 Basis Document [1], which were derived from IAEA SSG-25 [3] and CNSC REGDOC-2.3.3 [2]:

   RT1) Confirm there is a documented method and criteria for identifying safety related SSCs covered by the Aging Management Program.

   RT2) Ensure there is an effective Aging Management Program and dedicated organization with clearly defined roles and responsibilities with sufficient resources to continually assess aging effects in safety related SSCs.

   RT3) Establish a list of SSCs covered by the aging management program and records that provide information in support of the management of aging.

   RT4) Evaluate and document impact of potential aging degradation of safety-related SSCs.

   RT5) Confirm or develop understanding of dominant aging mechanisms of safety-related SSCs.

   RT6) Confirm existence of predictive maintenance program.

   RT7) Ensure existence of programs for timely detection and mitigation of aging mechanisms and/or aging effects of any SSCs important to safety, including obsolescence of technology used in the plant or obsolescence of services or supplies external to the plant.

   RT8) Establish acceptance criteria and required safety margin for safety-related SSCs for the period of PSR2 through reliability and risk assessments.

   RT9) Confirm adequacy of management of the effects of aging on those parts of the plant that will be required for safety when the reactor has ceased operation, for example the spent fuel storage facilities.

   RT10) Confirm the models used to predict the evolution and advancement of aging degradation are properly supported in accordance with current accepted practices pertaining to aging degradation.

2. Documenting assessments against applicable modern Laws, Regulations, Codes and Standards defined in the PSR2 Basis Document [1], per Appendix A.
3. Reviewing the effectiveness of the following applicable OPG Nuclear Programs that support the above assessments, through a review of audit and self-assessment results. Nuclear Program effectiveness reviews associated with Safety Factor 2, Actual Condition of SSCs Important to Safety, are also provided in the Safety Factor 4 Report.

Document Number	Document Title
N-PROG-MP-0008	Integrated Aging Management
N-PROG-MA-0025	Major Components
N-PROG-MA-0026	Equipment Reliability
N-PROG-MA-0017	Component and Equipment Surveillance
N-PROG-MA-0004	Conduct of Maintenance
N-PROG-OP-0004	Chemistry
N-PROG-MA-0019	Production Work Management
N-PROG-MA-0016	Fuel

The modern Laws, Regulations, Codes and Standards assessed in the review of this Safety Factor, along with the type of assessment conducted for each, are identified in Appendix A. The bases for the types of assessments performed are provided in Section 2.2 of the Safety Factor 4 Report [12].

6.1.4.3. Summary and Conclusions

Table 6 lists the eighteen PSR2 Gaps identified in the Safety Factor 4 report [12], together with the source of each Gap and the associated Global Issue number.

The Safety Factor 4 report states that the review "has confirmed that aging aspects affecting SSCs important to safety are being effectively managed and that an effective aging management program is in place at Pickering NGS" [12].

Table 6: PSR2 Gaps Identified for Safety Factor 4

Gap ID#[7]	Source of Gap			GI #
	Review Task	Modern Laws, Regulations, Codes and Standards	Other	
SF4-1	7			GI-15
SF4-2	9			GI-10
SF4-3		CSA N285.4-14		GI-50
SF4-4		CSA N285.4-14		GI-50
SF4-5		CSA N285.4-14		GI-50
SF4-6		CSA N285.4-14		GI-50
SF4-7		CSA N285.4-14		GI-50
SF4-8		CSA N285.4-14		GI-50
SF4-9		CSA N285.5-13		GI-16
SF4-10		CSA N285.5-13		GI-17
SF4-11		CSA N287.7-08		GI-18
SF4-12		CSA N287.7-08		GI-18
SF4-13		CSA N287.7-08		GI-19/GI-43
SF4-14		CNSC REGDOC-2.6.3		GI-20
SF4-15		CNSC REGDOC-2.6.3		GI-20
SF4-16		CSA N285.8-15		GI-1

[7] Each Gap ID# entry is hyperlinked to a description of the Gap and to the associated Global Issue in Appendix B.

Gap ID#[7]	Source of Gap			GI #
	Review Task	Modern Laws, Regulations, Codes and Standards	Other	
SF4-17			Audit and Self-Assessment Reviews	GI-20
SF4-18			COP Review in Support of Safety Factor 4	GI-22

The COP actions associated with Gap SF4-18 are listed in Table 12.

6.2. Safety Analysis

6.2.1. SF5: Deterministic Safety Analysis

6.2.1.1. Objective

The objective of the review of Safety Factor 5, Deterministic Safety Analysis, is to determine to what extent the existing deterministic safety analysis is complete and remains valid when the following aspects have been taken into account:

- The actual plant design, including all modifications of SSCs since the last update of the safety analysis report or PSR1.
- Current operating modes and fuel management.
- The actual condition of SSCs important to safety and their predicted state at the end of the period covered by PSR2.
- The use of modern, validated computer codes.
- Current deterministic methods.
- Current safety standards and knowledge (including research and development outcomes).
- The existence and adequacy of safety margins.

6.2.1.2. Scope of the Review

The PSR2 review of Safety Factor 5 contributes to confirming that the design, condition and operation of Pickering Units 1,4 and 5-8 (as well as common systems) will support continued safe operation for the period of PSR2 by:

1. Assessing compliance against the following Review Tasks identified in the PSR2 Basis Document [1], which were derived from IAEA SSG-25 [3] and CNSC REGDOC-2.3.3 [2]:

   RT1) Confirm the existence of current deterministic safety analyses and the assumptions used to perform these analyses.

   RT2) Evaluate the documentation and processes for defining, implementing and maintaining the Safe Operating Envelope (SOE).

   RT3) Perform assessment of OPG's Deterministic Safety Analysis to determine if the postulated events, event sequences and event combinations covered by the existing analysis are sufficient when compared against those for a modern nuclear power plant in accordance with the methodology in CNSC REGDOC-2.4.1, Deterministic Safety Analysis [15].

   RT4) Review adequacy of the documented guidelines for Deterministic Safety Analysis.

   RT5) Evaluate the supporting analyses for design extension conditions to confirm that the arrangements aimed at preventing or mitigating severe core damage meet regulatory requirements.

   RT6) Confirm that the impact of equipment failures and human errors, as well as the adequacy of engineering and administrative measures to prevent and mitigate accidents, have been analyzed and documented.

   RT7) Confirm that the capabilities of the plant in its current state, and where relevant with account taken of planned safety improvements, have been demonstrated to be within regulatory requirements and expectations for both normal operation and accident conditions. In addition, confirm that plans are in place to ensure that forecast operational conditions of the plant will meet acceptance criteria for the design basis, including adequacy of safety margins, throughout the period of PSR2.

2. Documenting assessments against applicable modern Laws, Regulations, Codes and Standards defined in the PSR2 Basis Document [1], per Appendix A.
3. Reviewing the effectiveness of the following applicable OPG Nuclear Program that supports the above assessments, through a review of audit and self-assessment results.

Document Number	Document Title
N-PROG-MP-0014	Reactor Safety Program

The modern Laws, Regulations, Codes and Standards assessed in the review of this Safety Factor, along with the type of assessment conducted for each, are identified in Appendix A. The bases for the types of assessments performed are provided in Section 2.2 of the Safety Factor 5 Report [16].

6.2.1.3. Summary and Conclusions

Table 7 lists the seven PSR2 Gaps identified in the Safety Factor 5 report [16], together with the source of each Gap and the associated Global Issue number.

The Safety Factor 5 report states that the review "has confirmed that the deterministic safety analysis programs and procedures at OPG are comprehensive, resulting in a systematic and disciplined approach to identifying, prioritizing and addressing any safety analysis related issues" [16].

Table 7: PSR2 Gaps Identified for Safety Factor 5

Gap ID#[8]	Source of Gap			GI #
	Review Task	Modern Laws, Regulations, Codes and Standards	Other	
SF5-1	7			GI-24
SF5-2	7			GI-25
SF5-3		CNSC REGDOC-2.4.1		GI-31
SF5-4		CNSC REGDOC-2.4.1		GI-31
SF5-5		CNSC REGDOC-2.5.2		GI-44

[8] Each Gap ID# entry is hyperlinked to a description of the Gap and to the associated Global Issue in Appendix B.

Gap ID#[8]	Source of Gap			GI #
	Review Task	Modern Laws, Regulations, Codes and Standards	Other	
SF5-6		CSA N288.2-14		GI-31
SF5-7			Additional Review Findings in Section 4.4 of Safety Factor 5 Report	GI-15

6.2.2. SF6: Probabilistic Safety Assessment

6.2.2.1. Objective

The objective of the review of Safety Factor 6, Probabilistic Safety Assessment (PSA), is to determine:

- The extent to which the existing PSA study remains valid as a representative model of the plant.
- Whether the results of the PSA show that the risks are sufficiently low and well balanced for all postulated initiating events and operational states.
- Whether the scope (which should include all operational states and identified internal and external hazards), methodologies and extent (i.e., Level 1, 2 or 3) of the PSA are in accordance with current national and international standards and good practices.
- Whether the existing scope and application of PSA are sufficient.

6.2.2.2. Scope of the Review

The PSR2 review of Safety Factor 6 contributes to confirming that the design, condition and operation of Pickering Units 1,4 and 5-8 (as well as common systems) will support continued safe operation for the period of PSR2 by:

1. Assessing compliance against the following Review Tasks identified in the PSR2 Basis Document [1], which were derived from IAEA SSG-25 [3] and CNSC REGDOC-2.3.3 [2]:

   RT1) Confirm existence of a PSA and the assumptions used, the fault schedule, the representations of operator actions and common cause events, the modelled plant configuration and consistency with other aspects of the safety case.

RT2) Confirm existence of processes to assess the impact of changes in plant design, operation, and plant specific failure data and update the PSA to reflect the current plant status as required.

RT3) Confirm there are guidelines to account for operator actions, common cause events, cross-link effects, redundancy and diversity.

RT4) Confirm that the accident management programs for accident conditions (design basis accident conditions and design extension conditions) are consistent with PSA models and results.

RT5) Confirm that the results of the PSA show that risks are sufficiently low and well balanced for all postulated initiating events and operational states, and meet relevant probabilistic safety criteria.

RT6) Review the extent to which hazards are represented in the PSA to verify that omissions are based on site specific justifications and that these omissions do not weaken the overall risk assessment for the plant.

2. Documenting assessments against applicable modern Laws, Regulations, Codes and Standards defined in the PSR2 Basis Document [1], per Appendix A.
3. Reviewing the effectiveness of the following applicable OPG Nuclear Program that supports the above assessments, through a review of audit and self-assessment results[9].

Document Number	Document Title
N-PROG-RA-0016	Risk and Reliability Program

The modern Laws, Regulations, Codes and Standards assessed in the review of this Safety Factor, along with the type of assessment conducted for each, are identified in Appendix A. The bases for the types of assessments performed are provided in Section 2.2 of the Safety Factor 6 Report [17].

6.2.2.3. Summary and Conclusions

Table 8 lists the five PSR2 Gaps identified in the Safety Factor 6 report [17], together with the source of each Gap and the associated Global Issue number.

[9] Although there may be content in Nuclear Programs that is applicable to multiple Safety Factors, each N-PROG review is only documented in one Safety Factor Report.

The Safety Factor 6 report states that the review "has confirmed that the PSA programs and procedures at OPG are comprehensive, resulting in a systematic and disciplined approach to identifying, prioritizing and addressing any PSA-related issues" [17].

Table 8: PSR2 Gaps Identified for Safety Factor 6

Gap ID#[10]	Source of Gap			GI #
	Review Task	Modern Laws, Regulations, Codes and Standards	Other	
SF6-1	5			GI-27
SF6-2	5			GI-27
SF6-3	5			GI-30
SF6-4		CNSC REGDOC-2.4.2		GI-32
SF6-5		CNSC REGDOC-2.5.2		GI-44

6.2.3. SF7: Hazard Analysis

6.2.3.1. Objective

The objective of the review of Safety Factor 7, Hazard Analysis, is to determine the adequacy of protection of the NPP against internal and external hazards, with account taken of the plant design, site characteristics, the actual condition of the SSCs important to safety and their predicted state at the end of the period covered by PSR2, and current analytical methods, safety standards and knowledge.

6.2.3.2. Scope of the Review

The PSR2 review of Safety Factor 7 contributes to confirming that the design, condition and operation of Pickering Units 1,4 and 5-8 (as well as common systems) will support continued safe operation for the period of PSR2 by:

[10] Each Gap ID# entry is hyperlinked to a description of the Gap and to the associated Global Issue in Appendix B.

1. Assessing compliance against the following Review Tasks identified in the PSR2 Basis Document [1], which were derived from IAEA SSG-25 [3] and CNSC REGDOC-2.3.3 [2]:

   RT1) Perform an assessment of the existing Deterministic and Probabilistic analyses to confirm existence of hazard analyses for hazards listed below. The following hazards are to be included in the assessment:

   (i) Internal Hazards: Fire, Pipe whip, Steam release, Toxic gas, Flooding, Missiles, Spray, Explosion.

   (ii) External Hazards: Changes in site characteristics, High winds (Tornado), Seismic, Toxic gas, Flooding, Extreme temperatures, Aircraft crash, Explosions.

   RT2) Confirm that the analyses and/or methods take into account the plant design and the condition of SSCs important to safety (both at present and predicted for the end of the period covered by PSR2).

   RT3) For each relevant hazard, verify, by means of current analytical techniques and data, that the frequency of occurrence and/or the consequences of the hazard are sufficiently low so that either no specific protective measures are necessary, or the preventive and mitigatory measures in place are adequate.

2. Documenting assessments against applicable modern Laws, Regulations, Codes and Standards defined in the PSR2 Basis Document [1], per Appendix A.

3. Reviewing the effectiveness of the following applicable OPG Nuclear Program that supports the above assessments, through a review of audit and self-assessment results[11].

Document Number	Document Title
N-PROG-RA-0012	Fire Protection

The modern Laws, Regulations, Codes and Standards assessed in the review of this Safety Factor, along with the type of assessment conducted for each, are identified in Appendix A. The bases for the types of assessments performed are provided in Section 2.2 of the Safety Factor 7 Report [18].

[11] Although there may be content in Nuclear Programs that is applicable to multiple Safety Factors, each N-PROG review is only documented in one Safety Factor Report.

6.2.3.3. Summary and Conclusions

Table 9 lists the PSR2 Gap identified in the Safety Factor 7 report [18], together with the source of the Gap and the associated Global Issue number.

The Safety Factor 7 report states that the review "has confirmed the adequacy of protection of Pickering NGS against internal and external hazards, with account taken of plant design (including confirmation that analyses/methods address the condition of SSCs important to safety), site characteristics, and current analytical methods, safety standards and knowledge" [18].

Table 9: PSR2 Gaps Identified for Safety Factor 7

Gap ID#[12]	Source of Gap			GI #
	Review Task	Modern Laws, Regulations, Codes and Standards	Other	
SF7-1	1			GI-25

6.3. Performance and Feedback from Operating Experience

6.3.1. SF8: Safety Performance

6.3.1.1. Objective

The objective of the review of Safety Factor 8, Safety Performance, is to determine whether the plant's safety performance indicators and records of operating experience, including the evaluation of root causes of plant events, indicate any need for safety improvements.

6.3.1.2. Scope of the Review

The PSR2 review of Safety Factor 8 contributes to confirming that the design, condition and operation of Pickering Units 1,4 and 5-8 (as well as common systems) will support continued safe operation for the period of PSR2 by:

[12] The Gap ID# entry is hyperlinked to a description of the Gap and to the associated Global Issue in Appendix B.

1. Assessing compliance against the following Review Tasks identified in the PSR2 Basis Document [1], which were derived from IAEA SSG-25 [3] and CNSC REGDOC-2.3.3 [2]:

   RT1) Confirm existence of a system for identifying, classifying and recording safety related incidents and operating experience including:

   - Safety related incidents, low level events and near misses.
   - Safety related operational data.
   - Maintenance, inspection and testing.
   - Replacements of SSCs important to safety owing to failure or obsolescence.
   - Modifications, either temporary or permanent, to SSCs important to safety.
   - Unavailability of safety systems.
   - Radiation doses (to workers, including contractors).
   - Off-site contamination and radiation levels.
   - Discharges of radioactive effluents.
   - Generation of radioactive waste.

   RT2) Confirm that safety-related incidents are investigated using root cause analysis and that lessons learned from investigation of these incidents are fed back into the conduct of Operations and Maintenance.

   RT3) Confirm that the results of the root cause analysis are used to minimize the chances of the same incident reoccurring.

   RT4) Confirm that information from trend analysis of safety related incidents is fed back into the conduct of Operations and/or Maintenance.

   RT5) Confirm there is an adequate set of performance indicators that provides a systematic and comprehensive method to record, trend and analyze safety related data including the major system parameters, and maintenance and inspection records. Performance indicators may include:

   - Frequency of unplanned trips while the reactor is critical.
   - Satisfactory performance of safety system tests within required limits.
   - Special Safety System unavailability.
   - Reliability of Systems Important to Safety.
   - Collective annual radiation dose of plant staff.
   - Amount of gaseous and liquid radioactive release relative to permitted limits.
   - Heavy water escape and loss rates.

- Fuel reliability.
- Chemistry index.
- Volume of Low Level radioactive waste.
- Change control index.
- Maintenance backlog.
- Training.
- Environment Index.
- Non-radioactive effluents, including hazardous substances.
- Non-radioactive wastes.
- Spills.

RT6) Confirm that for cases where performance indicators show an unsatisfactory trend, corrective action is taken.

RT7) Review the adequacy of:

- Records of the integrity of physical barriers for the containment of radioactive material.
- Records of radiation doses to persons on the site.
- Records of data from off-site radiation monitoring and records of the quantities of radioactive effluents.
- Records of non-radioactive effluents, including hazardous substances.
- Records of radioactive and non-radioactive waste.
- Records of spills.
- Records of other environmental impacts.

RT8) Consider the effects of any changes in operation at the plant on safety performance. In particular, confirm that current indicators and other safety performance methods continue to be relevant in the context of current and future operations, and confirm that only relevant data and records are used.

2. Documenting assessments against applicable modern Laws, Regulations, Codes and Standards defined in the PSR2 Basis Document [1], per Appendix A.
3. Reviewing the effectiveness of the following applicable OPG Nuclear Programs that support the above assessments, through a review of audit and self-assessment results[13].

[13] Although there may be content in Nuclear Programs that is applicable to multiple Safety Factors, each N-PROG review is only documented in one Safety Factor Report.

Document Number	Document Title
N-PROG-RA-0002	Conduct of Regulatory Affairs
N-PROG-RA-0003	Corrective Action
OPG-PROG-0010	Health & Safety Management System Program

The modern Laws, Regulations, Codes and Standards assessed in the review of this Safety Factor, along with the type of assessment conducted for each, are identified in Appendix A. The bases for the types of assessments performed are provided in Section 2.2 of the Safety Factor 8 Report [19].

6.3.1.3. Summary and Conclusions

There were no PSR2 Gaps identified in the Safety Factor 8 Report [19].

The Safety Factor 8 report states that the review "has confirmed that the safety performance indicators and records of operating experience, including the evaluation of root causes of plant events, exist and are utilized to ensure the safe operation of Pickering NGS" [19].

6.3.2. SF9: Use of Experience from Other NPPs and Research Findings

6.3.2.1. Objective

The objective of the review of Safety Factor 9, Use of Experience from Other NPPs and Research Findings, is to determine whether there is adequate feedback of relevant experience from other NPPs and from the findings of research and whether this is used to introduce reasonable and practicable safety improvements at the plant or in the operating organization.

6.3.2.2. Scope of the Review

The PSR2 review of Safety Factor 9 contributes to confirming that the design, condition and operation of Pickering Units 1,4 and 5-8 (as well as common systems) will support continued safe operation for the period of PSR2 by:

1. Assessing compliance against the following Review Tasks identified in the PSR2 Basis Document [1], which were derived from IAEA SSG-25 [3] and CNSC REGDOC-2.3.3 [2]:

   RT1) Confirm existence and adequacy of a program for the sending and receiving of experience relevant to safety to and from other nuclear power plants and relevant nonnuclear plants ("Other nuclear power plants" specifically include the IAEA, the Organization for Economic Cooperation and Development, Nuclear Energy Agency, the World Association of Nuclear Operators

(WANO), the Institute of Nuclear Power Operations (INPO) as well as CANDU Owners Group and experience within OPG at Darlington).

RT2) Confirm existence of a program for receiving information on the findings of relevant research programs.

RT3) Confirm there is a process for assessing the significance of operating experience from other plants and incorporating the lessons learned into improving safety performance at the station.

RT4) Confirm that there is a process for assessing the significance of research findings and technology developments and for incorporating relevant improvements into the station's design and operation.

RT5) Review adequacy and effectiveness of the feedback arrangements and timely implementation of assessment findings. (Assess program audit results).

RT6) List the major Operating Experience (OPEX) events and resulting plant changes that have resulted since PSR1 was completed.

2. Documenting assessments against applicable modern Laws, Regulations, Codes and Standards defined in the PSR2 Basis Document [1], per Appendix A.
3. Reviewing the effectiveness of applicable OPG Nuclear Programs that support the above assessments, through a review of audit and self-assessment results. There are no OPG Nuclear Programs assessed for the Use of Experience from Other NPPs and Research Findings in this report. Audit and self-assessment results for N-PROG-RA-0003, Corrective Action are provided in the Safety Factor 8 Report summary.

The modern Laws, Regulations, Codes and Standards assessed in the review of this Safety Factor, along with the type of assessment conducted for each, are identified in Appendix A. The bases for the types of assessments performed are provided in Section 2.2 of the Safety Factor 9 Report [20].

6.3.2.3. Summary and Conclusions

There were no PSR2 Gaps identified in the Safety Factor 9 Report [20].

The Safety Factor 9 report states that the review "has confirmed for Pickering NGS that there is adequate feedback of relevant experience from other nuclear power plants and from findings of research, and that this is used to introduce reasonable and practicable safety improvements at the plant or in the operating organization" [20].

6.4. Management

6.4.1. SF10: Organization, the Management System and Safety Culture

6.4.1.1. Objective

The objective of the review of Safety Factor 10, Organization, the Management System and Safety Culture, is to determine whether the organization, the management system and safety culture are adequate and effective for ensuring the safe operation of the plant.

6.4.1.2. Scope of the Review

The PSR2 review of Safety Factor 10 contributes to confirming that the design, condition and operation of Pickering Units 1,4 and 5-8 (as well as common systems) will support continued safe operation for the period of PSR2 by:

1. Assessing compliance against the following Review Tasks identified in the PSR2 Basis Document [1], which were derived from IAEA SSG-25 [3] and CNSC REGDOC-2.3.3 [2]:

   RT1) Review organization and administrative procedures to ensure they play a significant role in defining safety culture and evaluate the adequacy of safety culture indicators.

   RT2) Establish existence of a safety policy to ensure that safety takes precedence over production where a conflict between these two requirements exists.

   RT3) Identify the method for setting performance targets and confirm that these targets are regularly and systematically reviewed. Confirm that appropriate actions are initiated if safety targets are not met.

   RT4) Confirm that the published Nuclear organization, including any recent changes made to the organization, clearly defines the roles and responsibilities of all individuals and work groups who are involved in activities that could influence the safe operation of the station. Ensure that this organization is understood and that adequate and effective procedures are in place to ensure the availability of these resources and to control changes to this organization.

   RT5) Establish that mechanisms for maintaining configuration control of the plant and its documentation are effective and up-to-date.

   RT6) Confirm that there are formal arrangements for employing external technical, maintenance or other specialist staff, and confirm that the contracting procedures ensure that contract employees are qualified to do the work assigned to them.

RT7) Confirm that there is an approved Quality Assurance program and that regular Quality Assurance audits are conducted involving both internal and independent assessors.

RT8) Confirm that a program for self-assessment and continuous improvement has been adequately and effectively implemented including feedback of experience relating to organizational and management failures.

RT9) Confirm there is a system to ensure that comprehensive, easily retrievable, and auditable records exist of baseline design information, and operational and maintenance history.

RT10) Confirm there is an effective framework in place to support the management of regulatory affairs.

RT11) Confirm that the organization and management system include the processes and supporting information that explain how work is to be specified, prepared, reviewed, performed, recorded, assessed and improved.

RT12) Confirm there is control of purchasing of equipment and services where this affects plant safety.

RT13) Confirm there are comprehensive communication policies in place.

RT14) Confirm that a questioning attitude exists and conservative decision making is undertaken in the organization.

RT15) Verify that there is a process in place for prioritization of safety issues, with realistic objectives and timescales that ensures that these issues receive proper resources.

2. Documenting assessments against applicable modern Laws, Regulations, Codes and Standards defined in the PSR2 Basis Document [1], per Appendix A.
3. Reviewing the effectiveness of the following applicable OPG Nuclear Programs that support the above assessments, through a review of audit and self-assessment results[14].

Document Number	Document Title
N-PROG-AS-0001	Managed Systems
N-PROG-AS-0002	Human Performance

[14] Although there may be content in Nuclear Programs that is applicable to multiple Safety Factors, each N-PROG review is only documented in one Safety Factor Report.

Document Number	Document Title
N-PROG-RA-0010	Independent Assessment

The modern Laws, Regulations, Codes and Standards assessed in the review of this Safety Factor, along with the type of assessment conducted for each, are identified in Appendix A. The bases for the types of assessments performed are provided in Section 2.2 of the Safety Factor 10 Report [21].

6.4.1.3. Summary and Conclusions

There were no PSR2 Gaps identified in the Safety Factor 10 Report [21].

The Safety Factor 10 report states that the review "has confirmed that the Pickering NGS organization, the management system and safety culture are adequate and effective for ensuring the safe operation of the plant" [21].

6.4.2. SF11: Procedures

6.4.2.1. Objective

The objective of the review of Safety Factor 11, Procedures, is to determine whether the operating organization's processes for managing, implementing and adhering to operating and working procedures and for maintaining compliance with operational limits and conditions and regulatory requirements are adequate and effective and ensure plant safety.

6.4.2.2. Scope of the Review

The PSR2 review of Safety Factor 11 contributes to confirming that the design, condition and operation of Pickering Units 1,4 and 5-8 (as well as common systems) will support continued safe operation for the period of PSR2 by:

1. Assessing compliance against the following Review Tasks identified in the PSR2 Basis Document [1], which were derived from IAEA SSG-25 [3] and CNSC REGDOC-2.3.3 [2]:

   RT1) Determine if there is a process for the development, approval, and documenting of all safety related procedures.

   RT2) Confirm there is a formal process for modifying procedures affecting safety, including adequate arrangements for tracking changes.

   RT3) Confirm there is a program for assessing procedures and performance audits to determine if there is regular review and maintenance of these procedures.

RT4) Confirm that self-assessments are performed to ensure that the procedures are followed.

RT5) Establish that there is a means for assessing the adequacy of safety related procedures in comparison with industry good practices.

RT6) Confirm that there are operating procedures that apply comprehensively to normal, abnormal and emergency conditions (including anticipated operational occurrences, design basis accident conditions, post-accident conditions, and design extension conditions).

RT7) Confirm there is a means for assuring the clarity of procedures taking into account human factors.

RT8) Evaluate processes to update procedures to allow for changes in the assumptions made and/or the limits and conditions arising from the safety analysis, plant design and operating experience.

RT9) Verify that the analysis and justification of the accident management procedures are documented.

RT10) Verify that an appropriate process is in place for the categorization of procedures in accordance with their significance to safety.

RT11) Examine whether there is adequate involvement in the development of procedures by the staff that will use them.

RT12) Evaluate the distribution process for the control, copying and removal of obsolete versions of procedures, so that only the last approved edition is used.

RT13) Evaluate audits, self-assessments, safety performance and events to determine whether there is adequate understanding and acceptance of these procedures by managers and staff.

2. Documenting assessments against applicable modern Laws, Regulations, Codes and Standards defined in the PSR2 Basis Document [1], per Appendix A.
3. Reviewing the effectiveness of the following applicable OPG Nuclear Program that supports the above assessments, through a review of audit and self-assessment results[15].

[15] Although there may be content in Nuclear Programs that is applicable to multiple Safety Factors, each N-PROG review is only documented in one Safety Factor Report.

Document Number	Document Title
N-PROG-OP-0001	Nuclear Operations

The modern Laws, Regulations, Codes and Standards assessed in the review of this Safety Factor, along with the type of assessment conducted for each, are identified in Appendix A. The bases for the types of assessments performed are provided in Section 2.2 of the Safety Factor 11 Report [22].

6.4.2.3. Summary and Conclusions

There were no PSR2 Gaps identified in the Safety Factor 11 Report [22].

The Safety Factor 11 report states that the review "has confirmed that the Pickering NGS processes for managing, implementing and adhering to operating and working procedures and for maintaining compliance with operational limits and conditions and regulatory requirements are adequate and effective and ensure plant safety" [22].

6.4.3. SF12: Human Factors

6.4.3.1. Objective

The objective of the review of Safety Factor 12, Human Factors, is to evaluate the various human factors that may affect the safe operation of the NPP and to seek to identify improvements that are reasonable and practicable.

6.4.3.2. Scope of the Review

The PSR2 review of Safety Factor 12 contributes to confirming that the design, condition and operation of Pickering Units 1,4 and 5-8 (as well as common systems) will support continued safe operation for the period of PSR2 by:

1. Assessing compliance against the following Review Tasks identified in the PSR2 Basis Document [1], which were derived from IAEA SSG-25 [3] and CNSC REGDOC-2.3.3 [2]:

    RT1) Confirm that there are procedures to ensure that a minimum number of qualified staff, appropriate to the operating state of the plant, is available at all times.

    RT2) Confirm that adequate staff training facilities, training staff and training programs exist.

RT3) Confirm that the method of selecting staff for new positions and for promotions involves systematic and validated staff selection methods and a method for succession planning.

RT4) Confirm that there are appropriate programs for initial, refresher, and upgrade training. For operating staff, this should include the use of simulators.

RT5) Establish that there is training in safety culture, including for management staff, that includes work supervision practices and internal communication practices and expectations.

RT6) Confirm there are fitness for duty guidelines relating to hours of work, health and substance abuse.

RT7) Confirm that the human–machine interface is considered in the design of the control room and other workstations, that analysis of human information requirements and task workload is performed, and that there is linkage to the Probabilistic Safety Assessment, Deterministic Safety Analysis and Hazard Analysis. This review should include a discussion of how guidance such as U.S. NRC NUREG-0700 Revision 2, Human-System Interface Design Review Guidelines and NUREG-0711 Revision 2, Human Factors Engineering Program Review Model, identified in CNSC REGDOC-2.5.2 are relevant to the design of Pickering Units 1,4 and 5-8 (Note: Review Task #7 is addressed in the Plant Design Safety Factor 1 Report).

RT8) Confirm the style and clarity of procedures provides an appropriate level of detailed guidance for staff through a review of plant events identifying inadequate procedures as a contributing cause.

2. Documenting assessments against applicable modern Laws, Regulations, Codes and Standards defined in the PSR2 Basis Document [1], per Appendix A.
3. Reviewing the effectiveness of the following applicable OPG Nuclear Program that supports the above assessments, through a review of audit and self-assessment results[16].

Document Number	Document Title
N-PROG-TR-0005	Training

The modern Laws, Regulations, Codes and Standards assessed in the review of this Safety Factor, along with the type of assessment conducted for each, are identified in Appendix A.

[16] Although there may be content in Nuclear Programs that is applicable to multiple Safety Factors, each N-PROG review is only documented in one Safety Factor Report.

The bases for the types of assessments performed are provided in Section 2.2 of the Safety Factor 12 Report [23].

6.4.3.3. Summary and Conclusions

There were no PSR2 Gaps identified in the Safety Factor 12 Report [23].

The Safety Factor 12 report states that the review "has confirmed that the various human factors that may affect the safe operation of Pickering NGS have been appropriately addressed" [23].

6.4.4. SF13: Emergency Planning

6.4.4.1. Objective

The objective of the review of Safety Factor 13, Emergency Planning, is to determine whether:

a) The operating organization has in place adequate plans, staff, facilities and equipment for dealing with emergencies.

b) The operating organization's arrangements have been adequately coordinated with the arrangements of local and national authorities and are regularly exercised.

6.4.4.2. Scope of the Review

The PSR2 review of Safety Factor 13 contributes to confirming that the design, condition and operation of Pickering Units 1,4 and 5-8 (as well as common systems) will support continued safe operation for the period of PSR2 by:

1. Assessing compliance against the following Review Tasks identified in the PSR2 Basis Document [1], which were derived from IAEA SSG-25 [3] and CNSC REGDOC-2.3.3 [2]:

   RT1) Confirm the full range of accidents and radiation emergencies have been identified and studied.

   RT2) Confirm the appropriate response and mitigation strategies have been developed and have taken account of major changes at site and around the site (industrial, commercial, residential development).

   RT3) Confirm that the station organization includes dedicated Emergency Response personnel on duty at the plant at all times, to handle accidents and emergencies.

   RT4) Assess the adequacy of the training program for emergency response personnel including training, emergency exercises and qualification records.

RT5) Confirm there is a process for notification of staff that will be brought in to assist in the management of the response in the longer term.

RT6) Determine that there is a classification of accidents to guide the type of response.

RT7) Confirm there is a mechanism for notifying and informing relevant off-site organizations such as the police, fire departments, hospitals, ambulance services, regulatory bodies, local authorities, government, public welfare authorities and the news media.

RT8) Confirm the availability of sufficient communications equipment at the plant and at the off-site Emergency Centre to permit effective communications with Emergency Response Teams, both on and off site.

RT9) Assess adequacy of the emergency response procedures and training and exercises for all site staff. Confirm that integrated and partial emergency exercises have been conducted to check satisfactory function of the emergency organization and its equipment.

RT10) Confirm the adequacy of on-site equipment and facilities for emergencies and offsite emergency facilities or locations, including walkdowns of relevant areas on and off the site.

RT11) Confirm development or existence of a program for Severe Accident Management.

2. Documenting assessments against applicable modern Laws, Regulations, Codes and Standards defined in the PSR2 Basis Document [1], per Appendix A.
3. Reviewing the effectiveness of the following applicable OPG Nuclear Program that supports the above assessments, through a review of audit and self-assessment results[17].

Document Number	Document Title
N-PROG-RA-0001	Consolidated Nuclear Emergency Plan

The modern Laws, Regulations, Codes and Standards assessed in the review of this Safety Factor, along with the type of assessment conducted for each, are identified in Appendix A. The bases for the types of assessments performed are provided in Section 2.2 of the Safety Factor 13 Report [24].

[17] Although there may be content in Nuclear Programs that is applicable to multiple Safety Factors, each N-PROG review is only documented in one Safety Factor Report.

6.4.4.3. Summary and Conclusions

Table 10 lists the two PSR2 Gaps identified in the Safety Factor 13 report [24], together with the source of each Gap and the associated Global Issue number.

The Safety Factor 13 report states that the review "has confirmed that OPG Nuclear has: a) adequate plans, staff, facilities and equipment in place for dealing with emergencies, and b) there are adequate arrangements in place for regular emergency training and exercises, and interaction and coordination with local and national authorities" [24].

Table 10: PSR2 Gaps Identified for Safety Factor 13

Gap ID#[18]	Source of Gap			GI #
	Review Task	Modern Laws, Regulations, Codes and Standards	Other	
SF13-1		CNSC REGDOC-2.10.1		GI-41
SF13-2			Additional Review Findings in Section 4.4 of Safety Factor 13 Report	GI-26

6.5. Environment

6.5.1. SF14: Radiological Impact on the Environment

6.5.1.1. Objective

The objective of the review of Safety Factor 14, Radiological Impact on the Environment, is to determine whether the operating organization has an adequate and effective program for monitoring the radiological impact of the plant on the environment, which ensures that emissions are properly controlled and are As Low as Reasonably Achievable (ALARA).

[18] Each Gap ID# entry is hyperlinked to a description of the Gap and to the associated Global Issue in Appendix B.

6.5.1.2. Scope of the Review

The PSR2 review of Safety Factor 14 contributes to confirming that the design, condition and operation of Pickering Units 1,4 and 5-8 (as well as common systems) will support continued safe operation for the period of PSR2 by:

1. Assessing compliance against the following Review Tasks identified in the PSR2 Basis Document [1], which were derived from IAEA SSG-25 [3] and CNSC REGDOC-2.3.3 [2]:

    RT1) Confirm there are procedures in place to ensure that permitted release limits of radiological substances are not exceeded and, if they are, that appropriate corrective action is taken to minimize the possibility of limits being exceeded in the future.

    RT2) Confirm records of radiological effluent release are maintained in accordance with regulatory requirements.

    RT3) Confirm that a program exists to define the requirements for alarm systems to respond to unplanned effluent releases from on-site facilities.

    RT4) Confirm the environmental data recorded by the station is published and is available on request to the general public.

    RT5) Review the environmental data recorded by the station and compare with the values measured before the plant was put into operation.

    RT6) Confirm there is a process to address changes in the use of land external to the site with respect to the impact on public safety from facility releases.

    RT7) Confirm that the monitoring program is appropriate and sufficiently comprehensive. In particular, confirm that the radiological impact of the plant on the environment is not significant compared with that due to other sources of radiation.

2. Documenting assessments against applicable modern Laws, Regulations, Codes and Standards defined in the PSR2 Basis Document [1], per Appendix A.

3. Reviewing the effectiveness of the following applicable OPG Nuclear Programs that support the above assessments, through a review of audit and self-assessment results[19].

Document Number	Document Title
N-PROG-OP-0006	Environmental Management

[19] Although there may be content in Nuclear Programs that is applicable to multiple Safety Factors, each N-PROG review is only documented in one Safety Factor Report.

Document Number	Document Title
OPG-PROG-0005	Environmental Management System

The modern Laws, Regulations, Codes and Standards assessed in the review of this Safety Factor, along with the type of assessment conducted for each, are identified in Appendix A. The bases for the types of assessments performed are provided in Section 2.2 of the Safety Factor 14 Report [25].

6.5.1.3. Summary and Conclusions

There were no PSR2 Gaps identified in the Safety Factor 14 Report [25].

The Safety Factor 14 report states that the review "has confirmed that Pickering NGS has an adequate and effective program for monitoring the radiological impact of the plant on the environment, which ensures that emissions are properly controlled and are as low as reasonably achievable" [25].

6.6. Radiation Protection

6.6.1. SF15: Radiation Protection

6.6.1.1. Objective

The objective of the review of Safety Factor 15, Radiation Protection, is to confirm that:

- Radiation Protection has been adequately accounted for in the design and operation of Pickering NGS.
- Radiation Protection provisions (including design and equipment) provide adequate protection of persons from the harmful effects of radiation.
- Contamination and radiation exposures and doses to persons are monitored and controlled, and maintained ALARA.

6.6.1.2. Scope of the Review

The PSR2 review of Safety Factor 15 contributes to confirming that the design, condition and operation of Pickering Units 1,4 and 5-8 (as well as common systems) will support continued safe operation for the period of PSR2 by:

1. Assessing compliance against the following Review Tasks identified in the PSR2 Basis Document [1], which were derived from IAEA SSG-25 [3] and CNSC REGDOC-2.3.3 [2]:

    RT1) Confirm the adequacy of the reactor design features for Radiation Protection.

RT2) Confirm the adequacy of the Radiation Protection equipment and instrumentation for radiation monitoring.

RT3) Confirm that adequate provisions are in place to address Radiation Protection of the public and workers during nuclear emergencies.

RT4) Confirm that the Radiation Protection provisions have been improved as the result of external operating experience.

RT5) The review will demonstrate that the ALARA principle has been incorporated in any modifications of the reactor design and operational programs and arrangements.

2. Documenting assessments against applicable modern Laws, Regulations, Codes and Standards defined in the PSR2 Basis Document [1], per Appendix A.
3. Reviewing the effectiveness of the following applicable OPG Nuclear Program that supports the above assessments, through a review of audit and self-assessment results[20].

Document Number	Document Title
N-PROG-RA-0013	Radiation Protection

The modern Laws, Regulations, Codes and Standards assessed in the review of this Safety Factor, along with the type of assessment conducted for each, are identified in Appendix A. The bases for the types of assessments performed are provided in Section 2.2 of the Safety Factor 15 Report [26].

6.6.1.3. Summary and Conclusions

There were no PSR2 Gaps identified in the Safety Factor 15 Report [26].

The Safety Factor 15 report states that the review "has confirmed that Radiation Protection has been adequately accounted for in the design and operation of Pickering NGS, and that Radiation Protection provisions (including design and equipment) provide adequate protection of persons from the harmful effects of radiation, and that contamination and radiation exposures and doses to persons are monitored and controlled and maintained ALARA" [26].

[20] Although there may be content in Nuclear Programs that is applicable to multiple Safety Factors, each N-PROG review is only documented in one Safety Factor Report.

7. Review of the Pickering B ISR Continued Operations Plan

7.1. Objective

The objective of the Pickering B ISR COP Review in support of PSR2 [5] was to reassess the 88 applicable actions from the Pickering B ISR COP referred to in [27] and documented in [28] that need to be reassessed for PSR2, based on extended operation beyond 2020. The 88 COP actions were also reviewed for applicability to Pickering 1,4.

7.2. Scope of the Review

The Pickering B ISR COP was developed to document the actions that helped support continued operation of Pickering Units 5-8. The COP planning basis was operation to 2020. The COP was updated annually from 2010 through to 2015. A final update was provided in December 2015 ([29], [30]).

Per the PSR2 Basis Document [1], the COP actions were reviewed based on assumed operation to the end of 2028. The COP actions were also reviewed for applicability to Pickering 1,4. This approach is consistent with Reference [31] in that it takes into account CNSC staff comments on the 2013 version of the COP, as this version included the most comprehensive COP action listing. The 88 applicable actions were addressed in the Pickering B ISR COP Review in support of PSR2 [5].

The applicable COP actions are assigned action codes beginning with “F”, “G” or “I”. As described in the COP [30], “F” actions are from Life Cycle Management Plans or Condition Assessments, “G” actions arose from the Pickering B Global Assessment and “I” actions originated in the Pickering B ISR.

7.3. Summary of Gaps

The COP Review identified twenty-six PSR2 Gaps. Table 11 identifies the Global Issue number associated with each PSR2 COP Review Gap. The Pickering B ISR Integrated Implementation Plan (IIP) identification numbers (ID) are also included for completeness.

Table 11: PSR2 Gaps from Review of the Pickering B Continued Operations Plan

COP Review Gap ID#[21]	GI #	Pickering B ISR IIP ID Number
COP-1	GI-1	F02-1a
COP-2	GI-4	F04
COP-3	GI-5	F05
COP-4	GI-43	F06
COP-5	GI-29	F07
COP-6	GI-29	F08
COP-7	GI-28	F09
COP-8	GI-2	F14-1b
COP-9	GI-3	F14-2a
COP-10	GI-4	F14-3a
COP-11	GI-4	F14-3b
COP-12	GI-4	F14-3c
COP-13	GI-4	F14-3d
COP-14	GI-31	G01-06
COP-15	GI-11	G02-01
COP-16	GI-19	G04-03
COP-17	GI-13	I02
COP-18	GI-46	I03-1
COP-19	GI-23	I04

[21] Each Gap ID# entry is hyperlinked to a description of the Gap and to the associated Global Issue in Appendix B.

COP Review Gap ID#[21]	GI #	Pickering B ISR IIP ID Number
COP-20	GI-25	I09
COP-21	GI-31	I10, I10-01
COP-22	GI-5	I15-1d
COP-23	GI-21	I15-5d
COP-24	GI-49	I15-5e
COP-25	GI-19/GI-43	I15-6b
COP-26	GI-4	I15-7b

In the Pickering NGS PSR2 Safety Factor 4 Report [12], COP actions were also evaluated for applicability to PSR2. Two Gaps were identified. For completeness, the information on the related PSR2 Gaps is repeated below.

- PSR2 Gap SF4-13: Actions #31, #32, and #33 from the Pickering Units 5-8 Continued Operations Plan are related to N287.7 and although complete, need to be re-assessed for Pickering operation beyond 2020. (IIP Action #31 involved submission of Periodic Inspection Programs (PIPs) and Life Cycle Management Plans (LCMPs) for a number of safety-significant civil structures. IIP Action #32 involved submission of Aging Management Plans for Concrete Containment Structures to the CNSC for acceptance. IIP Action #33 involved revising the Reactor Building PIP and submitting to the CNSC for acceptance).
- PSR2 Gap SF4-18: Review of the Pickering Units 5-8 Continued Operations Plan identified the closed gaps from the Pickering B ISR outlined in Table 12 that will need to be revisited in the context of continued operation past 2020. The items included in SF4-18 are associated with Global Issue GI-22.

Table 12: Elements of PSR2 Gap SF4-18 – Global Issue GI-22

ID # from the COP ([29], [30])	Pickering B ISR IIP ID Number
Appendix A Item 4	G01-04
Appendix A Items 10, 11, 12, 13	F01 (including F01-1, F01-2, F01-3, F01-4)

ID # from the COP ([29], [30])	Pickering B ISR IIP ID Number
Appendix A Item 14	F02
Appendix A Item 19	F13
Appendix A Item 21	F14-1a
Appendix A Item 30	F14-4.1
Appendix A Item 52	I15-1a
Appendix A Item 69	I15-7a
Appendix C Item 5	F11
Appendix C Item 6	F12

7.4. Conclusions

Actions identified in the Pickering Units 5-8 COP were reviewed for PSR2 in the context of Pickering NGS operation beyond 2020 [5]. Twenty-six Gaps were identified for consideration in the PSR2 Global Assessment, in addition to 10 items from the Pickering B ISR that were identified for re-assessment as part of Gap SF4-18 during the review of Safety Factor 4.

8. Review of Fukushima Action Items

8.1. Objective

A review of FAIs in support of PSR2 was performed to consider the impact of extended operation on all FAIs assigned to OPG that are applicable to Pickering NGS. The PSR2 Basis Document [1] states that FAIs will be reviewed to determine if there are any impacts associated with operation past 2020. This is consistent with CNSC staff feedback in [31] that OPG should consider the FAIs that were closed on the basis that Pickering NGS units were to shut down in 2020 and determine if the basis of closure is affected by the potential to operate Pickering NGS beyond 2020.

The FAIs were originally identified in [32]. The status of all FAIs for OPG facilities was summarized in OPG's Fukushima Action Item Status Report [33]. As noted in [33], OPG has completed, and the CNSC has closed, all FAIs assigned to OPG.

8.2. Scope of the Review

The scope of the review of FAIs in support of PSR2 considered all FAIs that were listed in [32] and the follow-up actions that were created to monitor progress related to specific issues. The assessment considered Pickering Units 1,4 and 5-8. Common systems (Unit 0) are included in the Unit assessments as applicable.

8.3. Summary of Gaps

There were no PSR2 Gaps identified from a review of the FAIs [6].

8.4. Conclusions

All FAIs in [32] were assessed for the impact of operation beyond 2020 on Pickering Units 1,4 and 5-8. It was concluded that none of the FAIs are impacted by extended operation of Pickering NGS beyond 2020.

OPG's responses to the FAIs assigned to Pickering NGS remain valid in the context of extended operation beyond 2020. There are no additional Gaps identified from a review of the FAIs that must be considered in the PSR2 Global Assessment.

9. Findings from CNSC Staff Reviews of Safety Factor Reports and Complementary Reviews

The OPG-CNSC Protocol for the conduct of PSR2 is described in [4] and includes the agreed upon strategy and schedule for PSR2, including submission of the PSR2 deliverables to CNSC staff for review. This section describes CNSC staff findings from their independent reviews of PSR2 documents that are inputs to the Global Assessment Report, and describes how these findings have been incorporated into the Global Assessment Report.

From the CNSC staff review of the 15 Safety Factor Reports and two Complementary Reviews (COP and FAI reviews), 75 Additional Gaps were identified [34] to [43].

As stated in the CNSC review reports [34] to [43], CNSC staff were guided and informed by the following in performing their reviews:

- The Nuclear Safety and Control Act [44]
- General Nuclear Safety and Control Regulations [45]
- CNSC REGDOC-2.3.3, Periodic Safety Reviews [2]
- PSR2 Basis Document [1]

CNSC staff also used additional regulatory documents, standards and expert judgment to perform their assessments. In total, CNSC staff identified 2 Strengths during their assessments of the Safety Factor Reports. The Strengths identified by CNSC staff are listed in Table 13. These Strengths were considered in the Defence-in-Depth Assessment (refer to Section 18).

Table 13: Strengths Based on CNSC Staff Reviews

Safety Factor	Strength Title and Description	Reference
SF9	On page 18, Section 4.1.1, OPG discussion on and the extent of coverage of “Receiving OPEX” is considered by CNSC staff as a strength.	[40]
SF11	On page 8, Section 2.1, OPG discussion on and the extent of coverage of “Review Task Assessments” is considered by CNSC staff as a strength.	[40]

Based on correspondence documented in References [34] to [43], the identified Additional Gaps are primarily requests for additional information, or for additional supporting evidence to support statements made by OPG in the Safety Factor Reports or Complementary Reviews. The Additional Gaps have been grouped into three types as follows:

Type I. Provision of information. These are Additional Gaps related to CNSC staff requests for additional supporting evidence for statements made by OPG in the Safety Factor and Complementary Review Reports.

Type II. Provision of additional information to demonstrate the effectiveness of OPG Programs. These are Additional Gaps related to CNSC staff requests for additional evidence of program effectiveness or follow-up on specific requests for evidence of conformance.

Type III. Specific technical issues. These are Additional Gaps related to CNSC staff identification of technical concerns, or demonstration of adequacy of implementation/response to issues of concern to CNSC staff.

As documented in [46], OPG committed to address each of the Type I and II Additional Gaps by March 15, 2018. The PSR2 Database that is currently being developed by OPG will be used to assist with tracking the status and progress to completion of these Additional Gaps.

The Type III Additional Gaps are consolidated with other related PSR2 Gaps into appropriate Global Issues. Each Global Issue is prioritized and, depending on the prioritization, proposed Resolution Statements may or may not be developed, consistent with the process for developing proposed Resolution Plans described in Section 5.4. If proposed Resolution Statements are developed for Type III Additional Gaps, the Resolution Statements are ranked.

The Type III Additional Gaps are listed in Table 14, together with the source of each Additional Gap and the associated Global Issue number in Appendix B. The naming convention used in Table 14 for the Additional Gaps is:

- SF#-AG#, for Gaps based on correspondence related to the Safety Factor Reports
- COP-AG#, for Gaps based on correspondence related to the COP
- FAI-AG#, for Gaps based on correspondence related to FAIs

Table 14: List of Additional Gaps Identified Based on CNSC Staff Reviews

Safety Factor Report/Complementary Review	Type III Additional Gap ID #[22]	Associated GI #	Type III Additional Gap Source Reference
SF1	SF1-AG4	GI-40	[34]
	SF1-AG7	GI-35	[34]
	SF1-AG14	GI-31	[34]

[22] Each Type III Additional Gap ID# entry is hyperlinked to a description of the Additional Gap and to the associated Global Issue in Appendix B.

Safety Factor Report/Complementary Review	Type III Additional Gap ID #[22]	Associated GI #	Type III Additional Gap Source Reference
	SF1-AG15	GI-37	[34]
	SF1-AG16	GI-27	[34]
	SF1-AG17	GI-44	[34]
	SF1-AG18	GI-44	[34]
	SF1-AG19	GI-44	[34]
	SF1-AG20 (this is identified as SF1-AG3 in [34])	GI-38	[34]
SF2	SF2-AG1	GI-1/GI-4	[35]
	SF2-AG2	GI-22	[35]
	SF2-AG4	GI-8	[35]
	SF2-AG8	GI-8	[35]
	SF2-AG10	GI-50	[35]
SF3	No Type III Additional Gaps		
SF4	No Type III Additional Gaps		
SF5	SF5-AG1	GI-31	[38]
	SF5-AG3	GI-15	[38]
SF6	SF6-AG2	GI-27	[38]
	SF6-AG3	GI-27	[38]
SF7	No Type III Additional Gaps		
SF8	No Type III Additional Gaps		
SF9	No Type III Additional Gaps		
SF10	No Type III Additional Gaps		

Safety Factor Report/Complementary Review	Type III Additional Gap ID #[22]	Associated GI #	Type III Additional Gap Source Reference
SF11	No Type III Additional Gaps		
SF12	No Type III Additional Gaps		
SF13	No Type III Additional Gaps		
SF14	No Type III Additional Gaps		
SF15	No Type III Additional Gaps		
Review of COP	COP-AG1	GI-24	[42]
	COP-AG2	GI-1	[42]
	COP-AG3	GI-1	[42]
	COP-AG4	GI-1	[42]
Review of FAI	FAI-AG1	GI-27	[43]

Part III: Global Assessment

Section	
#	**Title**
10	Integrated Review of PSR2 Gaps
11	Development of Global Issues
12	Prioritization of Global Issues
13	Development of Proposed Resolution Plans
14	Ranking of Proposed Resolution Statements
15	Pickering PSR2 Expert Panel

Appendix
Appendix B – Global Issues and Proposed Resolution Plans
Appendix C – Ranking of Proposed Resolution Statements

10. Integrated Review of PSR2 Gaps

The elements of the Global Assessment process are described in Section 3.3.2 of the PSR2 Basis Document [1]. This section presents the results of completing the first step, which is a review and consolidation of the PSR2 Gaps. This step is a precursor to the identification of Global Issues.

PSR2 Gaps are identified from a review of the following sources, based on the guidance provided in Section 5.1:

- the Safety Factor Reports (93 Gaps identified in Sections 6.1.1 through 6.6.1, including 10 items from the Pickering B ISR (refer to Table 12) that were identified for re-assessment as part of Gap SF4-18 during the review of Safety Factor 4);
- the Complementary Review of the Pickering B ISR COP (26 Gaps identified in Section 7.3);
- the Complementary Review of the FAIs (no Gaps identified in Section 8.3); and
- CNSC Staff reviews of PSR2 deliverables (23 Additional Gaps identified in Section 9 for consideration in the Global Assessment).

A total of 142 PSR2 Gaps identified from the various sources are presented in Section 6 to Section 9, and one more Gap identified by the Expert Panel is presented in Section 15.5. The Gaps are grouped into Global Issues, as described in Section 11.

11. Development of Global Issues

The second element of the Global Assessment process (Section 3.3.2 of [1]) is the definition of Global Issues. Each Global Issue is derived from the grouped PSR2 Gaps described in Section 10.

There are two aspects to this step, as follows:

- Development of a Global Issue title and a description of the issue based on the related PSR2 Gaps; and
- Assessment of the interfaces between the various Safety Factors and the aggregate impact of Global Issues

This step follows the guidance provided in Section 5.2, using the Gaps listed in Section 6 to Section 9 and Section 15 as input.

11.1. Consolidation of Gaps into Global Issues

The Gaps listed in Section 6 to Section 9 and Section 15 are grouped into Global Issues in Appendix B according to their topical similarities, i.e., based on the related discipline, governing process or relevant modern codes and standards, with consideration of any interfaces, overlaps and similarities among the Gaps.

A total of 51 Global Issues are identified for the Pickering NGS PSR2. All 143 of the Gaps identified in Section 6 to Section 9 and Section 15 are mapped to one or more of the 51 Global Issues.

The Global Issue Titles are listed in Table 15. Full descriptions of each Global Issue are provided in Global Issue Tables in Appendix B, under the following section headings within each table:

Section 1 – Global Issue Summary

Section 2 – Associated Gaps

Section 3 – Background Information and Resolution Strategy

Section 4 – Priority Determination

Section 5 – Resolution Plan

The Safety Significance Level shown for each Global Issue in Table 15 is described in Section 12.

Table 15: PSR2 Global Issues

Global Issue #[23]	Global Issue Title	Safety Significance Level
GI-1	Fitness for Service for Fuel Channels	1
GI-2	Fitness for Service for Feeders	1
GI-3	Fitness for Service for Steam Generators	1
GI-4	Fitness for Service for Reactor Components and Structures	2
GI-5	Completeness of Class 1 Piping / Components Service Limits Assessment (Excluding Major Components)	2
GI-6	Impact of the Revised Criticality Coding on the Cable Surveillance Program	3
GI-7	Pickering Buried Piping Fitness for the Extended Operating Period	3
GI-8	Completion / Updating of the Condition Assessments	2
GI-9	Seismic Capacity of the Conveyor Tube and Fuel Basket Stacking Arrangement	3
GI-10	IFB Condition	3
GI-11	Fuel Management and Surveillance Software Upgrade	4
GI-12	Extending the Environmental Qualification of Equipment	3
GI-13	Seismic Qualification - N289	4
GI-14	Environmental Qualification Program Issues	4
GI-15	Governance Issues	4
GI-16	Concession Related to N285.5-M90	4
GI-17	FFS of Fiberglass Reinforced Plastic Material for the Extended Operating Period	2
GI-18	N287.7 – In-Service Examination and Testing Requirements for Concrete Containment Structures	4
GI-19	FFS of Containment for the Extended Operating Period	2
GI-20	Governance Implementation / Effectiveness Issues	4
GI-21	FFS of the Deaerator and the Deaerator Storage Tank for the Extended Operating Period	3

[23] Each Global Issue # entry is hyperlinked to a description of the Global Issue in Appendix B.

Global Issue #[23]	Global Issue Title	Safety Significance Level
GI-22	COP Actions Related to Aging Management from SFR4	2
GI-23	ASME N509-1980 and N510-1980 – Air Cleaning Systems	4
GI-24	Safety Analysis to Support the Extended Operating Period	2
GI-25	Category 3 CANDU Safety Issues	3
GI-26	Emergency Response Projection Software	3
GI-27	Pickering 1,4 Probabilistic Safety Assessment	3
GI-28	Reactivity Worth of Control Absorbers	3
GI-29	FFS of the Fuelling Machines and FM Bridge Ball Screws for the Extended Operating Period	2
GI-30	Evaluation of Instantaneous Risk	3
GI-31	Deterministic Safety Analysis	3
GI-32	Implementation of REGDOC-2.4.2 PSA Requirements	3
GI-33	N285.0-12, General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants	3
GI-34	CSA N290.1-13 - Requirements for the Shutdown Systems	3
GI-35	Human Factors Issues	4
GI-36	CSA N290.2 – Requirements for Emergency Core Cooling Systems	3
GI-37	N290.3-11 - Requirements for Containment System	3
GI-38	CSA N290.11 - Requirements for Reactor Heat Sinks	4
GI-39	CSA N290.14 -Qualification of Digital Hardware and Software for Use in I&C Applications	4
GI-40	Accident Management	3
GI-41	REGDOC-2.10.1 - Nuclear Emergency Preparedness and Response	4
GI-42	Examination and Testing Requirements for Design of Concrete Containment Structures	3
GI-43	Safety-Related Structures (Non-Containment) for Nuclear Power Plants	3
GI-44	REGDOC-2.5.2 - Design of Reactor Facilities: Nuclear Power	3

Global Issue #[23]	Global Issue Title	Safety Significance Level
	Plants	
GI-45	CRN Concession for Fire Protection Components	4
GI-46	Requirements of National Fire Code of Canada for Units 1 & 4 Standby Generator Fuel Tanks	4
GI-47	Fire Protection Code NFPA 24	3
GI-48	CSA N293-12 Fire Protection of Nuclear Power Plants	3
GI-49	FFS of PHT Auxiliary Piping Systems, and PHT Valves	2
GI-50	N285.4 PIP / Documentation Revision	3
GI-51	Fuelling with Pressure Tube Sag	4

11.2. Assessment of Interfaces Between the Various Safety Factors and Aggregate Impact of Global Issues

Following the consolidation of PSR2 Gaps into Global Issues (refer to Section 11.1), which is based on topical similarities between PSR2 Gaps, a further review of the Global Issues was performed to confirm that the interfaces between Safety Factors are appropriately considered in the development of the Global Issues.

The result of the aggregate assessment of Global Issues is presented in Table 16. The table shows that for a given functional topic comprising more than one Global Issue, there are no interfaces or aggregate effects that warrant a change to Global Issue descriptions or proposed Resolution Plans. Rather, in many cases the table identifies that OPG programs in place will ensure a consistent approach to the resolution of Global Issues on the same functional topic, and that lessons learned from the execution of proposed Resolution Plans will be applied to the resolution of other Global Issues within the same group. In addition, in some cases proposed Resolution Plans already cross-reference those for related Global Issues, indicating that interfaces are already explicitly considered. In other cases, Global Issues within a group are distinct and unrelated, again resulting in no aggregate effect or need to modify proposed Resolution Plans.

It is noted that many of the Global Issues are related to fitness for service and aging management of SSCs. This is fully expected and does not indicate an additional aggregate effect. Condition Assessment and aging management of SSCs is recognized by OPG as one of the key issues for Pickering NGS. Work is actively underway to update Condition Assessments and confirm assurance of fitness for service for safety-related SSCs for the extended operating period. Much of this work is already complete for 2024.

In summary, OPG's integrated management system provides a framework to ensure a consistent approach and sharing of lessons learned for Global Issues on the same functional topic, even those that originate from different Safety Factors or other sources. It is concluded that interfaces and aggregate effects of Global Issues do not result in a need to modify Global Issue descriptions or proposed Resolution Plans.

Table 16: Aggregate Impact of Global Issues

Aggregate Topic	GI #	Global Issue Title	Source of Gaps	Aggregate Assessment
Fitness for Service	GI-1	Fitness for Service for Fuel Channels	SF2, SF4, COP	For the purposes of assessing the aggregate impacts of Global Issues, the Global Issues related to fitness for service of SSCs are considered together as one aggregate. Nineteen Global Issues related to fitness for service are included in this group. Fourteen of these Global Issues are comprised of Gaps associated with Safety Factors SF1, SF2 and SF4, which are all related to "The Plant" Safety Factor Subject Area, as well as COP Review actions. Four Global Issues are comprised of COP Review actions, and one Global Issue is comprised of an Expert Panel Gap. Of the 19 Global Issues in this group, five are related to Major Components, i.e., Fuel Channels, Feeders, Steam Generators and Reactor Components and Structures. The remaining Global Issues are related to other safety related SSCs. Although these Global Issues are related to different SSCs, their proposed Resolution Plans include cross-references to other Global Issue proposed Resolution Statements, such as updating of Condition Assessments or Life Cycle Management Plans, to address interfaces where appropriate. OPG has a well-developed and effectively implemented program for continued validation of fitness for service of the Major Components, and has completed a significant amount of work related to Condition Assessments of SSCs for the extended operating period. Work is actively underway to complete Condition Assessments and confirm fitness for service for safety-related SSCs for the extended operating period. The proposed Resolution Plan actions related to fitness for service will all be completed under the umbrella of N-PROG-MA-0026, *Equipment Reliability* [47] which interfaces with N-PROG-MP-0008,
	GI-2	Fitness for Service for Feeders	SF2, COP	
	GI-3	Fitness for Service for Steam Generators	SF2, COP	
	GI-4	Fitness for Service for Reactor Components and Structures	SF2, COP	
	GI-5	Completeness of Class 1 Piping / Components Service Limits Assessment (Excluding Major Components)	SF2, COP	
	GI-6	Impact of the Revised Criticality Coding on the Cable Surveillance Program	SF2	
	GI-7	Pickering Buried Piping Fitness for the Extended Operating Period	SF1, SF2	
	GI-8	Completion / Updating of the Condition Assessments	SF2	
	GI-10	IFB Condition	SF2, SF4	
	GI-17	FFS of Fiberglass Reinforced Plastic Material for the Extended Operating Period	SF4	
	GI-19	FFS of Containment for	SF4, COP	

Aggregate Topic	GI #	Global Issue Title	Source of Gaps	Aggregate Assessment
		the Extended Operating Period		*Integrated Aging Management* [48] and N-PROG-MA-0025, *Major Components* [49] for life cycle management planning, to ensure a consistent approach and to facilitate valuable sharing of information so that lessons learned can be effectively implemented. In addition, N-CHAR-AS-0002, *Nuclear Management System* [50] provides an effective mechanism to drive excellence in fitness for service of SSCs. Therefore, the interfaces among and the aggregate impact of the Global Issues in this group are fully addressed, and no additional impacts need to be considered.
	GI-21	FFS of the Deaerator and the Deaerator Storage Tank for the Extended Operating Period	COP	
	GI-22	COP Actions Related to Aging Management from SFR4	SF4, SF2	
	GI-28	Reactivity Worth of Control Absorbers	COP	
	GI-29	FFS of the Fuelling Machines and FM Bridge Ball Screws for the Extended Operating Period	COP	
	GI-33	N285.0-12, General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants	SF1	
	GI-43	Safety-Related Structures (Non-Containment) for Nuclear Power Plants	SF1, SF2, SF4, COP	
	GI-49	FFS of PHT Auxiliary Piping Systems, and PHT Valves	COP	
	GI-51	Fuelling with Pressure Tube Sag	Expert Panel	

Aggregate Topic	GI #	Global Issue Title	Source of Gaps	Aggregate Assessment
Seismic Capacity	GI-9	Seismic Capacity of the Conveyor Tube and Fuel Basket Stacking Arrangement	SF2	This aggregate is comprised of two seismic related Global Issues. GI-9 is comprised of SF2 Gaps related to the seismic capacity of SSCs and GI-13 is comprised of SF3 Gaps and COP Review actions related to seismic qualification requirements. These Global Issues are unrelated. In addition, N-PROG-MP-0009, *Design Management* [51] is comprised of a series of engineering programs and procedures to ensure the adequacy of seismic qualification of SSCs [52]. Therefore, there is no additional aggregate impact.
	GI-13	Seismic Qualification - N289	SF3, COP	
Fuel Management	GI-11	Fuel Management and Surveillance Software Upgrade	COP	There is only one Global Issue related to fuel management and surveillance software, and therefore there is no additional aggregate impact.
Environmental Qualification	GI-12	Extending the Environmental Qualification of Equipment	SF3	There are two Global Issues, GI-12 and GI-14, related to environmental qualification of equipment that are considered as a group for the purposes of assessing the aggregate impact of Global Issues. These Global Issues are comprised of SF3 Gaps. Although these Global Issues pertain to separate Environmental Qualification issues, the work to address both of these issues will be governed by OPG's well-developed Environmental Qualification Program, N-PROG-RA-0006, *Environmental Qualification* [53], thus ensuring consistent updates of the relevant Environmental Qualification Assessment (EQA) documentation. Therefore, the aggregate impact of this group of Global Issues is fully addressed, and no additional impacts need to be considered.
	GI-14	Environmental Qualification Program Issues	SF3	
Conformance with modern Codes and Standards	GI-16	Concession Related to N285.5-M90	SF4	For the purposes of assessing the aggregate impact of Global Issues, 8 Global Issues related to conformance with safety-significant requirements of modern codes and standards are grouped together under this aggregate. There are other Global
	GI-23	ASME N509-1980	COP	

Aggregate Topic	GI #	Global Issue Title	Source of Gaps	Aggregate Assessment
		and N510-1980 – Air Cleaning Systems		Issues which pertain to modern codes and standards; however their aggregate impact is assessed in other groups according to the affected SSCs. The Global Issues in this group are comprised primarily of SF1, SF2 and SF4 Gaps, as well as SF5 and SF6 Gaps related to safety analysis, and COP Review actions. Although this group contains several Global Issues, the Global Issues are, for the most part, associated with separate and unrelated modern codes and standards or unrelated SSCs. OPG has well-developed and effectively implemented programs to ensure that the conditions of the Power Reactor Operating Licence are met. For the modern codes and standards that are not mandatory requirements of the licence or the Licence Conditions Handbook, the proposed Resolution Plans developed for the Global Issues ensure that appropriate actions are defined, or existing conditions (current practices, etc.) are adequately dispositioned. Therefore, there is no additional aggregate impact as a result of this group of Global Issues.
	GI-34	CSA N290.1-13 - Requirements for the Shutdown Systems	SF1	
	GI-36	CSA N290.2 – Requirements for Emergency Core Cooling Systems	SF1	
	GI-38	CSA N290.11 - Requirements for Reactor Heat Sinks	SF1	
	GI-39	CSA N290.14 - Qualification of Digital Hardware and Software for Use in I&C Applications	SF1	
	GI-44	REGDOC-2.5.2 - Design of Reactor Facilities: Nuclear Power Plants	SF1, SF5, SF6	
	GI-50	N285.4 PIP / Documentation Revision	SF2, SF4	
Governance Issues / Governance Implementation / Effectiveness	GI-15	Governance Issues	SF1, SF4, SF5	Two Global Issues related to governance issues or governance implementation and effectiveness, GI-15 and GI-20, are grouped together. The majority of Gaps associated with these Global Issues either require no further action or are assessed to be Acceptable Deviations. The Gaps are largely unrelated, and therefore there is no additional aggregate impact as a result of this group of Global Issues.
	GI-20	Governance Implementation / Effectiveness Issues	SF4	
Requirements for Containment	GI-18	N287.7 – In-Service Examination and Testing Requirements for	SF4	Two Global Issues related to requirements for Concrete Containment Structures, GI-18 and GI-42, are considered together as an aggregate. These Global Issues are

Aggregate Topic	GI #	Global Issue Title	Source of Gaps	Aggregate Assessment
		Concrete Containment Structures		comprised of SF1 and SF4 Gaps. The Gaps associated with GI-18 and GI-42 are not directly related. GI-18 relates to the requirements of CSA N287.7, "In-service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants", which is applicable to the operations phase. GI-42 relates to the requirements of CSA N287.5 "Examination and Testing Requirements for Concrete Containment Structures for Nuclear Power Plants", which is applicable to the design and construction phase. The proposed Resolution Plans for these Global Issues either require no further action or are assessed as Acceptable Deviations. Therefore, there is no aggregate impact for these Global Issues.
	GI-42	Examination and Testing Requirements for Design of Concrete Containment Structures	SF1	
Deterministic Safety Analysis	GI-24	Safety Analysis to Support the Extended Operating Period	SF5, COP	Three Global Issues, GI-24, GI-25 and GI-31, are considered as an aggregate related to deterministic safety analysis. Although GI-24 is also related to fitness for service, it is considered as part of this aggregate as it is mainly related to the impact of aging components on the safety analysis. Some of the Gaps associated with GI-25 and GI-31 are related to SF1, Plant Design, but they pertain to safety analysis issues. This aggregate contains only three Global Issues, and the issues are not related; GI-24 is related to the impact of aging components on the safety analysis, GI-25 is related to re-categorization of CANDU safety issues from Category 3 to Category 2, and GI-31 is related to the implementation of REGDOC-2.4.1 on deterministic safety analysis. Therefore, there is no additional aggregate impact.
	GI-25	Category 3 CANDU Safety Issues	SF1, SF5, SF7, COP	
	GI-31	Deterministic Safety Analysis	SF1, SF5, COP	
Emergency Preparedness / Response	GI-26	Emergency Response Projection Software	SF13	There are two Global Issues, GI-26 and GI-41, related to Emergency Preparedness / Emergency Response. These Global Issues are comprised of SF13 Gaps which are related to the "Management" Safety Factor Subject Area. Although both Global Issues are concerned
	GI-41	REGDOC-2.10.1 - Nuclear Emergency	SF13	

Aggregate Topic	GI #	Global Issue Title	Source of Gaps	Aggregate Assessment
		Preparedness and Response		with emergency preparedness and emergency response, the associated Gaps are not related. Therefore, there is no additional aggregate impact.
Probabilistic Safety Assessment	GI-27	Pickering 1,4 Probabilistic Safety Assessment	SF1, SF6, FAI	Three Global Issues, GI-27, GI-30 and GI-32 are related to Probabilistic Safety Assessment. In addition, GI-37 and GI-40, which are each comprised of one SF1 Gap and one SF1 Additional Gap, are considered in relation to PSA for aggregation purposes since they are related to proposed risk improvement initiatives and Phase 2 of the Emergency Mitigating Equipment (EME) Project. Gap SF6-5 associated with GI-44 is related to requirements for PSA risk limits, such as a Core Damage Frequency limit of less than 10^{-5}/y. Gap SF6-5 is assessed as an Acceptable Deviation. GI-44 is related to conformance with REGDOC-2.5.2, *Design of Reactor Facilities: Nuclear Power Plants*, and is therefore grouped under the aggregate associated with conformance with modern codes and standards. The Gaps associated with these Global Issues are from SF1, SF6 and the FAI Review. The Gaps associated with GI-30 and GI-32 are stand-alone Gaps that are not related to each other or to other Gaps in this group. PSA activities are governed by OPG's Risk and Reliability Program [54], which ensures that the reliability of systems important to nuclear safety meets requirements and that PSA risk goals are met. The resolutions of GI-27, GI-37 and GI-40 are synergistic in that in aggregate, they enhance Containment integrity for Beyond Design Basis Accidents. The specific cross references are GI-27-XRF-GI-40-RS1, GI-37-XRF-GI-40-RS1 and GI-40-XRF-GI-27-RS2. In summary, any aggregate impacts are already accounted for in the proposed Resolution Plans for these Global Issues
	GI-30	Evaluation of Instantaneous Risk	SF6	
	GI-32	Implementation of REGDOC-2.4.2 PSA Requirements	SF6	
	GI-37	N290.3-11 - Requirements for Containment System	SF1	
	GI-40	Accident Management	SF1	

Aggregate Topic	GI #	Global Issue Title	Source of Gaps	Aggregate Assessment
				and there is no additional aggregate impact.
Human Factors	GI-35	Human Factors Issues	SF1	There is one SF1 Gap associated with this Global Issue. There are no other Global Issues related specifically to Human Factors, and therefore, there is no additional aggregate impact.
Fire Protection	GI-45	CRN Concession for Fire Protection Components	SF1	There are four Global Issues related to fire protection, GI-45, GI-46, GI-47 and GI-48. These Global Issues are comprised of SF1 Gaps, except for GI-46 which is comprised of a COP Review action. OPG has a comprehensive Fire Protection Program, N-PROG-RA-0012, *Fire Protection* [55] that is based on the requirements of CSA N293, *Fire Protection for CANDU Nuclear Power Plants*, the National Fire Code of Canada and NFPA 24, and other applicable codes and standards. The Global Issues in this group are related to conformance with these fire codes and standards but are distinct issues. Therefore, this group of Global Issues has no additional aggregate impact.
	GI-46	Requirements of National Fire Code of Canada for Units 1 & 4 Standby Generator Fuel Tanks	COP	
	GI-47	Fire Protection Code NFPA 24	SF1	
	GI-48	CSA N293-12 Fire Protection of Nuclear Power Plants	SF1	

12. Prioritization of Global Issues

The PSR2 Global Issues are prioritized with respect to their overall impact on enhancing nuclear safety, using the process outlined in Section 5.3. This step is the fourth element of the Global Assessment process, as described in Section 3.3.2 of the PSR2 Basis Document [1].

As described in Section 5.3, the basis for prioritization of each Global Issue is provided in Appendices E and F of the PSR2 Basis Document [1], and comprises Deterministic and Probabilistic considerations.

The Deterministic considerations are:

- Defence in Depth (E1)
- Safety Significance Levels (E2)

Using the guidelines provided in Appendix E of the PSR2 Basis Document [1], a Safety Significance Level of 1, 2, 3 or 4 is assigned to each Deterministic consideration based on whether the Global Issue has a high, medium, low or very low impact on nuclear safety for the consideration being evaluated. A Safety Significance Level of 1, 2, 3 or 4 is then assigned to the overall Deterministic consideration based on the most safety significant result. For Deterministic considerations that are not relevant to the Global Issue, the prioritization is recorded as “N/A” or “Not Applicable”.

There are 7 Probabilistic considerations, as follows:

- Reactor Safety Core Damage Frequency (F1)
- Reactor Safety Defence in Depth (F2)
- Public Radiation Safety (F3)
- Plant Operability (F4)
- Occupational Radiation Safety (F5)
- Emergency Preparedness (F6)
- Environment (F7)

Using the guidelines provided in Appendix F of the PSR2 Basis Document [1], a Safety Significance Level of 1, 2, 3 or 4 is assigned to each Probabilistic consideration based on whether the Global Issue has a high, medium, low or very low impact on nuclear safety for the consideration being evaluated. A Safety Significance Level of 1, 2, 3 or 4 is then assigned to the overall Probabilistic consideration based on the most safety significant result. For Probabilistic considerations that are not relevant to the Global Issue, the prioritization is recorded as “N/A” or “Not Applicable”.

The overall Safety Significance Level for the Global Issue is then assigned based on the Safety Significance Level of whichever overall consideration, Deterministic or Probabilistic, has the highest nuclear safety impact.

The results of the prioritization of the Global Issues, including the Safety Significance Level assigned to each Global Issue and the accompanying rationale, are provided in Section 4 of each Global Issue Table in Appendix B, using the prioritization template shown below.

SECTION 4 – GLOBAL ISSUE PRIORITY DETERMINATION												
Safety Significance of Global Issue	**Deterministic Considerations**			**Probabilistic Considerations**								**Overall Safety Significance Level**
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
<u>Rationale:</u> {This part of the prioritization template presents a brief description of the Global Issue and describes the impact of the Global Issue on each of the individual Deterministic and Probabilistic considerations, as well as the impact on the overall considerations. The overall Safety Significance Level is also summarized.}												

The results of the prioritization of the Global Issues are shown in Table 15 and summarized below.

Three of the 51 Global Issues are assessed as having a high impact on nuclear safety and are accordingly assigned Safety Significance Level 1. These are Global Issues GI-1, GI-2 and GI-3, which are all related to fitness for service of Major Components for the extended operating period. Their proposed Resolution Plans are considered the most important in terms of supporting the safe operation of Pickering NGS for the extended operation period. The Gaps associated with these Global Issues are related to Safety Factor 2 and Safety Factor 4, as well as actions from the COP. OPG was already fully aware of the need to complete work related to demonstration of fitness for service over the extended operating period, and is actively working on addressing the remaining work in this area to cover the extended operating period.

Nine Global Issues are assessed as having a medium impact on nuclear safety, and as such are assigned Safety Significance Level 2. These Global Issues are related to aging management (GI-22), safety analysis (GI-24), and fitness for service of Reactor Structures (GI-

4) and SSCs other than Major Components (GI-5, GI-8, GI-17, GI-19, GI-29 and GI-49). Resolution of these Global Issues will support the safe operation of Pickering NGS for the extended operation period. The Gaps associated with these Global Issues are related to Safety Factor 2, Safety Factor 4, Safety Factor 5 and the COP.

Twenty-four Global Issues are assessed as having a low impact on nuclear safety and are accordingly prioritized as Safety Significance Level 3. These Global Issues are mostly related to specific requirement elements of modern codes and standards. The justification for considering these as low nuclear safety impact is provided in Section 4 of each Global Issue Table in Appendix B.

Fifteen Global Issues are assessed as having a very low impact on nuclear safety, and are accordingly prioritized as Safety Significance Level 4. The majority of these Global Issues are related to OPG governance or specific requirement elements of modern codes and standards, or are administrative in nature. The justification for considering these Global Issues as very low nuclear safety impact is also detailed in Section 4 of each Global Issue Table in Appendix B.

13. Development of Proposed Resolution Plans

Proposed Resolution Plans have been developed for all Global Issues following the methodology described in Section 5.4, and are presented in Section 5 of each Global Issue Table in Appendix B. Each Global Issue table includes the following elements relevant to the proposed Resolution Plan:

i). Background Information and Resolution Strategy presented in Section 3 of each Global Issue table, with an evaluation of the Global Issue describing the nature of the associated gaps and a summary of the status of any work already underway or completed to address the Global Issue.

ii). A proposed Global Issue Resolution Plan presented in Section 5 of each Global Issue table, and comprised of one or more of the following elements:

- Proposed Resolution Statements (RS): A proposed activity is defined to address the Gap(s).
- No Further Action (NFA): Work is completed or will be done outside of PSR2 to address the related Gap(s), or information has been found to obviate the Gap(s).
- Acceptable Deviation (AD): The deviation in the Gap(s) has been assessed to have a Very Low Safety Significance Level, or a Low Safety Significance Level and a practicable resolution is not readily evident.
- Cross Reference (XRF): An action that addresses the Gap(s) is covered by a proposed Resolution Statement under a different Global Issue.

A proposed Resolution Plan element may address more than one Gap, or portions of more than one Gap. A Gap may be addressed by more than one proposed Resolution Plan element.

In developing proposed Resolution Plans for the 51 Global Issues, each of the Gaps identified in Part II from the reviews of the Safety Factor Reports, Complementary Reviews and CNSC staff reviews, is addressed, as well as the Expert Panel Gap identified in Section 15. In total, 35 proposed Resolution Statements are identified.

As discussed in Section 5.4, development of the proposed Resolution Plans considers whether the proposed resolution activities could be different for operation to 2024 (the planning basis for the units), or beyond 2024. Table 17 summarizes the Global Issues identified as being potentially impacted if a decision is made to extend operation of Pickering NGS beyond 2024. These Global Issues are largely related to fitness for service, aging or Condition Assessments of SSCs. If a decision is made to operate beyond 2024, these Global Issues would be re-evaluated and their proposed Resolution Plans would be updated if/as appropriate.

Table 17: Global Issues Potentially Impacted by Pickering NGS Operation Beyond 2024

Global Issue #	Global Issue Title
GI-1	Fitness for Service for Fuel Channels
GI-2	Fitness for Service for Feeders
GI-3	Fitness for Service for Steam Generators
GI-4	Fitness for Service for Reactor Components and Structures
GI-5	Completeness of Class 1 Piping/Components Service Limits Assessment (Excluding Major Components)
GI-7	Pickering Buried Piping Fitness for the Extended Operating Period
GI-8	Completion / Updating of the Condition Assessments
GI-10	IFB Condition
GI-12	Extending the Environmental Qualification of Equipment
GI-17	FFS of Fiberglass Reinforced Plastic Material for the Extended Operating Period
GI-19	FFS of Containment for the Extended Operating Period
GI-21	FFS of the Deaerator and the Deaerator Storage Tank for the Extended Operating Period
GI-22	COP Actions Related to Aging Management from SFR4
GI-24	Safety Analysis to Support the Extended Operating Period
GI-29	FFS of the Fuelling Machines and FM Bridge Ball Screws for the Extended Operating Period
GI-31	Deterministic Safety Analysis
GI-32	Implementation of REGDOC-2.4.2 PSA Requirements
GI-33	N285.0-12, General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants
GI-43	Safety-Related Structures (Non-Containment) for Nuclear Power Plants
GI-49	FFS of PHT Auxiliary Piping Systems, and PHT Valves
GI-51	Fuelling with Pressure Tube Sag

The proposed Global Issue Resolution Statements presented in Section 5 of each Global Issue Table in Appendix B are inputs to the Integrated Implementation Plan phase of PSR2.

14. Ranking of Proposed Resolution Statements

Proposed Global Issue Resolution Statements presented in Section 5 of the Global Issue Tables in Appendix B are ranked using the methodology and Value Tree technique described in Section 5.5 and Appendix C. Acceptable Deviations and No Further Action statements do not go through the ranking process; only proposed Resolution Statements are ranked.

The results of ranking the proposed Global Issue Resolution Statements are summarized in Table 18 and detailed in Appendix C.

The proposed Global Issue Resolution Statements are sorted in ranked order, with the highest ranking first, in Table 18. Consistent with the methodology described in Section C.7, proposed Resolution Statements with the same ranking values are further sorted in order of those judged to be most important to the overall objective. For example, there are 5 proposed Resolution Statements with a ranking value of 100. The proposed Resolution Statement associated with Fuel Channel life cycle management (GI-1-RS3) is judged to be the most important among this group, whereas others such as measurements of the Calandria Tube/Liquid Injection Shutdown System nozzle gaps on Units 5-8 (GI-4-RS2) are judged to be of relatively lesser importance.

Table 18: Overall Ranking Results

GI-RS #	Proposed Resolution Statement Title	Normalized Ranking Value
GI-1-RS3	Update the Fuel Channels LCMP for Pickering 1,4 for the extended operating period.	100
GI-2-RS1	Update the Feeders Life Cycle Management Plan for Pickering 1,4 for the extended operating period based on updated fitness for service assessment.	100
GI-3-RS1	Update the Steam Generators Life Cycle Management Plan for Pickering 1,4 for the extended operating period based on updated fitness for service assessment.	100
GI-4-RS1	Update the Reactor Components and Structures Life Cycle Management Plan for Pickering 1,4 for the extended operating period based on updated fitness for service assessment.	100
GI-4-RS2	Perform measurements of Calandria Tube/Liquid Injection Shutdown System nozzle gaps on	100

GI-RS #	Proposed Resolution Statement Title	Normalized Ranking Value
	Units 5-8 to refine the gap closure rates. Using this new measurement data, update analyses as required, to demonstrate Fitness for Service.	
GI-24-RS1	Update Heat Transport System aging safety analysis models and perform the required safety analysis of events most impacted by aging (Small Break Loss of Coolant Accident (SBLOCA), Loss of Flow (LOF) and Neutron Overpower (NOP)) to support extended operation.	72
GI-27-RS2	Investigate and implement additional practicable design, operational and/or analytical enhancements to further improve Pickering 1,4 Severe Core Damage Frequency and Large Release Frequency (e.g., alternative emergency cooling water makeup).	72
GI-1-RS1	Complete CSA N285.8 Compliance Plan activities, including responding to comments.	70
GI-1-RS2	Review and revise if/as required the CSA N285.4 compliant Periodic Inspection Plans for Fuel Channels for Pickering NGS to cover the extended operating period.	70
GI-1-RS4	Update the structure of the Fuel Channels LCMP.	70
GI-5-RS1	Confirm the adequacy of the service limits assessments for Nuclear Class 1 Piping (Excluding Major Components) after accounting for any impact of environmental factors	70
GI-40-RS1	Ensure the completion of EME Phase 2 activities.	59
GI-48-RS1	Provide, as necessary, design and/or operational changes and commissioning/testing to facilitate required interconnection of Pickering 1,4 and Pickering 5-8 Fire Protection System water supplies to meet the safety intent of CSA N293-12 Clause 7.3.2.2 (d).	59

GI-RS #	Proposed Resolution Statement Title	Normalized Ranking Value
GI-8-RS1	Complete and update CAs for the piping systems and commodity groups in PSR2 scope for station operation for the extended operating period.	46
GI-8-RS2	Develop and implement a process to track and report aging-management-related actions from the Condition Assessment recommendations.	46
GI-10-RS1	Complete the Pickering 5-8 IFB Leakage Mitigation Project to mitigate leaks from IFB-B to the interspace.	46
GI-12-RS1	Complete EQA re-assessments to support the extended operating period.	46
GI-19-RS1	Demonstrate the FFS of the foundation steel H-piles for the Pickering A Reactor Building, Vacuum Building and Pressure Relief Duct at the Pickering site for the extended operating period.	46
GI-43-RS1	Perform the scope of inspections for non-Containment safety-significant civil structures as per the established Preventive Maintenance program (PM 00121151).	46
GI-26-RS1	Complete the emergency response projection enhancements identified in Action Item 2016-OPG-7469: Implementation of Emergency Response Projection Computer Code Upgrades.	38
GI-9-RS1	Complete the required assessment to support the current fuel basket stacking arrangements in the Pickering IFBs.	33
GI-25-RS2	Complete the re-categorization of CANDU Safety Issue CSI-IH6 for Pickering to Category 2. (Pickering 1,4 high-energy piping)	33
GI-27-RS1	Complete actions from PSA improvement Plan.	33

GI-RS #	Proposed Resolution Statement Title	Normalized Ranking Value
GI-6-RS1	Reassess the impact of the changes in the cable Criticality Coding and update the scope of the cable surveillance plan.	31
GI-7-RS1	Update the Buried Piping Program asset management plan and risk ranking for the extended operating period.	31
GI-43-RS3	Prepare Condition Assessments as appropriate for safety-significant civil structures for the extended operating period. Recommendations from these Condition Assessments will be tracked and reported along with those related to GI-8. This applies to non-Containment Safety-Related Civil Structures.	31
GI-47-RS1	Complete installation of locks on the 058 Yard Fire Protection System.	27
GI-43-RS2	Develop program governance using a risk based approach for aging management of safety-significant civil structures for the extended operating period. This applies to non-Containment Safety-Related Civil Structures.	26
GI-50-RS2	Assess the impact of extended operation on concessions against CSA N285.4	26
GI-31-RS1	Complete the Pickering NGS Implementation Plan for REGDOC-2.4.1.	22
GI-31-RS2	Prepare Implementation Plan update for REGDOC-2.4.1 including consideration of the impact of the extended operating period.	22
GI-32-RS1	Complete the activities in the REGDOC-2.4.2 Implementation Strategy and update the Strategy in the context of the additional operating period.	22
GI-50-RS1	Revise the N285.4 PIPs and governance to align with elements of N285.4-14	21

GI-RS #	Proposed Resolution Statement Title	Normalized Ranking Value
GI-25-RS1	Complete the re-categorization of the Large Break LOCA (LBLOCA) CANDU Safety Issues to Category 2.	19
GI-7-RS2	Update governance to reflect a graded approach in the event that leakage in fuel oil piping occurs.	5

The top five ranked proposed Resolution Statements in Table 18 contribute to meeting the fundamental objective of enhanced confidence in the fitness for service of SSCs (refer to Section 5.5 for the Value Tree fundamental objectives). These proposed Resolution Statements are determined to have the highest normalized ranking value of 100, and are associated with Global Issues related to fitness for service of Major Components, GI-1, GI-2, GI-3, and GI-4. Fitness for service of Pickering NGS SSCs is a key issue for OPG, and is being managed under the comprehensive programs that OPG has in place, including N-PROG-MA-0026, *Equipment Reliability* [47], N-PROG-MP-0008, *Integrated Aging Management* [48], and N-PROG-MA-0025, *Major Components* [49]. These programs ensure the condition of SSCs important to safety is well understood, the level of fitness for service is assessed, and effective actions are taken to maintain good plant condition. Work is actively underway to complete Condition Assessments and confirm assurance of fitness for service for safety-related SSCs for the extended operating period, and much of this work is already complete for 2024.

The next two highest ranked proposed Resolution Statements in Table 18 contribute to meeting the fundamental objective of enhanced confidence in the safety analyses. These are determined to have a normalized ranking value of 72 and are associated with Global Issues GI-24, which is related to safety analysis to support the extended operating period, and GI-27, which is related to Pickering 1,4 PSA. OPG has processes in place to ensure that aging effects associated with extended operation are accounted for in safety analysis, and the proposed Resolution Statement associated with GI-24 reflects the planned safety analysis to support Pickering NGS extended operation. The Pickering NGS PSA shows that the OPG risk-based Safety Goals for Core Damage Frequency and Large Release Frequency are met. The proposed Resolution Statements associated with GI-27 will enhance margins to these goals with emphasis on Pickering 1,4, and will consider physical and/or operational enhancements, in addition to analytical enhancements.

The proposed Resolution Statements with normalized ranking values less than 72 contribute, to a lesser extent, to meeting the fundamental objectives. All of the proposed Resolution Statements shown in Table 18 are inputs to the Integrated Implementation Plan phase of PSR2, and the Integrated Implementation Plan considers the ranking order shown in Table 18.

15. Pickering PSR2 Expert Panel

15.1. Overview

OPG established a third party technical Expert Panel to support the PSR2 Global Assessment process. The Expert Panel is comprised of experienced individuals who are familiar with the design and operation of Pickering NGS (and other nuclear plants) and have demonstrated leadership in the Nuclear Industry, participating in external review committees and initiatives.

The Expert Panel was given a broad mandate in terms of its activities, with its primary objective to provide guidance and counsel to both the Global Assessment Team and to OPG Pickering staff developing PSR2. The framework for these activities was based on the PSR2 Basis Document [1] and was further informed by, and was consistent with, the appropriate regulatory requirements [2] and industry guidance [3], [7], [8], [9].

The Expert Panel provided third party review of elements of the Global Assessment Report as they were being developed and subsequently of the Global Assessment Report itself, with support from other experienced individuals on an as-needed basis. The comments from the Expert Panel were reviewed and dispositioned by the Global Assessment Team and any findings were communicated to OPG for consideration in the Global Assessment Report.

The Expert Panel members, and their qualifications and experience, are discussed in Section 15.3.

15.2. Mandate

The Expert Panel completed specific tasks (e.g., document review) within the scope of the Pickering NGS Global Assessment Project as agreed to between OPG and the Global Assessment Team. Once assigned a task, the Expert Panel also provided third party technical feedback and counsel as appropriate to the Global Assessment Team or OPG. The Expert Panel was expected to identify and undertake additional tasks it deemed to be within the scope of its mandate, such as to identify additional review issues for consideration by the Global Assessment Team.

The Expert Panel focused on the Global Assessment Report, including:

1. The development of the Global Issues and Resolution Plans
2. The Global Issue Prioritization process
3. The Global Issue Resolution Statement ranking process
4. The defence-in-depth review
5. The overall justification for continued operation

Typically, the findings from the Expert Panel reviews were documented and discussed with the Global Assessment Team to come to an agreement on the dispositions. The agreed dispositions are incorporated into the Global Assessment Report.

Contact and communication with the Global Assessment Team and OPG was maintained during the Global Assessment Report development through both teleconferences and face to face meetings.

15.3. Members, Qualification and Expertise

The qualifications of the Expert Panel are given below.

J. Craig Sellers, B.A.Sc. (Chemical Engineering), P.Eng – Panel Chair

> Craig Sellers has over 37 years of extensive experience in the field of nuclear generation including positions as Commissioning Engineer, Authorized Shift Supervisor, Engineering Manager, Maintenance Manager, Senior Manager (Plant Design), and Director, Supply Planning. Craig retired in 2009 from the role of Chief Nuclear Engineer, OPG.
>
> Post retirement, Craig has filled numerous executive level positions including Vice President and Chief Engineer RCM Technologies and has actively participated in a variety of external activities with CSA, IAEA, and WANO/INPO. He is currently a member of the Bruce Power Nuclear Safety Review Board.

Dr. Keith C. Garel, PhD (Nuclear Engineering)

> Dr. Keith Garel has over 35 years of experience in the nuclear power industry including 25 years of nuclear safety and licensing experience where he performed a variety of functions including Safety Analysis Engineer, Technical Superintendent of Safety and Licensing at Pickering NGS, and Senior Manager – Operational Licensing – Nuclear Regulatory Affairs.
>
> Dr. Garel also possesses in excess of 10 years of experience performing engineering management functions that include managing engineering departments of disciplines varying amongst stress analysis, components and equipment, and project management of destiny projects for OPG as Project Manager Feeder Integrity Projects and the OPG Fuel Channel Life Management Project.
>
> Dr. Garel currently consults in a variety of areas including project management and project oversight, as well as managing task teams focused on resolving safety and licensing issues.

Michael K. O’Neill, B.Sc., M.Sc. (Physics), P.Phys, P.Eng

> Mike O’Neill has over 35 years of experience in nuclear engineering and operational support, including positions as Nuclear Safety Analysis Manager, Reactor Physics Manager and Reactor Safety Manager. Mike retired in 2011 from the role of Manager, Nuclear Safety and Technology, OPG.

Post retirement, Mike was heavily involved in OPG's post-Fukushima follow-up activities and currently is providing technical support in a variety of areas. Mike is the Chair of the CSA N290B Technical Committee on Reactor Safety and Risk Management. The scope of the N290B Technical Committee includes Periodic Safety Review, Probabilistic Safety Assessment, and BDBA standards.

15.4. Activities

The list of activities completed by the Expert Panel is given in Table 19.

Table 19: Expert Panel Activities Completed

Date	Description	Accountable	Status
Aug 2016	Review defence-in-depth methodology	Expert Panel	Complete
Sep 2016	Review Interim #1 Global Issue Development Methodology and proposed Resolution Plans[24]	Expert Panel	Complete
Oct 2016	Conduct Pairwise Comparison & Review Ranking Methodology	Expert Panel and Global Assessment Team	Complete
Nov 2016	Review Interim #2 Global Issue Development Methodology and proposed Resolution Plans	Expert Panel	Complete
Dec 2016	Review Interim #3 Global Issue Development Methodology and proposed Resolution Plans	Expert Panel	Complete
Feb 2017	Review Interim #4 Global Issue Development Methodology and proposed Resolution Plans	Expert Panel	Complete
Mar 2017	Review Interim Defence-in-Depth Assessment Rev 0	Expert Panel	Complete
Mar 2017	Review Preliminary Global Assessment Report (early draft)	Expert Panel	Complete
Apr 2017	Review Interim #5 Global Issue Development Methodology and proposed Resolution Plans	Expert Panel	Complete
May 2017	Review of the Pickering NGS PSR2 Strengths report	Expert Panel	Complete

[24] Global Issue Resolution Plans were developed iteratively as the Safety Factor Reports, Complementary Reviews and feedback from CNSC staff became available.

Date	Description	Accountable	Status
May – September 2017	Review draft Global Assessment Report	Expert Panel	Complete

Review of Global Assessment Methodologies

The Expert Panel reviewed all aspects of the Global Assessment Methodology.

Review of Global Issues

The Expert Panel provided technical advice and counsel on the development of the Global Issues. This included:

- Characterization of Global Issues
- Prioritization of Global Issues
- Development of Resolution Plans (Resolution Statements and Acceptable Deviations)
- Ranking of Global Issue Resolution Statements

Review of the Defence-in-Depth Assessment

The Expert Panel reviewed the defence-in-depth documentation, including refinement of Strengths and consideration of the aggregate impacts of Acceptable Deviations.

15.5. Expert Panel Findings

The PSR2 Expert Panel performed a detailed review of this version of the Global Assessment Report and supporting documents. The findings are as follows:

- The Global Assessment Report has been prepared in accordance with the PSR2 Basis Document [1], and in a manner consistent with regulatory requirements [2] and industry guidance [3], [7], [8], [9].
- The methodologies used to develop the Global Assessment Report are reasonable and aligned with previous Global Assessment Reports and international practices.
- One Expert Panel Gap, EP-1, is identified related to the impact of Pressure Tube sag on the ability to fuel the Fuel Channels for the period of extended operation.
- The identification of Global Issues is consistent with the Gaps identified in the PSR2 Safety Factor Reports and Complementary Reviews. The Global Issues are also consistent with the Expert Panel's knowledge of Pickering NGS and experience with Pickering NGS design and operation.
- The disposition of the Global Issues is reasonable, and the determination of "Resolution Statement", "No Further Action", or "Acceptable Deviation" is consistent with the PSR2 Basis Document and regulatory/international practices. The dispositions are also

aligned with the Expert Panel's knowledge of Pickering NGS and experience with Pickering NGS design and operation.

- The Defence-in-Depth Assessment accurately assesses the current provisions for defence-in-depth. The Defence-in-Depth Assessment also identifies the contribution of the enhancements proposed in the Resolution Statements.

The overall assessment of the Expert Panel is that this version of the PSR2 Global Assessment Report provides a balanced and accurate evaluation of Pickering station design, operation and supporting activities. The Expert Panel concurs with the overall conclusion of the Global Assessment Report, which is that the current plant design, operation, processes and management system will ensure continued safe operation of Pickering Units 1,4 and 5-8 both in the short term, and for extended operation.

Part IV: Assessment of Overall Acceptability of Operation of the Plant

Section	
#	**Title**
16	Pickering NGS Strengths Identified in PSR2
17	Acceptable Deviations Identified in PSR1
18	Defence-in-Depth Assessment
19	Conclusion of the Assessment of Overall Acceptability of Operation of the Plant

Appendix
Appendix D – Review of Safety Principles
Appendix E – Strengths Used in the Defence-in-Depth Assessment
Appendix F – Proposed Global Issue Resolution Statement Summaries
Appendix G – Grouping of PSR1 Acceptable Deviations
Appendix H – Aggregation of Acceptable Deviations by Defence-in-Depth Level

16. Pickering NGS Strengths Identified in PSR2

This section describes the identification of Strengths in Pickering NGS design, operations and performance. The Pickering NGS Strengths are used in the Defence-in-Depth Assessment in Section 18 to demonstrate the "extent to which the safety requirements of defence in depth are fulfilled", as required by CNSC REGDOC-2.3.3 [2], and to support mitigation of Global Issues.

The Defence-in-Depth Assessment (Section 18) also includes a discussion of Pickering NGS design features that support defence-in-depth and evaluates the proposed Pickering NGS enhancements in the Global Issue Resolution Statements (refer to Section 13) and Pickering NGS Strengths (presented in this section) for their contribution to each level of defence.

CNSC REGDOC-2.3.3 [2] defines strengths as current practices that are "equivalent to or better than those established in modern codes and standards, practices". Section 3.1 of the PSR2 Basis Document [1] states that from Safety Factor reviews:

> "The Compliances (and groups of Compliances) are taken into the Global Assessment for consideration as Strengths."

Section 3.2.3 of the PSR2 Basis Document [1] states that:

> "Compliances that are equivalent to or surpass PSR2 Assessment Basis requirements or practices will be forwarded into the Global Assessment process for consideration as Strengths."

Therefore, positive findings in PSR2 are identified as possible strengths if there is clear evidence that Pickering NGS and/or OPG programs are equivalent to or surpass conformance with the provisions of modern requirements and practices or review task objectives.

Pickering NGS Strengths are identified by the Global Assessment team through a review of the following sources:

- **Safety Factor Reports**

  Safety Factor reviews provide assessments for all aspects important to the safety of an operating nuclear power plant. There are 15 Safety Factors used in the PSR2 review; 14 are identified in IAEA SSG-25 [3], and one additional Safety Factor (Radiation Protection) is identified in CNSC REGDOC-2.3.3 [2].

  Section 4.4 of each Safety Factor Report ([11] to [14] and [16] to [26]) was reviewed for:

  - Explicit statements of strength or discussions in the review task assessments that indicate a strength (e.g., program(s) relevant to the review task that are fully developed and implemented may indicate a strength).
  - Explicit statements of strength or discussions in the program effectiveness assessments (summarized in Appendix B of each Safety Factor Report) that may indicate a strength (e.g., no safety significant issues identified against the programs could be an indication of a strength).

- **Codes and Standards Assessments**

  The PSR2 assessments of modern codes and standards include a history of compliance assessments, including a summary of results of PSR1 assessments and their applicability to PSR2, and where appropriate, an updated assessment specifically for PSR2 ([56] through [59]).

  The assessments of codes and standards were reviewed to identify those with positive findings (i.e., conformance with safety-significant requirements). Significant positive findings may be an indication of a strength. Codes and standards that address the same or similar topics were assessed for strengths as a group.

- **Complementary Reviews**

  The Pickering B COP [5] actions and the FAIs [6] were reviewed to support identification of strengths.

- **Independent Third Party Assessments**

  Additional insights from third-party assessments (e.g., CNSC's Regulatory Oversight Report for 2015 [60] and assessments/reviews by international organizations) of OPG/Pickering NGS programs were reviewed for indications of strength. While these assessment reports are not used as explicit references for identifying specific strengths, they are useful for indicating areas of station operation with candidate strengths and may confirm insights from other sources. Where the reports imply 'met or exceeded performance objectives' or rated a Safety and Control Area as Fully Satisfactory for Pickering NGS, evidence of a strength was sought within other OPG documentation (e.g., performance indicator trends).

- **CNSC Feedback on PSR2**

  CNSC staff feedback from their review of the PSR2 submissions ([34] through [43]) was assessed for explicit statements of strength or discussions that may indicate a strength.

Where an individual source provides an indication of a strength, the reviews of other sources are used to confirm or contradict the indication. This approach precludes over-reliance on any single source. An additional step involved seeking input from staff (PSR2 project internal as well as external subject matter experts in broad areas) for potential areas of strength. The methodology and the list of Strengths were reviewed by the Pickering NGS Global Assessment Expert Panel and other expert reviewers with extensive knowledge of the Pickering NGS PSR2 project and design/operation of the Pickering NGS, to confirm that the results presented are reasonable and representative. This process resulted in refinements of the Strengths. Overall, the process is robust and provides a list of Strengths that is supported by references and by specific Pickering NGS experience.

A total of 24 Strengths are identified for Pickering NGS, as listed in Table 20 and described in Appendix E. For each Strength identified, a rationale for its inclusion is provided in Appendix E, as well as the source(s) containing supporting information.

Table 20: Strengths Identified for Pickering NGS

Strength ID	Strength Title and Description
S-01	Management
S-02	Effective Equipment Reliability Program
S-03	Major Components Program
S-04	System and Component Health Reporting
S-05	Implementation of Fukushima Action Items
S-06	Deterministic Safety Analysis
S-07	Probabilistic Safety Assessment
S-08	Operationalization of Probabilistic Safety Assessment
S-09	Implementation of Safe Operating Envelope Program
S-10	Healthy Safety Culture
S-11	Relationship with Stakeholders and Public
S-12	Dose to Public
S-13	Radiation Exposure Performance
S-14	Heat Removal Systems
S-15	Electrical Power System
S-16	Human Factors Engineering Program
S-17	Environmental Qualification Program
S-18	Comprehensive Set of Performance Indicators
S-19	Use of Operating Experience and Research Findings
S-20	Minimum Staff Complement Management

Strength ID	Strength Title and Description
S-21	Training
S-22	Emergency Management
S-23	Environmental Protection Program
S-24	Advanced Technology to Support Radiation Protection

17. Acceptable Deviations Identified in PSR1

The aggregate impacts of Acceptable Deviations from PSR1 and PSR2 are assessed as part of the Defence-in-Depth Assessment methodology (described in Section 5.7). The PSR1 Acceptable Deviations are described in this section. The process for identifying PSR2 Acceptable Deviations during the development of proposed Resolutions Plans is presented in Section 13.

The 274 Acceptable Deviations from PSR1 [61] were assessed during the preparation of the Safety Factor Reports, to determine whether any are impacted by Pickering NGS operation beyond 2020. Accordingly, 34 of the PSR1 Acceptable Deviations were assessed in Reference [61] to be Gaps for PSR2. As these are already addressed in the PSR2 assessment, they are not included in the PSR1 Acceptable Deviation aggregate assessment described in this section. The remaining 240 PSR1 Acceptable Deviations, plus two additional PSR1 Acceptable Deviations identified during the PSR2 Global Assessment, are identified in Appendix G.

To facilitate the aggregate assessment, the PSR1 Acceptable Deviations are grouped according to topic, as shown in Appendix G. PSR1 Acceptable Deviations that are assessed as not having an impact, such as legacy document issues or minor document revisions, are identified as PSR1-AD-DOC and are not assessed for aggregate effects in the Defence-in-Depth Assessment in Section 18, as there is deemed to be no aggregate impact for these Acceptable Deviations. Thirty-three PSR1 Acceptable Deviations are assessed as PSR1-AD-DOC. PSR1 Acceptable Deviations that have already been addressed or that are not applicable to Pickering NGS are identified as PSR1-AD-NFA (No Further Action) and are not assessed for aggregate effects in the Defence-in-Depth Assessment in Section 18. Fifty-eight PSR1 Acceptable Deviations are assessed as PSR1-AD-NFA.

The remaining 151 PSR1 Acceptable Deviations are sorted into 22 groups and are confirmed to be applicable to PSR2. These PSR1 Acceptable Deviations are considered for their aggregate effects in the Defence-in-Depth Assessment in Section 18. The aggregate assessment is presented in Appendix H.

18. Defence-in-Depth Assessment

This section documents the assessment of the extent to which the safety requirements of defence-in-depth are fulfilled at Pickering NGS.

CNSC REGDOC-2.3.3 [2] indicates that the Global Assessment methodology "shall address and include... the extent to which the safety requirements of defence in depth are fulfilled". This assessment considers IAEA INSAG-10, *Defence in Depth in Nuclear Safety* [7], which describes the objectives, strategy, implementation and future development in the area of defence-in-depth in nuclear and radiation safety. Five levels of defence-in-depth are described in IAEA INSAG-10 [7], and an approach to assessment of defence-in-depth is described in IAEA SRS-46, *Assessment of Defence in Depth for Nuclear Power Plants* [8].

The Defence-in-Depth Assessment adopts the high-level elements of the approach to assessment of defence-in-depth described in IAEA SRS-46 [8], adapted appropriately considering that the assessment is performed in the context of a PSR and not as a stand-alone assessment.

The scope addresses the baseline plant and governance as of the freeze date of January 15, 2016, as identified in the PSR2 Basis Document [1]. The Defence-in-Depth Assessment is primarily based on this baseline, which takes into account the physical improvements and programmatic enhancements that have been implemented since Pickering NGS Units 1,4 and 5-8 first became operational.

The scope also considers the following elements of PSR2:

- The Strengths that have been identified in the PSR2 process, and how they support the baseline plant meeting the requirements of defence-in-depth.
- The positive impact on defence-in-depth of the enhancements associated with the proposed Resolution Plans.
- Confirmation that Acceptable Deviations do not have a significant adverse effect on defence-in-depth, either individually or when aggregated.

Defence-in-depth consists of a hierarchical deployment of different levels of equipment and procedures in order to maintain the effectiveness of physical barriers placed between radioactive material and workers, the public or the environment, during normal operation and postulated events at the plant.

Typically, five levels of defence-in-depth are considered. The objectives of each level and the means of achieving the objectives are described in IAEA INSAG-10 [7], as shown in Table 21.

Table 21: Defence-in-Depth Levels

Defence-in-Depth Level	Objective	Essential Means
Level 1	Prevention of abnormal operation and failures	Conservative design and high quality in construction and operation
Level 2	Control of abnormal operation and detection of failures	Control, limiting and protection systems and other surveillance features
Level 3	Control of accidents within the design basis	Engineered safety features and accident procedures
Level 4	Control of severe plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents	Complementary measures and accident management
Level 5	Mitigation of radiological consequences of significant releases of radioactive materials	Off-site emergency response

The objective of defence-in-depth Level 1 is the prevention of abnormal operation and system failures. Failure at this first level is an initiating event. Defence-in-depth Level 2 controls abnormal operation and detects failures. Level 3 ensures that specific safety systems and other safety features will be activated to limit the possible consequences of design basis accidents (DBAs). Level 4 limits accident progression by means of accident management measures in order to prevent or mitigate severe accident conditions with external releases of radioactive material. The objective of Level 5 is the mitigation of the radiological consequences of significant external releases through the off-site emergency response.

18.1. Methodology

IAEA SRS-46 [8] "describes a method for assessing the defence in depth capabilities of an existing plant, including both its design features and the operational measures taken to ensure safety." It is noted that the methodology described in IAEA SRS-46 [8] is not expressly for the purpose of a PSR. That is, IAEA SRS-46 [8] describes a way to perform a stand-alone assessment of defence-in-depth. Neither CNSC REGDOC-2.3.3 [2] nor IAEA SSG-25 [3] reference IAEA SRS-46 [8], yet both require that defence-in-depth be addressed within a PSR. Nevertheless, PSR2 uses the critical elements of IAEA SRS-46 [8], the safety principles, in conjunction with other information generated in PSR2, to execute a Defence-in-Depth Assessment, as described below.

The PSR2 Defence-in-Depth Assessment encompasses a systematic evaluation of the same aspects of defence-in-depth in a nuclear power plant as IAEA SRS-46 [8] but uses a topical approach that is based on the 15 Safety Factors set out in CNSC REGDOC-2.3.3 [2] (which includes a specific Safety Factor for Radiation Protection, in addition to the 14 from IAEA SSG-25 [3]). Therefore, the findings of the Safety Factor reviews and the proposed Resolution Plans (proposed enhancements) for the Global Issues are used together with defence-in-depth elements from IAEA SRS-46 [8] (based on the key areas, including the design, operation, maintenance, etc.) to develop an integrated approach to assess the defence-in-depth provisions of Pickering NGS[25].

Overall, the adequacy of the provisions for defence-in-depth is confirmed by demonstrating that the Pickering NGS design and operation are aligned with the specific safety principles covered in IAEA SRS-46 [8], taking into account the Strengths and proposed Resolution Plans identified in Appendix B, including the impact of Acceptable Deviations.

Considering the above, the assessment of the safety principles for the Global Assessment phase of PSR2 includes the following steps (described in more detail in Appendix D):

- Identification of the safety principles from IAEA SRS-46 [8] that are applicable to the defence-in-depth review.
- Establishment of the defence-in-depth levels impacted for each applicable safety principle (taken from IAEA SRS-46 [8]).
- Mapping of each safety principle to the relevant Safety Factor(s).
- Assessment of the adequacy of the Pickering NGS design and operation with respect to each safety principle.

As shown in Appendix D, the number of safety principles related to each level of defence is as follows:

- **Safety Principles Related to Level 1:** 36 of 52 safety principles
- **Safety Principles Related to Level 2:** 32 of 52 safety principles
- **Safety Principles Related to Level 3:** 38 of 52 safety principles
- **Safety Principles Related to Level 4:** 34 of 52 safety principles
- **Safety Principles Related to Level 5:** 7 of 52 safety principles

An overall assessment is then provided for each level of defence-in-depth, based on integration of the following:

- The conclusions from the assessment of the related safety principles (Appendix D).
- Consideration of Strengths identified in PSR2 and their impact on defence-in-depth (Appendix E).

[25] IAEA SRS-46 [8] provides this flexibility, as it states that the “assessment method described in this publication is not meant to replace the other evaluations required by national or international standards. Rather, it is intended to complement regulatory evaluations and to provide an additional tool for a better appreciation of the defence in depth capabilities of a plant.”

- The impact of proposed enhancements to defence-in-depth resulting from the proposed Resolution Statements (Appendix F).
- Assessment of the aggregate impact of the Acceptable Deviations on the defence-in-depth capability of the plant (Appendix G and Appendix H).

Finally, an overall summary is provided to confirm that Pickering NGS fulfills the safety requirements of defence-in-depth.

The main steps in the methodology and the sections of this report in which the results are documented are shown in Figure 2. Section 18.2 describes the current OPG management system and how it supports defence-in-depth, the current plant design features (including significant safety improvements that have been made since the start of operation of Pickering NGS Units 1,4 and 5-8) and the processes that support sustaining and enhancing defence-in-depth. This description establishes the baseline plant, processes and management system for the Pickering NGS Defence-in-Depth Assessment. Section 18.3 contains the assessment of each level of defence-in-depth, based on the baseline plant and taking into account the results of PSR2. The assessment is supported by information contained in various appendices, as shown in Figure 2. Finally, Section 18.4 provides the conclusions of this Defence-in-Depth Assessment.

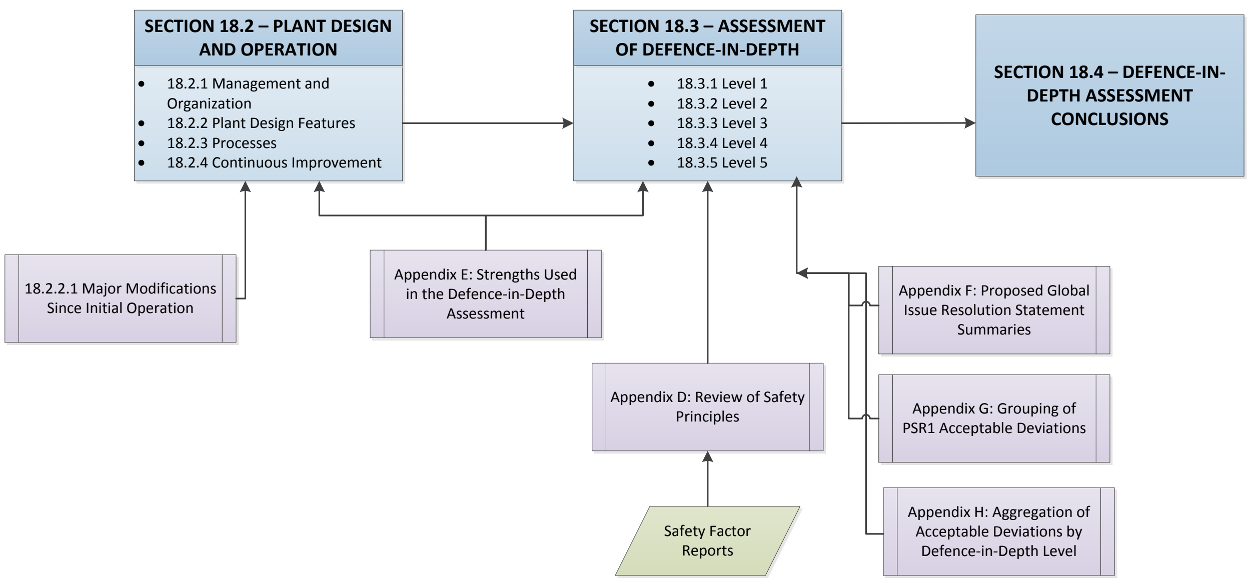

Figure 2: Methodology for Defence-in-Depth Assessment

18.2. Plant Design and Operation

This section describes the Pickering NGS management system and organization, the inherent and engineered design features, and the processes in place that maintain effective barriers between radioactive material and workers, the public and the environment, during normal operation and accidents.

18.2.1. Management and Organization

Nuclear safety is a core value at OPG. This is reflected in OPG Policy N-POL-0001, *Nuclear Safety Policy* [62] that is endorsed by OPG's Board of Directors. The policy places nuclear safety as the overriding priority above that of cost, schedule or production. It requires that all employees conduct themselves in a manner consistent with the behaviour of a healthy nuclear safety culture. Such conduct requires that staff always consider how their everyday activities can impact on the fundamental safety functions of the station.

OPG Charter N-CHAR-AS-0002, *Nuclear Management System* [50], which takes its authority from the *Nuclear Safety Policy* [62], gives authority to OPG Nuclear safety processes and defines responsibilities for the Nuclear Quality Program. OPG Standard N-STD-AS-0020, *Nuclear Management Systems Organizations* [63], outlines the implementation of the quality program and establishes the lines of authority and definition of duties. The well-defined organizational structure and strong lines of authority ensure that the *Nuclear Safety Policy* [62] is implemented effectively.

The safe operation and maintenance of Pickering NGS is under the direct accountability of the Director, Operations and Maintenance, who also ensures that the centre-led organizations effectively use resources to achieve performance targets. The quality and quantity of services provided by these centre-led support organizations are monitored by the Pickering Senior Site Vice President, who holds responsibility for establishing site requirements and priorities.

OPG identifies qualified and competent individuals for key positions with career development and succession planning being key elements in the management capability strategy. The corporate succession plan ensures that individuals with high leadership potential are identified to help continue excellence in nuclear safety.

Management responsibilities related to nuclear safety and protection of the environment are executed through a series of formal programs, as described in N-CHAR-AS-0002, *Nuclear Management System* [50]. The programs cover all aspects of operation including design modifications, engineering change control, maintenance and equipment reliability, integrated aging management, safety analysis, radiation protection, environmental management, security, conventional health and safety, fire protection, and public engagement. Key programs related to defence-in-depth are discussed in more detail in Section 18.2.3. PSR2 identifies that OPG's management system is a Strength (see Appendix E), as are Safety Culture and Use of Operating Experience and Research Findings. Strengths in these areas support defence-in-depth Levels 1, 2, 3, 4 and, for management, all five levels.

Managers at OPG ensure that tasks are executed as defined through programs, such as those listed above, that are specifically designed to achieve higher levels of nuclear and industrial safety, and higher unit reliability through event-free operation. This performance is accomplished through event-free tools such as pre-job briefings, post-job debriefings, self-checking programs, communications, self-assessments, and an observation and coaching program.

In summary, the existing corporate structure supports well-defined lines of responsibility throughout the organization. In particular, the appropriate functions are in place and adequately staffed to support and enhance nuclear safety at all levels of defence-in-depth.

18.2.2. Plant Design Features

Pickering NGS was designed and built to high standards using the principles of defence-in-depth. The design includes a number of robust active and passive safety characteristics, as well as engineered and administrative safety features. These characteristics and features prevent and mitigate accident progression.

18.2.2.1. Major Modifications Since Initial Operation

Numerous modifications have been made to Units 1,4 and Units 5-8 since the units commenced operation. These improvements reflect OPG's continuous improvement philosophy and they bring the station into closer alignment with modern codes and standards. The key modifications that have been made to Units 1,4 are as follows:

- **Fuel Channel Replacement:** As part of the Large Scale Fuel Channel Replacement Program, completed in 1987, all pressure tubes, end-fittings, shield plugs and garter springs were replaced in Units 1-4. This replacement has significantly extended the operating life of the units.
- **Shutdown System Enhancement:** To provide additional Shutdown System trip parameter coverage, additional trip parameters were added to the Unit 1,4 Shutdown System. These additional independent trip parameters, which use diverse instrumentation, are designated as SDSE and are independent from the original Shutdown System trip parameters and trip logic. Also, two additional shutoff rods were installed as part of SDSE bringing the complement of shutoff rods to 23 per reactor.
- **Control Computers:** The digital control computers used for the Reactor Regulating System and the Annunciation System were replaced on Unit 1 and Unit 4.
- **Fire Protection:** The Pickering NGS Units 1,4 Fire Protection Systems were comprehensively upgraded by the OPG Fire Protection Upgrade Program. These upgrades conform to the requirements of CAN/CSA N293-07, National Standard of Canada CAN/CSA-N293-07, *Fire Protection for CANDU Nuclear Power Plants*.
- **ECI Recovery System:** A design change to the configuration of the dump tank outlet piping was made to improve the performance and reliability of the Emergency Coolant Injection (ECI) Recovery System.

- **ECI Strainers:** Installation of additional large surface area strainers to reduce the probability of strainer blockage during recovery operation was completed.
- **Emergency Boiler Water Supply System:** The Emergency Boiler Water Supply (EBWS) System is designed to provide emergency water to the boilers of Pickering Units 1,4 following a postulated main steamline failure within the Pickering NGS A powerhouse. The emergency water is supplied from the discharge headers of the Pickering NGS Units 6 and 7 High Pressure Service Water System.
- **Containment Stack Monitors:** The design of the existing box-up activity monitoring arrangement at Pickering 1,4 was revised to improve Containment system reliability by separation of process and safety systems. This was achieved through the installation of additional radioactivity sensors, mounted external to Containment ventilation system pipework, to monitor instantaneous activity in the Reactor Building ventilation exhaust duct. The new design, which is based on a similar concept to that in place at Pickering 5-8 and Darlington, provides the capability to initiate box-up when required to limit off-site releases.
- **Service Water Systems:** Several design changes to significantly improve the capacity and reliability of the emergency Low and High Pressure Service Water Systems were implemented for Units 1 and 4. The design changes involve larger capacity emergency low and high pressure service water pumps and additional Class IV power service water load shedding components.
- **Inter Station Transfer Bus:** The Class III Inter-Station Transfer Bus (ISTB) is a standby power source to transfer power from Pickering Units 5-8 to the Pickering Units 1- 4 600 V Class II bus, which is normally fed via inverters. In the event inverters fail, the loads are transferred to the ISTB supplied by Pickering NGS Units 5-8.
- **Class II Inverters:** Replacement of the existing motor-generator sets used for providing continuous Class II power with electronic inverters was implemented on Units 1 and 4. This modification was implemented to improve the reliability of the Class II Power System.
- **Instrument Air System:** A design change to combine the low and high pressure instrument air compressors and relocate them to a higher elevation was implemented. Relocation of the compressors makes the instrument Air System less vulnerable to flooding. The new compressors supply instrument air to both Units 1 and 4.
- **Environmental Qualification:** As part of the Pickering A Return to Service Project, significant upgrades were made on operating units to satisfy Environmental Qualification requirements. Systems that are qualified include SDSA, SDSE, ECI (injection and recovery) and Moderator. Extensive PVC cable replacement was completed and all Hydrogen Igniters were replaced.

  A number of other Environmental Qualification initiatives have also been completed. The most significant is the major Environmental Qualification retrofit undertaken in the mid-1980s to improve mitigation of powerhouse harsh environment events. Modifications included strengthening the “H” line wall and the Control Equipment Rooms, protection of Class I and Class II electrical distribution systems, provision of a qualified heat sink and a Powerhouse Venting System with automatic initiation on temperature or pressure.

- **Seismic Assessment:** A seismic assessment was performed to evaluate the seismic capacity of SSCs required to perform the Control, Cool, Contain and Monitoring functions and to identify necessary seismic upgrades. Low seismic capacity components were replaced, structure and component anchorage was upgraded and potential seismic interactions were dispositioned for the return to service of Units 1 and 4. The list of seismic upgrades included:
  - Boiler Room shield wall upgraded
  - Switchgear and panel anchorage upgraded
  - Anchoring of heat exchangers in Vacuum Building basement enhanced
  - Standby Generator oil pump house masonry block wall reinforced
  - Supports for Standby Generator batteries enhanced
  - Supports for Class I batteries enhanced
  - Main Control Room (MCR) and Control Equipment Room panel anchorage upgraded
  - Control Equipment Room structural upgrades
  - Rerouting emergency air supply for airlocks
  - Anchoring Temporary Breathing Air System near emergency Low Pressure Service Water pumps
  - Enhancing supports for bleed valves
  - Enhancing spring hangers
  - Restraining gas bottles and fire extinguishers
  - Improving supports for Reactor Building Air Cooling Units
  - Reviewing proximity issues for various valves
  - Improving Deaerator Storage Tank Anchoring System
  - Providing lateral restraints for the high pressure feedwater heaters
  - Reviewing the Fuel Channel positioning assembly rod
  - Reviewing anchor bolts used on the fuelling machine support column
  - Upgrading mercoid switches
  - Upgrading relays

- A seismic Abnormal Incident Manual was issued and staff has been trained to respond to a seismic event
- Periodic testing of systems credited in seismic analysis
- Pressure Relief Duct analysis
- Piping supports on Recirculating Cooling Water piping

Also, administrative controls have been implemented to ensure that seismic qualification of Pickering NGS Units 1,4 seismic success path SSCs is maintained for the life of the facility.

The key modifications that have been made to Units 5-8 are as follows:

- **Third Emergency Power Generator:** An additional source of power to the Emergency Power System, consisting of three diesel generators capable of delivering a total of 2.5 MW, has been added to Units 5-8 to provide additional redundancy in the Emergency Power System.
- **SG Governor and Controls Upgrade:** This modification was a proactive strategic improvement initiative to maintain Class III power reliability/availability by replacing fuel governor, metering, delivery and control logic systems for the Standby Class Ill Power Generators.
- **Fire Protection Upgrades:** New Fire Detection Systems were installed in the Main Control Room, Control Equipment Rooms, and Cable Spreading Area. Turbine Generator and Transformer fire suppression systems were replaced with modern technology.

In addition, a number of other major improvements have been made that apply to Units 1,4 and Units 5-8. These are:

- **Electrical Power System:** The Electrical Power System at Pickering NGS has been enhanced through design modifications, such that the units are equipped with multiple sources of backup electrical power to ensure that controls and equipment important to safe operation are available during normal and abnormal conditions. These are:
    - Site Electrical System – This distribution system can be used for transferring back-up auxiliary power to any of the unit Class IV systems (High Pressure ECI pump motors, ECI-associated motorized valves for each of Units 1,4 and the electrical services of any single unit that has lost its Class IV power).
    - The Auxiliary Power System is a back-up power supply system that supplies power to selected Group 1 Class IV loads following a sustained loss of Class IV power across Pickering NGS. The system performs this function by supplying power to the Site Electrical System.

  These improvements are assessed as a Strength for the Electrical Power System (see Appendix E).

- **Hydrogen Igniters and Passive Auto-catalytic Recombiners:** A Hydrogen Ignition System has been provided in Units 1,4 and Units 5-8 to safely combine any hydrogen (or deuterium) gases that may be generated during certain scenarios. In addition, Passive Auto-catalytic Recombiners have been installed to control Containment hydrogen levels at reduced hydrogen generation rates, thereby eliminating reliance on the Hydrogen Ignition System in the longer term.
- **Critical Safety Parameter Monitoring:** The capability to monitor post-accident conditions remotely from the Main Control Room is provided in the SDSE Instrument Rooms for Units 1,4 and in the Unit Emergency Control Centre for Units 5-8.
- **Emergency Mitigating Equipment**: Emergency Mitigating Equipment has been provided for an additional makeup water supply for accidents beyond the design basis. Portable diesel-powered pumps can be deployed and can provide cooling water make-up to the secondary side of the boilers, to the Heat Transport System and to the Moderator. Portable diesel-powered generators can provide power for critical monitoring following an extended loss of all AC power event. Additional major Fukushima Project design modifications that have been installed include enhancements to water makeup/cooling capability for the IFBs. In addition, implementation of Phase 2 Emergency Mitigating Equipment modifications is in progress. Phase 2 Emergency Mitigating Equipment provides restoration of power to station equipment for sustained core cooling and restoration of Containment coolers to improve control of Containment pressure following BDBAs.

In summary, Pickering NGS has been continually upgraded and extensively modernized over its decades of operation to align it, as much as practicable, with current industry best practices. Many of the improvements listed in this section are further described in Section 18.3 in terms of how they support defence-in-depth.

18.2.2.2. Current Plant Design Features Important to Defence-in-Depth

Taking into account the major modifications described in the previous sub-section, the key design features of Pickering NGS that are important to the assessment of defence-in-depth are as follows:

- **Quality in Design:** The design and construction of Pickering NGS was undertaken under a formal quality assurance regime. Rigorous quality assurance processes were put in place and have been updated as improved standards become available for design analysis, stress analysis, material control and traceability, fabrication, in-process inspection, installation and welding, commissioning, non-destructive examination, and inspection.

  Currently, design modifications are performed under OPG Program N-PROG-MP-0001, *Engineering Change Control* [64], which meets modern rigorous quality assurance requirements. The engineering change control program defines a systematic process and methodology for controlling design modifications for plant SSCs.

- **Redundancy, Separation and Diversity:** In addition to high quality standards, critical safety functions are diverse and physically separated from process control functions. Redundant components are used where possible, so that the failure of a single component does not cause system failure. The Pickering NGS Units 1,4 and 5-8 designs include protection against common cause events; the design includes provisions to prevent the loss of safety functions due to damage to multiple SSCs.

  For Units 5-8, redundant equipment and circuits are separated and grouped (Group 1 and Group 2) to ensure the safety of the station following a common mode event. Each group is capable – independently of the other group – of safely shutting down the reactor, cooling the Fuel and providing the operator with indications of system conditions.

  Units 1,4 systems have been either qualified or retrofitted to function as required for a given common mode event by ensuring effective separation and diversity. Hazard assessments have been performed that confirm the capability of the design to perform the required safety functions following postulated common mode events.

- **Multiple Barriers:** Five physical barriers exist between the radionuclides and the public. They are the ceramic uranium dioxide fuel matrix, the metallic Zircaloy fuel cladding, the piping that comprises the Primary Heat Transport System boundary, the Containment boundary, and the exclusion zone around the station.

  The first four barriers prevent radioactive release to the environment. So long as they are intact, very little radioactive material will escape from the Containment. The exclusion zone utilizes the concept of distance from the source to mitigate radiological consequences if all of the first four barriers are breached.

- **Radiation Protection in Design:** During the initial design stage, emphasis was placed on the reduction of occupational radiation doses and limiting releases of radioactivity to the environment. The design of Pickering NGS ensures that the layout and operation of facility SSCs and processes are consistent with established radiation protection guidelines and contribute to maintaining occupational radiation exposures ALARA.

  Pickering NGS uses dynamic learning activities to provide workers an opportunity to practice radiation protection fundamentals in a simulated radioactive work environment using remotely controlled radiofrequency technology. OPG has implemented remote reading of radiation detection instrumentation and real-time data transmission to facilitate improved job planning and awareness of current radiological conditions.

  Specific design features at Pickering NGS to control radiation dose include the use of shielding, ventilation and emissions control, radiological zoning and the provision of area radiation monitoring equipment. The use of shielding and ventilation control reduces exposure to external radiation and airborne radioactive material, respectively. The use of emissions control reduces the doses to members of the public.

  The use of advanced technology to support radiation protection is a PSR2 Strength. This Strength demonstrates OPG's commitment to continuous improvement in radiation protection.

- **Reactor Control:** The plant systems are controlled to remain within their operating ranges. The control systems in Pickering NGS have been designed to maintain the reactor operating conditions within the normal operating range, and to effectively respond to transients to avoid the need for safety system actuation.

  Reactor control is highly reliable and prevents process upsets due to measurement failures, etc., with a high degree of redundancy in control devices and process measurements.

- **Automatic Shutdown Systems:** Pickering NGS has highly reliable Shutdown Systems for safety purposes that are independent of the equipment and processes used to control the reactor power and that are poised at all times during at-power operation.

  The Pickering NGS Units 5-8 have two independent, fast-acting and very effective Shutdown Systems. The independent Shutdown Systems are Shutdown System No. 1 (SDS1) (shutoff rods) and Shutdown System No. 2 (SDS2) (Gadolinium injection into the Moderator). Both Shutdown Systems are capable of shutting the reactor down fast enough for all DBAs, such that all limits are met. Both Shutdown Systems can also be manually activated. Both Shutdown Systems are fail safe and will shut down the reactors if electrical power to the system is lost.

  When Pickering NGS Units 1,4 began operation in the early 1970s, their design included a single Shutdown System with two diverse means of inserting negative reactivity into the core: (1) Dropping shutoff rods; and (2) Dumping the Moderator from the Calandria. The original Units 1,4 Shutdown System has been enhanced by retrofitting a Shutdown System Enhancement (SDSE) that provides sets of triplicated sensors and trip logic that are independent of those of the original Shutdown System (SDSA) to increase the reliability of actuation of the shutdown function, achieving system reliability targets for all events. Both systems can achieve their safety function without power or operator intervention, and they can also be manually activated.

- **Heat Removal:** Pickering NGS reactors are equipped with a Heat Transport System that effectively removes heat from the Fuel under normal operating conditions. The system is also effective in responding to anticipated transients and accidents within the design basis, supplemented by the emergency makeup water systems included in the design.

  In normal operation, heat is removed from the Fuel to the Steam Generators by the forced flow of coolant in the Heat Transport System. Steam generated in the Steam Generators is cooled in the condenser by the Condenser Cooling Water System, which rejects the heat to the ultimate heat sink (lake).

  For low power operation and when the reactor is shutdown, the Shutdown Cooling System cools down the Heat Transport System and maintains cooling for an indefinite period of time. The Shutdown Cooling System provides cooling for the Heat Transport System during outage operation and is designed to provide core cooling with the Heat Transport System depressurized to permit maintenance.

In the unlikely event of a pipe break in the Heat Transport System, the ECI System is designed to refill the Heat Transport System, and to provide coolant makeup and heat sink capability in the long term.

For postulated events involving a loss of Steam Generator inventory, a backup boiler water system (the Boiler Emergency Cooling System) is available to remove the decay heat generated in the Fuel. In the short term, the Boiler Emergency Cooling System provides water to Units 1,4 and Units 5-8 for reactor decay heat removal by the boilers for accident situations initiated by or leading to failure of the normal feedwater supply.

For long-term cooling for postulated events involving a loss of Steam Generator inventory, the Emergency Boiler Water Supply System provides emergency water to the boilers of Pickering NGS Units 1,4. In addition, the Emergency Water System is a system supplying long-term emergency makeup to the Units 5-8 boilers, Heat Transport System, ECI recovery heat exchangers, Containment air coolers and the Moderator. The Emergency Water System is powered from the Emergency Power System.

One of the operationally proven safety features at Pickering NGS is the ability to cool the Fuel through natural circulation in the event of a loss of forced flow in the Heat Transport System.

Emergency Mitigating Equipment provided as part of the implementation of lessons learned from the 2011 Fukushima accident can be deployed to prevent significant Fuel damage in the longer term (several hours) following a station blackout. Specifically, Emergency Mitigating Equipment has been provided for an additional makeup water supply for accidents beyond the design basis. Diesel-powered pumps are deployed to draw in lake water to provide cooling water make-up to the secondary side of the boilers, to the Heat Transport System and to the Moderator.

PSR2 identifies a Strength for Heat Removal due to the comprehensive and overlapping suite of heat removal provisions.

- **Containment:** Pickering NGS Containment is a robust Containment System structure that is provided to mitigate the effects of an unlikely Heat Transport System piping break, or other postulated initiating events, by limiting the release of radionuclides to the environment.

  The principle of Negative Pressure Containment is employed in the Pickering NGS. A Vacuum Building, maintained at a very low subatmospheric pressure, is linked to the Reactor Buildings by a pressure relief system. This arrangement ensures that the pressure within the Containment Boundary will be brought below the surrounding atmospheric pressure within a short time following a pressure rise within a Reactor Building. Long-term control of Containment pressure is accomplished through controlled venting via the Filtered Air Discharge System. The purpose of this system is to maintain Containment sufficiently sub-atmospheric following an accident and to provide a filtered and monitored pathway to the environment.

A Hydrogen Ignition System is provided to safely combine any hydrogen (or deuterium) gases generated in Containment following postulated events. The system is capable of addressing very low concentrations of hydrogen in air/steam mixtures, thus preventing the occurrence of high concentrations of hydrogen in Containment. In addition, Passive Auto-catalytic Recombiners have been installed to control Containment hydrogen levels at reduced hydrogen generation rates, and therefore eliminate reliance on the Hydrogen Ignition System in the longer term. The purpose of these two systems is to prevent a potential challenge to Containment integrity by keeping the overall concentration of hydrogen very low.

- **Monitoring Capability:** Values of important plant parameters are displayed in the Main Control Room to ensure that operators have clear and unambiguous indications of the status of plant conditions important for safety, especially for the purpose of identifying and diagnosing the automatic actuation and operation of a safety system or a challenge to defence-in-depth.

  Each Main Control Room contains the main control panels for all the generating units (Units 1,4 or Units 5-8). Each unit has its own control panels, and the panels for all units are of the same form. Two additional panels are provided for common equipment and electrical controls. All indications and controls essential for operation are located on the Main Control Room panels. When the Main Control Room is not available for any reason, the reactor can be monitored and can be safely shut down and maintained in that state indefinitely from the SDSE Instrument Rooms (Pickering Units 1,4) or the Unit Emergency Control Centre (Pickering Units 5-8).

  Also, post-accident monitoring capability of the critical safety parameters is available. By monitoring critical safety parameters and following critical safety parameters restoration procedures via field actions, it is possible to maintain the Control, Cool, Contain and Monitor safety functions externally from the Main Control Room. Capability to monitor post-accident conditions remotely from the Main Control Room is provided in the SDSE Instrument Rooms for Units 1,4 and in the Unit Emergency Control Centre for Units 5-8.

- **Standby Power Systems:** The reactors are equipped with multiple sources of electrical power to ensure that controls and equipment important to safe operation are available. The majority of the on-site power systems are supplied by the unit generator, Standby Generators and batteries, and are designated as Group 1. The Group 1 electrical power systems are divided into the following four classifications:

  - **Class I:** Direct Current (DC) supplies (backed up by batteries) for auxiliaries, control, protection, and safety related equipment.
  - **Class II:** Alternating Current (AC) supplies (backed up by batteries) for control power and supplies for safety-related auxiliaries, instrumentation, protection, and control equipment.
  - **Class III:** AC normally supplied from Class IV power, backed up by the Standby Generators, to safety-related equipment and auxiliaries.
  - **Class IV:** AC supplied from the bulk electrical grid and Unit generators, backed up by the Site Electrical System and the Auxiliary Power System.

The Auxiliary Power System is a back-up power supply system that supplies power to selected Group 1 Class IV loads following a sustained total loss of Class IV power across Pickering NGS.

- **Site Electrical System** – The Site Electrical System is a standby power supply for the Class IV distribution system at both Units 1,4 and Units 5-8. The power source for the Site Electrical System is either the Bulk Electrical System or one or more of the generating units at Pickering NGS or the Auxiliary Power System. The primary purpose of the Site Electrical System is to provide power to the High Pressure ECI pumps.

- **Emergency Power**: The purpose of the emergency power system is to provide power to support critical safety functions during Common Mode events that lead to the loss of Group 1 power. The following comprises emergency power at Pickering NGS:
  - The seismically-qualified Emergency Power System for Units 5-8 supplies power to a specific portion of the Safety-Related Systems in the station. There are 3 separate generator sources supplying power to the Emergency Power System.
  - The Inter Station Transfer Bus supplies power from Pickering Units 5-8 to essential Class II loads at Pickering Units 1,4.
  - The Standby Generators, both the Class II inverters, and the Class I rectifiers at Pickering Units 1,4 are seismically assessed and credited.

  As noted previously, the Pickering NGS electrical system is considered a Strength (see Appendix E) due to the multiple and overlapping provisions and redundant supplies.

- **Protection Against Common Mode Events:** The Pickering NGS Units 1,4 and 5-8 design includes protection against common cause events. The design includes provisions to prevent the loss of safety functions due to damage to multiple SSCs resulting from a common cause. For Units 5-8, redundant equipment and circuits are separated and grouped (Group 1 and Group 2) to ensure the safety of the station following a common mode event. Each group is capable, independently of the other group, of safely shutting down the reactor, cooling the Fuel, and providing the operator with indications of system conditions. For Units 5-8, seismically qualified Emergency Power and Water Systems are provided to support nuclear safety loads following significant seismic events. Units 1,4 systems have been either qualified or retrofitted to function as required for a given common mode event by ensuring effective separation, diversity, and robustness.

  The Pickering NGS design addresses dependent failures, using separation of redundant SSCs for responding to common mode events. Hazard assessments have been performed that confirm the capability of the design to perform the required safety functions following postulated common mode events.

- **Inspectability of Safety Equipment:** Periodic inspection is the non-destructive examination of nuclear equipment, the failure of which can have adverse effects. To support this activity, the safety systems and their support systems are designed to facilitate testing and inspection to ensure that the plant meets or exceeds safety standards.

- **Maintain Subcriticality:** A criticality hazard is not possible with fresh Fuel or with irradiated natural uranium Fuel bundles stored in the IFBs. This is because natural uranium Fuel bundles and light water cannot be made critical. Hence, no criticality concern exists in handling new Fuel or Fuel in the IFBs.
- **Beyond Design Basis Accident Management:** Pickering NGS has implemented and is in the process of further enhancing significant preventive and mitigating modifications to strengthen defence-in-depth against BDBAs, including severe accidents. Emergency Mitigating Equipment Guidelines and Severe Accident Management Guidelines are in place to mitigate accident progression and protect Containment integrity.

  A key line of defence for BDBAs is Emergency Boiler Makeup using Emergency Mitigating Equipment. An additional line of defence is Emergency Mitigating Equipment to provide makeup water to the Moderator. These features are capable of limiting core damage for most BDBAs.

  Additional major Fukushima Project design modifications include:

  - Passive Auto-catalytic Recombiners on all units to supplement the existing Hydrogen Igniters for control of hydrogen in Containment.
  - Enhancements to water makeup/cooling capability for the IFBs.

  In addition, implementation of Phase 2 Emergency Mitigating Equipment modifications is in progress. Phase 2 Emergency Mitigating Equipment includes restoration of power to key station equipment for core cooling and monitoring and to Containment coolers to enhance control of Containment pressure following BDBAs.

  Instrumentation and diagnostic aids to support deployment of Emergency Mitigating Equipment are available to the operators.

  PSR2 identifies a Strength for implementation of Fukushima Action Items (see Appendix E) on the basis that OPG has been proactive and a leading organization in identifying and implementing lessons learned from the 2011 Fukushima accident.

18.2.3. Processes

Pickering NGS is operated and maintained in accordance with current nuclear industry codes and standards consistent with regulatory and safety requirements and industry best practice. Normal plant operation is controlled by detailed, validated and formally approved procedures. In addition, emergency operating procedures are established and implemented to ensure effective operator response to abnormal events. Many of Pickering's operating, maintenance, engineering and other support programs and processes are assessed as Strengths.

OPG has established extensive programs and procedures and employs qualified staff to safely and effectively manage the nuclear plants. The programs and training were developed based on regulatory requirements, CSA standards, IAEA Guides, WANO recommendations and best nuclear industry practices from around the world. As part of continuous improvement, the programs and training are kept up to date, based on self-assessments, benchmarking, and ongoing use of industry OPEX.

OPG has implemented a modern Nuclear Management System (discussed in Section 18.2.1), that governs plant activities ranging from human performance, engineering, operations, maintenance, to environmental management, and support nuclear safety at all levels of defence-in-depth. These programs are aligned with modern industry best practice as evidenced by the few PSR2 Gaps identified in the related Safety Factor Reports, and they typically support multiple levels of defence-in-depth. Some of the key programs are listed and summarized below:

- **Quality Assurance Program:** OPG Charter N-CHAR-AS-0002, *Nuclear Management System* [50] and supporting documents referenced in the charter establish the Nuclear Management System for OPG Nuclear, which assures that systems, equipment and activities are of the required quality throughout the life of the nuclear facilities.
- **Engineering Change Control:** OPG Program N-PROG-MP-0001, *Engineering Change Control* [64] ensures that all modifications to plant SSCs, including software and station engineered tooling, are planned, designed, installed, commissioned, decommissioned, placed into service or removed from service within the SOE, design basis and plant licensing conditions. It defines a systematic process and methodology for controlling design modifications for plant SSCs to meet the requirements of CSA N285.0, *General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants* and CSA N286, *Management System Requirements for Nuclear Power Plants*.

  An aspect of this program, specifically the implementation of Human Factors Engineering, is identified as a Strength for Pickering NGS (see Appendix E).
- **Equipment Reliability:** OPG Program N-PROG-MA-0026, *Equipment Reliability* [47] defines the requirements for establishing and maintaining optimum levels of reliability for components important to nuclear safety, production, and environmental protection. Reliable performance of components means very low numbers of component failures, degraded equipment condition is minimized, and redundancy is maintained on key systems.

  The Equipment Reliability Program contains the following elements which ensure ongoing high levels of reliable performance of critical components:
  - Identifying critical components that require focused attention.
  - Specifying the required maintenance strategies to maintain high levels of reliability.
  - Executing predictive maintenance and preventive maintenance programs.
  - Monitoring system and component condition and implementing plans to restore and maintain system and component health.
  - Taking prompt and effective action, when critical equipment fails, to understand the technical and organizational causes and to prevent a recurrence.

- Identifying and predicting aging and obsolescence issues on important components and embedding mitigating strategies and actions into the business plan.

Pickering NGS's Equipment Reliability Program implementation and execution of the program is a station priority. This program is identified as a Strength for Pickering NGS. System Health reporting, which is an integral element of the Equipment Reliability Program, is also identified as a Strength.

- **Risk and Reliability Program:** OPG Program N-PROG-RA-0016, *Risk and Reliability Program* [54] establishes a framework for the development and use of PSA as a means to manage radiological risks from nuclear accidents and to contribute to safe operation of the reactors. Through execution of this program and its elements, risks from nuclear accidents are identified, monitored and controlled.

  An aspect of this program, specifically Operationalization of Probabilistic Safety Assessments, is identified as a Strength for Pickering. The PSA is used to support conduct of engineering, maintenance and operation at Pickering NGS.

- **Reactor Safety Program:** OPG Program N-PROG-MP-0014, *Reactor Safety Program* [65] defines organizational responsibilities and key program elements for the management of issues related to deterministic nuclear safety analysis, generic safety issues (CANDU Safety Issues) and the following components of safe operation:
  - Safety Analysis Basis
  - SOE
  - BDBA Management
  - Safety Reports

  Deterministic Safety Analysis for Pickering NGS is assessed as a Strength for Pickering NGS. In addition, implementation of the SOE program is identified as a separate Strength.

- **Human Performance Program:** OPG Program N-PROG-AS-0002, *Human Performance* [66] is executed through a series of documents that support management of human performance. Collectively, these documents lay the groundwork for improving and sustaining performance. Specifically, this program provides guidance to reduce the probability and consequences of human error associated with the worker – machine interface required to operate, maintain and support Pickering NGS.

  The Event Free Day Reset indicator is one of the means OPG uses to identify human performance events. The indicator reflects the effectiveness of management in reducing errors and improving organizational processes and activities to reduce the significance and frequency of human performance events.

  The indicators have continually improved at Pickering NGS. Performance improvement has been accomplished by effectively identifying human performance events, investigating the causes to determine corrective actions, performing trending and analysis to identify reoccurring and common issue areas and communicating the results throughout all levels of the organization.

Conventional Health and Safety, a key indicator of the Human Performance Program, has met or exceeded performance objectives and all applicable regulatory requirements.

A separate assessment has been performed to identify Strengths in Pickering NGS. These are listed in Appendix E. Some of the Strengths have also been recognized by external organizations, which indicates that Pickering not only has strong organizational structure and procedures, but also that staff perform in accordance with high safety standards, demonstrating an organization with a strong nuclear safety culture. This is also identified as a Strength.

18.2.4. Continuous Improvement

The continuous improvement process through which OPG strives to improve the safety and performance of its nuclear power plants, is longstanding, ongoing, and covers all aspects of operation. Current performance is compared to management expectations, industry standards of excellence, internal and external OPEX, and regulatory requirements to identify areas with opportunities for improvement, prepare action plans and incorporate enhancements.

Established programs and processes are used to identify and address areas for improvement. OPG participates with industry partners in developing new or revised codes and standards, in research and development activities, in the application of emerging technologies, and in the exchange of OPEX. This is done through membership in organizations such as WANO, INPO, the CANDU Owners Group, the CSA and the Electric Power Research Institute.

In particular, there is ongoing improvement as a result of operating experience from the 2011 Fukushima accident.

Following the March 2011 earthquake in Japan, the safety systems at the Fukushima Daiichi Nuclear Power Plant operated as designed and the reactors were automatically shut down. However, the tsunami that followed disabled power to critical support systems.

OPG acted promptly to understand what had happened at Fukushima Daiichi and confirmed that the OPG nuclear fleet remained safe for continued operation. OPG has completed additional assessments including those requested by the CNSC to review the impact of a similar event (i.e., an event resulting in a total loss of all AC power, subsequently resulting in a total loss of heat sinks) at OPG stations. Enhancements to provisions to maintain or re-establish the Control, Cool, Contain and Monitoring safety functions were assessed to determine those that are most practical to implement and also meet specified requirements. Several enhancements have been implemented and additional ones are being implemented, as discussed under BDBA Management in Section 18.2.2.2.

From a PSA perspective, the Pickering NGS Units 1,4 and 5-8 Level 1 and Level 2 At-Power Internal Events Risk Assessments have been updated as part of the Fukushima Action Item update, and demonstrate the benefits of the Fukushima enhancements.

A *Mutual Aid Agreement for Nuclear Emergency Support* [67] is in place with all Canadian nuclear utilities to provide support in the event of an emergency.

OPG continues to have a strong presence in international forums and with all operators of Canadian nuclear generating stations to ensure that the lessons learned from the 2011 Fukushima accident are applied at Pickering NGS.

As noted previously, implementation of Fukushima Action Items is identified as a Strength for Pickering NGS.

18.3. Assessment of Defence-in-Depth

The concept of defence-in-depth has been applied at the Pickering NGS design stage and throughout its operation over a period of several decades. At the design stage, the focus was on the first three levels of defence-in-depth, i.e., prevention of operation outside normal operating conditions, control of abnormal conditions, and provision of safety systems to effectively mitigate DBAs. For example, defence-in-depth Level 1 systems, i.e., process systems, are designed so that any failure in the system is not propagated to the control systems that control these processes. Similarly a failure in a control system does not propagate to the next level of defence-in-depth, i.e., the safety systems. This is accomplished through design of reliable systems and adequate separation of the control systems from the safety systems. Internationally this is achieved by ensuring adequate buffering of any components shared between the control and safety systems so that the failure cannot be propagated. In CANDU reactors, including Pickering NGS, this is achieved through separation of the control and safety systems. Defence-in-depth Level 2 is achieved by monitoring changes from normal operating conditions by both the Reactor Regulating System and the Special Safety Systems. Digital computerized monitoring of parameters important to safety is used in the design of the Reactor Regulating System. Level 3 includes the provision of inherent safety features, fail-safe design, engineered design features, and procedures that minimize the consequences of DBAs. As part of defence-in-depth, pressure retaining components in any safety system are required to meet the highest design and quality standards. In summary, the baseline plant incorporates extensive defence-in-depth at Levels 1, 2 and 3.

The fourth level of defence-in-depth makes use of the additional inherent capacity and capability of station systems and also relies on many additional systems that are not normally credited in safety analysis for DBAs. They are used to mitigate the consequences of BDBAs, including severe accidents. Such accidents have a very low frequency due to the high reliability of the safety systems and other mitigating provisions. Level 4 provisions are generally backup process systems and as such have been designed such that their failure would not affect the control or safety systems. The recent addition of Emergency Mitigating Equipment supplements the previous Level 4 provisions.

Comprehensive on-site and off-site plans and new facilities and processes have been implemented for response to emergencies as the fifth level of defence. Significant improvements have been implemented in the fourth and fifth levels of defence, based on international OPEX since Pickering NGS Units 1,4 and 5-8 were put into service, as well as new requirements from the CNSC.

A detailed assessment of safety principles for each level based on the assignment in IAEA SRS-46 [8] is provided in Appendix D of this report. The information on the provisions for each of the safety principles is based on the current Pickering NGS design and operation and considers information from sources such as the PSR2 Safety Factor Reports, Probabilistic Safety Assessments and the Safety Reports. The assessments in Appendix D provide overall confirmation that effective and overlapping provisions are in place to meet the objectives of the safety principles.

An assessment based on a detailed review of the safety principles, some of which are applicable to multiple levels of defence-in-depth, is presented for each level of defence in the following sub-sections. The Defence-in-Depth Assessment also takes into consideration the proposed Resolution Plans for the Global Issues (Appendix F), Strengths (Appendix E) and the aggregate impact of the Acceptable Deviations (Appendix H).

18.3.1. Level 1 – Prevention of Abnormal Operation and Failures

The aim of the first level of defence is to prevent deviations from normal operation and failures.

The first level of defence requires high quality in the design and construction of the plant with barriers to prevent the occurrence of abnormal operating conditions. This is particularly important for the physical barriers surrounding the radioactive material in the Fuel. Safe, conservative operation of the plant by qualified staff and a continued focus on preventive maintenance ensures reliable functionality of plant equipment under normal operation and therefore prevents process upsets and failures.

18.3.1.1. Level 1 – Provisions and Barriers

There are 36 safety principles that apply to defence-in-depth Level 1, as shown in Appendix D. The appendix confirms that all of these principles are fulfilled by the Pickering NGS plant design, processes and management system. The safety principles in each of these three areas that are most relevant to Level 1 (prevention of abnormal operation and failures) in the context of extended operation of Pickering NGS are discussed briefly below.

With respect to the plant, the key safety principles that ensure that the design is adequate as a Level 1 barrier are:

- D-158 – General Basis for Design
- D-154 – Proven Technology
- D-195 – Reactor Core Integrity
- D-209 – Reactor Coolant System Integrity
- D-203 – Normal Heat Removal
- D-205 – Startup, Shutdown, and Low Power Operation
- D-188 – Radiation Protection in Design
- S-136 – External Factors Affecting the Plant

- M&C-249 – Achievement of Quality

The key safety principles related to processes that ensure that design requirements continue to be met, that supporting processes are in place to prevent SSC failures, and that fitness for service of SSCs is maintained through the operational life of the station are:

- O-296 Engineering and Technical Support of Operations
- D-150 Design Management
- M&C-246 Safety Evaluation of Design
- O-305 Maintenance, Testing and Inspection
- O-272 Conduct of Operations
- O-288 Normal Operating Procedures
- C-258 Validation of Operating and Functional Test Procedures
- O-284 Operational Limits and Conditions
- O-312 Quality Assurance in Operation

The key safety principles related to the management system and that support the effectiveness of the Level 1 barrier are:

- O-265 Organization, Responsibilities and Staffing
- O-299 Feedback of Operating Experience
- O-278 Training
- M&C-249 Achievement of Quality

Alignment with these key safety principles supports the adequacy of the Level 1 barrier at Pickering NGS. The following discussion describes the specific aspects of Pickering NGS that implement these and the other safety principles related to defence-in-depth Level 1.

At Pickering NGS, defence-in-depth Level 1 includes conservative design and high-quality construction and commissioning which provides a baseline confidence that any unexpected failure of SSCs and changes from normal operations are minimized and accidents are prevented.

Pickering NGS is designed and built to high quality standards, with critical safety functions that are diverse and physically separated from process control functions.

The use of redundant components, so that the failure of a single component does not cause system failure, leads to the use of two out of three voting logic, or channels, in many poised systems (systems available to operate if called on). This requires two of three separate instruments to fail unsafe before the system logic fails. This type of logic also permits on power testing, channel by channel, without impairing the functionality of the system and prevents spurious initiation of a system if one instrument or channel fails. Also, diversity of functions (e.g., process and neutronic measurements) for important control and safety systems is used, so that a common fault in one type of component cannot cause failure of the function. To the extent possible, equipment is designed to fail-safe on loss of electrical power (e.g., shutoff rods drop when they lose power to their clutches). Similarly, pneumatic instruments and components such as air-operated valves, have redundancies and are designed to be fail-safe to the extent possible. Self-actuating devices are employed where possible.

The Pickering NGS Heat Transport System is designed to avoid pressure boundary failure through high quality design and maintenance, provision of adequate pressure relief capability, and comprehensive leak detection capability. Confidence in the integrity of the pressure boundary is supported through appropriate assessments, and research and development.

These design features are described in Section 18.2.2. In addition, a comprehensive and effective Nuclear Management System has been developed and implemented which assures that systems, equipment, and activities are of the required quality throughout the operational life of the plant. The Management System, executed by an effective management and organization as described in Section 18.2.1, governs plant activities ranging from human performance, design and engineering, operations, maintenance, to waste management, and support nuclear safety at all levels of defence-in-depth. These programs are described in Section 18.2.3. These programs support fulfilling the safety requirements of defence-in-depth Level 1.

The safety systems SSCs are maintained within their design basis by N-PROG-MA-0026, *Equipment Reliability* [47] and N-PROG-MP-0008, *Integrated Aging Management* [48], which ensures the condition of critical equipment is understood and that required activities are in place to ensure the ongoing health of these components and systems. The program includes requirements for in-service inspections, maintenance, engineering assessment and confirmatory research and development. These processes provide for the timely detection and mitigation of aging effects in SSCs that impact plant safety and reliability. Regular preventive and predictive maintenance, inspection, testing and servicing of SSCs important to safety and reliability are conducted to maintain SSCs within their design basis.

Operational improvements are implemented continuously based on national and international OPEX in addition to improvements driven by the evolving regulatory requirements.

18.3.1.2. Level 1 – PSR2 Assessment

PSR2 assessments have demonstrated that the plant has been designed conservatively using the appropriate design codes and materials, design procedures, equipment qualification, control of component fabrication, plant construction and commissioning. The PSR2 review has confirmed that the physical plant conditions such as SSC design and construction are sound. Some PSR2 Gaps are associated with update of documentation or analysis to address specific modern codes and standards, and in many cases this work was already in progress. The Gaps do not identify any weaknesses in Level 1 that require design changes.

Level 1 provides the initial basis for protection against external and internal hazards. The review of Safety Factor 7 has confirmed the adequacy of protection of Pickering NGS against internal and external hazards, with account taken of plant design (including confirmation that analyses/methods address the condition of SSCs important to safety), site characteristics, and current analytical methods, safety standards and knowledge.

In parallel with the PSR2 review, an extensive set of Condition Assessments were completed. The Condition Assessments, along with current system and component health programs, provide the station with sufficient information on the condition of SSCs to mitigate aging effects throughout extended operation. The detailed assessment confirmed no component or system aging compromises the current design basis or design basis assumptions. For Special Safety Systems for both Units 1,4 and 5-8, all assessed components were found to support the extended operating period.

The review concluded that the condition of SSCs is well understood and that plant safety and reliability are maintained through a set of systematic and planned surveillance, testing, inspection and maintenance activities using best industry practices and OPEX. The PSR2 review has confirmed that programs continue to be improved in the areas of Condition Assessment, aging management, and obsolescence management.

The conditions of the Pickering NGS SSCs are tracked in System Health Reports and Component Health Reports that are aligned with industry best practices.

In addition, rigorous programs and processes are in place to control and verify design compliance to ensure that the plant is operated within the design basis and within the bounds of the SOE.

Strengths are identified that are related to programs that ensure operation within normal ranges and prevent failures. The Strengths considered most relevant to Level 1 (see Appendix E), which support plant design and physical plant condition thus allowing for the most effective prevention of failures are:

- Heat Removal Systems
- Electrical Power System
- Human Factors Engineering Program
- Environmental Qualification Program
- Effective Equipment Reliability Program
- System and Component Health Reporting
- Major Components Program
- Implementation of Safe Operating Envelope Program
- Deterministic Safety Analysis
- Operationalization of Probabilistic Safety Assessment
- Probabilistic Safety Assessment

Strengths are also identified in broader areas of plant operation. These are:

- Comprehensive Set of Performance Indicators
- Use of Operating Experience and Research Findings
- Radiation Exposure Performance

- Healthy Safety Culture
- Management
- Minimum Staff Complement
- Training
- Advanced Technology to Support Radiation Protection

Note that most of these Strengths are also relevant to some of the other defence-in-depth levels.

As per the assessment in Appendix H, the aggregate impact of Acceptable Deviations associated with defence-in-depth Level 1 is assessed to be very low.

18.3.1.3. Level 1 – Improvement Initiatives

The proposed Resolution Statements to address the Global Issues (Appendix F) are relevant and significant for sustaining and enhancing defence-in-depth Level 1, particularly for the proposed extended operation period.

The Global Issues associated with the most significant proposed Resolution Statements relevant to defence-in-depth Level 1 are:

- Fitness for Service for Major Components and other SSCs for extended operation [GI-1 to GI-4, GI-5, and GI-19]
- Condition Assessments in PSR2 scope [GI-8 and GI-43]

The objective of these initiatives is to enhance confidence in the fitness-for-service for the SSCs for the extended operation period.

18.3.1.4. Level 1 – Conclusions

The assessment has confirmed that at Pickering NGS, effective Level 1 barriers are ensured through the original conservative design supplemented by design improvements implemented since initial operation, comprehensive programs in place, including strong operating and maintenance programs to ensure continued fitness for service and operation within the design basis, and ongoing continuous improvements based on national and international OPEX and evolving regulatory requirements. Given the focus and priority placed on equipment reliability to address the findings in the areas of the equipment condition, this level of defence will continue to be strong and effective for Pickering NGS.

Pickering NGS Units 1,4 and 5-8 design and operation aligns with the 36 safety principles related to defence-in-depth Level 1.

18.3.2. Level 2 – Control of Abnormal Operation and Detection of Failures

Defence-in-depth Level 2 concerns the control of abnormal operation and the detection of failures; a strong Level 2 barrier entails control of power, protective systems, and other surveillance features.

Provisions and barriers are provided to prevent or to control abnormal process conditions, with an objective to bring the plant back to normal operating conditions. Level 2 design features control abnormal plant operation, incorporating inherent characteristics, with account taken for protection against mechanisms capable of causing further deterioration.

18.3.2.1. Level 2 – Provisions and Barriers

There are 32 safety principles that apply to defence-in-depth Level 2, as shown in Appendix D. All but one of these also apply to Level 1, and many also apply to other levels. Appendix D confirms that all of the Level 2 safety principles are fulfilled by the Pickering NGS plant design, processes and management system. The safety principles in each of these three areas that are most relevant to defence-in-depth Level 2 (control of abnormal operation and detection of failures) in the context of extended operation of Pickering NGS are discussed briefly below.

With respect to the plant, the key safety principles that ensure that transients are detected and effectively controlled are:

- D-205 – Startup, Shutdown, and Low Power Operation
- D-230 – Preservation of Control Capability
- D-164 – Plant Process Control Systems
- D-192 – Protection Against Power Transient Accidents

The key safety principles related to processes and that have with a strong focus on the Level 2 barrier are:

- D-227 – Monitoring of Plant Safety Status
- O-312 – Quality Assurance in Operation
- O-284 – Operational Limits and Conditions
- O-290 – Emergency Operating Procedures

The key safety principles related to the management system and that are most closely related to the Level 2 barrier are:

- O-299 – Feedback of Operating Experience
- O-278 – Training
- O-269 – Safety Review Procedures

Alignment with these key Level 2 safety principles supports the adequacy of the Level 2 barrier at Pickering NGS. The following discussion describes the specific aspects of Pickering NGS that implement these and the other safety principles related to defence-in-depth Level 2.

Level 2 defence-in-depth is achieved by detecting changes from normal operating conditions by the Reactor Regulating System, plant process control systems and the Special Safety Systems. The control systems in Pickering NGS maintain the reactor operating conditions within the normal operating range, and effectively respond to anticipated transients to avoid the need for safety system action. Reactor control in Pickering NGS has a high degree of immunity to process upsets, measurement failures, etc., due to extensive redundancy in control devices and process measurements. The ability to maintain control in the presence of partial system failures, combined with high reliability of the Dual Computer Control System, leads to a very high availability of the Reactor Regulating System, which controls reactor power. The Reactor Regulating System prevents or minimizes transients for all but the most serious postulated initiating events.

The normal method of shutting down the reactor is by means of the Reactor Regulating System. Digital computerized monitoring of parameters important to safety is used in the design of the Reactor Regulating System. During plant upsets or potentially undesirable operating conditions, the reactor is shut down or derated automatically by the reactor setback function of the Reactor Regulating System (Units 5-8 include setback and a similar stepback function). The computer system reduces the reactor power setpoint until either the setback signal clears or the power is reduced to a specified low value.

Pickering NGS has the appropriate indications and alarms in the Main Control Room (and in secondary areas should the Main Control Room become uninhabitable) to inform operations staff of mitigating system action and the status of key plant parameters such that initiating events can be controlled.

Provisions are in place for safe handling of the Fuel from its arrival at the plant through to the IFBs and dry storage. A lattice of natural uranium and light water cannot be made critical in any configuration. Hence, no criticality concern exists in handling new Fuel or in the IFB.

The extensive use of Main Control Room simulators for training and validation of system modifications to assess their impact on other systems and human-machine interfaces provides OPG with a safe means of testing and training operators to be prepared for the detection and control of postulated initiating events, thus strengthening the Level 2 barrier.

The PROL requires that operation of Pickering NGS shall conform to the Safety Report and, hence, the safety analysis. SOE Limits and Conditions define the envelope for operating in compliance with safety analysis. A well-established framework of operating procedures is in place which includes actions on the detection of equipment malfunctions to take action ensuring that the plant stays within a well-defined SOE. This contributes significantly as a Strength to this level.

SSCs important to plant safety and reliability are continuously monitored and frequently tested to assure that they operate within their SOE and comply with associated reliability and performance requirements.

18.3.2.2. Level 2 – PSR2 Assessment

Strengths are identified that are related to programs that control changes in operation and provide means to detect failures before they occur. These identified Strengths are common between Level 1 and Level 2, and some of the other levels. The Strengths considered relevant to Level 2 (see Appendix E), which support plant physical condition and operation thus preventing progression to DBAs are:

- Heat Removal Systems
- Electrical Power System
- Human Factors Engineering Program
- Environmental Qualification Program
- Effective Equipment Reliability Program
- Major Components Program
- Implementation of Safe Operating Envelope Program
- Deterministic Safety Analysis
- Operationalization of Probabilistic Safety Assessment
- Probabilistic Safety Assessment

Strengths are also identified in broader areas of plant operation. These are:

- Comprehensive Set of Performance Indicators
- Use of Operating Experience and Research Findings
- Radiation Exposure Performance
- Healthy Safety Culture
- Management
- Minimum Staff Complement Management
- Training
- Advanced Technology to Support Radiation Protection

Note that most of these Strengths are also relevant to some of the other defence-in-depth levels.

As per the assessment in Appendix H, the aggregate impact of Acceptable Deviations associated with defence-in-depth Level 2 is assessed to be very low.

18.3.2.3. Level 2 – Improvement Initiatives

The proposed Resolution Statements to address the Global Issues (Appendix F) are relevant and significant for sustaining and enhancing the safety requirements of defence-in-depth Level 2, particularly for the proposed extended operation period.

The Global Issues associated with the most significant proposed Resolution Statements are common between Levels 1 and 2 of defence-in-depth. These improvements, which are already in progress, are:

- Fitness for Service for Major Components and other SSCs for extended operation [GI-1 to GI-4, GI-5 and GI-19]
- Condition Assessments in PSR2 scope [GI-8 and GI-43]
- Safety Analysis [GI-24, GI-31 and GI-32]

The objective of these initiatives is to enhance confidence in the fitness-for-service for the SSCs and the ability of systems to control transients.

18.3.2.4. Level 2 – Conclusions

The assessment of defence-in-depth Level 2 concludes that the provisions in place are mature and robust and will be enhanced by completion of proposed Resolution Plans related to Level 2.

Pickering NGS Units 1,4 and 5-8 design and operation aligns with the 32 safety principles related to defence-in-depth Level 2.

18.3.3. Level 3 – Control of Accidents Within the Design Basis

The third level of defence consists of barriers to minimize the consequences of accidents should they occur by providing inherent safety features, fail-safe design, additional equipment (including Emergency Mitigating Equipment), and mitigating procedures.

A strong Level 3 barrier is evidenced by the design and the robustness of engineered safety features (e.g., Special Safety Systems) coupled with correspondingly robust operating procedures. The PSR2 and the safety principles review presented in Appendix D identify several effective barriers and Strengths in the areas of Design, Safety Analysis, OPEX, and Management.

18.3.3.1. Level 3 – Provisions and Barriers

There are 38 safety principles that apply to defence-in-depth Level 3, as shown in Appendix D. Many of these apply to Levels 1 and 2 as well, some also apply to Level 4, and five safety principles apply only to Level 3. Appendix D confirms that all of these principles are fulfilled by the Pickering NGS plant design, processes and management system. The safety principles in each of these three areas that are most relevant to defence-in-depth Level 3 (control of accidents within the design basis) in the context of extended operation of Pickering NGS are discussed briefly below.

With respect to the plant, the key safety principles that assure control of accidents within the design basis are:

- D-237 – Control of Accidents within the Design Basis (unique to Level 3)
- D-182 – Equipment Qualification (unique to Level 3)
- D-168 – Automatic Safety Systems (unique to Level 3)
- D-177 – Dependent Failures (unique to Level 3)
- D-200 – Automatic Shutdown Systems
- D-207 – Emergency Heat Removal
- D-217 – Confinement of Radioactive Material
- D-221 – Protection of Confinement Structure

With respect to processes related to ensuring that accidents within the design basis are controlled, the key safety principles are:

- D-174 – Reliability Targets (unique to Level 3)
- M&C-246 – Safety Evaluation of Design
- EP-339 – Assessment of Accident Consequences and Radiological Monitoring
- O-290 – Emergency Operating Procedures

The key safety principles related to the management system and that are most closely related to the Level 3 barrier are:

- O-299 – Feedback of Operating Experience
- O-278 – Training
- O-269 – Safety Review Procedures

Alignment with these key Level 3 safety principles supports the adequacy of the Level 3 barrier at Pickering NGS. The following discussion describes the specific aspects of Pickering NGS that implement these and the other safety principles related to defence-in-depth Level 3.

In Pickering NGS, accidents within the design basis are classified as single failure (i.e., a single process failure) and dual failure (i.e., a process failure combined with a coincidental impairment of a Special Safety System). Safety system impairments considered include an assumed failure of the system to perform its safety function.

Accidents within the design basis include events that have a potential of occurring during the lifetime of the plant, events that would be classified as DBAs, and some dual failures that are BDBAs in the CNSC REGDOC-2.4.1 [15] terminology.

Pickering NGS's Special Safety Systems together with their support systems are capable of controlling accidents within the design basis, and preventing them from progressing to a severe accident.

Analyses demonstrating the effectiveness of the automatic response of the Special Safety Systems are documented in Part 3 of the Pickering 1-4 and Pickering 5-8 Safety Reports [68], [69]. Safety analyses documented in Part 3, Appendix 3 of the Pickering 1-4 and Pickering 5-8 Safety Reports demonstrate the effectiveness of the Shutdown Systems in mitigating accidents.

Unavailability and PSA models are used during the design of modifications to the plant to confirm that the systems meet their system reliability requirements. The models use component failure rates and test frequencies to arrive at predicted system unavailability. During operation, component fault data are collected, and predicted future unavailability is recalculated and reported to the CNSC on a yearly basis for Systems Important to Safety, using this actual component experience. Pickering NGS has highly reliable Shutdown Systems for safety purposes that are designed to be independent of the equipment and processes used to control the reactor power and that are poised at all times during at-power operation.

Pickering units have multiple heat removal systems of adequate capacity. Together with their support systems, these systems ensure that heat generated in the Fuel is transferred to the atmosphere or the lake under normal operating and shutdown/outage conditions, as well as in response to DBAs. These systems are:

- Steam Generators (boilers) supplied by inventory from normal or auxiliary feed water.
- Shutdown Cooling Heat Exchangers supplied by normal or Emergency High Pressure Service Water.
- Emergency heat removal with High Pressure ECI and ECI recovery, with Reactor Building Air Coolers.
- For events involving a loss of boiler inventory, in the short term, the Boiler Emergency Cooling System provides water to both Pickering 1,4 and Pickering 5-8 for reactor decay heat removal by the boilers for accident situations initiated by or leading to failure of the normal feedwater supply.
- For long-term cooling, the Emergency Boiler Water System is designed to provide emergency water to the boilers of Units 1,4.
- In Units 5-8, the Emergency Water System is a seismically qualified system supplying long-term emergency makeup to the boilers, Heat Transport System, ECI recovery heat exchangers, Containment air coolers and the Calandria.

The reactors are equipped with multiple sources of electrical power to ensure that controls and equipment important to safe operation are available. The sources of power are: the grid (off-site power), unit generator, other unit generators (Site Electrical System), Standby Generators, gas turbine/generator sets (Emergency Power System for Units 5-8), Combustion Turbine Unit driven generators (Auxiliary Power System) and batteries.

Standby power from six independent gas turbine driven Standby Generators for Units 1,4 and six for Units 5-8 is dedicated to those loads that are required for the safe shutdown of the reactor and heat sink/cooling of the Fuel core.

The Pickering NGS Units 1,4 and 5-8 designs include protection against common cause events; the design includes provisions to prevent the loss of safety functions due to damage to multiple SSCs. For Units 5-8, redundant equipment and circuits are separated and grouped (Group 1 and Group 2) to ensure the safety of the station following a common mode event. Each group is capable, independently of the other group, of safely shutting down the reactor, cooling the Fuel, and providing the operator with indications of system conditions. Units 1,4 systems have been either qualified or retrofitted to function as required for a given common mode event by ensuring effective separation and diversity. Hazard assessments have been performed that confirm the capability of the design to perform the required safety functions following postulated common mode events.

The seismically-qualified Emergency Power System for Units 5-8 supplies power to a specific portion of the Safety-Related Systems in the station. The purpose of the Emergency Power System is to independently perform the critical reactor safety functions, i.e., Control, Cool and Contain, on total loss of Group 1 distribution systems.

In addition, a third Emergency Power Generator set, consisting of three diesel generators capable of delivering a total of 2.5 MW, has been added to Units 5-8 to provide additional redundancy in the Emergency Power System.

The Auxiliary Power System is a back-up power supply system that supplies power to important Group 1 Class IV loads following a sustained total loss of Class IV power across Pickering NGS following a failure of the Bulk Electrical System. The system performs this function by supplying power to the Site Electrical System.

The ECI System is capable of restoring and maintaining fuel cooling following a LOCA.

One of the proven safety features is the ability to cool the Fuel through natural circulation that occurs in the Heat Transport System when forced circulation is lost.

The Pickering NGS Containment System is designed to retain radioactive material that might be released from the Fuel during an accident, and Containment effectiveness is demonstrated by safety analyses showing that public dose meets the applicable regulatory limits for the full range of accidents considered in the design.

Operating Manuals and Abnormal Incident Manuals are designed to allow the operator to respond to events. If actions identified in Operating Manuals and Abnormal Incident Manuals are unsuccessful in terminating the accident progression, actions will be taken per the Emergency Mitigating Equipment Guidelines to prevent the accident from progressing to a severe accident. Also, post-accident monitoring capability of the critical safety parameters is available. By monitoring critical safety parameters, and following critical safety parameter restoration procedures via field actions, it is possible to maintain the Control, Cool, Contain and Monitor safety functions, including from locations external to the Main Control Room.

A Hydrogen Ignition System is provided to safely combine any hydrogen (or deuterium) gases generated in Containment following DBAs that may produce hydrogen. The system is capable of addressing very low concentrations of hydrogen in air/steam mixtures, thus preventing the occurrence of high concentrations of hydrogen in Containment. In addition, the installation of Passive Auto-catalytic Recombiners provides additional redundancy to the Hydrogen Ignition System for control of hydrogen in Containment and thus ensures Containment integrity, therefore strengthening the capability of Pickering NGS to control accidents within the design basis. The Fukushima review and the actions to address events including the unlikely event of an extended loss of off-site power have further strengthened the robustness of the plant.

18.3.3.2. Level 3 – PSR2 Assessment

Confirmation of the effectiveness of the Level 3 barriers has been significantly enhanced with the completion of the CNSC S-294 [70] compliant PSA, which included internal hazards fire and flooding and external hazards seismic and high winds. This will be further enhanced by completion of the implementation strategy for CNSC REGDOC-2.4.2 [71]. The effectiveness of Level 3 barriers is also confirmed by conformance with CNSC S-98 [72] for specifying reliability targets for systems important to safety at Pickering NGS and monitoring the system reliability performance against those targets.

Strengths are identified that are related to programs that control accidents within the design basis. The Strengths considered most relevant to Level 3 (see Appendix E) and other levels, which support the plant design in meeting requirements for DBAs and for preventing progression to BDBA, are:

- Heat Removal Systems
- Electrical Power System
- Human Factors Engineering Program
- Environmental Qualification Program
- Effective Equipment Reliability Program
- Major Components Program
- Implementation of Safe Operating Envelope Program
- Deterministic Safety Analysis
- Operationalization of Probabilistic Safety Assessment
- Probabilistic Safety Assessment

Strengths are also identified in broader areas of plant operation. These are:

- Comprehensive Set of Performance Indicators
- Use of Operating Experience and Research Findings
- Radiation Exposure Performance
- Healthy Safety Culture

- Management
- Effective Training Programs
- Minimum Staff Complement

As per the assessment in Appendix H, the aggregate impact of Acceptable Deviations associated with defence-in-depth Level 3 is assessed to be very low.

18.3.3.3. Level 3 – Improvement Initiatives

The proposed Resolution Statements to address the Global Issues (Appendix F) are relevant for sustaining and enhancing the safety requirements of defence-in-depth Level 3.

The Global Issues associated with the most significant proposed Resolution Statements relevant to Level 3 are:

- PSA Risk Reduction Plan [GI-27]
- Buried Piping Risk [GI-9]
- Safety Analysis [GI-24, GI-31 and GI-32]

18.3.3.4. Level 3 – Conclusions

The PSR2 assessments and the review of safety principles show an effective Level 3 barrier. Adequate and effective provisions for the control of accidents within the design basis are provided at Pickering NGS. Operators have indications and alarms as well as the capability to perform actions from the Main Control Room for this purpose.

The review confirms that the Pickering NGS has strong Level 3 barriers due to the high quality of the design that includes extensive mitigating provisions, comprehensive accident management procedures, and a robust set of safety analyses.

18.3.4. Level 4 – Control of Severe Plant Conditions

Defence-in-depth Level 4 concerns the control of severe plant conditions, and includes the prevention of accident progression and the mitigation of severe consequences resulting from initial accidents. A strong Level 4 barrier is evidenced by strong complementary measures coupled with robust accident management strategies.

18.3.4.1. Level 4 – Provisions and Barriers

There are 34 safety principles that apply to defence-in-depth Level 4, as shown in Appendix D. Many apply to Levels 1, 2 and 3 as well. Some also apply to Level 5. Three safety principles apply only to Level 4. Appendix D confirms that all of these principles are fulfilled by the Pickering NGS plant design, processes and management system. The safety principles in each of these three areas that are most relevant to Level 4 (control of severe plant conditions) in the context of extended operation of Pickering NGS are discussed briefly below.

With respect to the plant, the key safety principles that are relevant to Level 4 are:

- AM-326 – Engineered Features for Accident Management (unique to Level 4)
- EP-336 – Emergency Response Facilities
- D-233 – Station Blackout
- D-221 – Protection of Confinement Structure
- D-207 – Emergency Heat Removal
- D-227 – Monitoring of Plant Safety Status

With respect to processes related to control of severe plant conditions, the key safety principles are:

- AM-318 – Strategy for Accident Management (unique to Level 4)
- AM-323 – Training and Procedures for Accident Management (unique to Level 4)
- O-290 – Emergency Operating Procedures
- EP-333 – Emergency Plans
- EP-339 – Assessment of Accident Consequences and Radiological Monitoring

The key safety principles related to the management system and that are most closely related to the Level 4 barrier are:

- O-299 – Feedback of Operating Experience
- O-278 – Training
- O-269 – Safety Review Procedures

Safety Principle O-299, Feedback of Operating Experience, is particularly important with respect to lessons learned from the 2011 Fukushima accident.

Alignment with these key Level 4 safety principles supports the adequacy of the Level 4 barrier at Pickering NGS. The following discussion describes the specific aspects of Pickering NGS that implement these and the other safety principles related to defence-in-depth Level 4.

Pickering NGS Units 1,4 and 5-8 have complementary design features for BDBAs. Some of these design features include redundant power systems, design adequacy for internal and external hazards, and hydrogen igniters and Passive Auto-catalytic Recombiners. If actions identified in Operating Manuals and Abnormal Incident Manuals are unsuccessful in terminating the accident progression, actions will be taken per the Emergency Mitigating Equipment Guidelines to prevent the accident from progressing to a severe accident.

Pickering NGS has implemented Severe Accident Management Guidelines for mitigating severe accident progression and protecting Containment. Severe Accident Management Guidelines are a set of written guidance to implement strategies should a BDBA progress to a severe accident. The goals of Severe Accident Management Guidelines are to terminate progression of core damage, if possible, by restoring cooling, and to protect Containment. The Severe Accident Management Guidelines aid in identifying longer-term actions that are required to address post-accident conditions once the station has been returned to a controlled, stable state.

In terms of maintenance of fuel cooling, a key line of defence for BDBAs is Emergency Boiler Makeup using Emergency Mitigating Equipment. An additional line of defence is Emergency Mitigating Equipment to provide water makeup to the Moderator. These features are capable of limiting core damage and support Containment pressure control for most BDBAs.

Emergency Mitigating Equipment has been provided for an additional makeup water supply for accidents beyond the design basis. Portable diesel-powered pumps can be deployed to provide cooling water make-up to the secondary side of the boilers, to the Heat Transport System and to the Moderator. Additional Fukushima Project design modifications have been installed to enhance water makeup/cooling capability for the IFBs.

In addition, implementation of Phase 2 Emergency Mitigating Equipment modifications is in progress. Phase 2 Emergency Mitigating Equipment includes restoration of power to key station equipment to support core cooling and monitoring and to Containment coolers to support control of Containment pressure following BDBAs.

Particular importance is placed on protecting Containment. A strong Level 4 barrier is evidenced by a robust Containment design as well as use of complementary design features and procedures to halt accident progression and mitigate the consequences of Beyond Design Basis conditions.

A Hydrogen Ignition System is provided to safely combine hydrogen (or deuterium) gases generated in Containment. The system is capable of addressing very low concentrations of hydrogen in air/steam mixtures, thus preventing the occurrence of high concentrations of hydrogen in Containment. In addition, Passive Auto-catalytic Recombiners have been installed to control Containment hydrogen levels at reduced hydrogen generation rates, and therefore eliminate reliance on the Hydrogen Ignition System in the longer term.

Emergency planning also encompasses many of the concepts of the Level 4 barrier. A mature emergency response infrastructure is in place, and the requisite qualified staff and expertise are maintained. As well, a *Mutual Aid Agreement for Nuclear Emergency Support* [67] among emergency support organizations from the Canadian nuclear operators has been established for inter-utility emergency support. This aspect of Level 4 is identified as a Strength.

18.3.4.2. Level 4 – PSR2 Assessment

The PSR2 review confirmed that committed Fukushima Action Items that control the progression and consequences of severe accidents have been or are being implemented, including:

- Installation of Passive Auto-catalytic Recombiners on all units (completed)
- Addition of Emergency Mitigating Equipment including portable diesel pumps and diesel generators (completed)
- Enhancements to water makeup/cooling capability for the IFBs (completed)
- Additional flood barriers installed around the Pickering A Standby Generator Fuel Forwarding Pump house (completed)
- Enhancements to Severe Accident Management Guidelines (completed)
- Additional Phase 2 Emergency Mitigating Equipment provisions (in progress)

The review in the Safety Factor 13 Report has confirmed that OPG Nuclear has: a) adequate plans, staff, facilities and equipment in place for dealing with a full range of emergencies, and b) there is regular emergency training and exercises, and adequate arrangements are in place for effective interaction and coordination with local and national authorities.

Strengths are identified that are related to programs that control accidents within and beyond the design basis. The Strengths considered most relevant to Level 4 (see Appendix E), which support plant design improvements thus allowing for the most effective control of severe accident conditions are:

- Implementation of Fukushima Action Items
- Use of Operating Experience and Research Findings
- Emergency Management

As per the assessment in Appendix H, the aggregate impact of Acceptable Deviations associated with defence-in-depth Level 4 is assessed to be very low.

18.3.4.3. Level 4 – Improvement Initiatives

The proposed Resolution Statements to address the Global Issues (Appendix F) are relevant for sustaining and enhancing the safety requirements of defence-in-depth Level 4.

The Global Issues associated with the most significant proposed Resolution Statements relevant to defence-in-depth Level 4 are:

- PSA Risk Resolution Plan [GI-27]
- Implementation of Emergency Mitigating Equipment [GI-40]

18.3.4.4. Level 4 – Conclusions

The measures considered at the first three levels will ensure maintenance of the structural integrity of the core and limit potential radiation hazards for members of the public. The PSR2 assessments and the review of safety principles show that additional design features and procedural provisions are in place and are effective for defence-in-depth Level 4.

The complete implementation of Severe Accident Management Guidelines and the OPEX from the 2011 Fukushima accident has also significantly strengthened this level. OPG is continuing to implement enhancements, such as the Phase 2 Emergency Mitigating Equipment, that will further strengthen defence-in-depth Level 4.

18.3.5. Level 5 – Mitigation of Radiological Consequences

Defence-in-depth Level 5 concerns the mitigation of radiological consequences of releases of radioactive material; a strong barrier indicates a robust emergency response program.

18.3.5.1. Level 5 – Provisions and Barriers

As shown in Appendix D, seven safety principles apply to defence-in-depth Level 5, and of these, three apply to all other levels. One safety principle also applies to Levels 3 and 4, two apply to Levels 4 and 5 and one applies only to Level 5. Appendix D confirms that all of these safety principles are fulfilled. The safety principles that apply to defence-in-depth Level 5 (mitigation of radiological consequences of significant releases of radioactive materials) are:

- S-140 – Feasibility of Emergency Plans (unique to Level 5)
- EP-333 – Emergency Plans
- EP-336 – Emergency Response Facilities
- EP-339 – Assessment of Accident Consequences and Radiological Monitoring
- S-138 – Radiological Impact on the Public and the Local Environment
- O-265 – Organization, Responsibilities and Training
- O-296 – Engineering and Technical Support of Operations

Alignment with these key Level 5 safety principles supports the adequacy of the Level 5 barrier at Pickering NGS. The following discussion describes the specific aspects of Pickering NGS that implement the safety principles related to Level 5.

In the event of an emergency, there are permanent on-site and off-site facilities appropriately equipped for effective emergency response. During the initial emergency phase, Main Control Room staff perform the assessment of plant status. Where possible, the Main Control Room staff also identify damage to plant equipment. Appropriate plant procedures are used and Main Control Room staff initiate an immediate operations response to move towards taking the plant to a safe and stable configuration. The Main Control Room team utilizes resources of the on-site Emergency Operations Centre to mobilize and deploy the necessary emergency teams. Main Control Room staff will seek, as appropriate, consultative, technical and resource assistance from the Site Management Centre who will in turn seek, as appropriate, assistance from the Corporate Emergency Operations Facility. During the response phase, the Main Control Room and Site Management Centre staff will, as a continuing action, evaluate the implementation of the emergency mitigation strategy and modify it as necessary.

The Provincial Emergency Operations Centre, located at the Ministry of Community Safety and Correctional Services in Toronto, is the provincial facility and organization that directs off-site emergency response operations.

Additional off-site emergency response facilities include:

- Durham Region Reception Centres and Emergency Worker Centres
- City of Toronto, Municipal Emergency Operations Centre

OPG Program N-PROG-RA-0001, *Consolidated Nuclear Emergency Plan* [73] provides a written basis that documents concepts, roles and resources required by OPG Nuclear to implement and maintain its emergency response capability to protect the public, employees and the environment in the event of a nuclear emergency. The *Consolidated Nuclear Emergency Plan* [73] defines a nuclear emergency as an emergency which poses an actual or potential hazard to public health and property or the environment from ionizing radiation, whose source is a major nuclear installation. Emergency plans have been prepared under the *Consolidated Nuclear Emergency Plan* [73], and emergency preparedness drills and exercises are mandated and are carried out for Pickering NGS.

The Provincial Nuclear Emergency Response Plan [74] requires Pickering NGS to procure adequate quantities of stable iodine tablets for their Primary Zone population. Potassium Iodide (KI) tablets have been distributed to residences and businesses in the Primary Zone. Designated Primary Zone municipalities are also required to establish and maintain a public alerting system in accordance with the Provincial Nuclear Emergency Response Plan.

The adequacy of the response and mitigation strategies that have been developed is demonstrated primarily through drills and exercises. On an annual basis, OPG's Emergency Preparedness Department assesses the Emergency Response Organization performance and reviews all drill and exercise related corrective actions to monitor status and ensure completeness. As part of the lessons learned from the 2011 Fukushima accident, Beyond Design Basis Emergency Response drills and exercises have been incorporated into the plans.

Integrated and partial emergency exercises have been conducted at Pickering NGS to confirm the satisfactory function of the emergency organization and its equipment.

18.3.5.2. Level 5 – PSR2 Assessment

Both the PSR2 and the safety principles review confirm that OPG has the following significant Strength related to defence-in-depth Level 5:

- Emergency Management.

The program is well established and coordinated both within and external to OPG.

As per the assessment in Appendix H, there are no Acceptable Deviations associated with defence-in-depth Level 5, so there is no impact of Acceptable Deviations on Level 5.

18.3.5.3. Level 5 – Improvement Initiatives

Completion of the Emergency Response Projection enhancements [GI-26] is relevant for sustaining and enhancing the safety requirements of defence-in-depth Level 5.

18.3.5.4. Level 5 – Conclusions

The coordinated emergency response capability of the various response organizations and the implementation of OPEX from the 2011 Fukushima accident support the strength of the Level 5 defence-in-depth provisions. Implementation of the planned improvement initiatives will further enhance the barriers for Level 5 at Pickering NGS.

18.4. Defence-in-Depth Assessment Conclusions

The detailed review of provisions for each level of defence presented above and in Appendix D shows that Pickering NGS Units 1,4 and 5-8 design and operation have adequate and effective barriers in all applicable levels of defence-in-depth and that significant improvements have been implemented since the plant was put into service.

The comprehensiveness of the assessment is assured by assessing each of the safety principles, which are supported by multiple and overlapping provisions, for each level of defence-in-depth. Furthermore, each defence-in-depth level is supported by multiple safety principles providing a second layer of overlap of provisions across levels of defence-in-depth.

The adequacy of these provisions has also been confirmed by the comprehensive PSAs. The Pickering Units 1,4 and 5-8 PSAs demonstrate that the overall plant design has a Core Damage Frequency and Large Release Frequency within the specified risk-based Safety Goals, indicating robustness in the design, and reliable equipment that is capable of responding effectively to accident scenarios. The defence-in-depth will be further strengthened with the implementation of the proposed Resolution Plans.

Implementation of lessons learned from the 2011 Fukushima accident, and installation of additional Emergency Mitigating Equipment for BDBAs has added further capability to the defence-in-depth. The development of Beyond Design Basis Emergency Mitigating Equipment Guidelines and enhancement to the Severe Accident Management Guidelines, to implement the requirements of OPG Standard N-STD-MP-0019, *Beyond Design Basis Accident Management* [75], have resulted in Pickering NGS strengthening its capability to mitigate and respond to low probability BDBAs. The continued implementation of the remaining post-Fukushima initiatives underway at Pickering will provide additional robustness to an already strong program. Implementation of the proposed Resolution Plans will further enhance the defence-in-depth for extended operation.

19. Conclusion of the Assessment of Overall Acceptability of Operation of the Plant

The Global Assessment demonstrates that Pickering NGS will operate safely during the extended operating period. In addition, activities are in progress and planned that will further enhance safe plant operation. The justification for this conclusion is based on the following:

Current Plant State:

i) The Pickering Station Leadership Team has effectively aligned the organization to significantly improve performance in a number of key focus areas. Station performance improvement has been recognized through industry reviews. The plant is safe, and is operated safely.

ii) OPG has comprehensive programs in place to ensure the condition of SSCs important to safety at Pickering 1,4 and Pickering 5-8 is well understood, to assess the level of fitness for service, and to effectively take action to maintain good plant condition. This has led to continuous improvement in the condition of the plant, and plant performance.

iii) OPG has made significant improvements to the Pickering plant design and processes. The plant design enhancements, discussed in Section 18, together with the process enhancements, closely align the plant with safety-significant requirements of modern codes and standards (which in some cases are beyond current requirements), and enhance defence-in-depth. In particular, enhancements made in response to the 2011 Fukushima accident have reduced, and will further reduce, the risk associated with BDBAs.

iv) Design and operation of the plant meet the current deterministic safety analysis dose limits, and processes are in place to ensure the safety analysis accounts for any additional aging effects associated with extended operation. The Probabilistic Safety Assessment shows that the OPG risk-based Safety Goals for Core Damage Frequency and Large Release Frequency are met. Initiatives have been proposed to further enhance the margins to these goals.

v) Radiological dose performance and environmental impact performance are significantly better than regulatory limits. Programs in place ensure the ongoing effectiveness of the radiological protection of workers, the public and the environment.

Results of the Periodic Safety Review:

i) The Global Assessment identifies 24 Strengths (refer to Section 16), indicating that Pickering NGS is well aligned with modern codes, standards and practices in key areas.

ii) The Global Assessment identifies 51 Global Issues (refer to Section 11). Proposed Resolution Plans for Global Issues are developed, and many are in progress to further enhance safety (refer to Section 13), including enhancements to further reduce the risk associated with BDBAs. Most of the proposed Global Issue Resolution Plan actions reflect existing work programs and plans at the station. In particular, for the Global

Issues of highest safety significance (i.e., fitness for service to cover the extended operating period), OPG was already fully aware of these issues and is actively working on addressing them for the extended operating period. None of the Global Issues identify a safety concern that requires additional planned or urgent action to be taken. The proposed Resolution Plans will be supported by specific actions in the Integrated Implementation Plan.

iii) The Global Issues of highest safety significance pertain to fitness for service of SSCs important to safety over the extended operating period. Units will be operated only if fitness for service of SSCs important to safety is assured. OPG has comprehensive programs in place to ensure the condition of SSCs important to safety is well understood, to assess the level of fitness for service, and to effectively take action to maintain good plant condition. The proposed Resolution Plans for these Global Issues will ensure the ongoing fitness for service of SSCs for the operational life of the plant, and these plans are actively being progressed.

iv) The Global Assessment includes a Resolution Plan that proposes the investigation and implementation of design, operational, and/or analytical options to further enhance margins to risk-based Administrative Safety Goals.

v) The assessment of Acceptable Deviations (refer to Appendix H) confirms there is no impact on the conclusion of the Global Assessment, either individually or in aggregate.

vi) The assessment of defence-in-depth of the plant (refer to Section 18) includes a detailed review and confirmation of the adequacy of the provisions for each level of defence. This is based on an assessment of how the related safety principles for each level of defence-in-depth are met, taking into account the plant design, the ongoing operations and maintenance activities at the plant, the identified Strengths, as well as the proposed enhancements identified in the Global Assessment process. The assessment also accounts for the aggregate effect of Acceptable Deviations. The Defence-in-Depth Assessment shows that Pickering Units 1,4 and Pickering Units 5-8 design and operation have adequate and effective barriers in all levels of defence-in-depth.

vii) OPG's organizational structure and management system provides the requisite processes, tools, resources and oversight that will ensure continued safe operation of the plant.

In summary, the current plant design, operation, processes and management system will ensure continued safe operation of Pickering 1,4 and Pickering 5-8 both in the short term, and for extended operation. OPG and the Pickering Station Leadership Team are committed to investing in the plant, and focusing the organization to strive for continued improvement in the plant condition, operation and performance.

Part V: References

Section	
#	**Title**
20	References

20. References

[1] OPG Report, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, P-REP-03680-00001-R002, June 2016.

[2] CNSC Regulatory Document, *Periodic Safety Reviews*, REGDOC-2.3.3, April 2015.

[3] IAEA Specific Safety Guide, *Periodic Safety Review of Nuclear Power Plants*, SSG-25, March 2013.

[4] OPG-CNSC Protocol, *OPG-CNSC Protocol for the Conduct of a Periodic Safety Review in Support of Pickering NGS Licence Renewal*, Revision 1, e-Doc 5143721, P-CORR-00531-04725 R001, January 17, 2017.

[5] OPG Report, *Pickering 5-8 Continued Operations Plan Review in Support of PNGS Periodic Safety Review 2 (PSR2)*, P-REP-03680-00024-R000, January 2017.

[6] OPG Report, *Fukushima Action Item Review in Support of PNGS Periodic Safety Review 2 (PSR2)*, P-REP-03680-00022 R000, February 6, 2017.

[7] IAEA Report, *Defence in Depth in Nuclear Safety*, INSAG-10, June 1996.

[8] IAEA Report, *Assessment of Defence in Depth for Nuclear Power Plants*, Safety Report Series No. 46, February 2005.

[9] IAEA Report, *Basic Safety Principles for Nuclear Power Plants*, 75-INSAG-3 Rev. 1, INSAG-12, October 1999.

[10] OPG Plan, *Pickering 2024 Securing Ontario's Clean Power Future Scope Review Board – Terms of Reference*, P-PLAN-09710-00001-R001, February 02, 2017.

[11] OPG Report, *Pickering NGS PSR2 Safety Factor 1 Report: Plant Design*, P-REP-03680-00008 R000, March 2017.

[12] OPG Report, *Pickering NGS PSR2 Safety Factor 4 Report: Aging*, P-REP-03680-00007 R000, July 2016.

[13] OPG Report, *Pickering NGS PSR2 Safety Factor 2 Report - Actual Condition of Structures, Systems, and Components Important to Safety*, P-REP-03680-00005 R001, March 2017.

[14] OPG Report, *Pickering NGS PSR2 Safety Factor 3 Report: Equipment Qualification (Seismic and Environmental)*, P-REP-03680-00006 R000, July 2016.

[15] CNSC Regulatory Document, *Deterministic Safety Analysis*, REGDOC-2.4.1, May 2014.

[16] OPG Report, *Pickering NGS PSR2 Safety Factor 5 Report: Deterministic Safety Analysis*, P-REP-03680-00009 R000, March 2017.

[17] OPG Report, *Pickering NGS PSR2 Safety Factor 6 Report: Probabilistic Safety Assessment*, P-REP-03680-00010 R000, March 2017.

[18] OPG Report, *Pickering NGS PSR2 Safety Factor 7 Report: Hazard Analysis*, P-REP-03680-00011 R000, March 2017.

[19] OPG Report, *Pickering NGS PSR2 Safety Factor 8 Report: Safety Performance*, P-REP-03680-00012 R000, December 2016.

[20] OPG Report, *Pickering NGS PSR2 Safety Factor 9 Report: Use of Experience from Other Nuclear Power Plants and Research Findings*, P-REP-03680-00013 R000, October 2016.

[21] OPG Report, *Pickering NGS PSR2 Safety Factor 10 Report: Organization, Management System, and Safety Culture*, P-REP-03680-00014 R000, December 2016.

[22] OPG Report, *Pickering NGS PSR2 Safety Factor 11 Report: Procedures*, P-REP-03680-00015 R000, October 2016.

[23] OPG Report, *Pickering NGS PSR2 Safety Factor 12 Report: Human Factors*, P-REP-03680-00016 R000, December 2016.

[24] OPG Report, *Pickering NGS PSR2 Safety Factor 13 Report: Emergency Planning*, P-REP-03680-00017 R000, December 2016.

[25] OPG Report, *Pickering NGS PSR2 Safety Factor 14 Report: Radiological Impact on the Environment*, P-REP-03680-00018 R000, December 2016.

[26] OPG Report, *Pickering NGS PSR2 Safety Factor 15 Report: Radiation Protection*, P-REP-03680-00019 R001, April 2017.

[27] CNSC Correspondence, M. Santini to B. Phillips, *Pickering NGS: CNSC Staff Assessment of 2013 COP, SOP and CAL Closure of AI 2013-48-4185*, e-Doc 4452163, OPG File No. P-CORR-00531-04272 R000, June 18, 2014.

[28] CNSC Correspondence, *CNSC staff Assessment of Pickering NGS 2013 EOL Consolidated Actions Log (CAL) (COP, SOP, SAP and SSP)*, e-Doc 4454610, Enclosure to OPG File No. P-CORR-00531-04272 R000, June 18, 2014.

[29] OPG Correspondence, B. McGee to M. Santini, *Pickering 5-8, Continued Operations Plan – 2015 Final Update*, P-CORR-00531-04470 R000, December 15, 2015.

[30] OPG Plan, *Pickering 5-8 Continued Operations Plan*, NK30-PLAN-00531-00001 R005, December 2015.

[31] CNSC Correspondence, H. Khouaja to B. McGee, *Pickering NGS: CNSC Staff Assessment of 2015 COP, SOP, SAP and CALs*, e-Doc 5024526, OPG File No. P-CORR-00531-04786 R000, July 6, 2016.

[32] CNSC Correspondence, G. Rzentkowski to W. M. Elliott, *Opening of Fukushima Action Items (FAIs) on Ontario Power Generation*, N-CORR-00531-05607, February 17, 2012.

[33] OPG Report, *Fukushima Action Item Status Report*, N-REP-03600-10003 R007, November 27, 2015.

[34] CNSC Correspondence, A. Viktorov to R. Lockwood, *Pickering NGS: PSR2 – CNSC Staff Assessment of OPG Safety Factor Report 1*, e-Doc 5305945, P-CORR-00531-05107, July 26, 2017.

[35] CNSC Correspondence, A. Viktorov to R. Lockwood, *Pickering NGS: PSR2 – CNSC Staff Assessment of OPG Safety Factor Report 2, Rev 1*, e-Doc 5295534, P-CORR-00531-05099, July 12, 2017.

[36] CNSC Correspondence, G. Frappier to B. McGee, *Pickering NGS: PSR2 – CNSC Staff Assessment of OPG SFR2 (Rev 0), SFR3 and SFR4*, e-Doc 5094505, OPG File No. P-CORR-00531-04864, October 21, 2016.

[37] CNSC Correspondence, A. Viktorov to B. McGee, *Pickering NGS: PSR2 – CNSC Staff Supplementary Assessment of OPG SFR3 and SFR4*, e-Doc 5137380, OPG File No. P-CORR-00531-04907, December 2, 2016.

[38] CNSC Correspondence, A. Viktorov to R. Lockwood, *Pickering NGS: PSR2 – CNSC Staff Assessment of OPG Safety Factor Reports 5, 6, and 7*, e-Doc 5289137, P-CORR-00531-05090, June 30, 2017.

[39] CNSC Correspondence, A. Viktorov to R. Lockwood, *Pickering NGS: Periodic Safety Review 2 (PSR2) – CNSC Staff Assessment of OPG Safety Factor Reports (SFR) 8, 10, 12, 13 and 14*, e-Doc 5233089, P-CORR-00531-05036, May 2, 2017.

[40] CNSC Correspondence, A. Viktorov to B. McGee, *Pickering NGS: PSR2 – CNSC Staff Assessment of OPG SFR9, SFR11, and SFR15*, e-Doc 5158719, OPG File No. P-CORR-00531-04950, January 25, 2017.

[41] CNSC Correspondence, A. Viktorov to R. Lockwood, *Pickering NGS: CNSC Staff Review of OPG Safety Factor Report 15, Rev 1*, e-Doc 5257495, P-CORR-00531-05059, May 26, 2017.

[42] CNSC Correspondence, A. Viktorov to B. McGee, *Pickering NGS: CNSC Staff Review of OPG's Reassessment of COP Actions for Consideration in the PSR2*, e-Doc 5189874, P-CORR-00531-04973, February 24, 2017.

[43] CNSC Correspondence, A. Viktorov to B. McGee, *Pickering NGS: CNSC Staff Review of OPG's Reassessment of Fukushima Action Items for Consideration in the Periodic Safety Review 2*, e-Doc 5210355, P-CORR-00531-05003, March 22, 2017.

[44] Government of Canada, *Nuclear Safety and Control Act*, S.C. 1997, c. 9, July 3, 2013.

[45] Government of Canada, *General Nuclear Safety and Control Regulations*, SOR/2000-202, June 12, 2015.

[46] OPG Correspondence, R. Lockwood to A. Viktorov, *Pickering Periodic Safety Review 2-Process for Addressing CNSC Identified Additional Gaps*, P-CORR-00531-05132, September 18, 2017.

[47] OPG Program, *Equipment Reliability*, N-PROG-MA-0026-R002, June 4, 2015.

[48] OPG Program, *Integrated Aging Management*, N-PROG-MP-0008-R006B, May 2, 2016.

[49] OPG Program, *Major Components*, N-PROG-MA-0025-R002, March 25, 2015.

[50] OPG Charter, *Nuclear Management System*, N-CHAR-AS-0002-R019, August 31, 2016.

[51] OPG Program, *Design Management*, N-PROG-MP-0009-R012, April 24, 2017.

[52] OPG, Procedure, *General Requirements for Seismic Qualification of OPG Nuclear Facilities*, N-STD-MP-0025-R002, October 27, 2016.

[53] OPG Program, *Environmental Qualification*, N-PROG-RA-0006-R008, May 28, 2015.

[54] OPG Program, *Risk and Reliability Program*, N-PROG-RA-0016-R009, May 27, 2016.

[55] OPG Program, *Fire Protection*, N-PROG-RA-0012-R011, July 28, 2015.

[56] OPG Report, *Pickering NGS Periodic Safety Review 2: Code and Standard Reviews for Safety Factors 2 (Actual Condition of SSCs), 3 (Equipment Qualification) and 4 (Aging)*, P-REP-03680-00004 R000, July 2016.

[57] OPG Report, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 9, 11, and 15*, P-REP-03680-0586480 R000, September 2016.

[58] OPG Report, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 8, 10, 12, 13, and 14*, P-REP-03680-00021 R000, December 2016.

[59] OPG Report, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 1, 5, 6, and 7*, P-REP-03680-00029 R000, March 2017.

[60] CNSC Report, *Regulatory Oversight Report for Canadian Nuclear Power Plants: 2015*, October 2016.

[61] J. Cheng to M. Ruffolo, *Re: PSR1 Acceptable Deviations Reassessed for Pickering PSR2*, P-CORR-03680-0620816 Rev 000, March 31, 2017.

[62] OPG Policy, *Nuclear Safety Policy*, N-POL-0001-R003, March 6, 2014.

[63] OPG Standard, *Nuclear Management Systems Organizations*, N-STD-AS-0020-R014, May 19, 2016.

[64] OPG Program, *Engineering Change Control*, N-PROG-MP-0001-R015, May 12, 2017.

[65] OPG Program, *Reactor Safety Program*, N-PROG-MP-0014-R006, June 1, 2016.

[66] OPG Program, *Human Performance*, N-PROG-AS-0002-R016, April 28, 2016.

[67] OPG File, *Mutual Aid Agreement for Nuclear Emergency Support*, N-LEGL-03490-0413370-R0, November 30, 2012.

[68] OPG Safety Report, *Pickering Nuclear 1-4 Safety Report: Part 3 – Accident Analysis*, NA44-SR-01320-00002-R004, October 31, 2013.

[69] OPG Safety Report, *Pickering Nuclear 5-8 Safety Report: Part 3 – Accident Analysis*, NK30-SR-01320-00003-R004, October 30, 2014.

[70] CNSC Regulatory Standard, *Probabilistic Safety Assessment (PSA) for Nuclear Power Plants*, S-294, April 2005.

[71] CNSC Regulatory Document, *Probabilistic Safety Assessment (PSA) for Nuclear Power Plants*, REGDOC-2.4.2, May 2014.

[72] CNSC Regulatory Standard, *Reliability Programs for Nuclear Power Plants*, S-98 REV.1, July 2005.

[73] OPG Program, *Consolidated Nuclear Emergency Plan*, N-PROG-RA-0001-R015, October 28, 2016.

[74] Province of Ontario, *Provincial Nuclear Emergency Response Plan*, 2009 (available at www.emergencymanagementontario.ca).

[75] OPG Standard, *Beyond Design Basis Accident Management*, N-STD-MP-0019-R002, July 13, 2016.

Appendix A – Modern Laws, Regulations, Codes and Standards Assessed in PSR2

The modern Laws, Regulations, Codes and Standards in the PSR2 Assessment Basis (refer to Section 2.6.2 of the PSR2 Basis Document [P-REP-03680-00001-R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016]) are listed in this appendix. The freeze date was January 15, 2016. Where a document was issued in January 2016 but the effective date is unknown, it was considered to have been issued prior to the freeze date. The process for selecting PSR2 Laws, Regulations, Codes and Standards is described in Section D.1.0 of the PSR2 Basis Document [P-REP-03680-00001-R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016]. The assessments were performed on the safety significant requirements.

Three types of assessments were considered for PSR2:

- Clause-by-Clause review (C): New Laws, Regulations, Codes and Standards referenced in Pickering PROL 48.02/2018 (listed in Appendix C of the Licence Conditions Handbook) were subjected to a clause-by-clause type review. In a clause-by-clause review, conformance with individual clauses is demonstrated by supporting evidence stating whether the requirements stipulated in the requirement document are met.
- High Level review (HL): New Laws, Regulations, Codes and Standards not referenced in Pickering PROL 48.02/2018 but which are in the PSR2 Assessment Basis were subjected to a high level review. In a high level review, the degree of conformance with clauses or groups of clauses in the Law, Regulation, Code or Standard is demonstrated by supporting evidence stating whether the intent of the requirements stipulated in the requirement document is met.
- Incremental review (I): For Laws, Regulations, Codes and Standards that have been reviewed in PSR1 but have had revisions since the last review, a topical review of the changes was performed.

The majority of modern Laws, Regulations, Codes and Standards in the PSR2 Assessment Basis were subjected to an incremental review as part of PSR2. The following table lists the modern Laws, Regulations, Codes and Standards considered in PSR2 and shows the Safety Factor Reports that contain the corresponding assessments, along with the PSR2 Gaps related to each modern Law, Regulation, Code or Standard.

#	Document Number	Document Title	Type of Review	Modern Version for PSR2	Safety Factor Report															PSR2 Gaps
					SFR1	SFR2	SFR3	SFR4	SFR5	SFR6	SFR7	SFR8	SFR9	SFR10	SFR11	SFR12	SFR13	SFR14	SFR15	
Documents Referenced in Pickering PROL 48.02/2018																				
1	CSA N286	Management System Requirements for Nuclear Power Plants	I	N286-12					x	x			x	x	x					-
2	CSA N290.15	Requirements for the Safe Operating Envelope of Nuclear Power Plants	I	N290.15-10					x			x								-
3	CSA N286.7	Quality Assurance Of Analytical, Scientific And Design Computer Programs For Nuclear Power Plants	I	N286.7-16	x				x	x	x			x						-
4	CSA N285.0	General Requirements For Pressure-Retaining Systems And Components in CANDU Nuclear Power Plants	I	N285.0-12	x															SF1-1 SF1-2
5	CSA N290.13	Environmental Qualification of Equipment for CANDU Nuclear Power Plants	I	N290.13-05			x	x												-
6	CSA N285.4	Periodic Inspection Of CANDU Nuclear Power Plant Components	I	N285.4-14	x	x		x												SF2-AG10 SF4-3 SF4-4 SF4-5 SF4-6 SF4-7 SF4-8
7	CSA N285.5	Periodic Inspection Of CANDU Nuclear Power Plant Containment Components	I	N285.5-13	x	x	x	x												SF4-9 SF4-10

#	Document Number	Document Title	Type of Review	Modern Version for PSR2	Safety Factor Report															PSR2 Gaps
					SFR1	SFR2	SFR3	SFR4	SFR5	SFR6	SFR7	SFR8	SFR9	SFR10	SFR11	SFR12	SFR13	SFR14	SFR15	
8	CSA N287.7	In-Service Examination and Testing Requirements for Concrete Containment Structures For CANDU Nuclear Power Plant Components	I	N287.7-08		x	x	x												SF4-11 SF4-12 SF4-13
9	CSA N288.1	Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities	I	N288.1-14								x						x		-
10	CSA N288.4	Environmental Monitoring Program at Class I Nuclear Facilities and Uranium Mines and Mills	I	N288.4-10								x						x		-
11	CSA N293	Fire Protection for CANDU Nuclear Power Plants	I	N293-12	x						x						x			SF1-3 SF1-4 SF1-5
12	CNSC RD-204	Certification of Persons Working at Nuclear Power Plants	I	2008										x						-
13	CNSC REGDOC 3.1.1	Reporting Requirements for Nuclear Power Plants	I	2014										x						-
14	CNSC REGDOC 2.4.1	Deterministic Safety Analysis	I	2014					x		x									SF1-AG14 SF5-3 SF5-4 SF5-AG1 COP-21
15	CNSC REGDOC 2.4.2	Probabilistic Safety Assessment (PSA) for Nuclear Power Plants	I	2014						x	x									SF6-4

#	Document Number	Document Title	Type of Review	Modern Version for PSR2	Safety Factor Report															PSR2 Gaps
					SFR1	SFR2	SFR3	SFR4	SFR5	SFR6	SFR7	SFR8	SFR9	SFR10	SFR11	SFR12	SFR13	SFR14	SFR15	
16	CNSC RD/GD-210[26]	Maintenance Programs for Nuclear Power Plants	I	2012			x	x												-
17	CNSC RD/GD-98	Reliability Programs for Nuclear Power Plants	I	2012			x	x												-
18	CNSC REGDOC 2.6.3[26]	Aging Management	I	2014			x	x												SF2-AG1 SF4-14 SF4-15
19	CNSC REGDOC 2.9.1[26]	Environmental Protection: Policies, Programs and Procedures	I	2013								x						x		-
20	CNSC REGDOC 2.10.1[26]	Nuclear Emergency Preparedness and Response	I	2014													x			SF13-1
Additional Documents (not referenced in Pickering PROL 48.02/2018)																				
21	CSA N287.1	General Requirements for Concrete Containment Structures for Nuclear Power Plants	I	N287.1-14	x															-
22	CSA N287.2	Material requirements for Concrete Containment Structures for CANDU Nuclear Power Plants	I	N287.2-08	x	x	x	x												-
23	CSA N287.3	Design Requirements for Concrete Containment Structures for Nuclear Power Plants	I	N287.3-14	x															-

[26] Superseding documents to those already in PROL 48.02/2018.

#	Document Number	Document Title	Type of Review	Modern Version for PSR2	Safety Factor Report															PSR2 Gaps
					SFR1	SFR2	SFR3	SFR4	SFR5	SFR6	SFR7	SFR8	SFR9	SFR10	SFR11	SFR12	SFR13	SFR14	SFR15	
24	CSA N287.5	Examination and Testing Requirements for Concrete Containment Structures for Nuclear Power Plants	I	N287.5-11	x	x														SF1-6
25	CSA N289.1	General Requirements for Seismic Design and Qualification of CANDU Nuclear Power Plants	I	N289.1-08	x		x													-
26	CSA N289.2	Ground Motion Determination for Seismic Qualification of Nuclear Power Plants	I	N289.2-10	x		x													-
27	CSA N289.3	Design Procedures for Seismic Qualification of Nuclear Power Plants	I	N289.3-10	x		x													SF3-2 COP-17
28	CSA N289.4	Testing Procedures for Seismic Qualification of Nuclear Power Plants Structures, Systems, and Components	I	N289.4-12	x		x													SF3-3
29	CSA N289.5	Seismic Instrumentation Requirements for Nuclear Power Plants and Nuclear Facilities	I	N289.5-12	x		x													SF3-4
30	CSA N290.0	General Requirements for Safety Systems of Nuclear Power Plants	I	N290.0-11	x															SF1-7 SF1-8 SF1-9
31	CSA N290.1	Requirements for the Shutdown Systems of Nuclear Power Plants	I	N290.1-13	x															SF1-10

#	Document Number	Document Title	Type of Review	Modern Version for PSR2	Safety Factor Report															PSR2 Gaps
					SFR1	SFR2	SFR3	SFR4	SFR5	SFR6	SFR7	SFR8	SFR9	SFR10	SFR11	SFR12	SFR13	SFR14	SFR15	
32	CSA N290.2	General Requirements for Emergency Core Cooling Systems for Nuclear Power Plants	I	N290.2-11	x															SF1-11 SF1-12
33	CSA N290.3	Requirements for Containment System of Nuclear Power Plants	I	N290.3-11	x															SF1-13 SF1-AG15
34	CSA N290.4	Requirements for Reactor Control Systems of Nuclear Power Plants	I	N290.4-11	x															SF1-14
35	CSA N290.5	Requirements for Electrical Power and Instrument Air Systems of CANDU Nuclear Power Plants	I	N290.5-06	x															SF1-15
36	CSA N290.6	Requirements for Monitoring and Display of Nuclear Power Plant Safety Functions in the Event of an Accident	I	N290.6-09	x															-
37	CSA N290.11	Requirements for Reactor Heat Removal Capability During Outage of Nuclear Power Plants	HL	N290.11-13	x															SF1-16 SF1-17 SF1-18 SF1-AG20
38	CSA N290.14	Qualification of Pre-Developed Software for Use in Safety-related Instrumentation and Control Applications in Nuclear Power Plants	HL	N290.14-15	x															SF1-19
39	CSA N291	Requirements for Safety-related Structures for CANDU Nuclear Power Plants	I	N291-15	x	x		x												SF1-20 SF1-21 SF1-22 SF4-13

#	Document Number	Document Title	Type of Review	Modern Version for PSR2	Safety Factor Report															PSR2 Gaps
					SFR1	SFR2	SFR3	SFR4	SFR5	SFR6	SFR7	SFR8	SFR9	SFR10	SFR11	SFR12	SFR13	SFR14	SFR15	
40	CSA N285.6 Series	Material Standards for Reactor Components for CANDU Nuclear Power Plants	I	N285.6 Series-12	x															-
41	ASME B31.1	Power Piping	I	B31.1-2014	x															-
42	ASME BVPC	Boiler and Pressure Vessel Code	I	BPVC 2015	x															-
43	CSA B51	Boiler, Pressure Vessel, and Pressure Piping Code	I	B51-14	x															-
44	CSA N285.8	Technical Requirements for In-Service Evaluation of Zirconium Alloy Pressure Tubes in CANDU Reactors	I	N285.8-15		x		x												SF4-16
45	CNSC G-323	Ensuring Presence of Sufficiently Qualified Staff at Class I Nuclear Facilities-Minimum Shift Complement	I	2007										x		x				-
46	CNSC G-278	Human Factors Verification and Validation Plans	I	2003	x											x				-
47	CNSC G-276	Human Factors Engineering Program Plans	I	2003	x											x				-
48	CNSC G-129	Keeping Radiation Exposures and Doses “As Low As Reasonably Achievable (ALARA)”	I	2004								x							x	-
49	CNSC G-228	Developing and Using Action Levels	I	2001								x						x	x	-

#	Document Number	Document Title	Type of Review	Modern Version for PSR2	Safety Factor Report															PSR2 Gaps
					SFR1	SFR2	SFR3	SFR4	SFR5	SFR6	SFR7	SFR8	SFR9	SFR10	SFR11	SFR12	SFR13	SFR14	SFR15	
50	S.C.1997, C.9	Nuclear Safety and Control Act (NSCA) and its associated Regulations	I	Amended in February 2015										x						-
51	SOR/ 2000-202	The General Nuclear Safety and Control Regulations	I	Amended in June 2015										x					x	-
52	SOR/ 2000-203	The Radiation Protection Regulations	I	Amended in June 2015								x							x	-
53	CSA N1600	General Requirements for Nuclear Emergency Management Programs	HL	N1600-14													x			-
54	CSA N288.6	Environment Risk Assessments at Class I Nuclear Facilities and Uranium Mines and Mills	I[27]	N288.6-12								x						x		-
55	CSA N288.5	Effluent Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills	I[27]	N288.5-11								x						x		-
56	NFPA 20	Standard for the Installation of Stationary Pumps for Fire Protection	I	NFPA-20 (2016)	x															-
57	NFPA 24	Standard for the Installation of Private Fire Service Mains and Their Appurtenances	I	NFPA-24 (2016)	x															SF1-23 SF1-24

[27] The assessment type for CSA N288.3.4, N288.5 and N288.6 was changed from High Level to Incremental since implementation plans with gap assessments were identified.

#	Document Number	Document Title	Type of Review	Modern Version for PSR2	Safety Factor Report															PSR2 Gaps
					SFR1	SFR2	SFR3	SFR4	SFR5	SFR6	SFR7	SFR8	SFR9	SFR10	SFR11	SFR12	SFR13	SFR14	SFR15	
58	CNSC REGDOC 2.5.2	Design of Reactor Facilities: Nuclear Power Plants	I	2014	x				x	x	x									SF1-25 SF1-26 SF1-27 SF1-28 SF1-29 SF1-30 SF1-31 SF1-32 SF1-AG17 SF1-AG18 SF1-AG19 SF5-5 SF5-AG1 SF6-5
59	CNSC REGDOC 2.2.2	Personnel Training	I	2014										x						-
60	CNSC REGDOC 2.2.3	Personnel Certification: Radiation Safety Officers	see note [28]	2014																-
61	CNSC REGDOC 2.3.2	Accident Management, Version 2	I	2015	x				x	x	x	x		x						SF1-33
62	CNSC REGDOC 2.3.3	Periodic Safety Reviews	HL	2015								x								-

[28] REGDOC-2.2.3 was intended to receive a high level review. However, it was determined that REGDOC-2.2.3 is not applicable to Pickering NGS.

#	Document Number	Document Title	Type of Review	Modern Version for PSR2	Safety Factor Report															PSR2 Gaps
					SFR1	SFR2	SFR3	SFR4	SFR5	SFR6	SFR7	SFR8	SFR9	SFR10	SFR11	SFR12	SFR13	SFR14	SFR15	
63	CSA N286.7.1	Guideline for the Application of N286.7-99, Quality Assurance of Analytical, Scientific, and Design Computer Programs for Nuclear Power Plants	see note [29]	N286.7.1-09					x	x	x			x						-
64	CSA N290.12	Human Factors in Design for Nuclear Power Plants	I[30]	N290.12-14	x											x				-
65	CNSC G-144	Trip Parameter Acceptance Criteria for the Safety Analysis of CANDU Nuclear Power Plants	I	2006					x											-
66	CNSC G-149	Computer Programs Used in Design and Safety Analyses of Nuclear Power Plants and Research Reactors	I	2000	x				x	x	x									-
67	CNSC R-77	Overpressure Protection Requirements for Primary Heat Transport Systems in CANDU Power Reactors Fitted with Two Shutdown Systems	I	1987	x															-
68	CSA N288.2	Guidelines for Calculating Radiological Consequences to the Public from a Release of Airborne Radioactive Material for Nuclear Reactor Accidents	I	N288.2-14					x											SF5-6

[29] The N286.7.1 guide has been amalgamated into the new (-16) edition of the N286.7 Standard. As a result, only N286.7-16 was reviewed for PSR2.

[30] Per CNSC's request in P-CORR-03680-0607223 [OPG Correspondence, P-CORR-03680-0607223-R000, *Pickering PSR2 – Change to Review Type for CSA N290.12*, July 25, 2016], the Review Type for CSA N290.12-14 was changed from High Level to Incremental.

#	Document Number	Document Title	Type of Review	Modern Version for PSR2	Safety Factor Report															PSR2 Gaps
					SFR1	SFR2	SFR3	SFR4	SFR5	SFR6	SFR7	SFR8	SFR9	SFR10	SFR11	SFR12	SFR13	SFR14	SFR15	
69	CSA N288.3.4	Performance Testing of Nuclear Air-Cleaning Systems at Nuclear Facilities	I[27]	N288.3.4-13								x						x		-
70	CSA N290.7	Cyber-Security for Nuclear Power Plants and Small Reactor Facilities	HL[31]	N290.7-14	x															-
71	-	National Building Code of Canada	I	NBCC 2010	x															-
72	-	National Fire Code of Canada	I	NFCC 2010	x															COP-18[32]
73	CSA N288.7	Groundwater Protection Programs at Class I Nuclear Facilities and Uranium Mines and Mills	HL	N288.7-15														x		-
74	CSA N290.8	Technical Specification Requirements for Nuclear Power Plant Components	HL	N290.8-15	x															SF1-34

[31] A gap analysis for N290.7-14 was completed by OPG and satisfies the intent of the PSR2 High Level Review. For reasons of security and confidentiality, the findings of the gap analysis are not discussed in this Global Assessment Report.

[32] The assessment of the National Fire Code of Canada in [P-REP-03680-00029 R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 1, 5, 6, and 7*, March 3, 2017] did not create a new SF1 Gap since the same gap is already identified as a COP Review Gap in [P-REP-03680-00024 R000, *Pickering 5-8 Continued Operations Plan Review in Support of PNGS Periodic Safety Review 2*, January 17, 2017].

Appendix B – Global Issues and Proposed Resolution Plans

B.1. GI-1 Fitness for Service for Fuel Channels

SECTION 1 - GI-1 GLOBAL ISSUE SUMMARY					
The goal of GI-1, Fitness for Service for Fuel Channels, is to ensure that Fuel Channels remain fit for service for the extended operating period. This Global Issue comprises four proposed Resolution Statements addressing nine Gaps: three SF2 review task Gaps, one SF4 code review Gap [CSA N285.8-15], one COP Review Gap, three COP Additional Gaps and one SF2 Additional Gap. GI-1 is Safety Significance Level 1 based on deterministic and probabilistic defence-in-depth considerations as well as plant operability considerations. OPG has in place robust FFS programs for major components, such as Fuel Channels, and will not operate a reactor unit unless there is high confidence in demonstrating the FFS of these major components and continued safe operation of Pickering NGS.					
Safety Significance Level:	1	**Category:**	Programmatic, Engineering, Analytical	**Reassessment Beyond 2024:**	Y

SECTION 2 - GI-1 ASSOCIATED GAPS	
SF2-1	Fitness for Service for Fuel Channels has not been demonstrated for station operation to 2028[33]. ***Review Task #1 Actual Condition of SSCs*** **Associated Resolutions:** GI-1-RS2, GI-1-RS4
SF2-2	OPG does not have approval to operate beyond the current Licence limit of 247,000 Effective Full Power Hours (EFPH) for Fuel Channels. ***Review Task #1 Actual Condition of SSCs*** **Associated Resolutions:** GI-1-RS4
SF2-3	The Fuel Channels LCMP has not been formally updated to address extended station operation to 2028[33]. ***Review Task #1 Actual Condition of SSCs***

[33] As noted in the PSR2 Basis Document [P-REP-03680-00001-R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016], the current planning basis for Pickering NGS is that Pickering NGS units will operate until the end of 2024. To align with the anticipated expiry date of the next PROL, for the purposes of PSR2 only the period of operation of Pickering NGS units is extended until the end of 2028.

SECTION 2 - GI-1 ASSOCIATED GAPS	
	Associated Resolutions: GI-1-RS2, GI-1-RS3
SF4-16	For the Pickering B ISR, no clause-by-clause review of CSA N285.8 was conducted on the basis that the pressure tubes will be replaced during the refurbishment outage for Pickering Units 5-8, and the condition of these components is well understood and managed through their own specific, detailed life cycle plans and fitness-for-service criteria. However, in November 2015, OPG issued Plan N-REP-31100-10061 R002 for Pickering NGS compliance with pressure tube in-service evaluation requirements in CSA N285.8-15. OPG had submitted a previous compliance plan for the long term use of the 2010 edition of CSA N285.8 and this compliance plan was accepted by the CNSC. The compliance plan was revised to document OPG's compliance to the 2015 edition of CSA N285.8. Since OPG has committed to fulfillment of the commitments in N-REP-31100-10061 R002, successful fulfillment by OPG of the commitments in the compliance plan is required for Pickering operation past 2020. This is therefore a Gap for Pickering PSR2. In particular, the significant changes to CSA N285.8-15 per the CSA Impact Statement will need to be reflected in Pickering procedures, including: • Implementation of statistically based fatigue crack initiation evaluation curves for axial flaws (Clauses D.4.2, D.4.3, and D.3.6); • Implementation of closed-form engineering relation for threshold peak stress for Delayed Hydride Cracking initiation (Clauses D.5 and 5.4.3.4); • Implementation of statistically based threshold relation for peak stress for crack initiation due to hydrided region overloads (Clause D.5); • Implementation of new fracture toughness models for axial through-wall flaws (Clause D.13.2); and • Implementation of Methods 1 and 2 Probabilistic Leak-Before-Break (Clauses 3.1, 7.3 and 7.4) ***Code Review N285.8-15*** **Associated Resolutions:** GI-1-RS1
COP-1	The probabilistic LBB assessments have not yet been fully completed for the entire extended operating period, as work is being performed to demonstrate fitness for service for the pressure tubes in a staged approach. ***Pickering PSR2 Gap COP-1*** **Associated Resolutions:** GI-1-RS1, GI-1-RS4
COP-AG2	Determine if reconfiguration of P7 & P8 Fuel Channels is required prior to 238k EFPH and if necessary, schedule reconfiguration for completion prior to 238k EFPH in the Fuel Channel Life Cycle Management Plan. ***Pickering PSR2 Gap formerly named COP-28*** **Associated Resolutions:** GI-1-RS4

SECTION 2 - GI-1 ASSOCIATED GAPS	
COP-AG3	Assess the susceptible flaws present in the inspected pressure tubes for crack initiation induced by Hydrided Region Overload for applicable loading conditions. ***Pickering PSR2 Gap formerly named COP-29*** **Associated Resolutions:** GI-1-RS1
COP-AG4	Assess and demonstrate the structural integrity of Zr-Nb-Cu loose fitting garter springs for operation during the PSR2 period. ***Pickering PSR2 Gap formerly named COP-30*** **Associated Resolutions:** GI-1-RS4
SF2-AG1	An SF2 Gap related to updating the Fuel Channels Life Cycle Management Plan as well as the Technical Basis used in the Fuel Channels Life Cycle Management Plan, documentation on mitigating actions such as R&D and extent of condition of each aging degradation mechanism, and the potential for CT/LISS contact prior to 2024 due to fuel channel sag, was identified in [P-CORR-00531-05099, e-Doc 5295534, July 12, 2017]. The associated resolution addresses Items (a) and (b) of the reference. Item (c) is addressed by GI-4. **Associated Resolutions:** GI-1-RS4

SECTION 3 - GI-1 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

This Global Issue contains nine Gaps (three SF2 Gaps, one SF4 Gap, one COP Review Gap, three COP Additional Gaps and one SF2 Additional Gap) that are related to Fitness for Service (FFS), the Life Cycle Management Plan (LCMP) and life extension approval for Fuel Channels.

OPG has well-established programmatic controls to demonstrate FFS for Fuel Channels. FFS is demonstrated through application of the OPG Integrated Aging Management Program (IAMP) [N-PROG-MP-0008-R006B, *Integrated Aging Management*, April 29, 2016], which is compliant with CNSC Regulatory Documents REGDOC-2.6.3 *Aging Management* (which superseded RD-334 *Aging Management for Nuclear Power Plants*), and REGDOC-2.3.3 *Periodic Safety Reviews* (which superseded RD-360 *Life Extension of Nuclear Power Plants*). Integrated Aging Management is based on understanding of the component and system design, materials, degradation mechanisms and operating environment. IAMP provides a Plan – Do – Check – Act cycle of activities to support continued fitness for service throughout the planned service life of the component. Application of the IAMP process, via the OPG Major Components Program [N-PROG-MA-0025-R002, *Major Components*, March 25, 2015], establishes a basis for demonstration of FFS for the Fuel Channels for the target operating life and development of the LCMP. The LCMP documents the strategies and actions planned to facilitate demonstration of FFS of the Fuel Channels throughout the planned operating period. OPG is currently updating the LCMP to comply with applicable requirements of REGDOC-2.6.3 for operation to 2024. FFS is demonstrated and re-assessed on an on-going basis through planned inspections and maintenance, and assessment of inspection results in accordance with the requirements of the CSA N285.4 and N285.8 Standards, and the IAMP and Major

SECTION 3 - GI-1 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

Components Program, using well established programmatic controls.

FFS assessments are based on the condition of the components throughout the life of the plant, usually as determined from the periodic inspections. The inspection results are assessed according to industry standard guideline documents that set out the permissible assessment methodologies and the mandatory requirements. The results are submitted to the regulator in accordance with the CSA N285.4 and N285.8 Standards' requirements, which indicate when regulatory acceptance is required. The inspection techniques and assessment methodologies continue to improve through the Research and Development (R&D) program supported by OPG.

The condition of Pickering Fuel Channels has been projected by OPG to be acceptable for the EFPH values identified in the Assurance of Fuel Channel Fitness for Service submission [P-CORR-00531-04953, *Pickering NGS- Assurance of Fuel Channel Fitness-for-Service for the Assumed Target Life of the Pickering Units*, April 4, 2017] using current assessment methods, models and acceptance criteria. This envelops operation to the end of 2024 for Pickering Units 1,4 and 5-8. OPG acknowledges that there will be continued discussions with CNSC staff regarding methodologies and inputs to the Fuel Channel FFS assessments and compliance with CSA N285.8 Standard that may result in a need to update the FFS assessments. Demonstration of Fuel Channel FFS will be ongoing throughout the planned operating period, following the programmatic controls in place.

LCMPs are prepared based on planned End-of-Operations dates (calendar dates), which are then converted to EFPH values, factoring in outage dates, and anticipated forced loss rates, etc. Each unit is unique in its End-of-Operation EFPH. The LCMPs address legal (regulatory/PIP) requirements, FFS and asset preservation activities. LCMPs are updated on an annual basis to include actions based on results of recent inspections, industry operating experience, R&D findings, and to address activities required due to changes in End-of-Operations dates. Work requirements are established and incorporated into outage and online plans. All LCMP updates are submitted to the CNSC for review, and a completion ratio report is provided to the CNSC annually per the Licence Conditions Handbook.

The current Fuel Channels LCMP [N-PLAN-01060-10002-R017, *Fuel Channels Life Cycle Management Plan*, October 2016] was submitted to the CNSC [N-CORR-00531-18390, OPG Correspondence, *CNSC Staff's Prior Notification of Document Changes: Life Cycle Management Plans*, December 21, 2016]. This LCMP covers Units 5-8 operation to 2024, and Units 1,4 operation to 2022.

The Fuel Channel Life Confirmation Project has completed most of the R&D work required to provide the information needed to support the demonstration of Fuel Channel Fitness for Service to 2024. R&D efforts to support validation of the fracture toughness model will continue as described in the annual update [N-CORR-00531-18461, *Update #5: Annual Update on Approach to Fitness-for-Service Assessment for Pressure Tubes - Action Item 2014-OPG-4782*, February 13, 2017].

Periodic Inspection Plans, in compliance with CSA N285.4, are complete for operation of Pickering 1,4 and, 5-8 to at least 2022, and will be reviewed and revised as required to cover the extended operating period. Application of the CSA N285.4 Standard, which is included in the PROL, provides a process to demonstrate FFS of the inspected Fuel Channels and to seek CNSC acceptance (via the disposition process) for continued operation if the inspected component conditions do not satisfy the CSA N285.4 acceptance criteria.

The CSA N285.8 compliance plan has been accepted by the CNSC [N-CORR-00531-18357, CNSC Correspondence, e-Doc 5126091, *Darlington and Pickering NGS: Revised CSA N285.8 Compliance*

SECTION 3 - GI-1 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

Plan Submission - New Action Item 2016-OPG-8975, December 5, 2016], on condition of inclusion of CNSC comments provided in the correspondence.

OPG will not operate a unit when Fuel Channel FFS has not been demonstrated.

The current Licence limit for Fuel Channel operation of 247,000 EFPH is expected to be removed or revised when the Pickering Power Reactor Operating Licence is renewed. OPG will provide more detail in the Fuel Channel Life Cycle Management Plan. The LCMP will continue to be submitted annually to CNSC staff as prescribed by the Licence Conditions Handbook. The LCMP will include a summary of relevant R&D and assessment methodology updates that may impact Fuel Channel FFS for Pickering NGS operation to 2024. The LCMP will include a table of all current Fuel Channel FFS assessments that have been provided to CNSC, including a summary of assessment results vs. acceptance criteria and the evaluation period addressed. On the basis of the reported results and the continued plans for inspections and implementation of identified/planned mitigations, OPG will state its level of confidence in demonstrating the continued fitness for service for Fuel Channels to the assumed service life targets or, if necessary, to an adjusted service life based on the results. In this way, with the revised LCMP structure in place, continued reporting will provide ongoing fitness for service assessment projections through the extended operating period, using latest Fuel Channel condition information, projection tools, and limits.

SECTION 4 - GI-1 PRIORITY DETERMINATION

Safety Significance of Global Issue	**Deterministic Considerations**			**Probabilistic Considerations**								**Overall Safety Significance Level**
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	1	N/A	**1**	4	1	4	1	N/A	N/A	N/A	**1**	**1**

Rationale:

Addressing this Global Issue assures the ongoing Fitness for Service of Fuel Channels for the operational life of the station. The Fuel Channels are part of the Heat Transport System pressure boundary and, as such, assurance of fitness for service is essential for the safe operation of the plant. As discussed in Section 3 of this Global Issue, OPG has an ongoing effective Major Components Program in place to ensure Fuel Channel fitness for service. In addition, OPG will operate a unit only with Fuel Channels that are fit for service.

SECTION 4 - GI-1 PRIORITY DETERMINATION

Nevertheless, this Global Issue has Safety Significance Level 1 for deterministic considerations with respect to Defence in Depth (E1), since addressing it will ensure the effectiveness of the pressure boundary barrier, which is consistent with the definition of Safety Significance Level 1 in Table E1 in the PSR2 Basis Document. Safety Significance Levels (E2) is considered not applicable, since this Global Issue potentially impacts a physical nuclear safety barrier, whereas E2 primarily relates to issues that impact other objectives or are indirectly related to nuclear safety. Hence, the overall Safety Significance Level of 1 for deterministic considerations is dictated by the E1 categorization.

With respect to probabilistic considerations, the Fuel Channels represent a safety barrier, the effectiveness of which can potentially be impacted by a postulated initiating event with a frequency of approximately 10^{-2}/y. Consistent with Table F2 in the PSR2 Basis Document, the Safety Significance Level is 1 for Defence-in-Depth (F2). Similarly, the Safety Significance Level with respect to Plant Operability (F4) is 1, since addressing this Global Issue will prevent an extended period of plant shutdown due to fitness for service issues.

A Safety Significance Level of 4 is assigned for Reactor Safety – Core Damage Frequency (F1). Potential failure of Fuel Channels is accounted for in the Probabilistic Safety Assessment, and resolution of this Global Issue will ensure that the assumptions in the Probabilistic Safety Assessment regarding Fuel Channel failure frequency remain valid. Therefore, the change in the Core Damage Frequency will be less than 10^{-7}/y, which corresponds to the fourth row of Table F1 in the PSR2 Basis Document, for which the Safety Significance Level is 4.

With respect to Public Radiation Safety (F3), the radiological consequences of a postulated Fuel Channel failure are already accounted for in the safety analysis and shown to be within regulatory limits. Therefore, this Global Issue will result in no adverse change in Public Radiation Safety (F3), which is the determining factor for the applicability of this consideration, and thus the Safety Significance Level is 4.

This Global Issue has no direct impact on the other probabilistic considerations, i.e., Occupational Radiation Safety (F5), Emergency Preparedness (F6) and Environment (F7). Therefore, these probabilistic considerations are not applicable.

In summary, both deterministic and probabilistic considerations dictate that this Global Issue has Safety Significance Level 1. As noted, OPG's existing programs address this issue on an ongoing basis to ensure Fuel Channel fitness for service for the operational life of the station.

SECTION 5 - GI-1 RESOLUTION PLAN

GI-1-RS1	Complete CSA N285.8 Compliance Plan activities, including responding to comments specified in [N-CORR-00531-18357, CNSC Correspondence, e-Doc 5126091, *Darlington and Pickering NGS: Revised CSA N285.8 Compliance Plan Submission - New Action Item 2016-OPG-8975*, December 5, 2016]. (SF4-16) (COP-1) (COP-AG3) This Resolution Statement also includes/addresses GI-22.
GI-1-RS2	Review and revise if/as required the CSA N285.4 compliant Periodic Inspection Plans for Fuel Channels for Pickering NGS to cover the extended operating period. (SF2-1) (SF2-3)

SECTION 5 - GI-1 RESOLUTION PLAN	
GI-1-RS3	Update the Fuel Channels LCMP [N-PLAN-01060-10002-R017, OPG Plan, *Fuel Channels Life-Cycle Management Plan*, October 2016] for Pickering Units 1,4 for the extended operating period. (SF2-3) This Resolution Statement also includes/addresses GI-22 and GI-33.
GI-1-RS4	Update the structure of the Fuel Channels LCMP [N-PLAN-01060-10002-R017, OPG Plan, *Fuel Channels Life-Cycle Management Plan*, October 2016] to demonstrate compliance with REGDOC-2.6.3 for operations to 2024, and to include a summary of relevant R&D and assessment methodology updates that may impact Fuel Channel FFS for Pickering NGS operation. The LCMP structure will include a table of all current Fuel Channel FFS assessments that have been provided to CNSC, as well as a summary of assessment results vs. acceptance criteria and the evaluation period addressed. The FFS for Fuel Channels includes demonstration of sufficient margin of the structural integrity of the pressure tubes, calandria tubes and garter springs (annulus spacers) during the continued operational life of the plant. Based on the reported results, R&D activities, and the continued plans for inspections as well as implementation of identified/planned mitigations, the LCMP will establish a basis to demonstrate the continued fitness for service of Fuel Channels. (SF2-1) (SF2-2) (COP-1) (COP-AG2) (COP-AG4) (SF2-AG1 Items (a) and (b)) OPG is actively progressing this work in support of extended operations at Pickering NGS.

B.2. GI-2 Fitness for Service for Feeders

SECTION 1 - GI-2 GLOBAL ISSUE SUMMARY					
The goal of GI-2, Fitness for Service for Feeders, is to ensure that Feeders remain fit for service for the extended operating period. This Global Issue comprises one proposed Resolution Statement addressing three Gaps: two SF2 review task Gaps and one COP Review Gap. GI-2 is Safety Significance Level 1 based on deterministic and probabilistic defence-in-depth considerations as well as plant operability considerations. OPG's Integrated Aging Management Program, N-PROG-MP-0008-R006, is supported by the Major Components Program, N-PROG-MA-0025-R002, and provides the systematic governance process to ensure ongoing Feeder fitness for service. Life Cycle Management Plans developed under this governance are in place. The Life Cycle Management Plans currently cover the period to 2022 for Pickering 1,4 and 2024 for Pickering 5-8. The proposed Resolution Plan primarily comprises activities to update the Life Cycle Management Plan, to document the basis for ensuring that Feeders will remain fit for service for the extended operating period. The update is to include a planned Feeder replacement plan/schedule to address the extended operating period. Completion of the proposed Resolution Plan will support and strengthen Level 1 defence-in-depth for the extended operating period.					
Safety Significance Level:	1	**Category:**	Programmatic, Engineering, Analytical	**Reassessment Beyond 2024:**	Y

SECTION 2 - GI-2 ASSOCIATED GAPS	
SF2-4	Fitness for Service for Feeders has not been demonstrated for station operation to 2028[33]. ***Review Task #1 Actual Condition of SSCs*** **Associated Resolutions:** GI-2-RS1
SF2-5	The Feeders LCMP has not been formally updated to address extended station operation to 2028[33]. ***Review Task #1 Actual Condition of SSCs*** **Associated Resolutions:** GI-2-RS1
COP-8	The Pickering Feeder replacement schedule/plan (including Feeder IDs and the supporting rationale, including life limits, for specific Feeder replacement) has not been updated to support extended operation for Pickering Units 1,4 and 5-8. ***Pickering PSR2 Gap COP-8*** **Associated Resolutions:** GI-2-RS1

SECTION 3 - GI-2 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

This Global Issue contains three Gaps (two SF2 Gaps and one COP Review Gap) that are related to FFS and the LCMP for Feeders.

In a process similar to that described in GI-1, OPG has well established programmatic controls to demonstrate FFS for Feeders. FFS is demonstrated through application of the OPG Integrated Aging Management Program (IAMP) [N-PROG-MP-0008-R006, *Integrated Aging Management*, April 29, 2016]. IAMP is based on understanding of the component and system design, materials, degradation mechanisms and operating environment. IAMP provides a Plan – Do – Check – Act cycle of activities to support continued fitness for service throughout the planned service life of the component. Application of the IAMP process, via the OPG Major Components Program [N-PROG-MA-0025-R002, *Major Components*, March 25, 2015] establishes a basis for demonstration of FFS for the Feeders for the target operating life and development of the LCMP. The LCMP documents the strategies and actions planned to facilitate demonstration of FFS of the Feeders throughout the planned operating period. FFS is demonstrated and re-assessed on an on-going basis through planned inspections and maintenance, and assessment of inspection results in accordance with the requirements of the CSA N285.4 Standard, and the IAMP and Major Components Program, using well established programmatic controls.

LCMPs are prepared based on planned End-of-Operations dates (calendar dates), which are then converted to EFPH values, factoring in outage dates, and anticipated forced loss rates etc. Each unit is unique in its End-of-Operation EFPH or its component end-of-life date. The current Feeders LCMP [N-PLAN-01060-10001-R018, OPG Plan, *Feeders Life Cycle Management Plan*, October 2016] was submitted to the CNSC [N-CORR-00531-18390, OPG Correspondence, *CNSC Staff's Prior Notification of Document Changes: Life Cycle Management Plans*, December 21, 2016]. This LCMP covers Units 5-8 operation to 2024, and Units 1,4 operation to 2022.

Periodic Inspection Plans, in compliance with CSA N285.4, are complete for operation of Pickering 1,4 and 5-8 to at least 2022, and will be reviewed and revised as required to cover the extended operating period. Application of the CSA N285.4 Standard, which is included in the PROL, provides a process to demonstrate FFS of the inspected Feeders and to seek CNSC acceptance (via the disposition process) for continued operation if the inspected component conditions do not satisfy the CSA N285.4 acceptance criteria.

All LCMP updates are submitted to the CNSC for information and a completion status report is provided to the CNSC annually.

SECTION 4 - GI-2 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								Overall Safety Significance Level
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	1	N/A	**1**	4	1	4	1	N/A	N/A	N/A	**1**	**1**

Rationale:

Addressing this Global Issue assures the ongoing fitness for service of Feeders for the operational life of the station. The Feeders are part of the Heat Transport System pressure boundary and, as such, assurance of fitness for service is important for the safe operation of the plant. As discussed in Section 3 of this Global Issue, OPG has an ongoing effective Major Components Program in place to ensure Feeders fitness for service. In addition, OPG will operate a unit only with Feeders that are fit for service.

Nevertheless, this Global Issue has Safety Significance Level 1 for deterministic considerations with respect to Defence in Depth (E1), since addressing it will ensure the effectiveness of the pressure boundary barrier, which is consistent with the definition of Safety Significance Level 1 in Table E1 in the PSR2 Basis document. Safety Significance Levels (E2) is considered not applicable, since this Global Issue potentially impacts a physical nuclear safety barrier, whereas E2 primarily relates to issues that impact other objectives or are indirectly related to nuclear safety. Hence, the overall Safety Significance Level of 1 for deterministic considerations is dictated by the E1 categorization.

With respect to probabilistic considerations, the Feeders represent a safety barrier, the effectiveness of which can potentially be impacted by a postulated initiating event of frequency of approximately 10^{-2}/y. Consistent with Table F2 in the PSR2 Basis Document, the Safety Significance Level is 1 for Defence in Depth (F2). Similarly, the Safety Significance Level with respect to Plant Operability (F4) is 1, since addressing this Global Issue will prevent an extended period of plant shutdown due to fitness for service issues.

A Safety Significance Level of 4 is assigned for Reactor Safety – Core Damage Frequency (F1). Potential failure of feeders is accounted for in the Probabilistic Safety Assessment, and resolution of this Global Issue will ensure that the assumptions in the Probabilistic Safety Assessment regarding Feeder failure frequency remain valid. Therefore, the change in the Core Damage Frequency will be less than 10^{-7}/y, which corresponds to the fourth row of Table F1 in the PSR2 Basis Document, for which the Safety Significance Level is 4.

With respect to Public Radiation Safety (F3), the radiological consequences of a postulated Feeder failure are already accounted for in the safety analysis and are within regulatory limits. Therefore, this

SECTION 4 - GI-2 PRIORITY DETERMINATION
Global Issue will result in no adverse change in Public Radiation Safety (F3), which is the determining factor for the applicability of this consideration, and thus the Safety Significance Level is 4. This Global Issue has no direct impact on the other probabilistic factors, i.e., Occupational Radiation Safety (F5), Emergency Preparedness (F6) and Environment (F7). Therefore, these probabilistic considerations are not applicable. In summary, both deterministic and probabilistic considerations dictate that this Global Issue has Safety Significance Level 1. As noted, OPG's existing programs address this issue on an ongoing basis to ensure fitness-for-service of the Feeders for the operational life of the station.

SECTION 5 - GI-2 RESOLUTION PLAN	
GI-2-RS1	Update the Feeders LCMP [N-PLAN-01060-10001-R018, OPG Plan, *Feeders Life Cycle Management Plan*, October 2016], based on updated Fitness for Service assessment for Pickering Units 1,4 for the extended operating period. The LCMP update is to support continued demonstration that predicted Feeder condition, with identified and planned mitigations, is acceptable for the intended operation, including any potential impact of Fuel Channel elongation on Feeder integrity. The Feeders LCMP update is to include a planned Feeder replacement plan/schedule to address the extended operating period. (SF2-4) (SF2-5) (COP-8) This Resolution Statement also includes/addresses GI-22 and GI-33. OPG is actively progressing this work in support of extended operation at Pickering NGS.

B.3. GI-3 Fitness for Service for Steam Generators

SECTION 1 - GI-3 GLOBAL ISSUE SUMMARY					
The goal of GI-3, Fitness for Service for Steam Generators, is to ensure that Steam Generators remain fit for service for the extended operating period. This Global Issue comprises one proposed Resolution Statement addressing three Gaps: two SF2 review task Gaps and one COP Review Gap. GI-3 is Safety Significance Level 1 based on deterministic and probabilistic defence-in-depth considerations as well as plant operability considerations. OPG's Integrated Aging Management Program, N-PROG-MP-0008-R006, is supported by the Major Components Program, N-PROG-MA-0025-R002, and provides the systematic governance process to ensure ongoing Steam Generator fitness for service. Life Cycle Management Plans developed under this governance are in place. The Life Cycle Management Plans currently cover the period to 2022 for Pickering 1,4 and 2024 for Pickering 5-8. The proposed Resolution Plan primarily comprises activities to update the Life Cycle Management Plan, to document the basis for ensuring that Steam Generators will remain fit for service for the extended operating period. Completion of the proposed Resolution Plan will support and strengthen Level 1 defence-in-depth for the extended operating period.					
Safety Significance Level:	1	**Category:**	Programmatic, Engineering, Analytical	**Reassessment Beyond 2024:**	Y

SECTION 2 - GI-3 ASSOCIATED GAPS	
SF2-6	Fitness for Service for Steam Generators has not been demonstrated for station operation to 2028[33]. ***Review Task #1 Actual Condition of SSCs*** **Associated Resolutions:** GI-3-RS1
SF2-7	The Steam Generators LCMP has not been formally updated to address extended station operation to 2028[33]. ***Review Task #1 Actual Condition of SSCs*** **Associated Resolutions:** GI-3-RS1
COP-9	Demonstration has not been completed for extended operation[33] to confirm that sufficient margin remains in the operating life of the Pickering Steam Generators (residual life based on the TLAA), Steam Generator tubes and tube supports, shell, attachment welds and other internals considering the original design requirements and the OPEX on in-service degradation with respect to: • thermal cyclic fatigue;

SECTION 2 - GI-3 ASSOCIATED GAPS

- mechanical fatigue;
- corrosion allowances; and
- other types of degradation mechanisms.

Pickering PSR2 Gap COP-9

Associated Resolutions: GI-3-RS1

SECTION 3 - GI-3 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

This Global Issue contains three Gaps (two SF2 Gaps and one COP Review Gap) that are related to FFS and the LCMP for Steam Generators.

In a process similar to that described in GI-1, OPG has well established programmatic controls to demonstrate FFS for Steam Generators. FFS is demonstrated through application of the OPG Integrated Aging Management Program (IAMP) [N-PROG-MP-0008-R006, *Integrated Aging Management*, April 29, 2016]. IAMP is based on understanding of the component and system design, materials, degradation mechanisms and operating environment. IAMP provides a Plan – Do – Check – Act cycle of activities to support continued fitness for service throughout the planned service life of the component. Application of the IAMP process, via the OPG Major Components Program [N-PROG-MA-0025-R002, *Major Components*, March 25, 2015] establishes a basis for demonstration of FFS for the Steam Generators for the target operating life and development of the LCMP. The LCMP documents the strategies and actions planned to facilitate demonstration of FFS of the Steam Generators throughout the planned operating period. FFS is demonstrated and re-assessed on an on-going basis through planned inspections and maintenance, and assessment of inspection results in accordance with the requirements of the CSA N285.4 Standard, and the IAMP and Major Components Program, using well established programmatic controls.

LCMPs are prepared based on planned End-of-Operations dates (calendar dates), which are then converted to EFPH values, factoring in outage dates, and anticipated forced loss rates etc. Each unit is unique in its End-of-Operation EFPH or its component end-of-life date. The current Steam Generators LCMP [N-PLAN-33110-10009-R007, OPG Plan, *Steam Generators Life Cycle Management Plan*, October 2016] was submitted to the CNSC [N-CORR-00531-18390, OPG Correspondence, *CNSC Staff's Prior Notification of Document Changes: Life Cycle Management Plans*, December 21, 2016]. This LCMP covers Units 5-8 operation to 2024, and Units 1,4 operation to 2022.

Periodic Inspection Plans, in compliance with CSA N285.4, are complete for operation of Pickering 1,4 and, 5-8 to at least 2022, and will be reviewed and revised as required to cover the extended operating period. Application of the CSA N285.4 Standard, which is included in the PROL, provides a process to demonstrate FFS of the inspected Steam Generators and to seek CNSC acceptance (via the disposition process) for continued operation if the inspected component conditions do not satisfy the CSA N285.4 acceptance criteria.

All LCMP updates are submitted to the CNSC for information and a completion status report is provided to the CNSC annually.

SECTION 4 - GI-3 PRIORITY DETERMINATION

Safety Significance of Global Issue	**Deterministic Considerations**			**Probabilistic Considerations**								**Overall Safety Significance Level**
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	1	N/A	**1**	4	1	4	1	N/A	N/A	N/A	**1**	**1**

Rationale:

Addressing this Global Issue assures the ongoing fitness for service of Steam Generators for the operational life of the station. The Steam Generators are part of the Heat Transport System pressure boundary and, as such, assurance of fitness for service is important for the safe operation of the plant. As discussed in Section 3 of this Global Issue, OPG has an ongoing effective Major Components Program in place to ensure Steam Generator fitness for service.

Nevertheless, this Global Issue has Safety Significance Level 1 for deterministic considerations with respect to Defence in Depth (E1) since addressing it will ensure the effectiveness of the pressure boundary barrier, which is consistent with the definition of Safety Significance Level 1 in Table E1 in the PSR2 Basis document. Safety Significance Levels (E2) is considered not applicable since this Global Issue potentially impacts a physical nuclear safety barrier, whereas E2 primarily relates to issues that impact other objectives or that are indirectly related to nuclear safety. Hence, the overall Safety Significance Level of 1 for deterministic considerations is dictated by the E1 categorization.

With respect to probabilistic considerations, the Steam Generator tubes represent a safety barrier, the effectiveness of which can be impacted by a postulated initiating event of frequency of approximately 10^{-2}/y. Consistent with Table F2 in the PSR2 Basis Document, the Safety Significance Level is 1 for Defence in Depth (F2). The Safety Significance Level with respect to Plant Operability (F4) is also 1, since addressing this Global Issue prevents an extended period of plant shutdown due to fitness for service issues.

Safety Significance Level 4 is assigned to Core Damage Frequency (F1) since this Global Issue has an insignificant impact on this consideration. With respect to Public Radiation Safety (F3), the radiological consequences of a Steam Generator tube failure are already accounted for in the safety analysis and are within regulatory limits. Therefore, this Global Issue will result in no adverse change in Public Radiation Safety (F3), which is the determining factor for the applicability of this consideration, and thus the Safety Significance Level is 4. This Global Issue has no direct impact on the other probabilistic factors: Occupational Radiation Safety (F5), Emergency Preparedness (F6) and Environment (F7). Therefore, these probabilistic considerations are not applicable.

SECTION 4 - GI-3 PRIORITY DETERMINATION
In summary, both deterministic and probabilistic considerations dictate that this Global Issue has Safety Significance Level 1. As noted, OPG's existing programs address this issue on an ongoing basis to ensure Steam Generators fitness for service for the operational life of the station.

SECTION 5 - GI-3 RESOLUTION PLAN	
GI-3-RS1	Update the Steam Generators LCMP [N-PLAN-33110-10009-R007, OPG Plan, *Steam Generators Life Cycle Management Plan*, October 2016], based on updated Fitness for Service assessment for Pickering Units 1,4 for the extended operating period. The LCMP update is to support continued demonstration that predicted Steam Generator condition, with identified and planned mitigations, is acceptable for the intended operation. (SF2-6) (SF2-7) (COP-9) This Resolution Statement also includes/addresses GI-22 and GI-33. OPG is actively progressing this work in support of extended operation at Pickering NGS.

B.4. GI-4 Fitness for Service for Reactor Components and Structures

SECTION 1 - GI-4 GLOBAL ISSUE SUMMARY					
The goal of GI-4, Fitness for Service for Reactor Components and Structures, is to ensure that the Reactor Components and Structures remain fit for service for the extended operating period. The major equipment within this category includes the Calandria Vessel, Calandria Tubes, Guide Tubes, Moderator Inlet Piping and Nozzles, Reactivity Control Units, Calandria Relief Ducts, Lattice Tubes and End Fittings, as well as Exposed Carbon Steel Components in the Calandria Vault. This Global Issue comprises two proposed Resolution Statements addressing nine Gaps: two SF2 review task Gaps, six COP Review Gaps and one SF2 Additional Gap. GI-4 is Safety Significance Level 2 based on deterministic and probabilistic defence-in-depth considerations as well as plant operability considerations. OPG's Integrated Aging Management Program, N-PROG-MP-0008-R006, is supported by the Major Components Program, N-PROG-MA-0025-R002, and provides the systematic governance process to ensure the ongoing fitness for service of Reactor Components and Structures. Life Cycle Management Plans developed under this governance are in place. The Life Cycle Management Plans currently cover the period to 2022 for Pickering 1,4 and 2024 for Pickering 5-8. The proposed Resolution Plan includes activities to update the Life Cycle Management Plan, to document the basis for ensuring that Reactor Components and Structures will remain fit for service for the extended operating period. In addition, the proposed Resolution Plan includes updating measurements and analysis of the gap between Liquid Injection Shutdown System nozzles and Calandria Tubes, and implementing mitigation strategies, if required. Completion of the proposed Resolution Plan will support and strengthen Level 1 defence-in-depth for the extended operating period.					
Safety Significance Level:	2	**Category:**	Programmatic, Engineering, Analytical	**Reassessment Beyond 2024:**	Y

SECTION 2 - GI-4 ASSOCIATED GAPS	
SF2-8	Fitness for Service for Reactor Components and Structures has not been demonstrated for station operation to 2028[33]. ***Review Task #1 Actual Condition of SSCs*** **Associated Resolutions:** GI-4-RS1, GI-4-RS2
SF2-9	The Reactor Components and Structures LCMP has not been formally updated to address extended station operation to 2028[33]. ***Review Task #1 Actual Condition of SSCs*** **Associated Resolutions:** GI-4-RS1

SECTION 2 - GI-4 ASSOCIATED GAPS	
COP-2	Demonstration of adequate margin to operate the Pickering Units 5-8 SDS2 LISS by performing piping fatigue and aging analysis has not been completed for the extended operation period[33] and for the period until LISS can be demonstrated to no longer be required. ***Pickering PSR2 Gap COP-2*** **Associated Resolutions:** GI-4-RS2
COP-10	Fitness for Service demonstration of the Pickering Units 1,4 and 5-8 Calandrias to address the full period of extended operation[33] has not been completed. ***Pickering PSR2 Gap COP-10*** **Associated Resolutions:** GI-4-RS1
COP-11	The Pickering Calandria Tube life assessment has not been updated for the full period of extended operation[33]. ***Pickering PSR2 Gap COP-11*** **Associated Resolutions:** GI-4-RS1, GI-4-RS2
COP-12	OPG requirements and plans for inspection and monitoring of Reactor Components have not been updated to address the full period of extended operation[33] of Pickering Units 1,4 and 5-8. ***Pickering PSR2 Gap COP-12*** **Associated Resolutions:** GI-4-RS1
COP-13	The Calandria and internal structures Technical Basis Document for Pickering Units 1,4 and 5-8, which includes OPEX, and FFS rationale to support that Calandria and internal components will remain fit has not been updated for the full period of extended operation[33]. ***Pickering PSR2 Gap COP-13*** **Associated Resolutions:** GI-4-RS1
COP-26	A review for the full period of extended operation[33] has not been completed of evidence that Calandria and internal structure guide tube springs for Pickering Units 1,4 and 5-8 will remain fit for service. ***Pickering PSR2 Gap COP-26*** **Associated Resolutions:** GI-4-RS1
SF2-AG1	An SF2 Gap related to updating the Fuel Channels Life Cycle Management Plan as well as the Technical Basis used in the Fuel Channels Life Cycle Management Plan, documentation on mitigating actions such as R&D and extent of condition of each aging degradation mechanism, and the potential for CT/LISS contact prior to 2024 due to fuel channel sag, was identified in [P-CORR-00531-05099, e-Doc 5295534, July 12, 2017].

SECTION 2 - GI-4 ASSOCIATED GAPS

The associated resolution addresses Item (c) of the reference. Items (a) and (b) are addressed by GI-1.

Associated Resolutions: GI-4-RS2

SECTION 3 - GI-4 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

This Global Issue contains nine Gaps (two SF2 Gaps, six COP Review Gaps and one SF2 Additional Gap) that are related to FFS and the LCMP for Reactor Components and Structures.

In a process similar to that described in GI-1, OPG has well established programmatic controls to demonstrate FFS for Reactor Components and structures. FFS is demonstrated through application of the OPG Integrated Aging Management Program (IAMP) [N-PROG-MP-0008-R006, *Integrated Aging Management*, April 29, 2016]. IAMP is based on understanding of the component and system design, materials, degradation mechanisms and operating environment. IAMP provides a Plan – Do – Check – Act cycle of activities to support continued fitness for service throughout the planned service life of the component. Application of the IAMP process, via the OPG Major Components Program [N-PROG-MA-0025-R002, *Major Components*, March 25, 2015] establishes a basis for demonstration of FFS for the Reactor Components and structures for the target operating life and development of the LCMP. The LCMP documents the strategies and actions planned to facilitate demonstration of FFS of the Reactor Components and structures throughout the planned operating period. FFS is demonstrated and re-assessed on an on-going basis through planned inspections and maintenance, and assessment of inspection results in accordance with the IAMP and Major Components Program, using well established programmatic controls.

LCMPs are prepared based on planned End-of-Operations dates (calendar dates), which are then converted to EFPH values, factoring in outage dates, and anticipated forced loss rates etc. Each unit is unique in its End-of-Operation EFPH or its component end-of-life date. The current Reactor Components and Structures LCMP [N-PLAN-01060-10003-R014, OPG Plan, *Reactor Components and Structures Life Cycle Management Plan*, October 2016] was submitted to the CNSC [N-CORR-00531-18390, OPG Correspondence, *CNSC Staff's Prior Notification of Document Changes: Life Cycle Management Plans*, December 21, 2016]. This LCMP covers Units 5-8 operation to 2024, and Units 1,4 operation to 2022.

Based on work performed under CANDU Owners Group (COG) Joint Project JP 4271 [COG-JP-4271-004, COG Report, *Calandria Fitness For Life Extension Guidelines: Phase 3*, May 2013], no component is identified as "must be replaced after 30 years operation" and there are no active degradation mechanisms or adverse conditions that would require imminent actions in the Pickering Units for operation extension for an additional 210,000 EFPH beyond 30 years of operation.

Baseline Calandria Tube - Liquid Injection Shutdown System (CT-LISS) nozzle gap assessments [N-PLAN-01060-10003-R014, OPG Plan, *Reactor Components and Structures Life Cycle Management Plan*, October 2016] show no contact is predicted to:

Unit 5: 268,000 EFPH

Unit 6: 256,000 EFPH

SECTION 3 - GI-4 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

Unit 7: >300,000 EFPH

Unit 8**:** 283,000 EFPH

These assessments use the baseline measurements of CT-LISS gaps in the units, except for Unit 7 where an additional measurement was done to obtain gap closure rate information. Additional measurements of CT-LISS nozzle gaps, per the Reactor Components and Structures LCMP [N-PLAN-01060-10003-R014, OPG Plan, *Reactor Components and Structures Life Cycle Management Plan*, October 2016], will be used to refine the gap closure rates. In the event that CT-LISS nozzle contact is predicted prior to the End-of-Operations, mitigation strategies will be developed and implemented.

All LCMP updates are submitted to the CNSC for information.

SECTION 4 - GI-4 PRIORITY DETERMINATION

Safety Significance of Global Issue	**Deterministic Considerations**			**Probabilistic Considerations**								**Overall Safety Significance Level**
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	2	N/A	**2**	4	2	4	2	N/A	N/A	N/A	**2**	**2**

<u>Rationale:</u>

Addressing this Global Issue assures the ongoing fitness for service of Reactor Components and structures for the operational life of the station. Reactor functions can be impacted by the aged conditions of these components and structures and, as such, assurance of fitness for service is important for the safe operation of the plant. As discussed in Section 3 of this Global Issue, OPG has an ongoing effective Major Components Program in place to ensure fitness for service of the Reactor Components and structures.

Nevertheless, this Global Issue has Safety Significance Level 2 for deterministic considerations with respect to Defence in Depth (E1) since ensuring fitness for service will prevent a reduction in margin of safety to the public or station personnel (Table E1 of the PSR2 Basis Document). Safety Significance Levels (E2) is considered not applicable since this Global Issue can have a direct impact on nuclear safety, whereas E2 primarily relates to issues without a direct impact on nuclear safety. Hence, the overall Safety Significance Level of 2 for deterministic considerations is dictated by the E1 categorization.

SECTION 4 - GI-4 PRIORITY DETERMINATION

With respect to probabilistic considerations, this Global Issue is assigned Safety Significance Level 2 for Defence in Depth (F2). This is because addressing the issues related to Calandria Tube/Liquid Injection Shutdown System nozzle gap or active aging mechanisms will prevent a potential adverse impact on the reactor Shutdown System, preventing a potential loss in the reliability of this system in mitigating events with frequency of approximately 10^{-2}/y. Similarly, the Safety Significance Level with respect to Plant Operability (F4) is also 2, since an extended period of plant shutdown as a result of fitness for service issues is expected to have low probability (less than 0.1). Safety Significance Level 4 is assigned to Core Damage Frequency (F1) and Public Radiation Safety (F3) since this Global Issue has an insignificant impact on these considerations. This Global Issue has no direct impact on the other probabilistic factors, i.e., Occupational Radiation Safety (F5), Emergency Preparedness (F6) and Environment (F7). Therefore, these probabilistic considerations are not applicable.

In summary, both deterministic and probabilistic considerations dictate that this Global Issue has Safety Significance Level 2. As noted, OPG's existing programs address this issue on an ongoing basis to ensure Reactor Components and structures fitness for service for the operational life of the station.

SECTION 5 - GI-4 RESOLUTION PLAN

GI-4-RS1	Update the Reactor Components and Structures LCMP [N-PLAN-01060-10003-R014, OPG Plan, *Reactor Components and Structures Life Cycle Management Plan*, October 2016], based on updated Fitness for Service assessment and an updated Technical Basis Document [N-PLAN-01060-10008 R00, *Reactor Components & Structures Life Cycle Management Plan: Technical Basis Document*, 2010] for Pickering Units 1,4 for the extended operating period. The LCMP update is to support continued demonstration that predicted Reactor Components and Structures condition, with identified and planned mitigations, is acceptable for the intended operation. (SF2-8) (SF2-9) (COP-10) (COP-11) (COP-12) (COP-13) (COP-26) This Resolution Statement also includes/addresses GI-22 and GI-33.
GI-4-RS2	Perform measurements, as required, of CT-LISS nozzle gaps on Units 5-8 to refine the gap closure rates. Using this new measurement data, update analyses as required, to demonstrate Fitness for Service. Implement mitigation strategies if CT-LISS nozzle contact is predicted within the extended operating period. (SF2-8) (COP-2) (COP-11) (SF2-AG1 Item (c)). This Resolution Statement also includes/addresses GI-22. OPG is actively progressing this work in support of extended operation at Pickering NGS.

B.5. GI-5 Completeness of Class 1 Piping / Components Service Limits Assessment (Excluding Major Components)

SECTION 1 - GI-5 GLOBAL ISSUE SUMMARY					
The goal of GI-5, Completeness of Class 1 Piping/Components Service Limits Assessment (excluding major components), is to confirm the adequacy of the service limits assessments for Nuclear Class 1 piping and components for the extended operating period. This Global Issue comprises one proposed Resolution Statement and one item requiring No Further Action, addressing three Gaps: one SF2 review task Gap and two COP Review Gaps. GI-5 is Safety Significance Level 2 based on deterministic and probabilistic defence-in-depth considerations as well as plant operability considerations. The service limits assessments for Nuclear Class 1 components (including piping) are complete to 2028 [P-CORR-33000-00001], but the updated piping assessments have not incorporated environmental factors. Therefore, the proposed Resolution Plan comprises activities to confirm the adequacy of the service limits assessments for Nuclear Class 1 Piping (excluding major components) after accounting for the impact of environmental factors such as irradiation, temperature, humidity, etc. Completion of the proposed Resolution Plan will support and strengthen Level 1 defence-in-depth for the extended operating period.					
Safety Significance Level:	2	**Category:**	Analytical	**Reassessment Beyond 2024:**	Y

SECTION 2 - GI-5 ASSOCIATED GAPS	
COP-3	Demonstration of adequate margin to operate the Pickering Primary Heat Transport piping has not been completed for the extended operation period and for the period until Primary Heat Transport piping integrity has been demonstrated to no longer be required. ***Pickering PSR2 Gap COP-3*** **Associated Resolutions:** GI-5-NFA1
COP-22	The final report of the Service Limits Assessment for Class 1 components has not been updated taking the full period of extended operation into account for Pickering Units 1,4 and 5-8. ***Pickering PSR2 Gap COP-22*** **Associated Resolutions:** GI-5-NFA1
SF2-10	Environmental Factors have not been incorporated into the Service Limits Assessment for Class 1 piping. ***Review Task #1 Actual Condition of SSCs***

SECTION 2 - GI-5 ASSOCIATED GAPS

Associated Resolutions: GI-5-RS1

SECTION 3 - GI-5 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

This Global Issue contains three Gaps (one SF2 Gap and two COP Review Gaps) that are related to the service limits of Class 1 Piping and Components (excluding Major Components).

SECTION 4 - GI-5 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								Overall Safety Significance Level
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	2	N/A	**2**	3	2	4	2	N/A	N/A	N/A	**2**	**2**

Rationale:

Addressing this Global Issue is part of assuring the fitness for service of Class 1 Piping/Components (excluding Major Components) for the operational life of the station. Class 1 Piping/Components Service Limits of the Major Components are addressed in the Major Components LCMPs (GI-1, GI-2, and GI-3). Resolution of this Global Issue will confirm that the original design of Class 1 Piping/Components (excluding Major Components) covers the extended operating period.

This Global Issue has Safety Significance Level 2 for deterministic considerations with respect to Defence in Depth (E1) since addressing it will contribute to preventing a margin reduction in fitness-for-service of the Heat Transport System barrier throughout the extended operation period (row 2 of Table E1 in the PSR2 Basis Document). Safety Significance Levels (E2) is considered not applicable, since this Global Issue potentially can have a direct impact on nuclear safety, whereas E2 primarily relates to issues without a direct impact on nuclear safety. Moreover, assignment of Safety Significance Level 2 for Defence in Depth (E1) subsumes any considerations for Safety Significance Levels (E2). Hence, the overall Safety Significance Level of 2 for deterministic considerations is dictated by the E1 categorization.

With respect to probabilistic considerations, for Defence in Depth (F2), this Global Issue has a Safety

SECTION 4 - GI-5 PRIORITY DETERMINATION
Significance Level 2 since confirming that service level C conditions for the Heat Transport System are met throughout the operational life is a means to confirm that safety margins are preserved. Therefore, row 3 of Table F2 in the PSR2 Basis Document is applicable, with 2 being the highest Safety Significance Level in this row. Similarly, the Safety Significance Level with respect to Plant Operability (F4) is 2, because the probability of an extended period of plant shutdown due to this Global Issue is less than 0.1. Safety Significance Level 3 is assigned to Core Damage Frequency (F1) since the failure probability of Class I piping is very low. Safety Significance Level 4 is assigned to Public Radiation Safety (F3) since this Global Issue has an insignificant impact on these considerations. This Global Issue has no direct impact on the other probabilistic considerations, i.e., Occupational Radiation Safety (F5), Emergency Preparedness (F6) and Environment (F7). Therefore, these probabilistic considerations are not applicable. In summary, both deterministic and probabilistic considerations dictate that this Global Issue has Safety Significance Level 2. As noted, OPG's existing programs address this issue on an ongoing basis to ensure fitness for service of Class 1 Piping/Components (excluding Major Components) for the operational life of the station.

SECTION 5 - GI-5 RESOLUTION PLAN	
GI-5-RS1	Confirm the adequacy of the service limits assessments for Nuclear Class 1 Piping after accounting for impact of environmental factors (for example: irradiation, temperature, humidity). Note – This Resolution Statement does not address Major Components. (SF2-10) This Resolution Statement also includes/addresses GI-33.
GI-5-NFA1	The service limits assessments for Nuclear Class 1 components (including piping) are complete to 2028 [P-CORR-33000-00001, *Pickering NGS Service Limits Assessment of Nuclear Class 1 Components for Potential Life Extension*, June 2016]. No further action is required. Note – This resolution does not address Major Components. (COP-3) (COP-22)

B.6. GI-6 Impact of the Revised Criticality Coding on the Cable Surveillance Program

SECTION 1 - GI-6 GLOBAL ISSUE SUMMARY					
The goal of GI-6, Impact of the Revised Criticality Coding on the Cable Surveillance Program, is to ensure that the on-going implementation of the Cable Surveillance Program takes into account the revised Criticality Coding for determining Cable Risk Rating for the extended operating period. This Global Issue comprises one proposed Resolution Statement addressing one SF2 review task Gap. GI-6 is Safety Significance Level 3 based on deterministic defence-in-depth considerations. The Cable Surveillance Program [N-PROC-MA-0099 R001] establishes scope for low and medium voltage electrical cables connected to Criticality Code 1 or Criticality Code 2 components. The proposed Resolution Plan comprises activities to reassess the impact of the changes in the Cable Criticality Coding and update the scope of the Cable Surveillance Plan accordingly. Completion of the proposed Resolution Plan will support and strengthen Levels 1, 2, 3 and 4 defence-in-depth for the extended operating period because the relevant cables are part of both process systems and mitigating systems.					
Safety Significance Level:	3	**Category:**	Analytical	**Reassessment Beyond 2024:**	N

SECTION 2 - GI-6 ASSOCIATED GAPS	
SF2-13	The Cable Surveillance Program risk assessment and Condition Assessments currently use out of date criticality coding. ***Review Task #1 Actual Condition of SSCs*** **Associated Resolutions:** GI-6-RS1

SECTION 3 - GI-6 BACKGROUND INFORMATION AND RESOLUTION STRATEGY
This Global Issue contains one SF2 Gap related to the Cable Surveillance Program. The Cable Surveillance Program [N-PROC-MA-0099 R001, OPG Nuclear Procedure, *Cable Surveillance*, August 19, 2014] establishes scope for low and medium voltage electrical cables connected to Criticality Code 1 (CC1) or Criticality Code 2 (CC2) components. Recently, changes in Criticality Code occurred as a result of the Criticality Code review project performed by OPG. The on-going implementation of the Cable Surveillance program needs to take into account the revised Criticality Coding for determining Cable Risk Rating for CC1 or CC2 components.

SECTION 4 - GI-6 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	**Overall Safety Significance Level**
	3	N/A	**3**	4	N/A	N/A	N/A	N/A	N/A	N/A	**4**	**3**

Rationale:

This Global Issue is related to the impact of the revised Criticality Coding on the Cable Surveillance Program. The ongoing and future implementation of the Cable Surveillance Program will take into account the updated equipment Criticality Coding when screening for potentially higher risk cables (Criticality Code 1 (CC1) or Criticality Code 2 (CC2)).

Regarding deterministic considerations, this Global Issue is assigned Safety Significance Level 3 for Defence in Depth (E1) since a safety function could be impacted by the issue (row 3 of Table E1 in the PSR2 Basis Document). Safety Significance Levels (E2) is assessed as not applicable since E2 primarily relates to issues without a direct nuclear safety impact. Therefore, Safety Significance Level 3 is selected for deterministic considerations.

Safety Significance Level 4 is assigned to Core Damage Frequency (F1) since the impact of this issue on core damage frequency is expected to be less than 10^{-7}/y. This Global Issue has no or insignificant impact on Defence in Depth (F2), Public Radiation Safety (F3), Plant Operability (F4), Occupational Radiation Safety (F5), Emergency Preparedness (F6) and Environment (F7). Therefore, these probabilistic considerations are not applicable.

In summary, the overall Safety Significance Level is 3. Work is in progress on addressing this Global Issue.

SECTION 5 - GI-6 RESOLUTION PLAN

GI-6-RS1	Reassess the impact of the changes in the cable Criticality Coding and update the scope of the cable surveillance plan accordingly. (SF2-13)

B.7. GI-7 Pickering Buried Piping Fitness for the Extended Operating Period

SECTION 1 - GI-7 GLOBAL ISSUE SUMMARY					
The goal of GI-7, Pickering Buried Piping Fitness for the Extended Operating Period, is to ensure the continued fitness of Pickering Buried Piping for the extended operating period. This Global Issue comprises two proposed Resolution Statements addressing two Gaps: one SF2 review task Gap and one SF1 Additional Review Finding Gap originating from the Darlington IIP. GI-7 is Safety Significance Level 3 based on deterministic and probabilistic defence-in-depth considerations as well as plant operability considerations. Assessments were completed [NA44/NK30-IR-58150-00001] confirming the adequacy of the buried piping to 2020 for Pickering 1,4 and to 2025 for Pickering 5-8, contingent on continuation of condition monitoring per the Buried Piping Program [N-PLAN-04916-10002-R003]. Action Request AR# 28175307 tracks implementation of the governance revision to apply a graded approach in the event that leakage from buried piping occurs. The proposed Resolution Plan comprises activities to update the Buried Piping Program asset management plan and risk ranking for the extended operating period, and to update governance to reflect a graded approach in the event that any leakage in fuel oil piping occurs. Completion of the proposed Resolution Plan will support and strengthen Levels 1, 3 and 4 defence-in-depth for the extended operating period because buried piping is associated with both process and mitigating systems.					
Safety Significance Level:	3	**Category:**	Analytical	**Reassessment Beyond 2024:**	Y

SECTION 2 - GI-7 ASSOCIATED GAPS	
SF2-14	The Buried Piping Program risk assessment and Condition Assessments have not been updated for extended operation to 2028[33]. ***Review Task #1 Actual Condition of SSCs*** **Associated Resolutions:** GI-7-RS1
SF1-35	Darlington Gap IIP-OI 063 was identified based on the requirement to replace single wall fuel oil piping with double wall piping if degraded piping is found. AR# 28175307 was initiated which required revision of N-PROC-MA-0088, *Buried Piping Program Requirements* to use a graded approach for the replacement of single walled piping with double walled material in instances of leakage. AR# 28175307 currently has corrective actions in place and is expected to be completed by Q1 2020. This issue is also applicable to Pickering NGS and is therefore a Gap for Pickering PSR2. ***Additional Review Findings*** **Associated Resolutions:** GI-7-RS2

SECTION 3 - GI-7 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

This Global Issue contains two Gaps (one SF1 Gap and one SF2 Gap) related to the Buried Piping Program.

Assessments were completed [NA44-IR-58150-00001, *Assessment of Cathodic Protect ion on Buried Piping Systems in Pickering A Nuclear Generating Station for Continued Operation until 2020*, June 2013] [NK30-IR-58150-00001, *Assessment of Cathodic Protection on Buried Piping Systems in Pickering B Nuclear Generating Station for Continued Operation until 2025*, December 2010] confirming the adequacy of the buried piping to 2020 for Pickering 1,4 and to 2025 for Pickering 5-8, contingent on continuation of condition monitoring per the Buried Piping Program [N-PLAN-04916-10002-R003, OPG Plan*, Buried Piping Program Asset Management Plan*, January 30, 2017]. An update to the Buried Piping Program is required for the extended operating period.

During the Darlington ISR, an assessment was completed [NK38-REP-03680-10204-R002, *Additional Analysis of DNGS ISR Fire Protection Related Issues*, January 12, 2015] that proposed a graded approach to addressing leakage in fuel oil piping. This graded approach recognizes the nuclear safety importance of the systems being supplied with fuel oil, and would allow these systems to be temporarily repaired while awaiting further corrective action, allowing the systems to remain in service. Action Request AR# 28175307 tracks implementation of the governance revision.

SECTION 4 - GI-7 PRIORITY DETERMINATION

	Deterministic Considerations			Probabilistic Considerations								
Safety Significance of Global Issue	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	**Overall Safety Significance Level**
	3	N/A	**3**	4	3	4	3	N/A	N/A	N/A	**3**	**3**

Rationale:

This issue is related to the Buried Piping Program. Addressing this Global Issue assures the ongoing fitness for service of Pickering buried piping for the operational life of the station. Safety functions can be impacted by the aged conditions of this piping and, as such, assurance of fitness for service is essential to ensure the adequacy of these safety functions.

This Global Issue has Safety Significance Level 3 for deterministic considerations with respect to

SECTION 4 - GI-7 PRIORITY DETERMINATION

Defence in Depth (E1) since assuring fitness for service prevents an adverse impact on a safety function (row 3 of Table E1 in the PSR2 Basis Document). Safety Significance Levels (E2) is considered not applicable because this Global Issue potentially impacts a nuclear safety barrier, whereas E2 primarily relates to issues that impact other objectives or are indirectly related to nuclear safety. Hence, the overall Safety Significance Level of 3 for deterministic considerations is dictated by the E1 categorization.

With respect to probabilistic considerations, this Global Issue is assigned Safety Significance Level 3 with respect to Defence in Depth (F2). This is because resolving this issue will prevent a partial loss of safety margin of a system important to safety in mitigating high probability events (frequency $>10^{-2}$/y). Similarly, the Safety Significance Level with respect to Plant Operability (F4) is also 3, since an extended period of plant shutdown as a result of fitness for service issues is expected to have a probability less than 0.01.

Safety Significance Level 4 is assigned for Reactor Safety – Core Damage Frequency (F1). The potential failure of safety related piping is insignificant in the Probabilistic Safety Assessment, and resolution of this Global Issue will ensure that the assumptions in the Probabilistic Safety Assessment regarding piping failure frequency remain valid. Therefore, the change in the Core Damage Frequency will be less than 10^{-7}/y, which corresponds to the fourth row of Table F1 in the PSR2 Basis Document, for which the Safety Significance Level is 4.

Resolution of this Global Issue is not expected to have any significant impact on Public Radiation Safety (F3) and accordingly this consideration is assigned Safety Significance Level 4. The other probabilistic considerations, i.e., Occupational Radiation Safety (F5), Emergency Preparedness (F6) and Environment (F7) are not applicable to this Global Issue. Hence, the overall Safety Significance Level of 3 for probabilistic considerations is dictated by F2 and F4 categorizations.

In summary, both deterministic and probabilistic considerations dictate that this Global Issue has Safety Significance Level 3. As noted, OPG is completing activities to ensure the fitness for service of Pickering buried piping for the operational life of the station.

SECTION 5 - GI-7 RESOLUTION PLAN

GI-7-RS1	Update the Buried Piping Program asset management plan [N-PLAN-04916-10002 R003, *Buried Piping Program Asset Management Plan*, January 2017] and risk ranking [P-MAN-04916-00001-R002, *Pickering Strategy Manual for Selection of Systems and Components for Inspection – Buried Piping*, March 2014] for the extended operating period. (SF2-14)
GI-7-RS2	Update governance to reflect a graded approach in the event that leakage in fuel oil piping occurs. This graded approach recognizes the nuclear safety importance of the systems being supplied with fuel oil, and would allow these systems to be temporarily repaired while awaiting further corrective action, allowing the systems to remain in service. This will involve document revision of Buried Piping Program Requirements [N-PROC-MA-0088-R003, *Buried Piping Program Requirements*, April 7, 2015]. (SF1-35)

B.8. GI-8 Completion / Updating of the Condition Assessments

SECTION 1 - GI-8 GLOBAL ISSUE SUMMARY					
The goal of GI-8, Completion/Updating of the Condition Assessments, is to confirm the completeness of the Pickering Condition Assessments (CAs) for the extended operating period. This Global Issue comprises two proposed Resolution Statements addressing four Gaps: two SF2 review task Gaps and two SF2 Additional Gaps. GI-8 is Safety Significance Level 2 based on deterministic and probabilistic defence-in-depth considerations as well as plant operability considerations. OPG has completed a significant amount (~95%) of CA work for the extended operating period. No issues have been found that significantly impact safety [P-REP-03680-00005 R001]. The proposed Resolution Plan comprises activities to complete and update the CAs for the piping systems and Commodity Groups for the extended operating period. Piping systems include many systems for which the CAs have been completed, e.g., the Heat Transport and Moderator Systems. Commodity Groups comprise components having common attributes encompassing all SSCs in Systems Important to Safety and Safe Operating Envelope systems. Recommendations resulting from updating the CAs will be assessed and included, as appropriate, in the CA action plans in the System and Component Health Reports per current governance. The final actions will also be tracked to completion as part of a process that will be created by OPG to track and report on progress to the CNSC. Completion of the proposed Resolution Plan will support and strengthen Level 1 defence-in-depth for the extended operating period.					
Safety Significance Level:	2	**Category:**	Analytical	**Reassessment Beyond 2024:**	Y

SECTION 2 - GI-8 ASSOCIATED GAPS	
SF2-12	Condition Assessments for in-scope piping systems are not complete for station operation to 2028[33]. ***Review Task #1 Actual Condition of SSCs*** **Associated Resolutions:** GI-8-RS1, GI-8-RS2
SF2-15	Updated Detailed Condition Assessments are not complete for Commodity Groups in the scope of PSR2 for station operation to 2028[33]. ***Review Task #1 Actual Condition of SSCs*** **Associated Resolutions:** GI-8-RS1, GI-8-RS2
SF2-AG4	An SF2 Gap related to knowledge of the actual condition of Structures, Systems, and Components Important to Safety and actions to address findings, was identified in [P-CORR-00531-05099, e-Doc 5295534, July 12, 2017]. **Associated Resolutions:** GI-8-RS2

SECTION 2 - GI-8 ASSOCIATED GAPS	
SF2-AG8	An SF2 Gap related to Obsolescence of Digital Controllers and a review of Critical Safety Parameters Monitoring Instrumentation was identified in [P-CORR-00531-05099, e-Doc 5295534, July 12, 2017]. The associated resolution addresses Item (b) of the reference. Item (a) is being addressed as described in [P-CORR-00531-05132, *Pickering Periodic Safety Review 2- Process for Addressing CNSC Identified Additional Gaps*, September 18, 2017]. **Associated Resolutions:** GI-8-RS1

SECTION 3 – GI-8 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

This Global Issue contains two SF2 Gaps and two SF2 Additional Gaps related to the Condition Assessments (CAs).

OPG has completed a significant amount (~95%) of CA work for the extended operating period. No issues have been found that significantly impact safety [P-REP-03680-00005 R001, *Pickering NGS PSR2 Safety Factor 2 Report – Actual Condition of Structures, Systems and Components Important to Safety*, March 2017].

The recommendations included in the Condition Assessments identify activities for current and extended operating life. The set of recommendations will be assessed and prioritized as per normal practice [N-PROC-MP-0060 R005B, OPG Nuclear Procedure, *Aging Management Process*, October 2015] associated with Aging Management assessments. When finalized, the set of actions will be tracked in System and Component Health Reports per current governance. The final actions will also be tracked to completion as part of a process that will be created by OPG to track and report on progress to the CNSC.

This overarching Global Issue will include the Irradiated Fuel Bay (IFB) Structures, Systems and Components; secondary side pressure retaining components; Deaerator and Deaerator Storage Tanks; Fuelling Machines and Fuelling Machine Bridge Ball Screws, as well as the Primary Heat Transport Auxiliary Piping Systems and Primary Heat Transport Valves.

The instrumentation and sensing devices used for CSPM have been assessed as part of the Condition Assessments of other systems, for example, under the Boiler Feed System for boiler level. Similarly, assessment of display and annunciation related components is included in systems such as Digital Control Computers, Boiler Steam and Water Systems, Electrical Systems etc.

As part of the Integrated Aging Management Program, the ongoing evaluation of the condition of critical SSCs, including the instrumentation related to CSPMs, is accomplished through the regular update of Condition Assessments. Critical components associated with CSPM are included in the Aging Management scope for PSR2.

SECTION 4 – GI-8 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								Overall Safety Significance Level
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	2	N/A	**2**	4	2	4	2	N/A	N/A	N/A	**2**	**2**

Rationale:

Addressing this Global Issue assures the completion of Condition Assessments for various systems and components to demonstrate fitness for service and good operating condition for the extended operating period. Safety functions can be impacted by the aged conditions of systems important to safety and, as such, completion of their Condition Assessment will identify required actions to ensure the equipment remains in good condition for the extended operating period. As discussed in Section 3 of this Global Issue, OPG has completed the majority of the Condition Assessments and no issues have been identified that significantly impact safety.

Nevertheless, this Global Issue has Safety Significance Level 2 for deterministic considerations with respect to Defence in Depth (E1), since ensuring fitness for service will preserve the integrity of the plant physical barriers (row 2 of Table E1 in the PSR2 Basis Document). Safety Significance Levels (E2) is considered not applicable because this Global Issue can have a direct impact on nuclear safety, whereas E2 primarily relates to issues without a direct impact on nuclear safety. Hence, the overall Safety Significance Level of 2 for deterministic considerations is dictated by the E1 categorization.

With respect to probabilistic considerations, this Global Issue is assigned Safety Significance Level 2 with respect to Table F2 in the PSR2 Basis Document. This is because resolving the issue preserves safety margins for systems important to safety in mitigating initiating events with a frequency $>10^{-2}$/y (row 3 of Table F2 in the PSR2 Basis Document). Similarly, the Safety Significance Level with respect to Plant Operability (F4) is also 2, because an extended period of plant shutdown as a result of fitness for service issues is expected to have a probability less than 0.1.

Safety Significance Level 4 is assigned to Core Damage Frequency (F1) and Public Radiation Safety (F3) since this Global Issue has an insignificant impact on these considerations. This Global Issue has no impact on the other probabilistic considerations, i.e., Occupational Radiation Safety (F5), Emergency Preparedness (F6) and Environment (F7). Therefore, these considerations are not applicable.

In summary, both deterministic and probabilistic considerations dictate that this Global Issue has Safety Significance Level 2. As noted, OPG's existing programs address this issue on an ongoing basis to ensure the Condition Assessments are completed and updated.

SECTION 5 – GI-8 RESOLUTION PLAN	
GI-8-RS1	Complete and update Condition Assessments (CA) for the piping systems and commodity groups in PSR2 scope for station operation for the extended operating period. Resulting recommendations will be assessed and included, as appropriate, in the CA action plans in the System and Component Health Reports. OPG is actively progressing with this work. (SF2-12) (SF2-15) (SF2-AG8 Item (b)) This Resolution Statement includes/addresses GI-10, GI-20, GI-21, GI-22, GI-29, and GI-49 for Condition Assessments.
GI-8-RS2	Develop and implement a process to track and report aging-management-related actions from the Condition Assessment recommendations. (SF2-12) (SF2-15) (SF2-AG4) As the aging-management related actions will be tracked using the process described, cross-references from Global Issues whose Resolution Statements may generate such actions, are not included in this report.

B.9. GI-9 Seismic Capacity of the Conveyor Tube and Fuel Basket Stacking Arrangement

SECTION 1 - GI-9 GLOBAL ISSUE SUMMARY					
The goal of GI-9, Seismic Capacity of the Conveyor Tube and Fuel Basket Stacking Arrangement, is to confirm the seismic capacity of the Pickering 5-8 IFB conveyor tube and fuel basket stacking arrangement in the Pickering IFBs for the extended operating period. This Global Issue comprises one proposed Resolution Statement and one item requiring No Further Action, addressing two SF2 review task Gaps. GI-9 is Safety Significance Level 3 based on deterministic and probabilistic defence-in-depth considerations. The seismic capacity assessment of the Pickering 5-8 IFB conveyer tube has been completed [NK30-CALC-35260-00001 R000], demonstrating the adequacy of the design. No Further Action is required for this Gap. The proposed Resolution Plan comprises activities to complete the required assessment to support the current fuel basket stacking arrangements in the Pickering IFBs. Completion of the proposed Resolution Plan will support and strengthen Level 1 defence-in-depth for the extended operating period.					
Safety Significance Level:	3	**Category:**	Analytical	**Reassessment Beyond 2024:**	N

SECTION 2 - GI-9 ASSOCIATED GAPS	
SF2-17	The seismic capacity of the current spent fuel basket stacking arrangements in the Pickering IFBs needs to be documented. ***Review Task #4 Spent Fuel Storage Facilities*** **Associated Resolutions:** GI-9-RS1
SF2-18	The seismic capacity of the Pickering 058 IFB conveyer tunnel needs to be documented. ***Review Task #4 Spent Fuel Storage Facilities*** • **Note – Per GI-9-NFA1, this Gap has been fully addressed.** **Associated Resolutions:** GI-9-NFA1

SECTION 3 - GI-9 BACKGROUND INFORMATION AND RESOLUTION STRATEGY
This Global Issue contains two SF2 Gaps related to the seismic capacity of the Pickering 5-8 IFB conveyor tube and fuel basket stacking arrangement in the Pickering IFBs. As per SCR #s P-2013-05015 and P-2015-11143 and AR # 28182003, the seismic capacities of the current spent fuel basket stacking arrangements in the Pickering IFBs and the Pickering 5-8 conveyer

SECTION 3 - GI-9 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

tube are to be documented.

SECTION 4 - GI-9 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								Overall Safety Significance Level
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	3	N/A	**3**	N/A	3	N/A	N/A	N/A	N/A	N/A	**3**	**3**

Rationale:

As discussed in Section 5 of this Global Issue, for this Global Issue, the resolution of this Global Issue will support the current spent fuel basket stacking arrangements in the Pickering IFBs. The seismic capacity assessment of the Pickering 5-8 IFB conveyer tube has been completed demonstrating the adequacy of the design.

Regarding deterministic considerations, the Safety Significance Level determined from Table E1 in the PSR2 Basis Document is Safety Significance Level 3 since resolution of this issue confirms the capability of the safety function (first column, third row of Table E1 in the PSR2 Basis Document). Table E2 in the PSR2 Basis Document is assessed as not applicable because this Global Issue can have a direct impact on nuclear safety, whereas E2 primarily relates to issues without a direct impact on nuclear safety. The overall Safety Significance Level for deterministic considerations is 3.

Regarding probabilistic considerations, Defence in Depth (F2) is the dominant consideration because of the potential impact on a barrier. Safety Significance Level 3 is assigned to Defence in Depth (F2) on the basis that seismic qualification of the spent fuel basket stacking arrangements precludes a partial loss of safety on a secondary parameter, i.e., the bottom row of Table F2 in the PSR2 Basis Document. The most conservative Safety Significance Level for this row is selected. This Global Issue has no impact on the other probabilistic considerations, i.e., Core Damage Frequency (F1), Public Radiation Safety (F3), Plant Operability (F4), Occupational Radiation Safety (F5), Emergency Preparedness (F6) and Environment (F7). Therefore, these considerations are not applicable.

In summary, the overall Safety Significance Level is 3. OPG has activities underway to fully address this issue.

SECTION 5 - GI-9 RESOLUTION PLAN	
GI-9-RS1	Complete the required assessment to support the current fuel basket stacking arrangements in the Pickering IFBs. This seismic related issue was noted in the response to Fukushima Action Item FAI 2.1.2. Additional investigation is required to support the current spent fuel basket stacking arrangements in the Pickering IFBs. (SF2-17)
GI-9-NFA1	The seismic capacity assessment of the Pickering 5-8 IFB conveyer tube has been completed [NK30-CALC-35260-00001 R000, OPG Engineering Calculation, *PNGSB Conveyor Tube Seismic Assessment*, March 2017], demonstrating the adequacy of the design. No further action is required. (SF2-18)

B.10. GI-10 IFB Condition

SECTION 1 - GI-10 GLOBAL ISSUE SUMMARY					
The goal of GI-10, IFB Condition, is to ensure the adequacy of the condition of the Pickering IFBs for the extended operating period. This Global Issue comprises one proposed Resolution Statement and one Cross Reference to GI-8, addressing two Gaps: one SF2 review task Gap and one SF4 review task Gap. GI-10 is Safety Significance Level 3 based on deterministic and probabilistic defence-in-depth considerations. The leakage from the IFB-B liner to the interspace is being tracked and OPG has already submitted a detailed action plan for addressing this issue [P-CORR-00531-04624, OPG Correspondence, *Response to CNSC Action Item 2014-48-5386 Status Update: CNSC Review of 2013 Groundwater Monitoring Results Report – Pickering B IFB Leak Mitigation*, February 26, 2016]. The status update of the project was provided in [P-CORR-00531-04865, OPG Correspondence, *Status Update: Pickering B Irradiated Fuel Bay Leak Mitigation Project #13-40703, Action Item 2014-48-5386*, November 17, 2016]. The proposed Resolution Plan comprises activities to complete the Pickering 5-8 IFB Leakage Mitigation Project to mitigate leaks from the IFB-B liner to the interspace. In addition, completion of the Condition Assessment of the IFB SSCs is cross-referenced to GI-8, Completion/Updating of the Condition Assessments.					
Safety Significance Level:	3	**Category:**	Engineering, Analytical	**Reassessment Beyond 2024:**	Y

SECTION 2 - GI-10 ASSOCIATED GAPS	
SF2-16	Action plans to correct the leakage in IFB-B are not complete. ***Review Task #4 Spent Fuel Storage Facilities*** **Associated Resolutions:** GI-10-RS1
SF4-2	Per Safety Factor 4 Review Task #9, there is a Gap with respect to the Aging Management practices of IFB facilities at Pickering NGS that is to specifically produce a CA for the Pickering 1-4 IFB SSCs and the Pickering Auxiliary IFB SSCs. It is noted that work to address this Gap is currently underway as part of the Pickering NGS Condition Assessments for the Pickering IFBs that will be addressed under Safety Factor Report 2, *Actual Condition of Structures, Systems and Components*. Since this work is not yet complete, this is identified as a Gap for Pickering PSR2. ***Review Task #9: Management of Aging for Spent Fuel Storage Facilities*** **Associated Resolutions:** GI-8-RS1

SECTION 3 - GI-10 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

This Global Issue contains two Gaps (one SF2 Gap and one SF4 Gap) that are related to the conditions of the IFBs.

The IFB-B leakage (CNSC Action Item 2014-48-5386) is being tracked by OPG. OPG has already submitted a detailed action plan for addressing this issue [P-CORR-00531-04624, OPG Correspondence, *Response to CNSC Action Item 2014-48-5386 Status Update: CNSC Review of 2013 Groundwater Monitoring Results Report – Pickering B IFB Leak Mitigation*, February 26, 2016]. OPG committed to provide the CNSC with a status update on Project # 13-40703 “Pickering B 90580 Irradiated Fuel Bay (IFB) Leak Mitigation” to address CNSC Action Item 2014-48-5386 by November 23, 2016. The status update of the project was provided in [P-CORR-00531-04865, OPG Correspondence, *Status Update: Pickering B Irradiated Fuel Bay Leak Mitigation Project #13-40703, Action Item 2014-48-5386*, November 17, 2016].

Work to assess the IFB SSCs is currently underway as part of the Pickering NGS Condition Assessments for the Pickering IFBs (see also GI-8).

SECTION 4 - GI-10 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								**Overall Safety Significance Level**
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	3	N/A	**3**	N/A	3	4	N/A	4	N/A	4	**3**	**3**

Rationale:

Resolution of this Global Issue will address leakage of the IFB liner to the interspace in IFB-B, as well as other aging-related and Condition Assessment issues for the IFBs. OPG has activities underway to address both aspects of this Global Issue.

Regarding deterministic considerations, Defence in Depth (E1) usually refers to barriers associated with the reactor. However, the IFBs also represent a physical barrier to the release of radioactive material from the station, so the Safety Significance Level is determined from Table E1 of the PSR2 Basis Document. On the basis that IFB issues, even liner leakage, do not impair the capability of safety provisions to effectively terminate an initiating event (first column of Table E1), Safety Significance Level 3 is selected. The last column of Table E1 is also relevant, as it refers to

SECTION 4 - GI-10 PRIORITY DETERMINATION

operational performance, for which Safety Significance Level 3 is also applicable since, from an operational perspective, improvements related to liner leakage are warranted. Table E2 is deemed not applicable, since this Global Issue can have a direct impact on nuclear safety, whereas E2 primarily relates to issues without a direct impact on nuclear safety. The overall Safety Significance Level for deterministic considerations is, therefore, 3.

Regarding probabilistic considerations, Defence in Depth (F2) is the dominant consideration due to the potential impact on a barrier to release. A Safety Significance Level of 3 is assigned on the basis that leakage into the interspace is at most only a partial loss of a barrier, so the bottom row of Table F2 is applicable. The most conservative Safety Significance Level for this row is selected, i.e., Safety Significance Level 3.

IFB leakage into the interspace is not expected to have any impact on Public Radiation Safety (F3), Occupational Radiation Safety (F5) or Environment (F7), but for conservatism, a Safety Significance Level of 4 is selected for these considerations.

As IFB issues do not affect the reactor core, Core Damage Frequency (F1) is not impacted. The potential for IFB events to contribute to the Large Release Frequency is addressed in the safety analysis program, so Table F1 is not applicable to this Global Issue. Plant Operability (F4) is not applicable either, since the issues associated with this Global Issue do not represent a loss of operating margin, which is the lowest condition for this consideration being assigned a Safety Significance Level. Emergency Preparedness (F6) is not applicable, since any minor IFB leakage to the interspace will not impact the response to a postulated initiating event.

In summary, the overall Safety Significance Level for this Global Issue is 3. Actions are in progress to address this Global Issue.

SECTION 5 - GI-10 RESOLUTION PLAN

GI-10-RS1	Complete the Pickering 5-8 IFB Leakage Mitigation Project [P-CORR-00531-04865, OPG Correspondence, *Status Update: Pickering B Irradiated Fuel Bay Leak Mitigation Project #13-40703, Action Item 2014-48-5386*, November 17, 2016] to mitigate leaks from IFB-B to the interspace. (SF2-16)
GI-10-XRF-GI-8-RS1	Complete the Condition Assessment of the IFB SSCs. (SF4-2)

B.11. GI-11 Fuel Management and Surveillance Software Upgrade

SECTION 1 - GI-11 GLOBAL ISSUE SUMMARY					
The goal of GI-11, Fuel Management and Surveillance Software Upgrade, is to ensure that CNSC Action Item 2016-OPG-8250 regarding the requirements for long-term core surveillance activities is addressed for the extended operating period. This Global Issue comprises one Acceptable Deviation, addressing one COP Review Gap. GI-11 is Safety Significance Level 4 based on deterministic safety significance levels considerations. OPG has provided a response to Action Item 2016-OPG-8250 [N-CORR-00531-18204] indicating that a status update will be provided with details of revised station documents, responses to suggestions for improvement, and a progress update on the fuel monitoring plan. The proposed Resolution Plan assesses this issue as an Acceptable Deviation because OPG is actively progressing completion of its commitment outside PSR2, and because of its very low Safety Significance.					
Safety Significance Level:	4	**Category:**	Analytical	**Reassessment Beyond 2024:**	N

SECTION 2 - GI-11 ASSOCIATED GAPS	
COP-15	There are remaining issues from Generic Action Item (GAI) 01G01 *Fuel Management and Surveillance Software Upgrade* (AI 2016-OPG-8250). ***Pickering PSR2 Gap COP-15*** **Associated Resolutions:** GI-11-AD1

SECTION 3 - GI-11 BACKGROUND INFORMATION AND RESOLUTION STRATEGY
This Global Issue contains one COP Review Gap related to Fuel Management and Surveillance Software. CNSC staff provided closure of Generic Action Item (GAI) 01G01, *Fuel Management and Surveillance Software Upgrade* and opened AI 2012-OPG-3465 to track associated actions. This AI was subsequently closed and a new AI (2016-OPG-8250) was opened to track the CNSC staff request that relevant OPG governance and/or station documents be revised to explicitly capture requirements for the long-term core surveillance activities identified. OPG has provided a response to this Action Item [N-CORR-00531-18204, OPG Correspondence, *Darlington and Pickering NGS: Fuel Management Surveillance Software Upgrade – New Action Item 2016-OPG-8250*, September 16, 2016] indicating that a status update will be provided with details of revised station documents, responses to suggestions for improvement, and a progress update on the fuel monitoring plan. This issue applies to Pickering Units 1,4 and Units 5-8.

SECTION 4 - GI-11 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								Overall Safety Significance Level
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	N/A	4	**4**	N/A	N/A	N/A	N/A	N/A	N/A	N/A	**N/A**	**4**

Rationale:

Resolution of this Global Issue will confirm the adequacy of certain approximations in the Fuel Management and Surveillance Software. Actions to address this Global Issue are underway.

Regarding deterministic considerations, this Global Issue is not associated with a physical barrier, so Defence in Depth (E1) is not applicable. For Safety Significance Levels (E2), Safety Significance Level 4 is determined to be applicable, since resolution of this issue may help identify areas that need more attention, in this case the station core model or fuel management software. Therefore, an overall Safety Significance Level of 4 is selected for deterministic considerations.

For this Global Issue, the probabilistic considerations are not applicable. This Global Issue has no impact on Core Damage Frequency (F1) or probabilistic Defence in Depth (F2), nor does it impact Plant Operability (F4), Public Radiation Safety (F3), Occupational Radiation Safety (F5), Emergency Preparedness (F6) or Environment (F7). The issue will result in confirmation of or enhancements to software that is already in service and is not addressing a deficiency. Therefore, probabilistic considerations have no impact on the overall Safety Significance Level.

In summary, the overall Safety Significance Level is 4. OPG is addressing this issue outside of PSR2.

SECTION 5 - GI-11 RESOLUTION PLAN

GI-11-AD1	OPG is actively progressing completion of the OPG commitment to provide additional information to address Action Item 2016-OPG-8250 (AR # 28193296) on fuel management and surveillance software, outside of PSR2. This issue is related to providing additional information on the Fuel Management and Surveillance Software and as per the very low safety significance (Safety Significance Level 4); it is assessed as an Acceptable Deviation. (COP-15)

B.12. GI-12 Extending the Environmental Qualification of Equipment

SECTION 1 - GI-12 GLOBAL ISSUE SUMMARY					
The goal of GI-12, Extending the Environmental Qualification of Equipment, is to confirm the environmental qualification of components for the extended operating period for components whose qualification would otherwise expire (referred to as life-limited components). This Global Issue comprises one proposed Resolution Statement, addressing one SF3 review task Gap. GI-12 is Safety Significance Level 3 based on deterministic and probabilistic defence-in-depth considerations. The proposed Resolution Plan comprises activities to update Environmental Qualification Assessments (EQAs) for life-limited components to support the extended operating period. Completion of the proposed Resolution Plan will support and strengthen Level 3 defence-in-depth for the extended operating period.					
Safety Significance Level:	3	**Category:**	Analytical	**Reassessment Beyond 2024:**	Y

SECTION 2 - GI-12 ASSOCIATED GAPS	
SF3-1	Per Review Task #3, the Environmental Qualification Program N-PROG-RA-0006, *Environmental Qualification Program* requires that whenever equipment on the EQL approaches the end of its qualified life, action must be taken to sustain the qualified status of that equipment regardless of what the station current life is taken to be. Hence, all EQAs will need to be re-assessed to ensure qualification is maintained in order to support continued operation of Pickering NGS beyond 2020. The current Environmentally Qualified life of all Pickering NGS SSCs may not necessarily extend to 2028[33] and a full review of Environmentally Qualified life-limited components impacted by operation past 2020 will need to be undertaken prior to life extension of Pickering NGS. This is therefore identified as a gap exists for Pickering PSR2. It is noted that work to address this gap is currently underway as part of the update of Pickering NGS Condition Assessments for Safety-Related Systems and Life Cycle Management Plans for Major Components. ***Review Task #3 Qualification of Installed Equipment*** **Associated Resolutions:** GI-12-RS1

SECTION 3 - GI-12 BACKGROUND INFORMATION AND RESOLUTION STRATEGY
This Global Issue contains one SF3 Gap related to the Environmental Qualification of equipment. A review of Environmentally Qualified life-limited components impacted by operation past 2020 will need to be undertaken. (See GI-1, GI-2, GI-3, GI-4 and GI-8).

SECTION 4 - GI-12 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								Overall Safety Significance Level
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	3	N/A	**3**	4	3	N/A	N/A	N/A	N/A	N/A	**3**	**3**

Rationale:

Resolution of this Global Issue will complete Environmental Qualification re-assessments to ensure the equipment will perform its safety functions in support of extended operation. OPG is actively progressing this work.

Regarding deterministic considerations, the Safety Significance Level determined from Table E1 in the PSR2 Basis Document is 3. This is because ensuring Environmental Qualification for extended operation will ensure the capability of safety provisions to effectively terminate an initiating event (first column, third row in Table E1 in the PSR2 Basis Document) and the issue does not affect the safety function capability for more than one level of protection (second column, third row in Table E1 in the PSR2 Basis Document). Safety Significance Levels (E2) is considered not applicable, since this Global Issue has a direct nuclear safety impact, whereas E2 primarily relates to issues that impact other objectives or are indirectly related to nuclear safety. The overall Safety Significance Level for deterministic considerations is 3.

Regarding probabilistic considerations, Defence in Depth (F2) is assigned Safety Significance Level 3. This is on the basis that confirming Environmental Qualification of the equipment associated with Systems Important to Safety precludes a reduction in the reliability of a System Important to Safety in responding to events of frequency $>10^{-3}$/y, or reduction in the reliability of a back-up system in responding to events of frequency $> 10^{-2}$/y. Safety Significance Level 4 is assigned for Reactor Safety – Core Damage Frequency (F1). Potential failure of equipment is accounted for in the Probabilistic Safety Assessment, and resolution of this Global Issue will ensure that the assumptions in the Probabilistic Safety Assessment regarding equipment failure frequency remain valid. Therefore, the change in the Core Damage Frequency will be less than 10^{-7}/y, which corresponds to the fourth row of Table F1 in the PSR2 Basis Document, for which the Safety Significance Level is 4.

This Global Issue has no impact on the other probabilistic considerations, i.e., Public Radiation Safety (F3), Plant Operability (F4), Occupational Radiation Safety (F5), Emergency Preparedness (F6) and Environment (F7). Therefore, these considerations are not applicable.

In summary, the overall Safety Significance Level is 3. OPG has activities underway to address this

SECTION 4 - GI-12 PRIORITY DETERMINATION
issue.

SECTION 5 - GI-12 RESOLUTION PLAN	
GI-12-RS1	Complete EQA re-assessments to support the extended operating period. OPG is actively progressing this work in support of extended operation at Pickering NGS. (SF3-1)

B.13. GI-13 Seismic Qualification - N289

SECTION 1 - GI-13 GLOBAL ISSUE SUMMARY					
The goal of GI-13, Seismic Qualification – N289, is to confirm that seismic qualification is adequately addressed with respect to the CSA N289 series of standards. This Global Issue comprises four Acceptable Deviations and one item requiring No Further Action addressing four Gaps: three SF3 code review Gaps and one COP Review Gap that also originates from a code review. GI-13 is Safety Significance Level 4 based on deterministic and probabilistic defence-in-depth considerations as well as core damage frequency considerations. Pickering NGS meets the intent of the codes and standards that were in place at the time of original design and construction. The Gaps related to modern requirements for seismic analysis, seismic testing procedures and seismic instrumentation requirements are resolved as three Acceptable Deviations based on the results of the PSA based Pickering Seismic Margin Assessments conducted as part of the Fukushima Integrated Action Plan, relevant assessments performed for Darlington, and their very low Safety Significance. The COP Review Gap identified with respect to the minimum number of cycles used for seismic fatigue analysis is assessed to require No Further Action for Units 5-8 because a review of actual design reports confirmed that the number of cycles used and continued to be used in seismic analyses meet or exceed CSA N289.3 requirements. For Units 1,4, the Gap with respect to the minimum number of cycles is assessed as an Acceptable Deviation because the systems and components are designed to American Society of Mechanical Engineers (ASME) requirements, including required cycles for fatigue analysis, and the Seismic Margin Assessment demonstrates that the risk associated with seismic hazards is sufficiently low.					
Safety Significance Level:	4	**Category:**	Analytical	**Reassessment Beyond 2024:**	N

SECTION 2 - GI-13 ASSOCIATED GAPS	
SF3-2	For N289.3-10, *Design Procedures for Seismic Qualification of Nuclear Power Plants* there is a gap associated with Safety Factor 3. Clause 4.4.4.5 of CSA N289.3-10 states: “The power spectral density (PSD) function of each time-history shall be calculated and shown to not have any significant gaps in energy over the frequency intervals outlined in Table 2….” The calculation of PSD is not addressed in the Pickering A or B PRA Based SMAs. The Pickering NGS A PRA Seismic Guide and the OPG PRA Guide do not identify any requirements for PSD. Also, evidence in the form of a calculation for time histories which represent the design ground motion was not found (which is a precursor for the PSD calculation). The lack of evidence of calculated time histories was also identified as a gap in the Darlington ISR (ISR Issues #D352 and #D617 – Documented evidence in the form of a calculation to show that the generated time history correctly represents the design ground response spectrum within the prescribed requirements has not been provided). The closure reference for #D352 and #D617 makes use of the detailed assessment performed in NK38-REP-03680-10224 R000 which is specific to Darlington. A similar assessment for Pickering NGS could not be found. As a result, there is a gap for PSR2 to provide similar evidence to show that: a) the generated time history used within seismic analyses of Safety-

SECTION 2 - GI-13 ASSOCIATED GAPS	
	Related Systems correctly represents the design ground response spectrum for the Pickering site in compliance with N289.3-10, and b) the PSD function of each time-history has been calculated and shown to not have any significant gaps in energy over the frequency intervals. ***Code Review N289.3-10*** **Associated Resolutions:** GI-13-AD1
SF3-3	For N289.4-12, *Testing Procedures for Seismic Qualification of Nuclear Power Plants Structures, Systems, and Components* there is a Gap associated with Safety Factor 3. Station-specific documents (including the Darlington seismic design guide, Darlington Reports and Darlington-specific technical specifications for seismic qualification) were used as the basis for compliance in the clause-by-clause Darlington code refresh review for clauses 4.2.1, 4.2.2.2, 4.2.3.1, 4.2.5, 4.3.2, 5.2.2.2.5, 5.7, 5.8.1, 5.8.1.2, 7.2.1, 7.7.1, 7.7.4 and 8.2. Pickering-specific seismic design guides, reports and technical specifications that are equivalent to those used to demonstrate Darlington compliance with the changes made in CSA N289.4-12 were identified. However, a detailed review to confirm that the Pickering-specific documents fully comply with the requirements of the clauses listed above is needed. As a result, this is a PSR2 Gap. ***Code Review N289.4-12*** **Associated Resolutions:** GI-13-AD2
SF3-4	For N289.5-12, *Seismic Instrumentation Requirements for Nuclear Power Plants and Nuclear Facilities* there is a gap associated with Safety Factor 3. Darlington ISR Issues #D622, D623 and D624 require no further action for Darlington as they were either classified as Acceptable Deviations or were closed. However, the issues are identified as a PSR2 gap for the following reasons: (Note: These gaps are closely related and are therefore identified as a single PSR2 gap.) • Darlington ISR Issue #624 refers to specific Darlington instrumentation in order to classify the gaps as Acceptable Deviations. It must be demonstrated that Pickering seismic instruments have the same capabilities as the Darlington instruments (fleet-wide or Pickering-specific standards that would ensure that the Pickering seismic instruments have the same capabilities as the Darlington instruments could not be found). Therefore, this is identified as a gap for PSR2. • Darlington ISR Issue #D622 was deemed to be of low safety significance. The same rationale may apply at Pickering, but first it must be demonstrated that Pickering has the same set up of seismic instruments as Unit 0 at Darlington. Therefore, this is identified as a gap for PSR2. • Darlington ISR Issue #D623 was deemed to be of low safety significance. The same rationale may apply at Pickering, but first it must be demonstrated that similar accelerometers are used at Pickering, and that their locations are not affected by strong ambient vibration. Therefore, this is identified as a gap for PSR2. ***Code Review N289.5-12***

SECTION 2 - GI-13 ASSOCIATED GAPS	
	Associated Resolutions: GI-13-AD3
COP-17	A review that considers the minimum number of cycles used for seismic fatigue analysis has not been completed for the level of compliance of Pickering NGS plant structures supporting the operation of Pickering reactors with CSA N289.3- M81, *Design Procedures for Seismic Qualification of CANDU Nuclear Power Plants*, Clause 5.13.2 in the context of extended operation. ***Pickering PSR2 Gap COP-17*** **Associated Resolutions:** GI-13-AD4, GI-13-NFA1

SECTION 3 - GI-13 BACKGROUND INFORMATION AND RESOLUTION STRATEGY
This Global Issue contains four Gaps (three SF3 Gaps and one COP Review Gap) that are related to the CSA N289 series (seismic qualification).

SECTION 4 - GI-13 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								Overall Safety Significance Level
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	4	N/A	**4**	4	4	N/A	N/A	N/A	N/A	N/A	**4**	**4**

Rationale:

This Global Issue relates to conformance with the safety-significant requirements of the CSA N289 series of standards on seismic qualification. The proposed Resolution Plan for this Global Issue comprises four Acceptable Deviations and one resolution with No Further Action.

Regarding deterministic considerations, the Safety Significance Level determined from Table E1 in the PSR2 Basis Document is 4, as the proposed Resolution Plan rationale shows that this issue does not affect a safety function (first column, row 4 of Table E1 in the PSR2 Basis Document). Safety Significance Levels (E2) is assessed as not applicable since this Global Issue has a direct nuclear

SECTION 4 - GI-13 PRIORITY DETERMINATION

safety impact, whereas E2 primarily relates to issues without a direct impact on nuclear safety. The overall Safety Significance Level for deterministic considerations is 4.

Regarding probabilistic considerations, the gaps associated with this Global Issue are analytical and/or related to instrumentation, testing and documentation, and their resolution does not impact the outcome of the Seismic PSA. Therefore, Safety Significance Level 4 is selected for Core Damage Frequency (F1), corresponding to a change in core damage frequency less than 10^{-7}/y. Defence in Depth (F2) is assigned Safety Significance Level 4 since this Global Issue has no significant impact on safety margin (refer to Table F2 in the PSR2 Basis Document). Given the nature of the gaps comprising this Global Issue, its resolution will have no impact on Public Radiation Safety (F3), Plant Operability (F4), Occupational Radiation Safety (F5), Emergency Preparedness (F6) and Environment (F7). Therefore, these considerations are not applicable.

In summary, the overall Safety Significance Level is 4.

SECTION 5 - GI-13 RESOLUTION PLAN

GI-13-AD1	The time-history method is a typical method for seismic qualification that has been employed at Pickering NGS. Time-history ground motion inputs at Pickering NGS were established based on the requirements of CSA N289.3-M81. [NA44-REP-02004-0119, Seismic Soil-Structure Interaction Analysis of Pickering NGS A Reactor Building, March 1996; P-REP-25140-00001-R02, Pickering Nuclear Vacuum Building Seismic Analysis, August 2010; NK30-REP-21 001-00002, Reactor Building Seismic Analysis, February 2012]. New power spectral density (PSD) requirements are stipulated in Clause 4.4.4.5 of the most recent version of the Standard (N289.3-10). However, in accordance with Clause 5.4 of CSA N289.1-08 R2013, an acceptable method of re-evaluating existing Nuclear Plants for seismic considerations is Seismic PSA. The Pickering NGS seismic qualification design response spectra were assessed to be appropriate and adequate as part of the Pickering NGS 1,4 and 5-8 Seismic Margin Assessments conducted as part of the Pickering A and Pickering B Probabilistic Safety Assessments [P-REP-03611-00006-R000, *Pickering NGS PSA Update to Include Enhancements from the Fukushima Integrated Action Plan*, April 30, 2014] and reviewed and concurred by the CNSC staff in [P-CORR-00531-04875, *Pickering NGS: Probabilistic Risk Assessment Based Seismic Margin Assessment,* November 2016]. In addition, the seismic spectra for Pickering and Darlington were prepared using similar methodologies, and the Darlington spectrum results were shown to be adequate in comparison with the new power spectral density requirements [NK38-REP-03680-10224 *Spectrum-enveloping N289.3 Code Compliance of Darlington NGS Seismic Time Histories,* July 25, 2014]. For these reasons, and as per the very low safety significance (Safety Significance Level 4), this is assessed as an Acceptable Deviation. (SF3-2)
GI-13-AD2	With respect to CSA N289.4-12 Testing Procedures for Seismic Qualification of Nuclear Power Plants, similar to the assessment conducted for Darlington as outlined in [P-REP-03680-00004 PNGS PSR2 Code and Standard Reviews for Safety Factors 2, 3 and 4, July 13, 2016], OPG's Aging Management governance

SECTION 5 - GI-13 RESOLUTION PLAN	
	[N-PROG-MP-0008-R006, Integrated Aging Management, April 29, 2016] applies equally for Pickering NGS, and this ensures that aging degradation effects are considered for Pickering equipment impacted by this Standard. In addition, the robustness of the Equipment Reliability Program at Pickering [N-PROG-MA-0026 Equipment Reliability, May 26, 2016], and the results of the Pickering Seismic Margin Assessments conducted as part of the Pickering A and Pickering B Probabilistic Safety Assessments [P-REP-03611-00006-R000, Pickering NGS PSA Update to Include Enhancements from the Fukushima Integrated Action Plan, April 30, 2014] confirm that the original design and seismic qualification of Pickering NGS provides an adequate level of safety from the earthquake hazard at the Pickering NGS site. OPG's commitment [N-CORR-00531-05661, Design Codes and Standards Effective Dates for OPG Nuclear Fleet, April 2012] to using CSA N289.4-12 for testing procedures for seismic qualification of nuclear power plant SSCs will ensure that the requirements will be followed. Furthermore, OPG governance is in place to ensure that CSA N289.4 is followed for seismic qualification of SSCs at OPG Nuclear Facilities [N-STD-MP-0025, *General Requirements For Seismic Qualification of OPG Nuclear Facilities*, October 27, 2016; Section 1.6]. As described in Sections 1.7.1 to 1.7.8 of N-STD-MP-0025, administrative controls have been developed to ensure that seismic qualification of SSCs performing safety-related functions during and following an earthquake is maintained for the life of the facility. For these reasons, and as per the very low safety significance (Safety Significance Level 4), this is assessed as an Acceptable Deviation. (SF3-3)
GI-13-AD3	Since the time of the Darlington NGS ISR, OPG has installed in-plant seismic instrumentation to monitor seismic activity at Pickering NGS (and at Darlington) in order to meet the intent of CSA standard N289.5-M91 [NK30-DM-61150-10001, *Pickering Nuclear: Seismic Monitoring* System, September 2013; N-GUID-02004-10000-R00, *Seismic Monitoring of OPG Nuclear Generating Stations*, December 2010]. Also, OPG has established procedures - Abnormal Incidents Manuals - that detail station response to earthquakes. Clear responsibilities are established to support monitoring and post-seismic response to an event [NK30-AIM-058-09013-6.0, *Abnormal Incident Manual: Seismic/Common Mode Event*; NA44-AIM-014-09013-06, *Abnormal Incident Manual: Seismic Event*]. Darlington ISR Issues #D622, D623 and D624 require no further action for Darlington as they were either classified as Acceptable Deviations or were closed. The Pickering Seismic Monitoring System was installed in 2012/2013 with the intent to meet the requirements of N298.4-M91, the same as those for Darlington seismic instrumentation; the rationales for acceptable deviations applied for Darlington NGS also apply to Pickering NGS. OPG is also a contributor to the operation of the Southern Ontario Seismic Network (SOSN) which provides detailed free-field seismic records covering Southern Ontario. These systems support in-plant monitoring of the station's response to seismic events. The results of the Pickering Seismic Margin Assessments conducted as part of the Pickering A and Pickering B Probabilistic Safety Assessments [P-REP-03611-00006-R000, *Pickering NGS PSA Update to Include Enhancements from the Fukushima Integrated Action Plan,* April 30, 2014] demonstrate the adequacy of the instrumentation and response to a seismic event at Pickering NGS. For these reasons, and as per the very low

SECTION 5 - GI-13 RESOLUTION PLAN	
	safety significance (Safety Significance Level 4), this is assessed as an Acceptable Deviation. (SF3-4)
GI-13-AD4	[Units 1,4] Pickering Units 1,4 Class 1 systems and components are designed to ASME Code Section III which betters the CSA N289.3 requirement, including required cycles for fatigue analysis. [For example: SR-30-33126-1, *Primary Heat Transport System (Stress Report) Feeder Pipes Units 5 to 8*, October 1982; NA44-DS-33126-00005-R00, *Pickering 'A' – Design Specification For Primary Heat Transport System (PHTS) Feeders For Units 1 and 4,* January 2009; NA44-REP-31100-2.0148, *Pickering GS 'A' Units 1 and 2 LSFCRP Feeder Connection Stress Analysis (West End),* August 1985]. Furthermore, Seismic Margin Assessment is an accepted method for assessing seismic qualification. The Pickering A seismic design basis [NA44-REP-02004-0073, *Seismic Assessment of Pickering 'A' Nuclear Generating Station*, February 1998; NA44-02004-0346-R1, *Seismic Margin Analysis Of Primary Heat Transport Main Circuit And Main Steam Piping At Pickering NGS A*, October 1997; NA44-REP-02004-0344-R1, *Seismic Margin Assessment Of Calandria And Fuelling Machine*, October 1997] was established using the EPRI Seismic Margin Assessment (SMA) methodology [Electric Power Research Institute, *A Methodology for Assessment of Nuclear Power Plant Seismic Margin*, Report NP-6041-SL, Revision 1, August 1991]. The SMA-based Probabilistic Safety Assessment for Pickering Units 1,4 [NA44-REP-03611-00022 R000, *PRA-Based Seismic Margin Assessment of PNGS-A*, January 2014] demonstrates that risk associated with seismic hazard is sufficiently low. For continuing analysis of potential new modifications, the requirements of CSA N289.3-10 Clause 5.13.2 would be met or bettered, consistent with the modelling methodology in place at OPG for such work as noted above. For these reasons, and as per the very low safety significance (Safety Significance Level 4), this is assessed as an Acceptable Deviation for Pickering 1,4. (COP-17)
GI-13-NFA1	[Units 5-8] With respect to the CSA N289.3-10 Clause 5.13.2 requirement on the minimum number of cycles used for seismic fatigue analysis, for Pickering 5-8, seismic structural analysis codes/modelling better the CSA N289.3-10 Clause 5.13.2 requirements for minimum number of cycles used for seismic fatigue analysis [OPG Memorandum, NK30-REF-68000-0476807, *Disposition of Continued Operations – 102 Action AR 28134792-07*, September 2013]. For continuing analysis of potential new modifications, the requirements of CSA N289.3-10 Clause 5.13.2 would be met or bettered, consistent with the modelling methodology in place at OPG for such work [DG-30-68000-2 R1, *Pickering NGS B Seismic Qualification of Safety Related Systems*]. No further action is required for Pickering 5-8. (COP-17)

B.14. GI-14 Environmental Qualification Program Issues

SECTION 1 - GI-14 GLOBAL ISSUE SUMMARY					
The goal of GI-14, Environmental Qualification Program Issues, is to ensure that complete Environmental Qualification documentation is in place for the extended operating period. This Global Issue comprises one item assessed as requiring No Further Action and one Acceptable Deviation, addressing two SF3 Audit and Self-Assessment Gaps. GI-14 is Safety Significance Level 4 based on deterministic safety significance levels considerations. CNSC staff has confirmed that the Action Notice on the Environmental Qualification documentation backlog has been completed by OPG [P-CORR-00531-04874]. For this reason, the proposed Resolution Plan assesses the documentation backlog Gap as requiring No Further Action. The proposed Resolution Plan also comprises activities in progress to revise the Environmental Qualification Assessment for Tefzel cables (AR # 28170757). This issue is related to an Environmental Qualification Assessment documentation update and, as per the very low Safety Significance, it is assessed as an Acceptable Deviation.					
Safety Significance Level:	4	**Category:**	Analytical	**Reassessment Beyond 2024:**	N

SECTION 2 - GI-14 ASSOCIATED GAPS	
SF3-5	The Environmental Qualification documentation backlog (e.g., Document Change Requests for Environmental Qualification Assessments) increased from 9% in Q4 2013 to 14% in Q4 2014. As a result, CNSC staff requested that OPG assess and create a corrective action plan to ensure that Environmental Qualification information remains current and more specifically to reduce and manage the document revision backlog. This is a Gap for PSR2 since the CNSC identified this issue in an Action Notice following a regulatory inspection and the associated action (AR#28179009) is due to be completed by Q3 2016. ***Audit and Self-Assessment Reviews*** • **Note – Per GI-14-NFA1, this Gap has been fully addressed.** **Associated Resolutions:** GI-14-NFA1
SF3-6	Pickering NGS is in non-compliance with N-PROC-RA-0044, *Environmental Qualification Assessment* for not correcting the documentation discrepancy for Unit 5-8 Vertical Flux Detector Tefzel cables with justification for a new qualified life value. As a result, CNSC staff requested that OPG revise the Environmental Qualification Assessment for Tefzel cables to reflect the change of qualified life of the Vertical Flux Detectors. This is a Gap for PSR2 since the CNSC identified this issue in an Action Notice following a regulatory inspection and the associated action (AR#28170757) is due to be completed by Q4 2016. ***Audit and Self-Assessment Reviews*** **Associated Resolutions:** GI-14-AD1

SECTION 3 - GI-14 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

This Global Issue contains two SF3 Gaps related to the Environmental Qualification Program.

CNSC staff identified the issue on the Environmental Qualification documentation backlog in an Action Notice (AN1) following a regulatory inspection [P-CORR-00531-04483, CNSC Correspondence, e-Doc 4766363, *Pickering NGS: CNSC Type II Compliance Inspection Report: PRPD-2015-005, Environmentally Qualified Equipment Inspection, New Action Item 2015-48-6459*, June 3, 2015]. Following implementation of corrective actions, this Action Notice (AN1) was subsequently confirmed complete by CNSC staff [P-CORR-00531-04874, e-Doc 5109031, *Pickering NGS: Followup to OPG Response to CNSC Type II Compliance Inspection: Environmentally Qualified Equipment, Action Item 2015-48-6459*, November 4, 2016].

SECTION 4 - GI-14 PRIORITY DETERMINATION

Safety Significance of Global Issue	**Deterministic Considerations**			**Probabilistic Considerations**								**Overall Safety Significance Level**
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	N/A	4	**4**	N/A	N/A	N/A	N/A	N/A	N/A	N/A	**N/A**	**4**

<u>Rationale:</u>

This Global Issue is related to an update of Environmental Qualification Assessment documentation, and is being addressed by OPG.

Regarding deterministic considerations, this Global Issue is not associated with a physical barrier, so Defence in Depth (E1) is not applicable. Safety Significance Levels (E2) is assigned Safety Significance Level 4 since resolution of this issue “may help identify areas that need more attention”, in this case the Environmental Qualification Assessment documentation update. Therefore, Safety Significance Level 4 is selected for deterministic considerations.

This Global Issue has no direct impact on the probabilistic considerations, i.e., Core Damage Frequency (F1), Defence in Depth (F2), Public Radiation Safety (F3), Plant Operability (F4), Occupational Radiation Safety (F5), Emergency Preparedness (F6) or Environment (F7). Therefore, probabilistic considerations are not applicable.

SECTION 4 - GI-14 PRIORITY DETERMINATION
In summary, the overall Safety Significance Level is 4. OPG is addressing this issue.

SECTION 5 - GI-14 RESOLUTION PLAN	
GI-14-AD1	Revision of the EQA for Tefzel cables is in progress (AR # 28170757). Visual inspection of the Vertical Flux Detector cables for Units 5-8 and review of ten sample EQAs for extent of condition have been completed. The revision of the EQA [NK30-EQA-31740-10000-R001, OPG Environmental Qualification Assessment, *HESIR Flux Detector Assembly*] is being actively progressed outside of PSR2. This issue is related to an EQA documentation update and as per the very low safety significance (Safety Significance Level 4); it is assessed as an Acceptable Deviation. (SF3-6)
GI-14-NFA1	The Action Plan (AR # 28179009) for Environmental Qualification Assessments (EQAs) to manage the document revision backlog is complete. The related CNSC Action Notice (AN1) was subsequently confirmed complete by CNSC staff [P-CORR-00531-04874, e-Doc 5109031, *Pickering NGS: Followup to OPG Response to CNSC Type II Compliance Inspection: Environmentally Qualified Equipment, Action Item 2015-48-6459*, November 4, 2016]. No further action is required. (SF3-5)

B.15. GI-15 Governance Issues

SECTION 1 - GI-15 GLOBAL ISSUE SUMMARY					
The goal of GI-15, Governance Issues, is to ensure that Gaps on various Governance issues are addressed for the extended operating period. This Global Issue comprises two items assessed as requiring No Further Action and one Acceptable Deviation, addressing four Gaps: one SF1 code review Gap [CSA N290.8-15], one SF4 review task Gap, one SF5 Gap from additional review findings and one SF5 Additional Gap. GI-15 is Safety Significance Level 4 based on deterministic safety significance levels considerations. The OPG Governance and the proposed Resolution Plans relevant to the identified Gaps are summarized as follows: • The Gap related to obsolescence of services or supplies external to the plant is addressed in OPG-PROC-0058, which specifies that any potential disruptions to the supply of items and/or services that may adversely impact OPG shall be addressed in a consistent and expedient manner. The proposed Resolution Plan, therefore, assesses this Gap as requiring No Further Action. • The Gap related to controls on the introduction of supplier-provided digital equipment into OPG nuclear power plants is addressed through processes outlined in N-PROG-MP-0001. The proposed Resolution Plan, therefore, assesses this Gap as an Acceptable Deviation. • The Gaps related to the use of the best estimate approach or a similarly conservative approach for analysis of operational events is addressed in Nuclear Safety Analysis governing document N-MAN-03600-10005-R006. These Gaps are assessed as requiring No Further Action because of their very low Safety Significance (Level 4) and because they are being addressed outside of PSR2.					
Safety Significance Level:	4	**Category:**	Programmatic	**Reassessment Beyond 2024:**	N

SECTION 2 - GI-15 ASSOCIATED GAPS	
SF4-1	The conclusion of Safety Factor 4 Review Task #7 is that programs for timely detection and mitigation of aging mechanisms and/or aging effects, including obsolescence of technology, have been established. However, N-STD-MA-0024, *Obsolescence Management* does not explicitly address obsolescence of services or supplies external to the plant. This is therefore identified as a Gap for Pickering PSR2. ***Review Task #7 Detection and Mitigation of Aging Systems*** • **Note – Per GI-15-NFA1, this Gap has been fully addressed.** **Associated Resolutions:** GI-15-NFA1
SF5-7	The Darlington Integrated Implementation Plan (IIP) [OPG Report, NK38-REP-03680-10185 R002, *Darlington NGS Integrated Implementation Plan (IIP)*, April 30, 2015] identified a Gap

SECTION 2 - GI-15 ASSOCIATED GAPS	
	(IIP-OI 055) related to use of the best estimate approach for analysis of operational events at Darlington NGS. The action for Darlington (AR 28175247, Target Completion Date Q1 2020) was to revise OPG governing document N-MAN-03600-10005, *Nuclear Safety Analysis*, to require the use of the best estimate approach or a similarly conservative approach for analysis of operational events. This action is also applicable for Pickering NGS and is therefore a Gap for Pickering PSR2. ***Additional Review Findings in Section 4.4 of Safety Factor 5 Report*** • **Note – Per GI-15-NFA2, this Gap has been fully addressed.** **Associated Resolutions:** GI-15-NFA2
SF1-34	Clause 4.7 of N290.8-15 mandates that the technical specification requires the supplier to identify and describe all digital items included in their equipment. In the event that the use of digital items is identified by OPG in advance of issuing a Request for Proposal or Request for Quotation, existing OPG procedures are adequate for ensuring that requirements related to digital items are documented in the technical specification. However, a requirement for a supplier to self-identify whether their product contains any digital items is not reflected in OPG governing documents This has therefore been identified as a PSR2 Gap. ***Code Review N290.8-15*** **Associated Resolutions:** GI-15-AD1
SF5-AG3	An SF5 Gap related to implementation of a Best Estimate Analysis or similarly conservative approach for Operational Events, or justification for the continued use of the existing conservative approach, was identified in [P-CORR-00531-05090, e-Doc 5289137, June 30, 2017]. **Associated Resolutions:** GI-15-NFA2

SECTION 3 - GI-15 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

This Global Issue consolidates four Gaps on various OPG governance issues. This includes one SF4 Gap, one SF5 Gap, one SF1 Gap and one SF5 Additional Gap.

SECTION 4 - GI-15 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								Overall Safety Significance Level
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	N/A	4	**4**	N/A	N/A	N/A	N/A	N/A	N/A	N/A	**N/A**	**4**

Rationale:

This Global Issue includes minor issues related to governance updates.

Regarding deterministic considerations, this Global Issue is not associated with a physical barrier, so Defence in Depth (E1) is not applicable. Safety Significance Levels (E2) is assigned Safety Significance Level 4 since resolution of this issue "may help identify areas that need more attention", in this case governance updates. Therefore, Safety Significance Level 4 is selected for deterministic considerations.

With respect to probabilistic considerations, this Global Issue has no impact on Core Damage Frequency (F1) or Defence in Depth (F2), nor does it impact Public Radiation Safety (F3), Plant Operability (F4), Occupational Radiation Safety (F5), Emergency Preparedness (F6) or the Environment (F7). Therefore, probabilistic considerations are not applicable.

In summary, the overall Safety Significance Level is 4.

SECTION 5 - GI-15 RESOLUTION PLAN

GI-15-AD1	OPG has established stringent controls on the introduction of digital equipment at its nuclear power plants. Technical specifications are prepared when performing design modifications, non-identical component replacements and item equivalencies. These processes, as outlined in N-PROG-MP-0001 *Engineering Change Control*, invoke detailed procedures which ensure appropriate specification of requirements with respect to digital equipment and embedded digital items, and include checks and confirmation of the form of equipment supplied via these processes. To add additional defence to the existing provisions, a Documentation Change Request has been filed (#140057) to update specification preparation governance to require suppliers to identify and describe digital items in equipment provided. The implementation of this change will be managed outside of PSR2. As per the very low safety significance (Safety

SECTION 5 - GI-15 RESOLUTION PLAN	
	Significance Level 4), it is assessed as an Acceptable Deviation. (SF1-34)
GI-15-NFA1	Obsolescence of services and supplies for any reason (business discontinuation/interruption, failure of a qualification audit, low volume interactions) is addressed by OPG-PROC-0058 [OPG-PROC-0058 R011, OPG Procedure, *Procurement Activities*, March 2017]. Section 1.20.5 of OPG-PROC-0058 specifies that any potential disruptions to the supply of items and/or services that may adversely impact OPG shall be addressed in a consistent and expedient manner. For nuclear and/or OPG wide suppliers: When OPG becomes aware of a potential supplier interruption a Station Condition Record (SCR) titled “Possible Supplier Interruption: [vendor name]” shall be initiated. Specific actions to address a supplier interruption of any type are then addressed by N-PROC-MM-0010 [N-PROC-MM-0010 R021, OPG Nuclear Procedure, *Establishing and Maintaining Ontario Power Generation Approved Suppliers List*, February 2017]. Among other things, this governance mandates an Approved Supplier List Oversight Committee meeting (ASLOC), which reviews impacts on internal stakeholders and initiates corrective actions as may be required. This governance ensures the risk posed by potential discontinuation of any external supplies or services is effectively mitigated. Furthermore, many of the same external services and supplies required for Pickering are also required for the Darlington station. Darlington is being refurbished to operate far beyond the Pickering extended operating period. OPG’s processes will provide assurance of supply of external services and supplies for the extended life of Darlington and Pickering stations. Accordingly, SF4-1 is adequately addressed by OPG-PROC-0058 [OPG-PROC-0058 R011, OPG Procedure, *Procurement Activities*, March 2017] and N-PROC-MM-0010 [N-PROC-MM-0010 R021, OPG Nuclear Procedure, *Establishing and Maintaining Ontario Power Generation Approved Suppliers List*, February 2017]. As a further enhancement, a Document Change Request has been filed to update N-PROC-MP-0060 on Aging Management to cross reference to the Procurement Activities Procedure OPG-PROC-0058. This work will be managed outside of PSR2. This review has identified that no change is required to N-STD-MA-0024 to address this issue. No further action is required in PSR2. (SF4-1)
GI-15-NFA2	Section 1.4.8.3 of OPG Nuclear Safety Analysis governing document [N-MAN-03600-10005-R006, OPG Manual, *Nuclear Safety Analysis*, September 2016] has been revised to include the use of the best estimate approach or a conservative approach for analysis of operational events. No further action is required. (SF5-7) (SF5-AG3)

B.16. GI-16 Concession Related to N285.5-M90

SECTION 1 - GI-16 GLOBAL ISSUE SUMMARY					
The goal of GI-16, Concession Related to N285.5-M90, is to confirm that concessions granted from CNSC for compliance with CSA N285.5 Periodic Inspection of CANDU Nuclear Power Plant Containment Components remain valid for the extended operating period. This Global Issue comprises two items assessed as requiring No Further Action and one Acceptable Deviation addressing one SF4 code review Gap [CSA N285.5]. GI-16 is Safety Significance Level 4 based on deterministic defence-in-depth considerations. Pickering is currently in compliance with CSA N285.5-08 Update No. 1 [P-CORR-00531-04186]. The approach to situations where components are deemed inaccessible for inspection is addressed in the OPG Periodic Inspection Program (PIP). The PIP Compliance Matrices state that full or partial disassembly of components will not be undertaken specifically for periodic inspection, as this may result in component damage. The CNSC have confirmed their finding that the revised PIPs satisfactorily meet the requirements of CSA N285.5-08 Update No. 1 [P-CORR-00531-04186]. This is not impacted by operation beyond 2020. The proposed Resolution Plan assesses the issue regarding the numerical rules for inspection of identical components as requiring No Further Action because Pickering follows the CSA N285.5-08 requirements. Similarly, the subject requirements of CSA N285.5-M90 for the timing of inspections are assessed as requiring No Further Action because they have been excluded from CSA N285.5-08. The issue regarding the extent of inspections for components deemed inaccessible is assessed as an Acceptable Deviation because of its very low Safety Significance.					
Safety Significance Level:	4	**Category:**	Programmatic	**Reassessment Beyond 2024:**	N

SECTION 2 - GI-16 ASSOCIATED GAPS	
SF4-9	There were a number of concessions granted from the CNSC for compliance with N285.5-M90 that will need to be reconciled for Pickering for the period of PSR2: (Since these Gaps are all concession-related and associated with N285.5-M90, they are tracked under a single PSR2 Gap). a) The Pickering B ISR Gap associated with N285.5-M90 clause 4.5.1 is closed. However, the disposition of the Gap refers to OPG receiving a concession from the CNSC on the inspection of components deemed to be inaccessible. A similar (updated) concession may be required for Pickering operation past 2020. Therefore, this is a Gap for PSR2. b) The Darlington ISR disposition of the Gaps for N285.5-M90 clauses 8.4.2.1 and 8.4.2.2 refer to OPG receiving a concession from the CNSC that insulation will not be removed in the absence of visible damage to a component, and only “light weight” access covers will be removed. The Darlington ISR states: “This is a concession from the regulator which is not assured in the case of refurbished plant.

SECTION 2 - GI-16 ASSOCIATED GAPS

As such, this represents a Gap". By the same logic it will need to be reconciled for Pickering for the period of PSR2 (life extension past 2020).

c) The Darlington ISR disposition of the Gap for N285.5-M90 for clause 8.5.2.2 refers to an exception of the numerical rules of this clause for reasons of practicality, and that a concession was received from the CNSC. The Darlington ISR stated "… it is categorized as a Gap, because a concession from the CNSC is not assured for a refurbished plant." By the same logic it will need to be reconciled for Pickering for the period of PSR2.

d) Per the Darlington ISR disposition of the Gap for N285.5-M90 clause 8.6.3, although CNSC acceptance was obtained, there is still a non-compliance with a portion of the clause related to the timing of inspections which is noted as needing to be reconciled for a refurbished station. The Darlington ISR stated "This represents a Gap that will need to be reconciled with the regulator for a refurbished station." By the same logic it will need to be reconciled for Pickering for the period of PSR2.

Code Review N285.5-13

Associated Resolutions: GI-16-AD1 (Gap Items a and b), GI-16-NFA1 (Gap Item c), GI-16-NFA2 (Gap Item d)

SECTION 3 - GI-16 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

This Global Issue contains one SF4 Gap related to CNSC concessions on N285.5-M90 [CSA N285.5-M90, *Periodic Inspection of CANDU Nuclear Power Plant Containment Components*], and whether they are affected by Pickering operation beyond 2020. Note: Pickering is currently in compliance with CSA N285.5-08 Update No. 1 [P-CORR-00531-04186].

The issues are related to:

a) CNSC concession on clause 4.5.1 on the inspection of components deemed to be inaccessible.

b) CNSC concession on clauses 8.4.2.1 and 8.4.2.2, that insulation will not be removed in the absence of visible damage to a component, and that only "light weight" access covers will be removed.

c) CNSC concession on clause 8.5.2.2 on the numerical rules of this clause for reasons of practicality.

The requirement in clause 8.5.2.2 of N285.5-08 states that the number of areas to be inspected for identical components in a multi-unit station shall not be fewer than that specified in Table 2 of the standard. Pickering follows the requirements identified in Table 2 of CSA N285.5-08 and the PIPs developed in accordance with CSA N285.5-08 have been accepted by the CNSC [P-CORR-00531-04186]. Therefore, this is no longer a Gap.

d) Clause 8.6.3: The requirements of N285.5-M90 Clause 8.6.3 have been excluded from N285.5-08. Therefore, this is no longer a Gap.

SECTION 4 - GI-16 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	**Overall Safety Significance Level**
	4	N/A	**4**	N/A	N/A	N/A	N/A	N/A	N/A	N/A	**N/A**	**4**

Rationale:

Resolution of this Global Issue will confirm applicability of a previously approved concession related to periodic inspection of Containment components.

Regarding deterministic considerations, Defence in Depth (E1) has Safety Significance Level 4 on the basis that no safety function is impacted by the issue. The Safety Significance Level determined from Table E2 in the PSR2 Basis Document is not applicable. The overall significance level for deterministic considerations is 4.

Regarding probabilistic considerations, this Global Issue has no impact on Core Damage Frequency (F1) or Defence in Depth (F2), nor does it impact Public Radiation Safety (F3), Plant Operability (F4), Occupational Radiation Safety (F5), Emergency Preparedness (F6) or the Environment (F7). Therefore probabilistic considerations are not applicable.

In summary, the overall Safety Significance Level is 4.

SECTION 5 - GI-16 RESOLUTION PLAN

GI-16-AD1	SF4-9 Gap Items a and b: CSA N285.5 Clauses 4.5.1, 8.4.2.1 and 8.4.2.2 are related to extent of inspection. The approach to situations where components are deemed inaccessible for inspection is addressed in the following OPG documents: Section 9.2 *Extent and Areas of Inspection* and Appendix F CAN/CSA-N285.5 *Compliance Matrices* of [NK30-PIP-03642.2-00001 R003, OPG Plan, *Pickering Nuclear Generating Station “B” Periodic Inspection Program for Containment Components*, July 31, 2012], [NA44-PIP-03642.2-00001 R002, OPG Plan, *Pickering Nuclear Generating Station “A” Periodic Inspection Program for Containment Components*, July 31, 2012] and Section 9.2 *Extent and Areas of Inspection* and Appendix G CAN/CSA-N285.5-08 *Compliance Matrices* of [P-PIP-03642.2-00001 R003, OPG Plan, *Pickering Nuclear Generating Station Periodic*

SECTION 5 - GI-16 RESOLUTION PLAN	
	Inspection Program for Unit 0 Containment Components, July 31, 2012]. The Compliance Matrices state that full or partial disassembly of components will not be undertaken specifically for periodic inspection, as this may result in component damage. The above PIPs have been accepted by the CNSC. In [NK30-CORR-00531-04876, CNSC Correspondence, e-Doc 3256609/ 2.01, *Pickering NGS-B Integrated Safety Review (ISR) - CNSC Review of Acceptable Deviations and Discrepancies for the Plant Design Safety Factor Report,* June 27, 2008], it is stated that the CNSC has accepted the reasonableness of the proposed solution to the above discrepancy and in [P-CORR-00531-04186, *Transition to 2008 Edition of CSA N285.5 Update No. 1 – Periodic Inspection of CANDU Nuclear Power Plant Containment Components – Submission of Periodic Inspection Programs,* November 2012], the CNSC confirmed their finding that the revised PIPs satisfactorily meet the requirements of CSA N285.5-08 Update No. 1. This is not impacted by operation beyond 2020. Appendix E of the Periodic Inspection Plans for Pickering B, e.g., [NK30-PIP-03641.2-00001, *Pickering Nuclear Generating Station B Periodic Inspection Plan for Unit 5*, August 2006] and Section 4 of the Periodic Inspection Plans for Pickering A, e.g., [NA44-PIP-03641.2-00001, *Pickering Nuclear Generating Station A Periodic Inspection Plan for Unit 1*, September 2006], state that "*Where the inspection of a system or component would necessitate the dismantling of equipment, the required inspection should be performed when the equipment is dismantled for other reasons (i.e., maintenance)*". As per the very low safety significance (Safety Significance Level 4), this is assessed as an Acceptable Deviation. (SF4-9, Items a and b)
GI-16-NFA1	The Pickering PIPs, accepted by the CNSC [P-CORR-00531-04186], are consistent with the requirements identified in Clause 8.5.2.2 and Table 2 of CSA N285.5-08, which relate to the number of areas to be inspected for identical components in a multi-unit station. No action is required. (SF4-9, Item c)
GI-16-NFA2	Clause 8.6.3 of the M90 edition of CSA N285.5 has been excluded from the 2008 edition of the standard. Pickering Containment PIPs have been accepted by the CNSC and the CNSC confirmed their finding that the PIPs satisfactorily meet the requirements of CSA N285.5-08 Update No. 1 [P-CORR-00531-04186, *Transition to 2008 Edition of CSA N285.5 Update No. 1 – Periodic Inspection of CANDU Nuclear Power Plant Containment Components – Submission of Periodic Inspection Programs,* November 14, 2012]. No action is required. (SF4-9, Item d)

B.17. GI-17 FFS of Fiberglass Reinforced Plastic Material for the Extended Operating Period

SECTION 1 - GI-17 GLOBAL ISSUE SUMMARY					
The goal of GI-17, FFS of Fiberglass Reinforced Plastic Material for the Extended Operating Period, is to confirm that the Fiberglass Reinforced Plastic (FRP) components remain fit for service for the extended operating period. This Global Issue comprises one item assessed as requiring No Further Action addressing one SF4 code review Gap [CSA N285.5-13]. GI-17 is Safety Significance Level 2 based on deterministic and probabilistic defence-in-depth considerations as well as plant operability considerations. The changes in N285.5-13 relative to N285.5-08 that are applicable to FRP material that is used at Pickering NGS have been assessed for fitness for service to 2024 [NA44-PLAN-34220-00002 R001, *Life Cycle and Aging Management Program Plan for Fiberglass-Reinforced Plastic Components in the Pickering NGS Vacuum Building*, September 27, 2012]. The proposed Resolution Plan assesses this issue as requiring No Further Action because the adequacy of the FRP material used at Pickering NGS has been assessed for fitness for service to 2024.					
Safety Significance Level:	2	**Category:**	Analytical	**Reassessment Beyond 2024:**	Y

SECTION 2 - GI-17 ASSOCIATED GAPS	
SF4-10	The changes in N285.5-13 relative to N285.5-08 that are applicable to Fiberglass Reinforced Plastic material that is used at Pickering NGS have only been assessed for fitness for service to 2024 in the Pickering Continued Operations Plan. These changes related to aging management (monitoring and test programs) for FRP materials. As a result, additional assessment is required for Pickering to address FRP aging management at Pickering for operation to 2028[33], and to confirm the current program aligns with N285.5-13 clauses 8.2, 8.3.3, 8.3.4 and A.6.1.2 (Note: This Gap only exists if Pickering NGS intends to operate past 2024). ***Code Review N285.5-13*** **Associated Resolutions:** GI-17-NFA1

SECTION 3 - GI-17 BACKGROUND INFORMATION AND RESOLUTION STRATEGY
This Global Issue contains one SF4 gap related to FFS of Fiberglass Reinforced Plastic (FRP). The changes in N285.5-13 relative to N285.5-08 that are applicable to FRP material that is used at Pickering NGS have been assessed for fitness for service to 2024 [NA44-PLAN-34220-00002 R001, *Life Cycle and Aging Management Program Plan for Fiberglass-Reinforced Plastic Components in the*

SECTION 3 - GI-17 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

Pickering NGS Vacuum Building, October 2012], [P-CALC-34000-00006 R000, *Material Properties for FRP Components Used in Pickering Vacuum Building Based on 2015 Test Results*, March 2017].

SECTION 4 - GI-17 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								Overall Safety Significance Level
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	2	N/A	**2**	4	2	4	2	N/A	N/A	N/A	**2**	**2**

Rationale:

Addressing this Global Issue will ensure the fitness for service of Fiberglass-Reinforced Plastic Components in the Pickering NGS Vacuum Building for the operational life of the station. As discussed in Section 3 of this Global Issue, the fitness for service of these components is currently assured to 2024.

This Global Issue is assigned Safety Significance Level 2 for deterministic considerations with respect to Defence in Depth (E1) since ensuring ongoing fitness for service will avoid a reduction in margin of safety to public or station personnel. This is consistent with the definition of Safety Significance Level 2 in Table E1 in the PSR2 Basis Document. Safety Significance Levels (E2) is considered not applicable because this Global Issue can have an impact on nuclear safety, whereas E2 primarily relates to issues without a direct impact on nuclear safety. Hence, the overall Safety Significance Level is 2 for deterministic considerations.

With respect to probabilistic considerations, this Global Issue is conservatively assigned Safety Significance Level 2 with respect to Table F2 in the PSR2 Basis Document, since resolution of this issue will exclude a possible reduction in the reliability of System Important to Safety for any probable relevant event (third row of Table F2 in the PSR2 Basis Document). Similarly, the Safety Significance Level with respect to Plant Operability (F4) is also 2, since an extended period of plant shutdown as a result of fitness for service issues is expected to have a probability less than 0.1.

Safety Significance Level 4 is assigned to Reactor Safety - Core Damage Frequency (F1) and Public Radiation Safety (F3) since this Global Issue will have an insignificant impact on these probabilistic considerations.

This Global Issue has no impact on the other probabilistic considerations, i.e., Occupational Radiation

SECTION 4 - GI-17 PRIORITY DETERMINATION
Safety (F5), Emergency Preparedness (F6) and Environment (F7). Therefore, these probabilistic considerations are not applicable. In summary, both deterministic and probabilistic considerations dictate that this Global Issue has Safety Significance Level 2. As noted, the fitness for service of these components has been demonstrated to 2024.

SECTION 5 - GI-17 RESOLUTION PLAN	
GI-17-NFA1	The adequacy of the FRP material used at Pickering NGS has been assessed for fitness for service to 2024. (SF4-10)

B.18. GI-18 N287.7 - In-Service Examination and Testing Requirements for Concrete Containment Structures

SECTION 1 - GI-18 GLOBAL ISSUE SUMMARY					
The goal of GI-18, N287.7 – In-Service Examination and Testing Requirements for Concrete Containment Structures, is to confirm that the requirements of CSA N287.7 are satisfied for the extended operating period. This Global Issue comprises one item assessed as requiring No Further Action and one Acceptable Deviation, addressing two SF4 code review Gaps [CSA N287.7]. GI-18 is Safety Significance Level 4 based on deterministic defence-in-depth considerations. This Global Issue addresses two specific issues related to CSA N287.7. The first relates to the accuracy and repeatability capability of standard available commercial sensors in testing equipment. OPG will continue to request concessions pending resolution of this issue through the CSA committee that is working to update CSA N287.7 with respect to this issue. The second issue relates to compliance with the standard for a sealant used in repairs. OPG has addressed this issue and the related Action Item 2013-8-4515 has been closed by the CNSC [P-CORR-00531-04787]. The proposed Resolution Plan assesses the issue regarding the accuracy and repeatability requirements in testing equipment as an Acceptable Deviation because of its very low Safety Significance. The issue regarding the Dow Corning sealant is assessed as requiring No Further Action because it has been addressed to the satisfaction of the CNSC.					
Safety Significance Level:	4	**Category:**	Analytical, Programmatic	**Reassessment Beyond 2024:**	N

SECTION 2 - GI-18 ASSOCIATED GAPS	
SF4-11	N287.7-08 clause 7.11.2 Table 1 involving non-compliance with accuracy and repeatability requirements for dew point temperature was a Gap for Darlington. No evidence can be found that this has been addressed for Pickering NGS. This is therefore a Gap for Pickering PSR2. ***Code Review N287.7-08*** **Associated Resolutions:** GI-18-AD1
SF4-12	OPG initiated a Regulatory Management action to provide the CNSC with the latest Dow Corning 995 material test report in response to an Action Notice raised in the CNSC Type II Inspection. The work is currently in progress. Therefore, this is a Gap for Pickering PSR2. ***Code Review N287.7-08*** • **Note – Per GI-18-NFA1, this Gap has been fully addressed.** **Associated Resolutions:** GI-18-NFA1

SECTION 3 - GI-18 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

This Global Issue contains two SF4 Gaps related to in-service examination and testing requirements for Concrete Containment Structures.

The Gap SF4-11 is related to N287.7-08 clause 7.11.2 on the accuracy requirement for dew point temperature measurement and repeatability requirement for pressure transmitter measurements for leakage rate testing. Current instrumentation data does not meet the accuracy requirement for dew point temperature of ± 1°C and for pressure transmitter measurement repeatability requirement of ± 0.001% of full scale. Currently industry can meet ± 2°C for dew point temperature accuracy, and ± 0.05% full scale for pressure transmitter measurement repeatability.

As per Action Notice 3 (AN3) of CNSC Action Item 2013-8-4515, OPG committed to develop and implement corrective actions to address the repairs that were performed with the Dow Corning sealant in order to become compliant with sub-section 6.6.1 of the Periodic Inspection Program for the Reactor Building [NK30-PIP-03643.2-00001-R003, OPG Plan, *Pickering Nuclear GSB – Reactor Building Periodic Inspection Program*, February 2014]. Based on OPG's AN3 update and request for closure [NK30-CORR-00531-07245, OPG Correspondence, *CNSC Action Item 2013-8-4515, AN3 Update and Request for Closure – Pickering Units 5 to 8: CNSC Type II Compliance Inspection Report, # PRPD-2013-182*, June 29, 2016], CNSC staff confirmed that OPG meets the closure criteria for (AN3) and has closed the Action Item 2013-8-4515 [P-CORR-00531-04787, CNSC Correspondence, e-Doc 5034577, *Pickering NGS: Closure of CNSC Action Item 2013-8-4515, Type II Compliance Inspection – Implementation of CSA N285.4, N285.5 and N287.7*, July 7, 2016].

SECTION 4 - GI-18 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								Overall Safety Significance Level
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	4	N/A	**4**	N/A	N/A	N/A	N/A	N/A	N/A	N/A	**N/A**	**4**

Rationale:

This Global Issue is related to in-service examination and testing for Concrete Containment Structures.

Regarding deterministic considerations, this Global Issue has Safety Significance Level 4 with respect

SECTION 4 - GI-18 PRIORITY DETERMINATION
to Defence in Depth (E1). This is because, as discussed in Section 3 of this Global Issue, the gap is against a requirement for accuracy and repeatability of measurements that cannot be met. CNSC has accepted applicable concessions for the issue, and the CSA committee is working to update CSA N287.7. E2 Safety Significance Levels is considered not applicable. Therefore, Safety Significance Level 4 is selected for deterministic considerations. This Global Issue has no direct impact on probabilistic considerations, i.e., Core Damage Frequency (F1), Defence in Depth (F2), Public Radiation Safety (F3), Plant Operability (F4), Occupational Radiation Safety (F5), Emergency Preparedness (F6) or Environment (F7). Therefore, probabilistic considerations are not applicable. In summary, the overall Safety Significance Level is 4.

SECTION 5 - GI-18 RESOLUTION PLAN	
GI-18-AD1	A CSA committee is working to update CSA N287.7 to account for accuracy and repeatability capability of standard available commercial sensors in testing equipment. OPG is actively participating on this committee. Until such time as N287.7 is revised, OPG will continue to request appropriate concessions from the CNSC when leak rate testing is required. Examples of concession approvals are [NK30-CORR-00531-07225, *Pickering NGS: Concession Request for Instrumentation Related to Unit 8 Containment Leak Rate Testing*, e-Doc 4970098, April 2016] and [NA44-CORR-00531-07599, *Pickering NGS: Concession Request for Instrumentation Related to Unit 4 Containment Leak Rate Testing*, e-Doc 4937932, February 2016]. As per the very low safety significance (Safety Significance Level 4), this is assessed as an Acceptable Deviation. (SF4-11).
GI-18-NFA1	The Action Notice (AN3) of Action Item 2013-8-4515 has been addressed to the satisfaction of the CNSC [P-CORR-00531-04787, CNSC Correspondence e-Doc 5034577, *Pickering NGS: Closure of CNSC Action Item 2013-8-4515, Type II Compliance Inspection – Implementation of CSA N285.4, N285.5 and N287.7*, July 7, 2016]. No further action is required. (SF4-12)

B.19. GI-19 FFS of Containment for the Extended Operating Period

SECTION 1 - GI-19 GLOBAL ISSUE SUMMARY					
The goal of GI-19, FFS of Containment for the Extended Operating Period, is to ensure that the safety-significant civil structures of Containment remain fit for service for the extended operating period. This Global Issue comprises one proposed Resolution Statement and one item assessed as requiring No Further Action, addressing 3 Gaps: one SF4 code review Gap [CSA N287.7-08], and two COP Review Gaps. GI-19 is Safety Significance Level 2 based on deterministic and probabilistic defence-in-depth considerations. Containment building structures are covered by Administrative Requirements for In Service Inspection and Testing for Concrete Containment Structures [N-PROC-MA-0066 R005]. The Periodic Inspection Programs (PIPs) cover the Vacuum Building and the Pressure Relief Duct, the Vacuum Building Post Tensioning Rods and the Reactor Buildings. The final report for the PIP inspection program incorporates all results, and any repairs and recommendations, arising from the PIP inspections to ensure fitness for service of the Reactor Buildings, Pressure Relief Duct and Vacuum Building structures. OPG's path forward with regard to demonstrating whether the foundation steel H-piles at the Pickering site will withstand their design loads for all civil structures that they support for operation beyond 2020 was communicated to the CNSC [P-CORR-00531-04896]. The proposed Resolution Plan comprises activities to demonstrate the FFS of the foundation steel H-piles at the Pickering site for the extended operating period. The remaining issues are assessed as requiring No Further Action because the Containment PIPs are in place to cover the extended operating period. Completion of the proposed Resolution Plan will support and strengthen Level 1 defence-in-depth for the extended operating period.					
Safety Significance Level:	2	**Category:**	Analytical	**Reassessment Beyond 2024:**	Y

SECTION 2 - GI-19 ASSOCIATED GAPS	
SF4-13	Actions #31, #32, and #33 from the Pickering Units 5-8 Continued Operations Plan are related to N287.7 and although complete, need to be reassessed for Pickering operation past 2020. (COP Action #31 involved submission of Periodic Inspection Plans and Life Cycle Management Plans for a number of safety-significant civil structures. COP Action #32 involved submission of Aging Management Plans for Concrete Containment Structures to the CNSC for acceptance. COP Action #33 involved revising the Reactor Building Periodic Inspection Plan and submitting to the CNSC for acceptance.) ***Code Review N287.7-08*** Note: Actions #32 and #33 are covered under this Global Issue. **Associated Resolutions:** GI-19-NFA1

SECTION 2 - GI-19 ASSOCIATED GAPS	
COP-16	Reassessment of the Vacuum Building Outage schedule and basis for maintaining a fully serviceable Containment boundary has not been completed for extended operation and until Negative Pressure Containment can be demonstrated to no longer be required. ***Pickering PSR2 Gap COP-16*** • **Note – Per GI-19-NFA1, this Gap has been fully addressed.** **Associated Resolutions:** GI-19-NFA1
COP-25	An assessment of margin to operate all the Pickering Reactor Building foundations has not been completed for the period of extended operation and until Reactor Building integrity can be demonstrated to no longer be required. This issue applies also to the Vacuum Building and Pressure Relief Duct for the extended operation period and for the period until the Negative Pressure Containment System integrity can be demonstrated to no longer be required. ***Pickering PSR2 Gap COP-25*** **Associated Resolutions:** GI-19-RS1

SECTION 3 - GI-19 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

This Global Issue contains three Gaps (one SF4 Gap and two COP Review Gaps) that are related to fitness for service of Containment.

Non-Containment safety-related civil structures are covered in GI-43.

SF4-13 includes three actions (#31, #32, and #33) from the Pickering Units 5-8 COP [NK30-PLAN-00531-00001 R005, OPG Plan, *Pickering 5-8 Continued Operations Plan*, November 2015]. Action #31 states “Include the periodic inspection programs and LCMPs for the safety-significant civil structures that are under the scope of CSA N291-08, but not covered by the N287.7 standard”. This applies to non-Containment structures and accordingly is not dealt with in this Global Issue but is considered in GI-43. Action #32 involved submission of Aging Management Plans for Concrete Containment Structures to the CNSC for acceptance. Action #33 involved revising the Reactor Building Periodic Inspection Plan and submitting to the CNSC for acceptance. Accordingly, both Actions #32 and #33 pertain to issues that are within the scope of this Global Issue (GI-19).

Containment building structures are covered by Periodic Inspection Programs per [N-PROC-MA-0066 R005, OPG Nuclear Procedure, *Administrative Requirements for In Service Inspection and Testing for Concrete Containment Structure*, April 24, 2014].

NA44-PIP-03643.2-00002-R002 [NA44-PIP-03643.2-00002 R002, OPG Plan, *Pickering Nuclear GS – PRD & VB Periodic Inspection Program*, February 4, 2014] covers the Vacuum Building and the Pressure Relief Duct, NA44-PIP-03643.2-00003-R002 [NA44-PIP-03643.2-00003 R002, OPG Plan, *Pickering Nuclear GS – Vacuum Building Post Tensioning Rods Periodic Inspection Program*, April 30, 2014] is the PIP program for the Vacuum Building post tensioning rods, and the Periodic Inspection

SECTION 3 - GI-19 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

Plans for the Reactor Buildings are:

- [NA44-PIP-03643.2-00001 R003, *Pickering Nuclear GSA – Reactor Building Periodic Inspection Program*, February 2014]
- [NK30-PIP-03643.2-00001 R003, *Pickering Nuclear GSB – Reactor Building Periodic Inspection Program*, February 2014]

The in-service leakage rate test requirements of the Reactor Buildings and Pressure Relief Duct, in accordance with CSA N287.7, are specified in NA44-REP-34200-00017 [NA44-REP-34200-00017 R000, OPG Report, *Pickering NGS "A" Reactor Building and Pressure Relief Duct In-Service Leakage Rate Test Requirements in Accordance with CSA N287.7-08*, September 20, 2011], while those for the Vacuum Building are specified in NA44-REP-25100-00009-R000 [NA44-REP-25100-00009 R000, OPG Report, *Pickering NGS Vacuum Building In-Service Leakage Rate Test Requirements in Accordance with CSA N287.7-08*, November 2, 2012]. Leakage rate testing of the Reactor Buildings, the Pressure Relief Duct and the Vacuum Building is conducted on a schedule approved by the CNSC. The final report for the PIP inspection program incorporates all results, and any repairs and recommendations, arising from the PIP inspections to ensure fitness for service of the Reactor Buildings, Pressure Relief Duct and Vacuum Building structures.

SECTION 4 - GI-19 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								Overall Safety Significance Level
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	2	N/A	**2**	N/A	2	4	4	N/A	N/A	N/A	**2**	**2**

Rationale:

Resolution of this Global Issue will confirm the fitness for service of Containment for the operational life of the station.

Regarding deterministic considerations, Defence in Depth (E1) is assigned Safety Significance Level 2 since assuring fitness for service confirms the effectiveness of the Containment barrier (column 1, second row of Table E1 in the PSR2 Basis Document). Safety Significance Levels (E2) is assessed as not applicable because this Global Issue can have a direct nuclear safety impact, whereas E2 primarily relates to issues without a direct impact on nuclear safety. The overall Safety Significance Level for

SECTION 4 - GI-19 PRIORITY DETERMINATION

deterministic considerations is 2.

Regarding probabilistic considerations, Defence in Depth (F2) is assigned Safety Significance Level 2 since resolving this Global Issue precludes a challenge to the safety barrier (Containment) for events with initiating frequency less than 10^{-3}/y.

Safety Significance Level 4 is assigned for Public Radiation Safety (F3) on the basis that ensuring fitness for service of Containment precludes a potential change in public dose of less than 1 mSv for events with initiating frequency > 10^{-5}/y (row 4, column 4 in Table F3 in the PSR2 Basis Document). This Global Issue has insignificant impact on Plant Operability (F4), i.e., ~ 0.001 probability that the issue requires or leads to an extended period of plant shutdown. This Global Issue has no direct impact on the other probabilistic considerations, i.e., Core Damage Frequency (F1), Occupational Radiation Safety (F5), Emergency Preparedness (F6) and Environment (F7). Therefore, these probabilistic considerations are not applicable.

In summary, the overall Safety Significance Level is 2. Containment Periodic Inspection Plans are in place to cover the extended operating period, and work is under way to demonstrate the fitness for service of the foundation steel H-piles at the Pickering site for the extended operating period.

SECTION 5 - GI-19 RESOLUTION PLAN

GI-19-RS1	Demonstrate the FFS of the foundation steel H-piles for the Pickering A Reactor Building, Vacuum Building, and Pressure Relief Duct at the Pickering site for the extended operating period, as specified in [P-CORR-00531-04896, *Pickering NGS: Continued Operations Plan (COP) Actions F06 and I15-6B - Periodic Safety Review Reassessment for Operation Beyond 2020*, January 23, 2017] and in [P-CORR-00531-04973, *Pickering NGS: CNSC Staff Review of OPG's Reassessment of COP Actions for Consideration in the PSR2*, February 24, 2017]. (COP-25) This Resolution Statement also includes/addresses GI-43.
GI-19-NFA1	Containment PIPs were updated to comply with CSA N287.7-08, submitted to the CNSC for approval, and subsequently accepted by the CNSC in [P-CORR-00531-04265, *Pickering NGS: Acceptance of CSA N287.7-08 Periodic Inspection Program Documents, Action Item 2010-8-10 (RIB 2471), 2010-8-03 (RIB 2458), and 2010-4-18 (RIB 2478)*, June 2014]. The current PIPs include inspections required to be performed during a Vacuum Building Outage. The current Pickering planning basis is to perform the Vacuum Building Outage in 2020. There is no further PSR2 action required. (SF4-13 Actions #32 and #33) (COP-16)

B.20. GI-20 Governance Implementation / Effectiveness Issues

SECTION 1 - GI-20 GLOBAL ISSUE SUMMARY					
The goal of GI-20, Governance Implementation / Effectiveness Issues, is to ensure that Pickering Governance is effectively implemented for the extended operating period. This Global Issue comprises two Acceptable Deviations and one Cross Reference to GI-8 addressing two SF4 code review Gaps [REGDOC-2.6.3] and one SF4 Gap from Audits and Self-Assessment Reviews. GI-20 is Safety Significance Level 4 based on deterministic safety significance levels considerations. A plan to ensure the implementation of OPG's Aging Management Process, N-PROC-MP-0060, requirements for reviewing and updating the Condition Assessments, has been developed and provided to CNSC staff [P-CORR-00531-04805]. Completion of the actions addressing oversight and implementation of the Integrated Aging Management Program (IAMP) is being actively addressed outside of PSR2. The issue related to ensuring qualified individuals perform Aging Management engineering activities is closed and corrective actions are complete. The proposed Resolution Plan assesses the implementation of N-PROC-MP-0060 requirements and oversight and implementation of the IAMP as Acceptable Deviations because of their very low Safety Significance and because they are being addressed outside of PSR2. Completion of the Condition Assessments consistent with the revised Reactor Safety Criticality Codes is Cross Referenced to GI-8, Completion / Updating of the Condition Assessments.					
Safety Significance Level:	4	**Category:**	Programmatic, Analytical	**Reassessment Beyond 2024:**	N

SECTION 2 - GI-20 ASSOCIATED GAPS	
SF4-14	OPG is not compliant with N-PROC-MP-0060 Aging Management Process, Section 1.7 for "not reviewing and updating the Component Condition Assessments within the review cycle of the component, and when new information or feedback from the program was received." OPG has since revised these CAs, which are now valid until 2020. OPG has stated they will develop an implementation plan to prevent reoccurrence of: a) not reviewing and revising the CAs within the review cycle, and b) not updating the CAs when pertinent new information becomes available. OPG stated they will provide an update and a target implementation date on this action to the CNSC by October 30, 2016. This is a Gap for Pickering PSR2. ***Code Review CNSC REGDOC-2.6.3*** **Associated Resolutions:** GI-20-AD1
SF4-15	OPG is not compliant with N-PROC-MA-0077, *Critical Equipment Identification and Categorization*, Section 1.2 because "the Reactor Safety (RS) category code and rationale for critical components was not always accurate or consistently applied in the CCAs." OPG has stated they have since completed a review and update of the RS category code and rationale for a portion of the components to become fully compliant with N-PROC-MA-0077. However, OPG has stated that a review of the CAs will be conducted to ensure consistency

SECTION 2 - GI-20 ASSOCIATED GAPS	
	with the revised Reactor Safety codes and that an update will be provided to the CNSC by October 30, 2016. This is a Gap for Pickering PSR2. ***Code Review CNSC REGDOC-2.6.3*** **Associated Resolutions:** GI-8-RS1
SF4-17	Per Section B.1, Nuclear Oversight conducted a performance based audit (NO-2016-027) of the IAM Program in March 2016. The purpose of the audit was to determine whether IAM program requirements are being met and are effectively implemented to support safe and reliable operation. The audit concluded that the managed system controls are not fully effective and identified the following two open findings applicable to Pickering NGS which result in a PSR2 Gap (Note: These Gaps are closely related and are therefore identified as a single PSR2 Gap): • The IAM program requires that the interfacing programs affecting critical component condition should be comprehensive and sufficiently integrated to ensure critical information and assumptions used in completing Condition Assessments and Aging Management activities are valid and effective. However, a lack of integrated life cycle initiatives has been identified, which has the potential to impact equipment health. In addition, the program defines the requirements for program oversight and implementation. However, issues were identified in the completion of Condition Assessments and the execution of related recommendations due to ineffective oversight and implementation of the IAM program. SCR N-2016-08041 (AR#28189056) has been raised to address this issue and is expected to be completed by Q4 2017. This is a Gap for PSR2 since the SCR is not yet closed, and missing information in Condition Assessments and incomplete actions may lead to ineffective management of the aging equipment and impact the reliability of SSCs. • The IAM implementing procedure identifies the requirement for qualified individuals to perform Aging Management engineering activities such as preparing and reviewing Condition Assessments and screening reports. The audit identified that some Engineering Support Personnel performed engineering work independently while they were not qualified in the Training Information Management System. SCR P-2016-08008 (AR#28189028) has been raised to address this issue and is expected to be completed by Q3 2016. This is a Gap for PSR2 since the SCR is not yet closed, and unqualified staff performing work independently could impact the quality of Engineering work including Aging Management work activities. ***Audit and Self-Assessment Reviews*** **Associated Resolutions:** GI-20-AD2

SECTION 3 - GI-20 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

This Global Issue consolidates a total of three SF4 Gaps related to effective implementation of OPG governance.

SECTION 4 - GI-20 PRIORITY DETERMINATION												
Safety Significance of Global Issue	**Deterministic Considerations**			**Probabilistic Considerations**								**Overall Safety Significance Level**
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	N/A	4	**4**	N/A	N/A	N/A	N/A	N/A	N/A	N/A	**N/A**	**4**

Rationale:

This Global Issue includes minor issues related to the implementation or effectiveness of OPG governance.

Regarding deterministic considerations, Defence in Depth (E1) is not applicable since this Global Issue is not associated with a physical barrier. Safety Significance Levels (E2) is assigned Safety Significance Level 4 since resolution of this issue “may help identify areas that need more attention”, in this case the adequacy of governance implementation or its effectiveness. Therefore, Safety Significance Level 4 is selected for deterministic considerations.

This Global Issue has no direct impact on the probabilistic considerations, i.e., Core Damage Frequency (F1), Defence in Depth (F2), Public Radiation Safety (F3), Plant Operability (F4), Occupational Radiation Safety (F5), Emergency Preparedness (F6) and Environment (F7). Therefore, probabilistic considerations are not applicable.

In summary, the overall Safety Significance Level is 4. OPG is addressing this issue outside of PSR2.

SECTION 5 - GI-20 RESOLUTION PLAN	
GI-20-AD1	An implementation plan to ensure the implementation of N-PROC-MP-0060 requirements [N-PROC-MP-0060 R005B, OPG Nuclear Procedure, *Aging Management Process,* October 1, 2015] for reviewing and updating the Condition Assessments within the review cycle of the component, and when new information or feedback from the program is received, has been developed and provided to CNSC staff [P-CORR-00531-04805, OPG Correspondence, *CNSC Action Item 2015-48-7043, Action Notice AN1 and AN2 Update – Type II Compliance Inspection Report, Integrated Aging Management Program #PRPD- 2015-015*, October 28, 2016]. OPG is actively progressing the implementation plan activities outside of PSR2. As per the very low safety significance (Safety Significance

SECTION 5 - GI-20 RESOLUTION PLAN	
	Level 4), this is assessed as an Acceptable Deviation. (SF4-14)
GI-20-AD2	AR # 28189028 is closed and corrective actions are complete. Completion of the actions in AR # 28189056 addressing oversight and implementation of the IAM program is being actively addressed outside of PSR2. As per the very low safety significance (Safety Significance Level 4), this is assessed as an Acceptable Deviation. (SF4-17)
GI-20-XRF-GI-8-RS1	The Condition Assessments being completed are consistent with the revised Reactor Safety Criticality Codes. Complete the Condition Assessments. (SF4-15)

B.21. GI-21 FFS of the Deaerator and the Deaerator Storage Tank for the Extended Operating Period

SECTION 1 - GI-21 GLOBAL ISSUE SUMMARY					
The goal of GI-21, FFS of the Deaerator and the Deaerator Storage Tank for the Extended Operating Period, is to ensure that the Deaerator and the Deaerator Storage Tank remain fit for service for the extended operating period. This Global Issue comprises one Cross Reference to GI-8 addressing one COP Review Gap. GI-21 is Safety Significance Level 3 based on deterministic and probabilistic defence-in-depth considerations. The COP Review Gap action to respond to a CNSC inquiry regarding this issue has been completed, and is documented in the *Pickering 5-8 Continued Operations Plan* [NK30-PLAN-00531-00001-R004]. Fitness for service of the Deaerator and Deaerator Storage Tanks at Pickering NGS for the extended operating period is addressed in GI-8. Therefore, in the proposed Resolution Plan, completion of the Condition Assessment of the Deaerator and the Deaerator Storage Tank is Cross Referenced to GI-8, Completion / Updating of the Condition Assessments. Completion of the proposed Resolution Plan will support and strengthen Level 1 defence-in-depth for the extended operating period.					
Safety Significance Level:	3	**Category:**	Analytical	**Reassessment Beyond 2024:**	Y

SECTION 2 - GI-21 ASSOCIATED GAPS	
COP-23	A review for the full period of extended operation has not been performed to confirm that corrosion fatigue will not affect the welds in the deaerator and the deaerator storage tank for Pickering Units 1,4 and 5-8. ***Pickering PSR2 Gap COP-23*** **Associated Resolutions:** GI-8-RS1

SECTION 3 - GI-21 BACKGROUND INFORMATION AND RESOLUTION STRATEGY
This Global Issue contains one COP Review Gap related to FFS of the Deaerator and Deaerator Storage Tank. This COP Review Gap action was to provide a response to a specific CNSC inquiry. This action was completed. The issue is documented, with references, under Item #69 of Appendix A in [NK30-PLAN-00531-00001-R004, *Pickering 5-8 Continued Operations Plan*, December 2014]. This issue has not been assessed in the context of extended operation and for Pickering Units 1,4. Fitness for service of the Deaerator and Deaerator Storage Tanks at Pickering NGS for the extended operating period is addressed in GI-8.

SECTION 4 - GI-21 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								Overall Safety Significance Level
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	3	N/A	**3**	4	3	N/A	4	N/A	N/A	N/A	**3**	**3**

Rationale:

Resolution of this Global Issue will ensure fitness for service of the Deaerator and Deaerator Storage Tank for the extended operating period.

Regarding deterministic considerations, Defence in Depth (E1) is assigned Safety Significance Level 3 on the basis that the issue does not adversely impact the capability of safety provisions to terminate an anticipated process failure, and that resolving the issue will ensure the specific safety function of providing feed water from the Deaerator. Safety Significance Levels (E2) is assessed as not applicable because this Global Issue can have a direct nuclear safety impact, whereas E2 primarily relates to issues without a direct nuclear safety impact. The overall Safety Significance Level for deterministic considerations is 3.

Regarding probabilistic considerations, Defence in Depth (F2) is similarly assigned Safety Significance Level 3 on the basis that resolving the issue precludes a partial loss of safety margin for events with initiating frequency less than 10^{-3}/y.

This Global Issue has insignificant impact on Reactor Safety – Core Damage Frequency (F1) and Plant Operability (F4), and therefore Safety Significance Level 4 is assigned for these considerations. This Global Issue has no direct impact on Public Radiation Safety (F3), Occupational Radiation Safety (F5), Emergency Preparedness (F6) and Environment (F7), and therefore these considerations are not applicable.

In summary, the overall Safety Significance Level is 3. OPG has activities underway to address this Global Issue.

SECTION 5 - GI-21 RESOLUTION PLAN

GI-21-XRF-GI-8-	Complete the Condition Assessments for the Deaerator and Deaerator Storage

SECTION 5 - GI-21 RESOLUTION PLAN	
RS1	Tanks in all units. (COP-23)

B.22. GI-22 COP Actions Related to Aging Management from SFR4

SECTION 1 - GI-22 GLOBAL ISSUE SUMMARY					
The goal of GI-22, COP Actions Related to Aging Management, is to ensure that a group of 13 Pickering Units 5-8 COP actions related to aging management and one SF2 Additional Gap are addressed in PSR2. A number of these issues are also applicable to Pickering Units 1,4. This Global Issue comprises two Cross References to GI-1, one Cross Reference to GI-2, one Cross Reference to GI-3, two Cross References to GI-4, one Cross Reference to GI-8, one Cross Reference to GI-24, and one Cross Reference to GI-43, addressing one SF4 Gap and one SF2 Additional Gap. As the issues relate to demonstration of fitness for service of SSCs for the extended operating period, GI-22 is assigned Safety Significance Level 2. All of the proposed Resolution Statements for this Global Issue are cross-references to proposed Resolution Statements for other Global Issues (GI-1, GI-2, GI-3, GI-4, GI-8, GI-24 and GI-43). Completing the activities identified in the cross referenced proposed Resolution Statements will effectively address the COP actions and SF2 Additional Gap identified in this Global Issue. Completion of the cross referenced proposed Resolution Statements of GI-1, GI-2, GI-3, GI-4, GI-8 and GI-43 will support and strengthen Level 1 defence-in-depth for the extended operating period while those cross referencing GI-24 will support and strengthen Levels 2 and 3 defence-in-depth for the extended operating period.					
Safety Significance Level:	2	**Category:**	Programmatic, Analytical	**Reassessment Beyond 2024:**	Y

SECTION 2 - GI-22 ASSOCIATED GAPS	
SF4-18	Review of the Pickering Units 5-8 Continued Operations Plan [OPG Plan, NK30-PLAN-00531-00001 R005, *Pickering 5-8 Continued Operations Plan*, November 2015] identified the following closed gaps from the Pickering B ISR that will need to be revisited in the context of continued operation past 2020 for PSR2 Safety Factor 4. Appendix A Item 4 Demonstrate adequate safety margins to operate Pickering B units from a Heat Transport System aging perspective to Jan 31, 2021. The 2015 strategy update to CNSC staff provided a progress report on Heat Transport System (HTS) Aging Safety Analysis and related activities, and an updated revision of the HTS Aging Management Strategy for the period 2015-2020. This needs to be expanded to cover operation past 2020 for Pickering Units 1,4 and Units 5-8. Therefore, this is a gap for Pickering PSR2. Appendix A Items 10, 11, 12, 13 Develop a strategy to provide evidence that the Calandria Tube (CT) - Liquid Injection Shutdown System (LISS) nozzle gap will be maintained beyond 240,000 Effective Full Power Hours (EFPH) for all Pickering B units.

SECTION 2 - GI-22 ASSOCIATED GAPS	
	The strategy for CT - LISS nozzle gap preservation may apply beyond 2025, but this needs to be confirmed. Therefore, this is a gap for Pickering PSR2. <u>Appendix A Item 14</u> Develop R&D justification for extending Fuel Channel design life beyond 240,000 EFPH in the areas of hydrogen ingress, fracture toughness, spacer mobility and integrity. Actions were constrained by the shutdown date of 2020 assumed in the 2011 business plan. This needs to be expanded to cover operation past 2020 and is therefore a gap for Pickering PSR2. Note: An interim LCMP update for Major Components is documented in [P-CORR-01060-0587604 R000, OPG Memorandum, *Fitness for Service of Major Components*, March 29, 2016], which describes life cycle management strategies for Major Components to achieve extended operation to 2024. Strategies in this document may apply beyond 2024, but this needs to be confirmed as part of the resolution of this COP related PSR2 gap. <u>Appendix A Item 19</u> Update the NOP analysis for Pickering B. Actions were constrained by the shutdown date of 2020 assumed in the 2011 business plan. This is primarily relevant to Safety Factor 5 but is also of relevance to Safety Factor 4. This needs to be expanded to cover operation past 2020 for Pickering Units 1,4 and Units 5-8. Therefore, this is a gap for Pickering PSR2. <u>Appendix A Item 21</u> With respect to the Feeder LCMPs, clarify the impact of Fuel Channel axial elongation during operation beyond the Fuel Channel assumed design life of 210,000 EFPH on Feeder stress analysis and acceptable Feeder thickness. This was only addressed to 2025. This needs to be expanded to cover operation to 2028[33]. Therefore, this is a gap for Pickering PSR2. Note: An interim LCMP update for Major Components is documented in [P-CORR-01060-0587604 R000 OPG Memorandum, *Fitness for Service of Major Components*, March 29, 2016], which describes life cycle management strategies for Major Components to achieve extended operation to 2024. Strategies in this document may apply beyond 2024, but this needs to be confirmed as part of the resolution of this COP related PSR2 gap. <u>Appendix A Item 30</u> Include the Periodic Inspection Programs and LCMPs for the secondary side pressure retaining components and submit them for CNSC review. Although the action to submit PIPs and LCMPs for the secondary side pressure retaining components is complete, these documents will need to be extended to cover operation past 2020 for Pickering Units 1,4 and Units 5-8. Therefore, this is a gap for Pickering PSR2. <u>Appendix A Item 52</u> Perform Time Limiting Aging Analysis (TLAAs) and include such TLAAs in the LCMPs and in the CAs. OPG to provide commitment that TLAAs necessary to determine the actual conditions of components will be completed. Although the action is complete, this will need to be updated to cover operation past 2020 for

SECTION 2 - GI-22 ASSOCIATED GAPS	
	Pickering Units 1,4 and Units5-8. Therefore, this is a gap for Pickering PSR2. Appendix A Item 69 Include relevant information from COG JP 4271 Calandria and internals. Fitness for Life Extension Guidelines in N-PLAN-01060-10003 *Reactor Components and Structures Life Cycle Management Plan (LCMP)* and submit the LCMP to the CNSC in accordance with Pickering B PROL 08.20/2013 LC 1.2. Although the action is complete, this will need to be updated to cover operation past 2020 for Pickering Units 1,4 and Units 5-8. Therefore, this is a gap for Pickering PSR2. Note: An interim LCMP update for Major Components is documented in P-CORR-01060-0587604 R000, which describes life cycle management strategies for major components to achieve extended operation to 2024. Strategies in this document may apply beyond 2024, but this needs to be confirmed as part of the resolution of this COP related PSR2 gap. Appendix C Item 5 Update the Pickering B HTS aging model. This action is complete but needs to be reviewed to assess impact of operation past 2020 for Pickering Units 1,4 and Units 5-8. Therefore, this is a gap for Pickering PSR2. Appendix C Item 6 Update the Pickering B HTS aging management strategy. This action is complete but needs to be reviewed to assess impact of operation past 2020 for Pickering Units 1,4 and Units 5-8. Therefore, this is a gap for Pickering PSR2. ***COP Review in Support of Safety Factor 4*** **Associated Resolutions:** GI-1-RS1, GI-1-RS3, GI-2-RS1, GI-3-RS1, GI-4-RS1, GI-4-RS2, GI-8-RS1, GI-24-RS1, GI-43-RS3
SF2-AG2	An SF2 Gap related to Time Limiting Elements of Aging Analysis for passive long-lived SSCs for the period of extended operation was identified in [P-CORR-00531-05099, e-Doc 5295534, July 12, 2017]. **Associated Resolutions:** GI-1-RS3, GI-2-RS1, GI-3-RS1, GI-4-RS1, GI-8-RS1, GI-43-RS3

SECTION 3 - GI-22 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

This Global Issue contains one SF2 Additional Gap and one SF4 Gap related to COP Appendix A Actions #4, 10, 11, 12, 13, 14, 19, 21, 30, 52, 69, and Appendix C Items 5 and 6. A number of these issues are also applicable to Pickering Units 1,4. As shown in the Resolution Plan, all of the items identified in the SF4-18 Gap and the SF2-AG2 Additional Gap are addressed in other GIs.

SECTION 4 - GI-22 PRIORITY DETERMINATION												
Safety Significance of Global Issue	**Deterministic Considerations**			**Probabilistic Considerations**								**Overall Safety Significance Level**
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
												2

Rationale:

This Global Issue consists of gaps arising from reassessment of 13 COP actions for the extended operating period and one SF2 Additional Gap. These gaps are related to fitness for service and safety analysis for events impacted by aging. All of the Resolution Statements for this Global Issue are cross-references to Resolution Statements for other Global Issues (GI-1, GI-2, GI-3, GI-4, GI-8, GI-24 and GI-43). There are no unique Resolution Statements for this Global Issue. As the issues relate to demonstration of fitness for service of SSCs for the extended operating period, and as most fitness for service Global Issues have a Safety Significance Level of 2, the Safety Significance Level assigned to this Global Issue is also 2. OPG activities addressing the Global Issue are underway and they are detailed in GI-1, GI-2, GI-3, GI-4, GI-8, GI-24 and GI-43.

SECTION 5 - GI-22 RESOLUTION PLAN	
GI-22-XRF-GI-1-RS1	The R&D needed to support assurance of Fuel Channel Fitness for Service to 2024 is complete. Complete the additional R&D to support validation of the fracture toughness model. (SF4-18 – COP Appendix A Item #14)
GI-22-XRF-GI-1-RS3	Update the Fuel Channels LCMP [N-PLAN-01060-10002-R017, OPG Plan, *Fuel Channels Life-Cycle Management Plan*, October 2016] for Pickering Units 1,4 for the extended operating period. (SF4-18 – COP Appendix A Item #52) (SF2-AG2)
GI-22-XRF-GI-2-RS1	Update the Feeders LCMP [N-PLAN-01060-10001-R018, OPG Plan, *Feeders Life Cycle Management Plan*, October 2016], based on updated Fitness for Service assessment for Pickering Units 1,4 for the extended operating period. This is covered by GI-2. (SF4-18 – COP Appendix A Item #21 and #52) (SF2-AG2)
GI-22-XRF-GI-3-RS1	Update the Steam Generators LCMP [N-PLAN-33110-10009-R007, OPG Plan, *Steam Generators Life Cycle Management Plan*, October 2016] for Pickering Units 1,4 for the extended operating period. (SF4-18 – COP Appendix A Item #30 and

SECTION 5 - GI-22 RESOLUTION PLAN	
	#52) (SF2-AG2)
GI-22-XRF-GI-4-RS1	Update the Reactor Components and Structures LCMP [N-PLAN-01060-10003-R014, OPG Plan, *Reactor Components and Structures Life Cycle Management Plan*, October 2016] for Pickering Units 1,4 for the extended operating period. (SF4-18 – COP Appendix A Item #52 and #69) (SF2-AG2)
GI-22-XRF-GI-4-RS2	Perform measurements, as required, of CT-LISS nozzle gaps on Units 5-8 to refine the gap closure rates. Using this new measurement data, update analyses as required, to demonstrate Fitness for Service. Implement mitigation strategies if CT-LISS nozzle contact is predicted within the extended operating period. (SF4-18 – COP Appendix A Items #10, #11, #12, #13)
GI-22-XRF-GI-8-RS1	Complete and update Condition Assessments for secondary side pressure retaining components for the extended operating period. (SF4-18 – COP Appendix A Item #30) (SF4-18 – COP Appendix A Item #52) (SF2-AG2)
GI-22-XRF-GI-24-RS1	Update Heat Transport System aging analysis models and perform the required safety analysis of the events most impacted by aging (SBLOCA, LOF and Neutron Overpower (NOP)) to support extended operation as per the existing practices [N-CORR-00531-18427, OPG Correspondence, *Progress Report on OPG Heat Transport System Aging Safety Analysis*, February 24, 2017]. (SF4-18 – COP Appendix A Items #4, #19, Appendix C Items #5, #6)
GI-22-XRF-GI-43-RS3	Prepare Condition Assessments as appropriate for safety-significant civil structures for the extended operating period. Recommendations from these Condition Assessments will be tracked and reported along with those related to GI-8. This applies to non-Containment Safety-Related Civil Structures. (SF4-18 – COP Appendix A Item #52) (SF2-AG2)

B.23. GI-23 ASME N509-1980 and N510-1980 - Air Cleaning Systems

SECTION 1 - GI-23 GLOBAL ISSUE SUMMARY

The goal of GI-23, ASME N509-1980 and N510-1980 – Air Cleaning Systems, is to demonstrate conformance with the requirements of these standards that are applicable to PSR2. This Global Issue comprises one item requiring No Further Action which addresses one COP Review Gap that was identified to be specific to Pickering Units 1,4, since a review of these standards for these Units was not performed. Safety Significance Level 4 is selected for this Global Issue based on deterministic safety significance levels considerations.

The proposed Resolution Plan for this Global Issue is specified as No Further Action, based on the applicability of the closure of this COP item for Pickering Units 5-8. This is because the scope of the PSR2 review specifically relates to nuclear safety, hence the Filtered Air Discharge System (as part of Containment) is the only filtered air or air cleaning system that is within the scope of PSR2, and it was addressed in the Pickering Units 5-8 review.

Safety Significance Level:	4	**Category:**	Programmatic	**Reassessment Beyond 2024:**	N

SECTION 2 - GI-23 ASSOCIATED GAPS

COP-19	A review of Pickering Units 1,4 Non-SOE air cleaning systems against ANSI/ASME N509-1980 and N510-1980 has not been completed. ***Pickering PSR2 Gap COP-19*** **Associated Resolutions:** GI-23-NFA1

SECTION 3 - GI-23 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

This Global Issue contains one COP Review Gap related to ASME N509-1980 and N510-1980.

The issue is that the Pickering Units 1,4 non-SOE air cleaning systems have not been assessed against ANSI/ASME N509-1980 and N510-1980. Given the scope of the PSR2 review specifically relates to nuclear safety, the Filtered Air Discharge System (FADS) (as part of Containment) is the only filtered air or air cleaning system that is within the scope of PSR2.

For Pickering Units 5-8, a review was completed and COP action closure accepted by the CNSC (refer to page 18 of Appendix A of [NK30-PLAN-00531-00001 R005, OPG Plan, *Pickering 5-8 Continued Operations Plan*, November 2015]). The closure is based on the assessment presented on pages 4 & 5 of Appendix B of [P-LIST-09314-00001-R000, OPG List, *Pickering Consolidated End of Life Action Log*, December, 2012] that concluded that these standards were considered and followed in the design, installation, testing and operation of the FADS and other normal operating ventilation systems, and that Pickering 5-8 is compliant to a high degree to ASME N509/ N510. FADS, which is common

SECTION 3 - GI-23 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

for Pickering 1,4 and 5-8, has already been assessed as part of the Pickering 5-8 compliance review for N509 and N510.

In addition, OPG Report [N-REP-03480-0601454, *Code Review of CSA N288.3.4 for PN, PWMF, DWMF and WWMF High Efficiency Air Cleaning Assemblies*, January 2016], referenced in the PSR2 N288.3.4 code review [P-REP-03680-00021, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 8, 10, 12, 13, 14*, December 2016], did not identify any Gaps specifically related to FADS. The identified Gaps relate to generic test program design and documentation issues. Since FADS is a Containment sub-system and part of a Special Safety System, the filter system has explicit design basis requirements to ensure that the nuclear safety credits are satisfied [NK30-DR-34230-10001 R000, OPG System Design Requirements, *Filtered Air Discharge System*, March 2004]. These are documented in the SOE for Pickering 1,4 and Pickering 5-8, in the Containment Operational Safety Requirements (OSRs) [NA44-OSR-08131.02-00002 R003, OPG Operational Safety Requirements, *Pickering 1-4 Operational Safety Requirements: Negative Pressure Containment*, March 2015], [NK30-OSR-08131.02-00003 R004, OPG Operational Safety Requirements, *Pickering 5-8 Operational Safety Requirements: Negative Pressure Containment*, March 2015], and in the associated OSR Compliance Tables. The Compliance Tables identify the routine operational tests and periodic surveillances specified to provide assurance of filter availability and effectiveness.

SECTION 4 - GI-23 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								Overall Safety Significance Level
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	N/A	4	**4**	N/A	N/A	N/A	N/A	N/A	N/A	N/A	**N/A**	**4**

Rationale:

This Global Issue is related to Pickering Units 1,4 non-SOE air cleaning systems and ANSI/ASME N509-1980/N510-1980. As noted in Section 3 of this Global Issue, the scope of the PSR2 review specifically relates to nuclear safety, so FADS (as part of Containment) is the only filtered air or air cleaning system that is within the scope of PSR2, and closure of the original COP item 19 on this item for Pickering Units 5-8 is also applicable to Pickering Units 1,4. Therefore, no further action is required.

SECTION 4 - GI-23 PRIORITY DETERMINATION
Regarding deterministic considerations, this Global Issue is not associated with a physical nuclear safety barrier, so Defence in Depth (E1) is not applicable. Safety Significance Levels (E2) is assigned Safety Significance Level 4 since resolution of this issue “may help identify areas that need more attention”. Therefore, Safety Significance Level 4 is selected for deterministic considerations. Regarding probabilistic considerations, this Global Issue has no impact on Core Damage Frequency (F1), Defence in Depth (F2), Public Radiation Safety (F3), Plant Operability (F4), Occupational Radiation Safety (F5), Emergency Preparedness (F6) or the Environment (F7). Therefore, probabilistic considerations are not applicable. In summary, the overall Safety Significance Level is 4. No further action is required with respect to this Global Issue.

SECTION 5 - GI-23 RESOLUTION PLAN	
GI-23-NFA1	Given the scope of the PSR2 review specifically relates to nuclear safety, FADS (as part of Containment) is the only filtered air or air cleaning system that is within the scope of PSR2. Closure of the original COP item for Pickering Units 5-8 is thus also applicable to Pickering Units 1,4. No further action is required. (COP-19)

B.24. GI-24 Safety Analysis to Support the Extended Operating Period

SECTION 1 - GI-24 GLOBAL ISSUE SUMMARY					
The goal of GI-24, Safety Analysis to Support the Extended Operating Period, is to demonstrate the adequacy of the safety margins of the plant with aged conditions covering the extended operating period. This Global Issue comprises one proposed Resolution Statement addressing two Gaps: one SF5 review task Gap and one COP Additional Gap. Safety Significance Level 2 is selected for this Global Issue based on deterministic defence-in-depth and safety significance levels considerations, as well as probabilistic defence-in-depth considerations. The proposed Resolution Plan for this Global Issue comprises one proposed Resolution Statement to update the aging safety analysis model and perform the required safety analysis. Completion of the proposed Resolution Plan will support and strengthen Levels 2 and 3 defence-in-depth for the extended operating period.					
Safety Significance Level:	2	**Category:**	Analytical	**Reassessment Beyond 2024:**	Y

SECTION 2 - GI-24 ASSOCIATED GAPS	
SF5-1	Per Review Task #7, the current safety analysis for Pickering 1,4 and 5-8, demonstrates adequate Shutdown System trip coverage until 2017 and 2018 respectively for the Small Break Loss of Coolant Accident (SBLOCA) and Loss of Flow (LOF) scenarios. However, the impact of Heat Transport System (HTS) component aging on the SBLOCA, LOF and Slow Loss of Regulation (LOR) accident scenarios, will need to be further assessed in order to demonstrate adequate safety margins exist beyond 2020, and therefore a Gap exists for Pickering PSR2. It is noted in [N-CORR-00531-16444, *Progress Report on OPG Heat Transport System Aging Safety Analysis*, February 23, 2016], that work is currently underway to perform Safety Analysis to support the initiative to extend Pickering commercial operation to 2024, accounting for possible mitigation strategies of life-limiting aging mechanisms. Note a related Gap has been captured in the Aging Safety Factor Report (PSR2 Gap SF4-18). ***Review Task #7: Capabilities of the Plant in its Current State*** **Associated Resolutions:** GI-24-RS1
COP-AG1	• Demonstrate adequate safety margins to operate Pickering B units from an HTS ageing perspective to Jan 31, 2021. • Update the Pickering B HTS aging model with respect to reactor safety analysis. • Develop the Pickering B HTS Aging Management Strategy and ensure the strategy describes a path forward to 2020. • Update the Neutron Over Power (NOP) analysis for Pickering B. Analysis should

SECTION 2 - GI-24 ASSOCIATED GAPS

incorporate HTS aging effects and impact, if any, on trip set points.

Pickering PSR2 Gap formerly named COP-27

Associated Resolutions: GI-24-RS1

SECTION 3 - GI-24 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

This Global Issue contains two Gaps related to aging safety analysis.

The most recent progress report on Heat Transport System (HTS) aging safety analysis [N-CORR-00531-18427, OPG Correspondence, *Progress Report on OPG Heat Transport System Aging Safety Analysis*, February 24, 2017] indicates that the analyses for Pickering 1,4 SBLOCA and LOF are valid to January 31, 2019 while those for Pickering 5-8 are valid to June 30, 2019. The progress report [N-CORR-00531-18427, OPG Correspondence, *Progress Report on OPG Heat Transport System Aging Safety Analysis*, February 24, 2017] also indicates planning to perform safety analyses to support Pickering commercial operation to 2024, accounting for possible mitigation strategies of life-limiting aging mechanisms.

SECTION 4 - GI-24 PRIORITY DETERMINATION

	Deterministic Considerations			Probabilistic Considerations								
Safety Significance of Global Issue	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	**Overall Safety Significance Level**
	2	2	**2**	3	2	3	3	N/A	N/A	N/A	**2**	**2**

<u>Rationale:</u>

Resolution of this Global Issue will ensure that deterministic safety analysis accounts for aging effects for the operational life of the station, confirming that deterministic safety analysis acceptance criteria continue to be met when aging effects are accounted for. The ongoing safety analysis program updates the relevant accident analysis to account for aging effects, as well as any physical or operational enhancements that mitigate the effects of aging on safety margins. OPG will ensure that the Pickering safety analysis accurately accounts for the aging effects of the Heat Transport System.

SECTION 4 - GI-24 PRIORITY DETERMINATION

Regarding deterministic considerations, safety analysis does not directly affect the Defence in Depth (E1) barriers. However, the effectiveness of the safety functions which protect against DBAs needs to be confirmed for the extended operating period. Therefore, Safety Significance Level 2 is assigned to Defence in Depth (E1). For Safety Significance Levels (E2), updating the safety analysis to account for aging effects over the operational life of the station will confirm the adequacy of safety margins. Based on past experience, future updates of the safety analysis may identify some potential reduction in margin. Table E2 in the PSR2 Basis Document associates “some reduction in margin” with Safety Significance Level 2, which, therefore, is assigned. Consequently, the overall Safety Significance Level for deterministic considerations is 2.

Regarding probabilistic considerations, Defence in Depth (F2) is assigned a Safety Significance Level of 2. This is based on the third row of Table F2 in the PSR2 Basis Document, which is similar to some potential reduction in margin for deterministic considerations. For conservatism, the highest Safety Significance Level in this row, which is 2, is applied.

With respect to Plant Operability (F4), aging has the potential to impact operating limits, but such impacts are not expected to impact the complexity of plant operation. Therefore, addressing aging effects in the safety analysis places this Global Issue in row 3 of Table F4 of the PSR2 Basis Document, and for conservatism, the highest Safety Significance Level in this row, which is 3, is selected.

Core Damage Frequency (F1) and Public Radiation Safety (F3) are assigned Safety Significance Levels 3, since the initiating events primarily affected by aging have a low contribution to Core Damage Frequency. These initiating events are Small Loss of Coolant, Loss of Flow, and Loss of Regulation. For these events, the calculated doses to the public in the Safety Report are very low and well within regulatory limits, such that the impact of aging is not significant.

Finally, the impact of aging on safety analysis has no direct relation to Occupational Radiation Safety (F5), Emergency Preparedness (F6) or Environment (F7); the safety analysis to address aging does not affect these, so they are not applicable for this Global Issue.

In summary, both deterministic and probabilistic considerations dictate that this Global Issue has a Safety Significance Level of 2. OPG’s ongoing safety analysis program identifies potential aging impacts and addresses them in advance of potential adverse effects on safety margins.

SECTION 5 - GI-24 RESOLUTION PLAN

GI-24-RS1	Update Heat Transport System aging safety analysis models and perform the required safety analysis of the events most impacted by aging (SBLOCA, LOF and Neutron Overpower (NOP)) to support extended operation as per the existing practices [N-CORR-00531-18427, OPG Correspondence, *Progress Report on OPG Heat Transport System Aging Safety Analysis*, February 24, 2017]. (SF5-1) (COP-AG1) This Resolution Statement also includes/addresses GI-22.

B.25. GI-25 Category 3 CANDU Safety Issues

SECTION 1 - GI-25 GLOBAL ISSUE SUMMARY					
The goal of GI-25, Category 3 CANDU Safety Issues, is to resolve outstanding issues within the scope of PSR2 related to Category 3 CANDU Safety Issues. This Global Issue comprises two proposed Resolution Statements addressing four Gaps: one SF1 code review Gap [CSA N290.0-11], one SF5 review task Gap, one SF7 review task Gap, and one COP Review Gap. Safety Significance Level 3 is assigned for this Global Issue based on deterministic safety significance levels considerations. The proposed Resolution Plan primarily comprises completion of ongoing activities to facilitate the re-categorization of these issues to Category 2. Completion of the proposed Resolution Plan will support and strengthen Level 3 defence-in-depth for the extended operating period.					
Safety Significance Level:	3	**Category:**	Analytical	**Reassessment Beyond 2024:**	N

SECTION 2 - GI-25 ASSOCIATED GAPS	
SF5-2	Per Review Task #7, for the Large Break Loss of Coolant Accident (LBLOCA) CANDU Safety Issues (CSIs), while the development of the industry's proposed Composite Analytical Approach (CAA) is on-going, the licensing basis of existing CANDU reactors for the LBLOCA scenario will continue to be based on conservative safety analysis for which acceptance criteria are established. For the Category 3 non-LBLOCA CSI, the industry has applied to re-categorize the issue into a lower category based on analytical evidence and actions taken. Since four CSIs applicable to Pickering NGS (three LBLOCA/one non-LBLOCA) are currently in Category 3 and are undergoing further assessment in order to re-classify into a lower category and address operation past 2020, a Gap exists for Pickering PSR2. Note, the 3 LBLOCAs CSIs are also captured as a Gap in the PSR2 Continued Operations Plan (COP) Review Report (PSR2 Gap COP-20) as they relate to Pickering B Integrated Implementation Plan (IIP) Item I09. The 1 non-LBLOCA CSI is also identified as a Gap in the Hazards Analysis Safety Factor Report (PSR2 Gap SF7-1) as it relates to pipe whip. ***Review Task #7: Capabilities of the Plant in its Current State*** **Associated Resolutions:** GI-25-RS1, GI-25-RS2
SF7-1	To address CNSC safety issue CSI-IH 6, the potential for, and possible impacts of, high-energy piping failures in Pickering Units 1,4 must be assessed. This assessment is currently underway as per [OPG Report, P-REP-04960-00001 R002, *Methodology of High-Energy Line Break Assessment for Piping Inside the Pickering Reactor Buildings*, June 14, 2016] and [OPG Correspondence, N-CORR-00531-18288 R000, *Re-Categorization Request for CANDU Safety Issue IH6 for Pickering NGS 5-8 and Status for Pickering NGS 1-4*, December 2016]. Therefore, this has been identified as a PSR2 Gap. ***Review Task #1: Internal and External Hazards in Deterministic and Probabilistic***

SECTION 2 - GI-25 ASSOCIATED GAPS	
	Analyses **Associated Resolutions:** GI-25-RS2
SF1-9	Clause 4.13 of N290.0-11 identifies requirements to address dynamic piping effects. OPG is currently in the process of completing the High Energy Line Break Assessment (HELBA) for Pickering NGS. Preliminary results show that there would be no consequential damage caused by the rupture of high energy pipes inside Containment to safety related equipment, beyond that already accounted for in the Safety Reports. The final HELBA reports for both Pickering Units 5-8 have been completed, while Pickering Units 1,4 are expected to be completed in 2017. Since this work has not been completed for Pickering 1,4, this is identified as a PSR2 Gap. ***Code Review N290.0-11*** **Associated Resolutions:** GI-25-RS2
COP-20	Three LBLOCA CANDU Safety Issues that are applicable to Pickering NGS remain in Category 3 and have not been fully reassessed in order to re-classify into a lower risk category and cover operation past 2020. ***Pickering PSR2 Gap COP-20*** **Associated Resolutions:** GI-25-RS1

SECTION 3 - GI-25 BACKGROUND INFORMATION AND RESOLUTION STRATEGY
This Global Issue contains four Gaps (one SF1 Gap, one SF5 Gap, one SF7 Gap and one COP Review Gap) related to the remaining Category 3 CANDU Safety Issues (CSIs). These gaps are related to LBLOCA CSIs and CSI-IH6 on high energy piping.

SECTION 4 - GI-25 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								Overall Safety Significance Level
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	4	3	**3**	N/A	N/A	N/A	N/A	N/A	N/A	N/A	**N/A**	**3**

Rationale:

Resolution of this Global Issue will facilitate the reclassification of Category 3 CANDU Safety Issues, namely the CANDU Safety Issues related to Large Break Loss of Coolant Accidents and the CANDU Safety Issue related to high energy piping (IH6). Given the recent progress in addressing the findings of CNSC staff reviews, it is expected that the remaining Category 3 CSIs will be re-categorized to Category 2 (lower significance).

Regarding deterministic considerations, this Global Issue is not directly related to Defence in Depth (E1). However, although no safety barrier is directly impacted, completing the closure criteria for these CANDU Safety Issues will facilitate re-categorization of these CANDU Safety Issues to Category 2. Therefore Safety Significance Level 4 is assigned to Defence in Depth (E1). Safety Significance Levels (E2) is assigned Safety Significance Level 3 since this issue is considered not significant by itself and has been supported by analytical evidence as discussed in Section 5 of this Global Issue. Accordingly, the overall Safety Significance Level for deterministic considerations is 3.

This Global Issue has no direct impact on the probabilistic considerations, i.e., Core Damage Frequency (F1), Defence in Depth (F2), Public Radiation Safety (F3), Plant Operability (F4), Occupational Radiation Safety (F5), Emergency Preparedness (F6) and Environment (F7). Therefore, these probabilistic considerations are not applicable.

In summary, the overall Safety Significance Level is 3. OPG is progressing the required work to complete the re-categorization of IH6 and is working with the industry on completing the re-categorization of the CANDU Safety Issue related to Large Break Loss of Coolant Accidents.

SECTION 5 - GI-25 RESOLUTION PLAN

GI-25-RS1	Complete the re-categorization of the Large Break LOCA (LBLOCA) CANDU Safety Issues to Category 2. OPG submitted an update to CNSC staff on the resolution of the LBLOCA issue [N-CORR-00531-18022, OPG Correspondence, *Resolution of Large Break LOCA (LBLOCA) Safety Analysis Margin Issue*, April

SECTION 5 - GI-25 RESOLUTION PLAN	
	25, 2016]. An OPG update on the status of CSIs and their resolution is submitted to the CNSC annually, the latest being [N-CORR-00531-18052, *Progress Update On Category 3 CANDU Safety Issues - Implementation of Risk Control Measures*, June 15, 2016]. Given the recent progress by industry in addressing the findings of CNSC staff reviews, it is expected that the remaining Category 3 CSIs will be re-categorized to Category 2 in 2017. OPG is actively progressing this work. (SF5-2) (COP-20)
GI-25-RS2	Complete the re-categorization of CANDU Safety Issue CSI-IH6 for Pickering to Category 2. Complete the assessment of the layout of high-energy piping and Safety-Related Systems inside of the Reactor Buildings of Pickering Units 1 and 4 as per [P-REP-04960-00001 R002, OPG Report, *Methodology of High-Energy Line Break Assessment for Piping Inside the Pickering Reactor Buildings*, June 14, 2016]. For Pickering Units 5-8, the assessment is complete [N-CORR-00531-18052, OPG Correspondence, *Progress Update on Category 3 CANDU Safety Issues – Implementation of Risk Control Measures*, June 15, 2016] and a request for re-categorisation has been made [N-CORR-00531-18288, OPG Correspondence, *Re-Categorization Request for CANDU Safety Issue IH6 for Pickering NGS 5-8 and Status for Pickering NGS 1-4*, December 5, 2016]. For Pickering 1,4, a re-categorization request is planned for June 2018 [N-CORR-00531-18618, OPG Correspondence, *Progress Update on Category 3 CANDU Safety Issues – Implementation of Risk Control Measures*, June 23, 2017]. OPG is actively progressing this work. (SF5-2) (SF7-1) (SF1-9)

B.26. GI-26 Emergency Response Projection Software

SECTION 1 - GI-26 GLOBAL ISSUE SUMMARY					
The goal of GI-26, Emergency Response Projection Software, is to enhance the capability of the Emergency Response Projection (ERP) software. This Global Issue comprises one proposed Resolution Statement addressing one SF13 additional review finding Gap. Safety Significance Level 3 is assigned for this Global Issue based on deterministic defence-in-depth and safety significance levels considerations, as well as probabilistic emergency preparedness considerations. The proposed Resolution Plan comprises completion of enhancements to the ERP software capability. Completion of the proposed Resolution Plan will support and strengthen Level 5 defence-in-depth for the extended operating period.					
Safety Significance Level:	3	**Category:**	Analytical	**Reassessment Beyond 2024:**	N

SECTION 2 - GI-26 ASSOCIATED GAPS	
SF13-2	Darlington Gap IIP-OI 046 was identified to assess the ERP code for potential enhancements to address multi-unit Beyond Design Basis Event scenarios. The action assigned to this Gap (AR 28175339, TCD Q1 2017) is also applicable for Pickering NGS and is therefore a Gap for Pickering PSR2. ***Additional Review Findings*** **Associated Resolutions:** GI-26-RS1

SECTION 3 - GI-26 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

This Global Issue contains one SF13 Gap related to Emergency Response Projection software.

Darlington Gap IIP-OI 046 was identified to assess the Emergency Response Projection (ERP) code for potential enhancements to address multi-unit BDBA scenarios. The above assessment is complete and enhancements, which are also applicable to PNGS, are in progress [N-CORR-00531-18136, OPG Correspondence, *Status Update for Action Item 2016-OPG-7469: Implementation of Emergency Response Projection Computer Code Upgrades*, July 22, 2016].

OPG and Bruce Power are currently performing a project to update their emergency response projection tools. The project is adopting the Unified RASCAL Interface (URI) which is widely used by USA utilities. The project main tasks include:

- The development and deployment of a URI model for each of Darlington, Pickering and Bruce sites for use by the Emergency Response Organization (ERO) to project off-site consequences while an accident is in progress.

SECTION 3 - GI-26 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

- Development and delivery of training for the URI model.

SECTION 4 - GI-26 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								Overall Safety Significance Level
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	3	3	**3**	N/A	N/A	N/A	N/A	N/A	3	N/A	**3**	**3**

Rationale:

Resolution of this Global Issue will enhance the capability of the Emergency Response Projection software. This issue is being addressed and, as discussed in Section 5 of this Global Issue, an updated tool for emergency response projection is currently under development.

Regarding deterministic considerations, this Global Issue is related to enhancement of software and is not directly related to Defence in Depth (E1). However, the level of operational performance in responding to emergency situations would be impacted (improved) by addressing this issue and accordingly Defence in Depth (E1) is assigned Safety Significance Level 3. Similarly, Safety Significance Levels (E2) is Safety Significance Level 3 since this issue is not considered significant by itself. Accordingly, the overall Safety Significance Level for deterministic considerations is 3.

With respect to probabilistic considerations, Emergency Preparedness (F6) is assigned Safety Significance Level 3 since Emergency Preparedness procedures will be updated to incorporate the new updated software for emergency response projection.

This Global Issue has no direct impact on the other probabilistic considerations, i.e., Core Damage Frequency (F1), Defence in Depth (F2), Public Radiation Safety (F3), Plant Operability (F4), Occupational Radiation Safety (F5), and Environment (F7). Therefore, these probabilistic considerations are not applicable.

In summary, the overall Safety Significance Level is 3. OPG is progressing the required work to update the tool for emergency response projection.

SECTION 5 - GI-26 RESOLUTION PLAN	
GI-26-RS1	Complete the emergency response projection enhancements identified in OPG Correspondence [N-CORR-00531-18136, *Status Update for Action Item 2016-OPG-7469: Implementation of Emergency Response Projection Computer Code Upgrades*, July 22, 2016], which are currently underway. (SF13-2)

B.27. GI-27 Pickering 1,4 Probabilistic Safety Assessment

SECTION 1 - GI-27 GLOBAL ISSUE SUMMARY					
The goal of GI-27, Pickering 1,4 Probabilistic Safety Assessment (PSA), is to further enhance margins to the PSA Safety Goals. This Global Issue comprises two proposed Resolution Statements and one Cross Reference to GI-40 (Accident Management) which address six Gaps: two SF6 review task Gaps, one FAI Additional Gap, two SF6 Additional Gaps and one SF1 Additional Gap related to achieving the Administrative Safety Goals for Pickering 1,4 (Pickering 5-8 meet these goals). Safety Significance Level 3 is assigned for this Global Issue based on the probabilistic considerations of core damage frequency and public radiation safety. The proposed Resolution Plan comprises activities to further improve Pickering 1,4 Severe Core Damage Frequency and Large Release Frequency and to complete the planned Phase 2 Emergency Mitigating Equipment implementation. Completion of the proposed Resolution Plan will support and strengthen Level 3 and 4 defence-in-depth for the extended operating period.					
Safety Significance Level:	3	**Category:**	Engineering, Analytical	**Reassessment Beyond 2024:**	N

SECTION 2 - GI-27 ASSOCIATED GAPS	
SF6-1	When each hazard is considered individually by reactor, the time-average Severe Core Damage Frequency Probabilistic Safety Assessment PSA results for certain S-294 PSA elements, in Table 3, are above the OPG specified Administrative Safety Goal (10^{-5}/y), but within the OPG Safety Goal (10^{-4}/y). After incorporating the PSA updates that include certain Fukushima enhancements, only the Pickering Units 1,4 at-power fire risk remains above the OPG specified Administrative Safety Goal for Severe Core Damage Frequency. The PSA results described in this PSR2 report, however, do not incorporate certain proposed analysis enhancements and do not necessarily reflect the latest risk reduction activities currently underway at OPG. ***Review Task #5: Compliance with Safety Criteria*** **Associated Resolutions:** GI-27-RS1, GI-27-RS2, GI-40-RS1
SF6-2	When each hazard is considered individually by reactor, the time-average Large Release Frequency Probabilistic Safety Assessment (PSA) results for certain S-294 PSA elements, in Table 4, are above the OPG specified Administrative Safety Goal (10^{-6}/y) but within the OPG Safety Goal (10^{-5}/y). After incorporating the PSA updates that include certain Fukushima enhancements, only the Pickering Units 1,4 at-power internal events and at-power fire risks remain above the OPG specified Administrative Safety Goal for Large Release Frequency. The PSA results described in this PSR2 report, however, do not incorporate certain proposed analysis enhancements and do not necessarily reflect the latest risk reduction activities currently underway at OPG. ***Review Task #5: Compliance with Safety Criteria***

SECTION 2 - GI-27 ASSOCIATED GAPS	
	Associated Resolutions: GI-27-RS1, GI-27-RS2, GI-40-RS1
FAI-AG1	An FAI Gap related to the need for further safety enhancements to control long term radiological releases and, to the extent practicable unfiltered releases, following a BDBA, was identified in [P-CORR-00531-05003, e-Doc 5210355, March 22, 2017]. **Associated Resolutions:** GI-27-RS2
SF6-AG2	An SF6 Gap related to Further Safety Enhancements to Support Severe Accident Management, particularly hydrogen source term management, computational tools and venting effectiveness, was identified in [P-CORR-00531-05090, e-Doc 5289137, June 30, 2017]. **Associated Resolutions:** GI-27-RS2
SF6-AG3	An SF6 Gap related to station-specific investigations on Calandria vessel integrity during core degradation and in-vessel retention, and structural and stress analyses of the Pickering Calandria vessel under severe accident conditions, was identified in [P-CORR-00531-05090, e-Doc 5289137, June 30, 2017]. **Associated Resolutions:** GI-27-RS2
SF1-AG16	An SF1 Gap related to installation of additional means of Hydrogen Monitoring in Containment, in order to benchmark computational aids used in SAMG, was identified in [P-CORR-00531-05107, e-Doc 5305945, July 26, 2017]. **Associated Resolutions:** GI-27-RS2

SECTION 3 - GI-27 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

This Global Issue contains two SF6 Gaps and four Additional Gaps.

The following Pickering Units 1,4 Probabilistic Safety Assessments are identified as not achieving the OPG specified Administrative Safety Goals but complying with the OPG Safety Goals:

- At-Power Fire for Severe Core Damage Frequency
- At-Power Internal Events and At-Power Fire for Large Release Frequency

A review of the Fukushima Actions performed for PSR2 is documented in [P-REP-03680-00022, OPG Report, *Fukushima Action Item Review in Support of PNGS Periodic Safety Review 2 (PSR2)*, February 6, 2017] and [P-CORR-03680-0586481, OPG Memorandum, *Pickering NGS Extended Operations – Reassessment of Fukushima Action Item 1.3.1 & 1.3.2 – Containment Integrity*, November 30, 2016]. The report considered enhancements to safety that had already been made as well as enhancements that were underway or committed. These actions, discussed in the reports, will also address the six Gaps associated with this Global Issue, and are described in the two proposed Resolution Statements for this Global Issue and the Cross-Reference to a GI-40 proposed Resolution Statement.

SECTION 4 - GI-27 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								Overall Safety Significance Level
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	N/A	N/A	**N/A**	3	N/A	3	N/A	N/A	N/A	N/A	**3**	**3**

Rationale:

Resolution of this Global Issue will further improve the calculated Pickering 1,4 Severe Core Damage Frequency and Large Release Frequency.

This Global Issue pertains specifically to the Pickering NGS Probabilistic Safety Assessment. Therefore, deterministic considerations, Defence in Depth (E1) and Safety Significance Levels (E2), are not applicable to this Global Issue.

Regarding probabilistic considerations, this issue is assigned Safety Significance Level 3 with respect to Core Damage Frequency (F1) since the expected reduction in Core Damage Frequency as a result of addressing this issue is 10^{-7} to 10^{-6}/y. Public Radiation Safety (F3) is not directly related to this issue and limited to issues with potential change in dose for events with initiating frequencies > 10^{-5}/y. However, Safety Significance Level 3 is selected on the basis that the expected reduction in Large Release Frequency for Pickering 1,4 as a result of addressing this issue is in the order of 10^{-7}/y and consequently a corresponding risk reduction will be achieved with respect to public dose for severe accidents. This Global Issue has no direct impact on the other probabilistic considerations, i.e., Defence in Depth (F2), Plant Operability (F4), Occupational Radiation Dose (F5), Emergency Preparedness (F6), and Environment (F7). Therefore, these probabilistic considerations are not applicable.

In summary, the overall Safety Significance Level is 3. OPG is actively progressing this work in support of extended operation at Pickering NGS.

SECTION 5 - GI-27 RESOLUTION PLAN

GI-27-RS1	Complete actions from PSA improvement Plan [P-CORR-00531-04946, OPG Correspondence, *Pickering NGS: Risk Improvement Plan Update*, February 28, 2017]. OPG is progressing this work. (SF6-1) (SF6-2)

SECTION 5 - GI-27 RESOLUTION PLAN	
GI-27-RS2	Investigate and implement additional practicable design, operational and/or analytical enhancements to further improve Pickering 1,4 Severe Core Damage Frequency and Large Release Frequency (e.g., alternative emergency cooling water makeup). (SF6-1) (SF6-2) (FAI-AG1) (SF6-AG2) (SF6-AG3) (SF1-AG16) This Resolution Statement is related to GI-40.
GI-27-XRF-GI-40-RS1	Complete the planned Phase 2 EME implementation. This includes supplying cooling water, and power to essential loads via EME generators, to allow for operation of Air Cooling Units (ACUs) and Hydrogen Igniters [P-CORR-00531-04945, OPG Correspondence, *Pickering NGS – CNSC Action Item 2016-48-7470 Status Update on Emergency Mitigating Equipment and Telecommunications Projects*, February 16, 2017]. OPG is actively progressing this work in support of extended operation at Pickering NGS. (SF6-1) (SF6-2)

B.28. GI-28 Reactivity Worth of Control Absorbers

SECTION 1 - GI-28 GLOBAL ISSUE SUMMARY					
The goal of GI-28, Reactivity Worth of Control Absorbers, is to confirm that the Pickering 5-8 Control Absorbers will have adequate reactivity worth through the period of extended operation (Pickering 1,4 reactors do not have Control Absorbers). This Global issue comprises one item requiring No Further Action which addresses one COP Review Gap. Safety Significance Level 3 is assigned for this Global Issue based on both deterministic and probabilistic defence-in-depth considerations. The proposed Resolution Plan is No Further Action based on an assessment which demonstrates acceptable reactivity worth of the Control Absorbers over the extended operating period.					
Safety Significance Level:	3	**Category:**	Analytical	**Reassessment Beyond 2024:**	N

SECTION 2 - GI-28 ASSOCIATED GAPS	
COP-7	Demonstration of adequate margin to operate the Pickering 5-8 RRS Control Absorbers through the period of extended operation by performing engineering analysis of RRS control absorber cadmium to demonstrate adequate reactivity-worth, has not been completed. ***Pickering PSR2 Gap COP-7*** **Associated Resolutions:** GI-28-NFA1

SECTION 3 - GI-28 BACKGROUND INFORMATION AND RESOLUTION STRATEGY
This Global Issue contains one COP Review Gap related to reactivity worth of the Pickering Units 5-8 Reactor Regulating System (RRS) Control Absorbers. This gap identifies the need to demonstrate that the reactivity worth of the Control Absorber cadmium is adequate for the extended operating period.

SECTION 4 - GI-28 PRIORITY DETERMINATION

	Deterministic Considerations			Probabilistic Considerations								
Safety Significance of Global Issue	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	**Overall Safety Significance Level**
	3	N/A	**3**	4	3	N/A	4	N/A	N/A	N/A	**3**	**3**

Rationale:

Resolution of this Global Issue confirms that the reactivity worth of the Control Absorber cadmium is adequate for the extended operating period. As discussed in Section 5 of this Global Issue, a review of previous work has confirmed that the change in the reactivity worth of the Reactor Regulating System Control Absorbers over the extended operating period is acceptable. Nevertheless, the Safety Significance of this Global Issue is determined here in accordance with the Global Assessment process.

This Global Issue has Safety Significance Level 3 for deterministic considerations with respect to Defence in Depth (E1) since it is related to confirmation of the adequacy of the reactor control safety function, and the issue does not impair the capability of safety provisions to terminate an anticipated event. Safety Significance Levels (E2) is considered not applicable, since this Global Issue potentially impacts a physical nuclear safety barrier, whereas E2 primarily relates to issues that impact other objectives or are indirectly related to nuclear safety. Hence, the overall Safety Significance Level of 3 for deterministic considerations is dictated by the E1 categorization.

With respect to probabilistic considerations, Defence in Depth (F2) is assigned Safety Significance Level 3 on the basis that resolution of this issue prevents at most a partial loss of safety margin, so the bottom row of Table F2 in the PSR2 basis document is applicable. The most conservative Safety Significance Level for this row is selected, i.e., Safety Significance Level 3. This Global Issue has insignificant impact (less than 10^{-7}/y) on the Reactor Core Damage Frequency (F1) and this consideration is accordingly assigned Safety Significance Level 4. Similarly, this Global Issue has insignificant impact on Plant Operability (F4), i.e., ~ 0.001 probability that the issue requires or leads to an extended period of plant shutdown. Therefore, Safety Significance Level 4 is assigned to Plant Operability (F4).

This Global Issue has no direct impact on the other probabilistic considerations, i.e., Public Radiation Safety (F3), Occupational Radiation Safety (F5), Emergency Preparedness (F6) and Environment (F7). Therefore, these probabilistic considerations are not applicable. Based on this, the overall Safety Significance Level for probabilistic considerations is 3.

In summary, both deterministic and probabilistic considerations dictate that this Global Issue has

SECTION 4 - GI-28 PRIORITY DETERMINATION
Safety Significance Level 3. However, as discussed in Section 5 of this Global Issue, the change in the reactivity worth of the Reactor Regulating System Control Absorbers over the extended operating period has been confirmed to be acceptable and no further work is required on this issue.

SECTION 5 - GI-28 RESOLUTION PLAN	
GI-28-NFA1	The change in the reactivity worth of the RRS Control Absorbers over the extended operating period is acceptable [P-CORR-03680-0620822, *Pickering NGS PSR2 – Reactivity Worth of Pickering 5-8 Control Absorbers*, June 9, 2017]. The Control Absorbers are positioned outside of the reactor core for the large majority of the time. No further action is required. (COP-7)

B.29. GI-29 FFS of the Fuelling Machines and FM Bridge Ball Screws for the Extended Operating Period

SECTION 1 - GI-29 GLOBAL ISSUE SUMMARY					
The goal of GI-29, FFS of the Fuelling Machines and FM Bridge Ball Screws for the Extended Operating Period, is to ensure that the Fueling Machines remain fit for service for the extended operating period. This Global Issue comprises one Cross Reference to GI-8 which addresses two COP Review Gaps. Safety Significance Level 2 is assigned for this Global Issue based on both deterministic and probabilistic defence-in-depth and plant operability considerations. The proposed Resolution Plan consists of a cross-reference to one GI-8 (Completion/Updating of the Condition Assessment) proposed Resolution Statement to complete the Condition Assessments for the FM and FM Bridge Ball Screws. Completion of the proposed Resolution Plan will support and strengthen Level 1 defence-in-depth for the extended operating period.					
Safety Significance Level:	2	**Category:**	Analytical	**Reassessment Beyond 2024:**	Y

SECTION 2 - GI-29 ASSOCIATED GAPS	
COP-5	The fatigue analysis of the Pickering Fuelling Machine components has not been completed to demonstrate adequate margin to operate the Fuelling Machines for the extended operation period and for the period beyond where required for defueling activities. ***Pickering PSR2 Gap COP-5*** **Associated Resolutions:** GI-8-RS1
COP-6	Demonstration of adequate margin to operate the Pickering Fuelling Machine Bridge ball screws for the extended operation period and for the period beyond as required for defueling activities by performing engineering analysis of fatigue and aging of bridge ball screws, has not been completed. ***Pickering PSR2 Gap COP-6*** **Associated Resolutions:** GI-8-RS1

SECTION 3 - GI-29 BACKGROUND INFORMATION AND RESOLUTION STRATEGY
This Global Issue contains two COP Review Gaps related to the condition of Fuel Machines and Machine Bridge ball screws.

SECTION 4 - GI-29 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								Overall Safety Significance Level
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	2	N/A	**2**	4	2	4	2	4	N/A	N/A	**2**	**2**

Rationale:

Resolution of this Global Issue will ensure the fitness for service of Fuelling Machines for the extended operating period. As discussed in Section 3 of this Global Issue, Condition Assessments, including those for Fuelling Machines and Fuelling Machine Ball Screws, are in progress.

This Global Issue has Safety Significance Level 2 for deterministic considerations with respect to Defence in Depth (E1). This is because the fitness for service of Fuelling Machines is required to ensure that the issue does not impact the Primary Heat Transport pressure boundary. Safety Significance Levels (E2) is considered not applicable because this Global Issue potentially impacts a physical nuclear safety barrier, whereas E2 primarily relates to issues that impact other objectives or that are indirectly related to nuclear safety. Hence, the overall Safety Significance Level of 2 for deterministic considerations is dictated by the E1 categorization.

With respect to probabilistic considerations, the Fuelling Machines become part of Primary Heat Transport pressure boundary during fuelling. Resolving this Global Issue would preclude a potential reduction in the reliability of the Fuelling Machines for anticipated events (third row of Table F2 in the PSR2 Basis Document). The highest Safety Significance Level of 2 is assigned to Defence in Depth (F2) to conservatively cover initiating events of frequency of approximately 10^{-2}/y. Similarly, the Safety Significance Level with respect to Plant Operability (F4) is also 2, because an extended period of plant shutdown as a result of this issue is expected to have a probability less than 0.1. The issue has insignificant impact on Public Radiation Safety (F3) and Occupational Radiation Safety (F5), and therefore Safety Significance Level 4 is assigned for these considerations.

A Safety Significance Level of 4 is assigned to Core Damage Frequency (F1) since the Global Issue has insignificant impact on CDF. The other probabilistic considerations, i.e., Emergency Preparedness (F6), and Environment (F7) are not directly applicable to this Global Issue.

In summary, the overall Safety Significance Level is 2. OPG is progressing the required work to ensure the fitness for service of the Fuelling Machines for the extended operating period.

SECTION 5 - GI-29 RESOLUTION PLAN	
GI-29-XRF-GI-8-RS1	Complete the Condition Assessments for the Fuelling Machines and FM Ball Screws. (COP-5) (COP-6)

B.30. GI-30 Evaluation of Instantaneous Risk

SECTION 1 - GI-30 GLOBAL ISSUE SUMMARY					
The goal of GI-30, Evaluation of Instantaneous Risk, is to enhance Safety Goal monitoring. This Global Issue comprises one Acceptable Deviation addressing one SF6 review task Gap. Safety Significance Level 3 is assigned for this Global Issue based on deterministic defence-in-depth considerations and probabilistic defence-in-depth and core damage frequency considerations. The proposed Resolution Plan comprises an Acceptable Deviation because OPG continues to participate in an industry initiative on evaluation of instantaneous risk, and because of the low safety significance of this issue and the existence of other deterministic rules at Pickering NGS that provide assurance of defence-in-depth during temporary outage or maintenance alignments.					
Safety Significance Level:	3	**Category:**	Analytical	**Reassessment Beyond 2024:**	N

SECTION 2 - GI-30 ASSOCIATED GAPS	
SF6-3	Although the OPG instantaneous risk Safety Goals apply to Level 1 and Level 2 Probabilistic Safety Assessment (PSA) for all hazards, current practice at Pickering NGS is that instantaneous risk is only evaluated using the Level 1 at-power internal events and Level 1 outage internal events PSA models. This is similar to current practices at other Canadian utilities. There are other deterministic rules at Pickering NGS that provide assurance of defence-in-depth during temporary outage or maintenance alignments, and these deterministic considerations apply regardless of the potential hazard. Work is underway, via an industry initiative, to develop methodologies for assessment of instantaneous risk from other hazards included in the PSA. ***Review Task #5: Compliance with Safety Criteria*** **Associated Resolutions:** GI-30-AD1

SECTION 3 - GI-30 BACKGROUND INFORMATION AND RESOLUTION STRATEGY
This Global Issue contains one SF6 Gap related to the assessment of instantaneous risk. The current scope of OPG instantaneous risk evaluation involves use of Level 1 Internal Events At-Power and Outage PSA models.

SECTION 4 - GI-30 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								Overall Safety Significance Level
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	3	N/A	**3**	3	3	N/A	N/A	N/A	N/A	N/A	**3**	**3**

Rationale:

The current scope of OPG instantaneous risk evaluation involves use of Level 1 Internal Events At-Power and Outage Probabilistic Safety Assessment models. As discussed in Section 3 of this Global Issue, At-Power and Outage instantaneous risk Safety Goal monitoring is already proceduralized and implemented regularly for online maintenance and during each unit outage. OPG is continuing to participate in the industry initiative on evaluation of instantaneous risk.

Regarding deterministic considerations, this Global Issue is assigned Safety Significance Level 3 for Defence in Depth (E1), since the issue is related to potential enhancement in the level of operational performance. Safety Significance Levels (E2) is considered not applicable since this Global Issue can have a direct impact on nuclear safety, whereas E2 primarily relates to issues that are indirectly related to nuclear safety. Hence, the overall Safety Significance Level of 3 for deterministic considerations is dictated by the E1 categorization.

With respect to probabilistic considerations, Core Damage Frequency (F1) is assigned Safety Significance Level 3 on the basis that enhancements to the method for evaluating instantaneous risk will have a small impact on the overall core damage frequency, not exceeding a change greater than 10^{-6}/y. Defence in Depth (F2) is assigned Safety Significance Level 3 on the basis that the issue is related to the reliability/availability of the heat sink during online maintenance or a unit outage. The highest level is conservatively selected (the bottom row of Table F2 in the PSR2 Basis Document). This Global Issue has no direct impact on the other probabilistic considerations, i.e., Public Radiation Safety (F3), Plant Operability (F4), Occupational Radiation Safety (F5), Emergency Preparedness (F6) and Environment (F7). Hence, the overall Safety Significance Level of 3 for probabilistic considerations is dictated by the Core Damage Frequency (F1) and Defence in Depth (F2) categorizations.

In summary, the overall Safety Significance Level is 3. OPG is continuing to participate in the industry initiative on methods for evaluating instantaneous risk.

SECTION 5 - GI-30 RESOLUTION PLAN	
GI-30-AD1	An industry initiative is underway to develop methodology and processes for evaluation of instantaneous risk. The current instantaneous risk Safety Goal monitoring is developed using the At-Power and Outage Internal Events PSA models. At-Power and Outage instantaneous risk Safety Goal monitoring is already proceduralized and implemented regularly for online maintenance and during each unit outage. This is similar to current practices at other Canadian utilities. There are other deterministic rules at Pickering NGS that provide assurance of defence-in-depth during temporary outage or maintenance alignments, and these deterministic considerations apply regardless of the potential hazard. OPG is continuing to participate in the industry initiative on evaluation of instantaneous risk. Per the low safety significance (Safety Significance Level 3), this is assessed as an Acceptable Deviation. (SF6-3)

B.31. GI-31 Deterministic Safety Analysis

SECTION 1 - GI-31 GLOBAL ISSUE SUMMARY					
The goal of GI-31, Deterministic Safety Analysis, is to complete the implementation of [N-PLAN-03500-0500515 R003, *REGDOC-2.4.1 Implementation Plan*, May 2015] for CNSC Regulatory Document REGDOC-2.4.1, *Deterministic Safety Analysis*, using a graded approach as permitted by the REGDOC, and update the plan in the context of extended operation. This Global Issue comprises two proposed Resolution Statements, two Acceptable Deviations, one item requiring No Further Action and one cross-reference to GI-44, which address 12 Gaps: five SF1 Gaps related to requirements for Anticipated Operational Occurrences (AOOs) in design-related codes, three SF5 Gaps and one COP Review Gap related to the implementation of REGDOC-2.4.1 requirements, one COP Review Gap related to the Boiling Length Average Critical Heat Flux correlation for 28-element fuel and two Additional Gaps. GI-31 is Safety Significance Level 3 based on deterministic safety significance levels considerations. The proposed Resolution Plan for GI-31 primarily comprises activities to update the REGDOC-2.4.1 Implementation Plan to include consideration of the extended operating period. These activities address two of the SF5 Gaps and one of the COP Review Gaps. Two Acceptable Deviations address the SF1 Gaps, the remaining SF5 Gap and one of the SF5 Additional Gaps. The other SF5 Additional Gap is addressed by a Cross-Reference to GI-44. No Further Action is required to address the COP Review Gap on the critical heat flux correlation because this item has been closed by the CNSC. Completion of the proposed Resolution Plan will support and strengthen Levels 2 and 3 defence-in-depth for the extended operating period.					
Safety Significance Level:	3	**Category:**	Programmatic, Analytical	**Reassessment Beyond 2024:**	Y

SECTION 2 - GI-31 ASSOCIATED GAPS	
SF5-3	The REGDOC-2.4.1 Implementation Plan and associated gap assessments capture all gaps related to REGDOC-2.4.1 and incorporate a systematic selection of the scope of work to address the most pertinent gaps in accordance with the graded approach to upgrading existing analyses. REGDOC-2.4.1 compliant analysis activities and progress related to REGDOC-2.4.1 implementation in the Pickering Licence Conditions Handbooks are tracked according to the CNSC Compliance Verification Criteria. Since the implementation is in progress, this has been identified as a PSR2 gap for Pickering NGS REGDOC-2.4.1 compliance. ***Code Review CNSC REGDOC-2.4.1*** **Associated Resolutions:** GI-31-RS1
SF5-4	As described in the REGDOC-2.4.1 Implementation Plan: "Limited upgrades are proposed in the Pickering A and B Plan, which has been developed with consideration for demonstration of continued safe operation while accounting for the limited remaining operating life of the Pickering Units". The REGDOC-2.4.1 Implementation Plan for Pickering did not consider

SECTION 2 - GI-31 ASSOCIATED GAPS	
	operation past 2020 and therefore the need for review and update of the Implementation Plan in the context of operation of Pickering NGS beyond 2020 is identified as a PSR2 gap. This will be informed by the timeline of the Darlington REGDOC-2.4.1 Implementation Plan, and the limited additional years of Pickering NGS operation. ***Code Review CNSC REGDOC-2.4.1*** **Associated Resolutions:** GI-31-RS2
SF1-8	Clause 4.2 of N290.0-11 requires that Plant States be grouped into several categories, including Anticipated Operational Occurrences (AOOs). This is consistent with clauses of REGDOC-2.4.1 and REGDOC-2.5.2 related to identification and classification of initiating events. Since AOOs have not been identified and analyzed in the current Pickering Safety Reports, the requirements and credits attributed to the Special Safety Systems for AOOs, if any, cannot be readily ascertained. This issue has therefore been identified as a PSR2 gap. It is being addressed as part of REGDOC-2.4.1 implementation. ***Code Review N290.0-11*** **Associated Resolutions:** GI-31-AD1
COP-21	The updates to the Safety Reports are being conducted in accordance with the REGDOC-2.4.1 Implementation Plan (REGDOC-2.4.1 superseded CNSC RD-310). This plan did not consider operation beyond 2020, for Pickering Units 1,4 and Units 5-8. ***Pickering PSR2 gap COP-21*** **Associated Resolutions:** GI-31-RS2
SF5-6	Safety Report upgrades currently underway for Pickering as part of REGDOC-2.4.1 implementation for the period of 2017-2021 will utilize methods consistent with N288.2-14. The REGDOC-2.4.1 Implementation Plan update will consider the incremental implications of Pickering operation beyond 2020, including any considerations of N288.2 revisions. This issue has therefore been identified as a PSR2 gap. It is being addressed as part of REGDOC-2.4.1 implementation. ***Code Review N288.2-14*** **Associated Resolutions:** GI-31-AD2
COP-14	There are remaining issues from AI 201113-2297 follow-up related to providing a detailed assessment report of the uncertainty in the implementation and use of BLA CHF correlation for 28 element fuel. ***Pickering PSR2 Gap COP-14*** • **Note – Per GI-31-NFA1, this gap has been fully addressed.** **Associated Resolutions:** GI-31-NFA1
SF1-14	Clause 4.2 and Clause 5.19 of CSA N290.4-11 require the capability of the Reactor Regulating System (RRS) to be assessed to deal with the Anticipated Operational

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	Occurrences (AOOs), by preventing them from escalating into Design Basis Accidents (DBAs) that would require Shutdown System action. In general the setback function (and stepback in Pickering Units 5-8) addresses this requirement; however AOOs have not been identified and analyzed in the current Pickering Safety Reports. Therefore, this has been identified as a PSR2 gap. It is being addressed as part of REGDOC-2.4.1 implementation. Note: There are also additional clauses which refer to requirements of RRS during AOOs (Clauses 5.6.2, 5.19, 5.16.1); however, for convenience, all issues related to AOO requirements for RRS in N290.4-11 are captured under this one PSR2 gap. ***Code Review N290.4-11*** **Associated Resolutions:** GI-31-AD1
SF1-15	A gap exists for the Pickering Units 1,4 and 5-8 Instrument Air and Electrical Systems on clauses 7.1 and 7.4.2 of N290.5-06 (R2011) including Update No. 1 dealing with requirements for Anticipated Operational Occurrences (AOOs). These clauses introduce the requirement for components to be qualified to perform their required functions during normal operation and AOOs. Only the portion of this clause on AOOs is pertinent to nuclear safety. It is likely that AOOs, due to their nature, do not result in a challenge to the qualification of systems, including Instrument Air and Electrical systems. However, AOOs have not been identified and analyzed in the current Pickering Safety Reports. This issue has therefore been identified as a PSR2 gap. It is being addressed as part of REGDOC-2.4.1 implementation. ***Code Review N290.5-06*** **Associated Resolutions:** GI-31-AD1
SF1-16	The CSA N290.11-13 Clause 5.1.1 to 5.1.5 requirement for back-up heat sinks to mitigate the conditions following an Anticipated Operational Occurrence (AOO) is not specified in governance/procedures. Loss of a division of power, a single component failure, etc., which are likely to be in the set of AOOs, are accounted for in the specification of heat sinks. However, AOOs have not been identified and analyzed in the current Pickering Safety Reports. This issue is therefore a PSR2 gap. It is being addressed as part of REGDOC-2.4.1 implementation. ***Code Review N290.11-13*** **Associated Resolutions:** GI-31-AD1
SF1-18	Clause 5.6.1 of CSA N290.11-13 requires that the designed reliability for process heat sinks be consistent with Anticipated Operational Occurrence (AOO) frequency limits, such that an emergency heat sink does not need to be used for an AOO. AOOs have not been identified and analyzed in the current Pickering Safety Reports. This issue is therefore a PSR2 gap and is being addressed as part of REGDOC-2.4.1 implementation. ***Code Review N290.11-13*** **Associated Resolutions:** GI-31-AD1
SF1-	An SF1 Gap related to addressing analysis of Anticipated Operational Occurrences was

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AG14	identified in [P-CORR-00531-05107, e-Doc 5305945, July 26, 2017]. **Associated Resolutions:** GI-31-AD1
SF5-AG1	An SF5 Gap related to requirements for the Credit of Operator Action Times in REGDOC-2.4.1 and REGDOC-2.5.2, and justification for which operation action times are credited in safety analysis, was identified in [P-CORR-00531-05090, e-Doc 5289137, June 30, 2017]. **Associated Resolutions:** GI-44-AD8

SECTION 3 - GI-31 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

This Global Issue contains 12 Gaps (five SF1 gaps, three SF5 gaps and 2 COP Review Gaps, one SF1 Additional Gap and one SF5 Additional Gap) related to Deterministic Safety Analysis and REGDOC-2.4.1 implementation. This Global Issue includes a number of gaps arising from AOOs requirements in design-related codes. These gaps have been consolidated into this Global Issue as AOOs have not been identified and analyzed in the current Pickering Safety Reports. Consideration of the identification and analysis of AOOs is addressed under REGDOC-2.4.1 implementation.

Compliance with REGDOC-2.4.1 is currently a licensing requirement for Pickering NGS (per PROL 48.03/2018) as indicated in Appendix C.2 of the R05 Pickering Licence Conditions Handbook. REGDOC-2.4.1 allows a graded approach to upgrading existing analyses.

OPG submitted a progress report on OPG's Safety Analysis Improvement and REGDOC-2.4.1 implementation activities and acknowledged the closing of Action Item 2014-OPG-5461 as per CNSC Correspondence [N-CORR-00531-18016, e-Doc 4947467, *Darlington & Pickering NGS: Safety Analysis Improvement and REGDOC-2.4.1 Implementation - Closure of Action Item 2014-OPG-5461*, March 24, 2016]. Attachment 1 of [N-CORR-00531-18239, OPG Correspondence, *Progress Report on OPG Safety Analysis Improvement and REGDOC-2.4.1 Implementation*, October 17, 2016] provides a progress update on REGDOC-2.4.1 compliant safety analyses.

As stated in the Licence Conditions Handbook [P-CORR-00531-04886, e-Doc 5121102, *Pickering NGS: Licence Conditions Handbook, LCH-PNGS-R005*, November 10, 2016], the current REGDOC-2.4.1 implementation plan defines the REGDOC-2.4.1 compliant analyses to be undertaken in the 2014-2017 timeframe. OPG is to update its REGDOC-2.4.1 Implementation Plan at the end of 2017, applying a graded, safety significant approach. The updated plan will identify any changes required to support the continued safe operation of Pickering NGS, and will be informed by the timeline of the Darlington REGDOC-2.4.1 Implementation Plan and the limited additional years of Pickering NGS operation.

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Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								Overall Safety Significance Level
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	N/A	3	**3**	N/A	N/A	N/A	N/A	N/A	N/A	N/A	**N/A**	**3**

Rationale:

This Global Issue is related to Deterministic Safety Analysis and REGDOC-2.4.1 implementation. As discussed in Section 3 of this Global Issue, OPG will update its REGDOC-2.4.1 Implementation Plan by the end of 2017, including consideration of the impact of the extended operating period. OPG's existing deterministic safety analysis is comprehensive and robust.

Regarding deterministic considerations, this Global Issue pertains to analysis and is not directly related to any physical barriers. Therefore, this Global Issue is not directly applicable to Defence in Depth (E1). Safety Significance Levels (E2) is assigned Safety Significance Level 3 since the issue is not significant by itself (the definition of Safety Significance Level 3 for E2) and OPG will update its REGDOC-2.4.1 Implementation Plan by the end of 2017. Hence, the overall Safety Significance Level of 3 for deterministic considerations is dictated by the E2 categorization.

This Global Issue pertains to deterministic safety analysis, so has no direct impact on the probabilistic considerations, i.e., Core Damage Frequency (F1), Defence in Depth (F2), Public Radiation Safety (F3), Plant Operability (F4), Occupational Radiation Safety (F5), Emergency Preparedness (F6) and Environment (F7). Therefore, the probabilistic considerations are not directly applicable to this issue.

In summary, the overall Safety Significance Level is 3. OPG continues to update Pickering deterministic safety analysis as mandated by the Licence Conditions Handbook, and work is progressing to update the REGDOC-2.4.1 Implementation Plan using a graded approach with consideration of the impact of the extended operating period.

SECTION 5 - GI-31 RESOLUTION PLAN

GI-31-RS1	Complete the Pickering NGS Implementation Plan for REGDOC-2.4.1 [N-PLAN-03500-0500515 R003, *REGDOC-2.4.1 Implementation Plan*, May 25, 2015]. The Implementation Plan at Pickering NGS was summarized in the PROL Amendment request as follows: "In alignment with current Pickering licensing requirements, and with the graded approach permitted by REGDOC-2.4.1 requirements, OPG

SECTION 5 - GI-31 RESOLUTION PLAN	
	will be upgrading the Pickering safety reports only to the extent that a new appendix will be included to address the development and analysis of common mode events in 2017. The analysis of common mode events represents the single largest gap in the Pickering Safety Reports with respect to REGDOC-2.4.1." OPG is progressing this activity. (SF5-3)
GI-31-RS2	Prepare Implementation Plan update for REGDOC-2.4.1 including consideration of the impact of the extended operating period. OPG is progressing this activity. (SF5-4) (COP-21)
GI-31-AD1	Required system performance under accident conditions is addressed in the Pickering Safety Reports. Also, a full range of initiating events, including AOO-type sequences, is considered in the PSAs. However, AOOs have not been identified and analyzed in the current Pickering Safety Reports. Pickering has a comprehensive list of events (Initiating Events) defined for the safety analyses that are modelled in the PSA and the risk is acceptably low. Consideration of the identification and analysis of AOOs is addressed under the REGDOC-2.4.1 implementation plan which will be updated in accordance with the Licence Conditions Handbook and will identify any changes required to support the continued safe operation of Pickering NGS. These changes will be informed by the timeline of the Darlington REGDOC-2.4.1 Implementation Plan and the limited additional years of Pickering NGS operation. A practicable solution to addressing the AOOs requirements identified in the design-related code gaps, and implementing any enhancements within the time available during extended operation, is not readily evident. On this basis, and given the low safety significance (Safety Significance Level 3), this is assessed as an Acceptable Deviation. (SF1-8) (SF1-14) (SF1-15) (SF1-16) (SF1-18) (SF1-AG14)
GI-31-AD2	The analysis code ADDAM (Atmospheric Dispersion and Dose Analysis Method) is an Industry Standard Toolset code used, as required, for analysis updates that include atmospheric dispersion and dose calculations. ADDAM has been evaluated against the 2014 version of CSA N288.2 under COG Project WP 50109 [N-CORR-00531-06905, *REGDOC 3.1.1 Research and Development Annual Reporting*, June 16, 2015], outside of PSR2. The section by section review showed that no major modifications to the ADDAM 1.4.2 code, methodology or manuals would be required [COG report ISTO-15-5057, *Assessment of Impact of CSA N288.2-14 on ADDAM*, March 2017]. The ADDAM 1.4.2 code is being used in the update of the Pickering Safety Report Common Mode Events appendices, under the REGDOC-2.4.1 Implementation Plan. As the REGDOC-2.4.1 Implementation Plan does not identify other updates of the Pickering Safety Reports' analyses to address changes in N288.2-14, the existing Safety Report analyses related to atmospheric dispersion and dose calculations continue to support the safety case for Pickering. On this basis, and given the low safety significance (Safety Significance Level 3), this is assessed as an Acceptable Deviation. (SF5-6)
GI-31-NFA1	AI 201113-2297 is related to the modified 37-Element Fuel Bundle and was closed

SECTION 5 - GI-31 RESOLUTION PLAN	
	by the CNSC in April 2014 [NK38-CORR-00531-16762, e-Doc 4410455, *Action Item 201113-2297: Modified 37-Element Fuel Bundle CHF/PDO Test Results And CHF Correlation Development - New Action Item 2014-13-4926,* April 1, 2014]. The CNSC raised follow-up actions under AI 2014-13-4926 associated with assessment and quantification of Critical Heat Flux (CHF) data uncertainties and CHF/Critical Channel Power simulation uncertainties. These actions too were dispositioned by OPG in [NK38-CORR-00531-17038, *Darlington NGS - Modified 37-Element Fuel Bundle CHF/PDO Test Results and CHF Correlation Development - Action Item 2014-13-4926* October 30, 2014] and accepted by the CNSC in [NK38-CORR-00531-17353, e-Doc 4658016, *Darlington NGS: Modified 37-Element Fuel Bundle CHFIPDO Test Results and CHF Correlation Development - Closure of Action Item 2014-13-4926* March 26, 2015]. The relevant Action Item to Pickering is AI 2007-4-08 on 28-element CHF and it was closed on August 21, 2012 [NA44-CORR-00531-07002, e-Doc 3991677, *Pickering NGS-A -Safe Operation Following Findings From 28-Element Critical Heat Flux Tests -Action Item 2007-4-08 (RIB #2420),* August 21, 2012]. Follow-up Action Item 2012-OPG-3464 was also closed on February 13, 2013 [N-CORR-00531-06063, e-Doc 4054739, *Request for additional information on PHTS Aging Fuel Channel And Plant Thermalhydraulics and CCP uncertainty aspects of the new NOP Analysis Methodology – Closure of Action Item 2012-OPG-3464*, February 13, 2013]. There are no outstanding Action Items related to this issue. (COP-14)
GI-31-XRF-GI-44-AD8	Operator Action Time Credits. REGDOC-2.5.2 requirements for allowable times for operator action from the MCR or the field are more limiting than the corresponding requirements of REGDOC-2.4.1. Pickering A and B Safety Report credits for operator actions from the MCR and in the field are consistent with REGDOC-2.4.1 requirements of 15 and 30 minutes, respectively. The ability to execute required actions within these time limits has been demonstrated through decades of operation, through effective training and testing programs. The gap against the corresponding REGDOC 2.5.2 requirements is a low safety significance issue (Safety Significance Level 3) and has been addressed to the extent practicable. Therefore, it is assessed as an Acceptable Deviation. (SF5-AG1)

B.32. GI-32 Implementation of REGDOC-2.4.2 PSA Requirements

SECTION 1 - GI-32 GLOBAL ISSUE SUMMARY

The goal of GI-32, Implementation of REGDOC-2.4.2 PSA Requirements, is to complete the Implementation Strategy for CNSC Regulatory Document REGDOC-2.4.2, Probabilistic Safety Assessment, and to update the Strategy in the context of the extended operating period. This Global Issue comprises one proposed Resolution Statement addressing one SF6 code review Gap. Safety Significance Level 3 is assigned for this Global Issue based on probabilistic core damage frequency considerations.

The proposed Resolution Plan comprises a proposed Resolution Statement to complete the activities in the REGDOC-2.4.2 Implementation Strategy as identified in the Licence Conditions Handbook, and update the Strategy in the context of the extended operating period. Completion of the proposed Resolution Plan will support and strengthen Levels 2, 3, and 4 defence-in-depth for the extended operating period.

Safety Significance Level:	3	**Category:**	Programmatic, Analytical	**Reassessment Beyond 2024:**	Y

SECTION 2 - GI-32 ASSOCIATED GAPS

SF6-4	The REGDOC-2.4.2 Pickering Implementation Plan agreed to with the CNSC did not consider operation beyond 2020 and therefore, the review and update of the Implementation Plan in the context of operation of Pickering NGS beyond 2020 is required. Therefore, this has been identified as a PSR2 gap. ***Code Review CNSC REGDOC-2.4.2*** **Associated Resolutions:** GI-32-RS1

SECTION 3 - GI-32 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

This Global Issue contains one SF6 Gap related to the implementation of REGDOC-2.4.2.

This gap (SF6-4) is identified to consider the Implementation Strategy for REGDOC-2.4.2 documented in [P-CORR-00531-04557, *Pickering NGS-Request for Amendment to Pickering PROL 48.01/2018 to Implement New Regulatory Documents REGDOC-2.4.1 and REGDOC-2.4.2*, October 23, 2015], in the context of extended operation beyond 2020.

SECTION 4 - GI-32 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								Overall Safety Significance Level
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	N/A	N/A	**N/A**	3	N/A	N/A	N/A	N/A	N/A	N/A	**3**	**3**

Rationale:

This Global Issue is related to completion of the Implementation Strategy for REGDOC-2.4.2, and update of the Strategy with consideration of the extended operating period.

This Global Issue pertains specifically to the Pickering NGS Probabilistic Safety Assessment. Therefore, deterministic considerations, Defence in Depth (E1) and Safety Significance Levels (E2), are not applicable to this Global Issue.

With respect to probabilistic considerations, Safety Significance Level 3 is assigned the Core Damage Frequency (F1). The impact of resolving this Global Issue is not expected to cause a change in Core Damage Frequency greater than 10^{-6}/y. This corresponds to Safety Significance Level 3 in Table F1 of the PSR2 Basis Document. OPG has a comprehensive and robust Probabilistic Safety Assessment.

This Global Issue has no direct impact on the other probabilistic considerations, i.e., Defence in Depth (F2), Public Radiation Safety (F3), Plant Operability (F4), Occupational Radiation Safety (F5), Emergency Preparedness (F6) and Environment (F7). Therefore, these other probabilistic considerations are not directly applicable to this issue.

In summary, the overall Safety Significance Level is 3. OPG is progressing this activity in support of extended operation at Pickering NGS.

SECTION 5 - GI-32 RESOLUTION PLAN

GI-32-RS1	Complete the activities in the REGDOC-2.4.2 Implementation Strategy, as identified in Section 5.1, Safety Analysis Program, of [P-CORR-00531-04886, CNSC Correspondence, e-Doc 5121102, *Pickering NGS: Licence Conditions Handbook, LCH-PNGS-R005*, November 10, 2016] and update the Strategy in the context of the additional operating period. OPG is progressing this activity in support of extended operation at Pickering NGS. (SF6-4)

B.33. GI-33 N285.0-12, General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants

SECTION 1 - GI-33 GLOBAL ISSUE SUMMARY					
The goal of GI-33, N285.0-12 General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants, is to ensure compliance with the requirements of CSA N285.0-12. This Global Issue comprises one Acceptable Deviation, one Cross Reference to GI-1, one Cross Reference to GI-2, one Cross Reference to GI-3, one Cross Reference to GI-4, and one Cross Reference to GI-5, which address two SF1 code review Gaps [CSA N285.0-12]. Safety Significance Level 3 is assigned to this Global Issue based on deterministic safety significance levels considerations. The cross references to GI-1, GI-2, GI-3 and GI-4 are to confirm that the allowable cycles for fatigue of the Class 1 major components will not be exceeded. The cross reference to GI-5 is to confirm the adequacy of the service limits assessments for Class 1 Piping after accounting for the impact of environmental factors. The Acceptable Deviation addresses Liquid Injection Shutdown System classification and it is based on a rationale that was accepted and a code classification concession that was granted by the CNSC. Completion of the proposed Resolution Plan for the cross references to other GIs will support and strengthen Level 1 defence-in-depth for the extended operating period.					
Safety Significance Level:	3	**Category:**	Analytical	**Reassessment Beyond 2024:**	Y

SECTION 2 - GI-33 ASSOCIATED GAPS	
SF1-1	Clause A.2.3.1 of CSA N285.0-06 identifies that for Shutdown Systems, pressure-retaining portions shall be classified as Class 1, except for three listed exceptions. It was identified during the Pickering B Integrated Safety Review (ISR) that a limited number of Liquid Injection Shutdown System (LISS) components, which should have been Class 1, were purchased and installed as Class 3. In follow-up, OPG proposed four actions to address the deficiency. When refurbishment was not pursued, a code classification concession was accepted for continued operations. This code classification concession and the four actions identified in the Pickering B ISR gap resolution need to be reconsidered in the context of operation of Pickering NGS beyond 2020. Therefore, this has been identified as a PSR2 gap. ***Code Review N285.0-12*** **Associated Resolutions:** GI-33-AD1
SF1-2	The Pickering A Return to Service review against CSA-N285.0-95 identified two Acceptable Deviations relating to Clause 7.0 requiring confirmation that the allowable cycles for fatigue would not be exceeded. For Pickering Units 1,4 and Units 5-8 operation beyond 2020, further confirmation is required that the allowable cycles for fatigue will continue to bound current

SECTION 2 - GI-33 ASSOCIATED GAPS

service limits for extended operation. Therefore, this has been identified as a PSR2 gap.

Code Review N285.0-12

Associated Resolutions: GI-1-RS3, GI-2-RS1, GI-3-RS1, GI-4-RS1, GI-5-RS1

SECTION 3 - GI-33 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

This Global Issue contains two SF1 Gaps related to CSA N285.0-12.

Compliance with CSA N285.0-08 (including Updates No. 1 and No. 2) is currently a licence requirement for Pickering NGS (per PROL 48.03/2018) as indicated in Section 6.2 and Appendix C.1 of the R05 Pickering Licence Conditions Handbook. Two gaps were identified: one is related to LISS component classification and the other gap is related to a required confirmation that the allowable cycles for fatigue will continue to bound current service limits for the extended operating period.

The service limits assessment requirement is limited to Nuclear Code Class 1 components, as a requirement under the current ASME Boiler and Pressure Vessel code [NK30-CORR-00531-06324, e-Doc 3947907, *Pickering NGS-B - CNSC Staff Assessment Of OPG's 2011 Continued Operations Plan (action Item 2010-8-05 (2461)) and path forward*, June 19, 2012, Attachment 1 – Part B (Table B-1), item 2.2.8].

For the Major Components, the OPG Life Cycle Management Plans (LCMPs) include confirmation that the allowable cycles for fatigue will continue to bound current service limits. This is complete to 2022 and 2024 for Pickering 1,4 and 5-8 Major Components respectively, and is addressed in GI-1, GI-2, GI-3 and GI-4.

For all other Code Class 1 components and piping, the confirmation that the allowable cycles for fatigue will continue to bound current service limits for the extended operating period is complete, and is addressed in GI-5. The activity to confirm the adequacy of the Class 1 piping service limits assessment after accounting for impact of environmental factors is also addressed in GI-5.

SECTION 4 - GI-33 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								Overall Safety Significance Level
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	N/A	3	**3**	N/A	N/A	N/A	N/A	N/A	N/A	N/A	**N/A**	**3**

Rationale:

This Global Issue is related to a code classification concession for Liquid Injection Shutdown System components, and confirmation that the allowable cycles for fatigue bound the service limits for extended operation for Nuclear Code Class I components. The Safety Significance Level of the LISS requirement is assessed in this Global Issue. The requirements for the other components (e.g., Fuel Channels) are cross referenced to proposed Resolution Statements in GI-1, GI-2, GI-3, GI-4 and GI-5.

Regarding deterministic considerations, this Global Issue is not directly applicable to Defence in Depth (E1). Safety Significance Levels (E2) is assigned Safety Significance Level 3 on the basis that the issue is not significant. The Liquid Injection Shutdown System deviation from Class 1 requirements was previously assessed to be acceptable and accepted by the CNSC. Hence, the overall Safety Significance Level of 3 for deterministic considerations is dictated by the E2 categorization.

This Global Issue has no direct impact on the probabilistic considerations, i.e., Core Damage Frequency (F1), Defence in Depth (F2), Public Radiation Safety (F3), Plant Operability (F4), Occupational Radiation Safety (F5), Emergency Preparedness (F6) and Environment (F7). Therefore, the probabilistic considerations are not directly applicable to this issue.

In summary, the overall Safety Significance Level is 3 with respect to the LISS components.

SECTION 5 - GI-33 RESOLUTION PLAN

GI-33-AD1	A limited number of LISS components that should have been code Class 1 were purchased and installed as code Class 3. However, a rationale was accepted and a code classification concession was granted by the CNSC to allow the system to remain as-is for the legacy modifications. Further installations would be done using code Class 1 materials [NK30-CORR-00531-02663, OPG Correspondence, *Pickering NGS 'B' Units 5-8 Liquid Injection Shutdown System (34700) Request for Code Classification Approval*, December 13, 2004] and [NK30-CORR-00531-03047, CNSC Correspondence File: 26-1-8-3-0, *Pickering NGS-B Code*

SECTION 5 - GI-33 RESOLUTION PLAN	
	Classification Approval – Legacy Modifications to Liquid Injection Shutdown System (USI 34700), Units 5-8, February 11, 2005]. In NK30-CORR-00531-02663, OPG provided rationale to show that consequences following any failure of Class 3 or Class 6 portions of the systems continue to satisfy the requirements of N285.0. On this basis, this issue is low safety significance (Safety Significance Level 3) and is being managed to the extent practicable. Therefore it is assessed as an Acceptable Deviation. (SF1-1)
GI-33-XRF-GI-1-RS3	Update the Fuel Channels LCMP [N-PLAN-01060-10002-R017, OPG Plan, *Fuel Channels Life-Cycle Management Plan*, October 30, 2016] for Pickering Units 1,4 for the extended operating period. (SF1-2)
GI-33-XRF-GI-2-RS1	Update the Feeders LCMP [N-PLAN-01060-10001-R018, OPG Plan, *Feeders Life Cycle Management Plan*, October 31, 2016] for Pickering Units 1,4 for the extended operating period. (SF1-2)
GI-33-XRF-GI-3-RS1	Update the Steam Generators LCMP [N-PLAN-33110-10009-R007, OPG Plan, *Steam Generators Life Cycle Management Plan*, October 24, 2016] for Pickering Units 1,4 for the extended operating period. (SF1-2)
GI-33-XRF-GI-4-RS1	Update the Reactor Components and Structures LCMP [N-PLAN-01060-10003-R014, OPG Plan, *Reactor Components and Structures Life Cycle Management Plan*, October 30, 2016], based on updated Fitness for Service assessment and an updated Technical Basis Document [N-PLAN-01060-10008 R00, *Reactor Components & Structures Life Cycle Management Plan: Technical Basis Document*, June 25, 2010] for Pickering Units 1,4 for the extended operating period. (SF1-2)
GI-33-XRF-GI-5-RS1	Confirm the adequacy of the service limits assessments for Nuclear Class 1 Piping after accounting for impact of environmental factors (for example: irradiation, temperature, humidity). (SF1-2)

B.34. GI-34 CSA N290.1-13 - Requirements for the Shutdown Systems

SECTION 1 - GI-34 GLOBAL ISSUE SUMMARY					
The goal of GI-34, CSA N290.1-13- Requirements for the Shutdown Systems, is to demonstrate an appropriate degree of conformance with the requirements of CSA N290.1-13. This Global Issue comprises one Acceptable Deviation addressing one SF1 code review Gap related to remote tripping and monitoring capability. Safety Significance Level 3 is assigned for this Global Issue based on deterministic defence-in-depth considerations. The proposed Resolution Plan comprises an Acceptable Deviation based on the low safety significance level and the availability of dedicated remote tripping and monitoring for Pickering 1,4 SDSE at the Instrument Rooms and for Units 5-8 SDS2 at the Unit Emergency Control Centres. This arrangement meets the intent of the requirements of CSA N290.1-13 to the extent practicable.					
Safety Significance Level:	3	**Category:**	Analytical	**Reassessment Beyond 2024:**	N

SECTION 2 - GI-34 ASSOCIATED GAPS	
SF1-10	Clause 4.1.8.2 of CSA N290.1-13 is for a new plant and requires remote tripping and monitoring capability for both Shutdown Systems. Pickering Units 1,4 only have one Shutdown System with tripping capability from separate logic (SDSA and SDSE). Remote tripping capability is available for Pickering 5-8 SDS2 and Pickering 1,4 SDSE. However, Pickering Units 5-8 and 1,4 do not have remote tripping and monitoring capability for SDS1 or SDSA respectively. Therefore, this has been identified as a PSR2 gap. ***Code Review N290.1-13*** **Associated Resolutions:** GI-34-AD1

SECTION 3 - GI-34 BACKGROUND INFORMATION AND RESOLUTION STRATEGY
This Global Issue contains one SF1 gap related to CSA N290.1-13. Compliance with N290.1 is not currently a licence requirement for Pickering NGS (PROL 48.03/2018) and is not referenced in the Pickering Licence Conditions Handbook. The gap (SF1-10) is related to Clause 4.1.8.2 of CSA N290.1-13 which requires remote tripping and monitoring capability to be available for both Shutdown Systems in a secondary control room. Both Pickering 5-8 and 1,4 have remote tripping capability for SDS2 and SDSE, respectively. Pickering 5-8 has SDS2 tripping capability in the Unit Emergency Control Centres (UECCs). Pickering 1,4 has remote SDSE tripping capability in the SDSE Instrument Room. Neither SDSA at Pickering 1,4

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nor SDS1 at Pickering 5-8 have remote tripping and monitoring capability.

SECTION 4 - GI-34 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								Overall Safety Significance Level
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	3	N/A	**3**	N/A	N/A	N/A	N/A	N/A	N/A	N/A	**N/A**	**3**

Rationale:

This Global Issue is related to remote tripping and monitoring capability for both Shutdown Systems.

Regarding deterministic considerations, this Global Issue is assigned a Safety Significance Level 3 for Defence in Depth (E1), since the issue is related to, but does not impair, shutdown capability (row 3 of Table E1 in the PSR2 Basis Document). There is dedicated remote tripping and monitoring capability at Pickering 1,4 with SDSE and at Pickering 5-8 with SDS2. Safety Significance Levels (E2) is not directly applicable since this Global Issue can have a direct impact on nuclear safety, whereas E2 primarily relates to issues that are indirectly related to nuclear safety. Hence, the overall safety Significance Level of 3 for deterministic considerations is dictated by the E1 categorization.

This Global Issue has no direct impact on the probabilistic considerations, i.e., Core Damage Frequency (F1), Defence in Depth (F2), Public Radiation Safety (F3), Plant Operability (F4), Occupational Radiation Safety (F5), Emergency Preparedness (F6) and Environment (F7). Therefore, the probabilistic considerations are not directly applicable to this issue.

In summary, the overall Safety Significance Level is 3.

SECTION 5 - GI-34 RESOLUTION PLAN

GI-34-AD1	There is dedicated remote tripping and monitoring capability at Pickering 1,4 for SDSE in the SDSE Instrument Rooms [NA44-SR-01320-00001-R016, *Pickering A Safety Report*, July 20, 2017] and at Pickering 5-8 for SDS2 in the Unit Emergency Control Centres [NK30-SR-01320-00002-R004, *Pickering B Safety Report – Part*

SECTION 5 - GI-34 RESOLUTION PLAN	
	2, October 10, 2012], and this meets the CSA N290.1 intent and requirements to the extent practicable. In addition, the failure to shutdown probability, as demonstrated in the PSAs, is very low [P-REP-03611-00006-R000, *Pickering NGS PSA Update to Include Enhancements from the Fukushima Integrated Action Plan*, April 30, 2014]. On this basis, and as per the low safety significance level (Safety Significance Level 3), this issue is assessed as an Acceptable Deviation. (SF1-10)

B.35. GI-35 Human Factors Issues

SECTION 1 - GI-35 GLOBAL ISSUE SUMMARY					
The goal of GI-35, Human Factors Issues, is to demonstrate an appropriate degree of conformance with specific Human Factors requirements of CSA N290.0-11 General requirements for safety systems of nuclear power plants. This Global Issue includes one SF1 code review Gap that is assessed to be an Acceptable Deviation and one SF1 Additional Gap related to NUREG-0700 that is assessed as No Further Action. Safety Significance Level 4 is assigned for this Global Issue based on deterministic defence-in-depth considerations. The SF1 Gap is related to Human Factors Engineering (HFE) activities not being formally documented when the Main Control Rooms were originally designed and constructed and is considered an Acceptable Deviation based on the extensive operating experience and the established processes and instructions to ensure that human-system interface elements for a modification are addressed.					
Safety Significance Level:	4	**Category:**	Programmatic	**Reassessment Beyond 2024:**	N

SECTION 2 - GI-35 ASSOCIATED GAPS	
SF1-7	The Darlington Integrated Safety Review (ISR) identified a gap against Clause 4.14.10 of N290.0-11 as a result of the lack of design standards related to Human Factors Engineering (HFE) or HFE activities being formally documented when the Main Control Rooms were originally designed and constructed. Pickering NGS has many years of successful Special Safety System (SSS) operation and the absence of formal HFE in the original design is not expected to have any nuclear safety significance relating to SSSs. However, the Darlington gap is also applicable to Pickering NGS and is therefore identified as a PSR2 gap. ***Code Review N290.0-11*** **Associated Resolutions:** GI-35-AD1
SF1-AG7	An SF1 Gap related to a review of Pickering B against NUREG-0700 was identified in [P-CORR-00531-05107, e-Doc 5305945, July 26, 2017]. **Associated Resolutions:** GI-35-NFA1

SECTION 3 - GI-35 BACKGROUND INFORMATION AND RESOLUTION STRATEGY
This Global Issue contains two Gaps, an SF1 report Gap (SF1-7) and an SF1 Additional Gap (SF1-AG7). SF1-7 is against Clause 4.14.10 of N290.0-11 as a result of minimal design standards being available related to Human Factors Engineering (HFE) or HFE activities not being formally documented when the Main Control Rooms were originally designed and constructed. SF1-AG7 is related to consideration of [NUREG-0700, Rev. 2, *Human-System Interface Design Review Guidelines*, May

SECTION 3 - GI-35 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

2002] and [NUREG-0711, Rev. 3, *Human Factors Engineering Program Review Model*, November 2012] in PSR2 reviews.

When the Main Control Rooms were designed, Ontario Hydro employed electrical standards that included indicating lamp conventions, handswitch conventions, and labeling conventions. For design modifications that require an HFE plan, the plans prepared by OPG meet the requirements of CNSC G-276 [G-276, *Human Factors Engineering Program Plans*, June 2003] and CNSC G-278 [G-278, *Human Factors Verification and Validation Plans*, June 2003].

SECTION 4 - GI-35 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								Overall Safety Significance Level
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	4	N/A	**4**	N/A	N/A	N/A	N/A	N/A	N/A	N/A	**N/A**	**4**

Rationale:

This Global Issue is related to minimal design standards being available for Human Factors Engineering activities, or these activities not being formally documented, when the Main Control Rooms were originally designed and constructed. As discussed in Section 5 of this Global Issue, the original design phase of Pickering NGS recognized the need for focus on the operator interfaces in the control centres, and the related human-systems interfaces. The Pickering units as a result have decades of safe operation and operating experience with monitoring, operation, testing, maintenance and training (including simulators).

Regarding deterministic considerations, this Global Issue is assigned Safety significance Level 4 for Defence in Depth (E1). This is because this issue is not significant by itself and it has no impact on the level of operational performance and safety culture. Safety Significance Levels (E2) is not directly applicable since this Global Issue can have a direct impact on nuclear safety, whereas E2 primarily relates to issues that are indirectly related to nuclear safety. Hence, the overall safety Significance Level of 4 for deterministic considerations is dictated by the E1 categorization.

This Global Issue has no direct impact on the probabilistic considerations, i.e., Core Damage Frequency (F1), Defence in Depth (F2), Public Radiation Safety (F3), Plant Operability (F4), Occupational Radiation Safety (F5), Emergency Preparedness (F6) and Environment (F7). Therefore,

SECTION 4 - GI-35 PRIORITY DETERMINATION
the probabilistic considerations are not directly applicable to this issue. In summary, the overall Safety Significance Level is 4.

SECTION 5 - GI-35 RESOLUTION PLAN	
GI-35-AD1	The original design phase of Pickering NGS recognized the need for focus on the operator interfaces in the control centres and on recognition of the integrated whole of the control centre and the related human-systems interfaces. The Pickering units as a result have decades of safe operation and operating experience with monitoring, operation, testing, maintenance and training (including simulators). With regards to plant modifications, all plant modifications are required to be completed in compliance with N-PROG-MP-0001 [N-PROG-MP-0001 R015, OPG Nuclear Program, *Engineering Change Control*, May 12, 2017], which includes the requirements for Human Factors and Ergonomics to be considered in design. If the modification is judged to have an HFE impact, a Human Factors Engineering Specialist must concur with the Human Factors Level of Activity. The modification may require the preparation of the Human Factors Engineering Plan or the Human Factors Worksheet. These instructions and processes ensure that the human-system interface elements for the modification are addressed. The technical, design and operator reviews, during and following the design process and via the Availability for Service (AFS) process, ensure the usability requirements will be achieved. Based on the extensive operating experience and modifications processes, there is no justification for revisiting the overall Main Control Room design from a Human Factors perspective. On this basis, and as per the very low safety significance (Safety Significance Level 4), this is assessed as an Acceptable Deviation. (SF1-7)
GI-35-NFA1	The review task assessment in the Safety Factor Report [P-REP-03680-00008 R000, *Pickering NGS PSR2 Safety Factor 1 Report: Plant Design*, March 3, 2017] confirmed that Human Factors Engineering Program Plans prepared by OPG meet the requirements of CNSC G-276 [G-276, *Human Factors Engineering Program Plans*, June 2003] and CNSC G-278 [G-278, *Human Factors Verification and Validation Plans*, June 2003] and the applicable elements from [NUREG-0711, Rev. 3, *Human Factors Engineering Program Review Model*, November 2012]. Per OPG governance, specifically OPG Manual [N-MAN-06700-10002, *Guide for OPG Human Factors Engineering Process*, December 18 2015], the Human Factors Engineering Program Plans are expected to meet the intent of NUREG-0711. With respect to human-system interface in design, the Pickering units have years of safe operating experience with special safety system monitoring, operation (including testing), maintenance and training (including simulator). Improvements based on operation and maintenance experience have also been incorporated into processes and the design to improve the human-machine interface. In addition, training and qualification processes (and certification processes for control room staff) for Operations positions ensure that the staff are competent to carry out

SECTION 5 - GI-35 RESOLUTION PLAN	
	functions assigned to them. Any potentially significant human interaction deficiencies due to the design would be identified during these activities and be addressed. Hence, the extent of application of human factors engineering in the original design is not expected to have an impact on nuclear safety relating to the SSCs. Therefore, a review of the design of human-system interfaces to assess the extent to which Pickering NGS meets [NUREG-0700, Rev. 2, *Human-System Interface Design Review Guidelines*, May 2002] is not considered necessary. A gap analysis against mandatory requirements of CSA N290.12-14, Human Factors in Design for Nuclear Power Plants, was performed for PSR2 in [P-REP-03680-00021, Pickering PSR2 law, Regulation, Code and Standard Reviews Associated with Safety Factors 8, 10, 12, 13, and 14] and found only minor issues that do not have a nuclear safety impact, and there are no PSR2 gaps. Also, versions of CSA N290.0, General Requirements for Safety Systems of Nuclear Power Plants, subject to previous PSR1 reviews, as well as their applicability to Pickering PSR2, are assessed in [P-REP-03680-00029, Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 1, 5, 6, and 7], and a gap was identified against clause 4.14.10 of CSA N290.0 on human-machine interface requirements. The gap (SF1-7) from CSA N290.0 is assessed as an Acceptable Deviation under GI-35. Therefore, based on the years of safe operation, incorporation of OPEX into Pickering NGS modifications, procedures, training and the results of assessments of CSA N290.12-14 and CSA N290.0, Pickering NGS can maintain its safe operation for the extended period, and a review of NUREG-0700 and NUREG-0711 is not considered necessary. No further action is required. (SF1-AG7)

B.36. GI-36 CSA N290.2 - Requirements for Emergency Core Cooling Systems

SECTION 1 - GI-36 GLOBAL ISSUE SUMMARY					
The goal of GI-36, CSA N290.2- Requirements for Emergency Core Cooling Systems, is to demonstrate an appropriate degree of conformance with the requirements of this standard. This Global Issue comprises two Acceptable Deviations addressing two SF1 code review Gaps. Safety Significance Level 3 is assigned for this Global Issue based on the deterministic and probabilistic defence-in-depth considerations. The proposed Resolution Plan comprises two Acceptable Deviations, each addressing one Gap. The first Acceptable Deviation addresses the Gap related to the requirement to assess ECI effectiveness based on the least effective of the Shutdown Systems. The Pickering 1,4 Shutdown System arrangement has previously been accepted by the CNSC and the ECI capability is assessed using the available SDS and hence this meets the CSA N290.2 requirement to the extent practicable. The second Gap relates to instrumentation to monitor emergency cooling debris strainer effectiveness post-accident. This Gap is assessed as an Acceptable Deviation because monitoring ECI recovery pump performance and Reactor Building water level for any adverse trend in performance expected to be caused by debris loading meets the intent of this requirement.					
Safety Significance Level:	3	**Category:**	Engineering	**Reassessment Beyond 2024:**	N

SECTION 2 - GI-36 ASSOCIATED GAPS	
SF1-11	Clause 5.2.1.2 of CSA N290.2-11 requires that Emergency Coolant Injection System (ECIS) design requirements be based on the assumption that the least effective of the Shutdown Systems has operated successfully. The Pickering Units 5-8 Safety Report analysis does address this requirement and the requirement is also contained in the Pickering Units 5-8 Design Requirements. However, this requirement cannot be met for Pickering Units 1,4 since there is only one Shutdown System (albeit with tripping capability from separate SDSA and SDSE logic). Therefore, this has been identified as a PSR2 gap. ***Code Review N290.2-11*** **Associated Resolutions:** GI-36-AD1
SF1-12	Clause 5.14.11 of CSA N290.2-11 requires instrumentation to be available to monitor post-accident effectiveness and to determine the extent of plugging of Emergency Coolant Injection System (ECIS) debris interceptors (strainers). While relative health of a strainer can be inferred by a combination of ECIS recovery pump performance and Reactor Building water level, there is no direct correlation between these conditions and debris loading available. Therefore, this has been identified as a PSR2 gap (which is applicable to both Pickering Units 5-8 and 1,4). ***Code Review N290.2-11***

SECTION 2 - GI-36 ASSOCIATED GAPS

Associated Resolutions: GI-36-AD2

SECTION 3 - GI-36 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

This Global Issue contains two SF1 Gaps related to CSA N290.2.

CSA N290.2 is identified in Appendix E.1 of the R05 Pickering Licence Conditions Handbook as "Guidance or Criteria".

SECTION 4 - GI-36 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								Overall Safety Significance Level
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	3	N/A	**3**	4	3	N/A	N/A	N/A	N/A	N/A	**3**	**3**

Rationale:

This Global Issue includes two gaps, one related to Pickering 1,4 ECI System capability with respect to Shutdown System operation and the other gap is related to monitoring of ECI System debris strainer effectiveness.

Regarding deterministic considerations, this Global Issue is assigned Safety Significance Level 3 for Defence in Depth (E1) on the basis that the issue is related to but does not impact ECI Injection System capability (row 3 of Table E1 in the PSR2 Basis Document). This is because for the first gap, although there is only one Shutdown System on Pickering 1,4, the tripping capability from separate SDSA and SDSE logic exists. For the second gap, the intent of the requirement is met by monitoring ECI recovery pump performance and Reactor Building water level. Safety Significance Levels (E2) is not directly applicable since this Global Issue can have a direct impact on nuclear safety, whereas E2 primarily relates to issues that are indirectly related to nuclear safety. Hence, the overall Safety Significance Level of 3 for deterministic considerations is dictated by the E1 categorization.

Regarding probabilistic considerations, the issue has Safety Significance Level 3 for Reactor Safety - Defence in Depth (F2). This is because the issue is related to the reliability of ECI System

SECTION 4 - GI-36 PRIORITY DETERMINATION

operation which is required for events with initiating frequency on the order of 10^{-3}/y or less. For Reactor Safety – Core Damage Frequency (F1), the issue has Safety Significance Level 4 since any impact on Core Damage Frequency is expected to be insignificant (less than 10^{-7}/y).

This Global Issue has no direct impact on the other probabilistic considerations, i.e., Public Radiation Safety (F3), Plant Operability (F4), Occupational Radiation Safety (F5), Emergency Preparedness (F6) and Environment (F7). Therefore, these probabilistic considerations are not applicable, and the overall Safety Significance Level of 3 for probabilistic considerations is dictated by the F2 categorization.

In summary, the overall Safety Significance Level is 3. The issue is mitigated by alternative design features.

SECTION 5 - GI-36 RESOLUTION PLAN

GI-36-AD1	This issue is related to ECIS capability with respect to Shutdown System operation. The design of the Pickering 1,4 Shutdown System has previously been accepted by CNSC and the ECIS capability is assessed using the available SDS and hence this meets the CSA N290.2 requirement to the extent practicable. The Pickering 1,4 ECI System reliability meets the licensing target, as demonstrated in [NA44-REP-09051.1-00014, R000, *2014 Annual Reliability Report – Pickering Units 1 & 4*, March 10, 2015]. On this basis, and as per the low safety significance of this issue (Safety Significance Level 3), this is assessed as an Acceptable Deviation. (SF1-11)
GI-36-AD2	This issue is related to monitoring of ECIS debris strainer effectiveness. The detailed assessment of the potential sources of strainer debris and contaminants for Pickering 5-8 [NK30-CORR-00531-05194 R001, *Pickering B – Generic Action Item 06G01 Emergency Core Cooling System Strainer Deposits – Status Update and Request for Closure*, June 30, 2009] demonstrated sufficient margin to ensure post-accident ECI recovery strainer effectiveness. The issues identified in the Pickering 1,4 assessment [NA44-CORR-00531-06062, *GAI 06G01: Emergency Core Cooling System Strainer Deposits – Status Update*, June 30, 2009] resulted in the installation of new strainer modules in Units 1 and 4. Since Pickering 1,4 and 5-8 comply with the requirements for new plant debris interceptors, with the exception of clause 5.14.11, the benefit of developing and implementing new instrumentation for post-accident effectiveness is assessed to be small. This is because the intent of the requirement is met by monitoring ECI recovery pump performance and Reactor Building water level for any adverse trend in performance expected to be caused by debris loading. Additional mitigating factors include (a) the demonstrated margin that ensures post-accident effectiveness of the ECI recovery phase with the installed strainer modules, (b) the low contribution of ECIS recovery strainer plugging to the PSA results [NA44-CORR-33350-0265268-R000, *Pickering A Risk Assessment ECI Strainer Plugging Following a Large LOCA*, September 23, 2008], and (c) mandatory inspections to ensure the ECI recovery flowpath is free from debris prior to restart of a Pickering 5-8 or Pickering 1,4 unit following a maintenance outage [NK30-SRS-E-082-R004, *ECI Recovery Flowpath Inspection*, March 27, 2014], [NA44-SRS-E-026-U14-

SECTION 5 - GI-36 RESOLUTION PLAN	
	R019, *ECI Recovery Flowpath Inspection*, February 21, 2017]. Given the low safety significance (Safety Significance Level 3), this is assessed as an Acceptable Deviation. (SF1-12)

B.37. GI-37 N290.3-11 - Requirements for Containment System

SECTION 1 - GI-37 GLOBAL ISSUE SUMMARY					
The goal of GI-37, N290.3-11 Requirements for Containment System, is to demonstrate an appropriate degree of conformance with the requirements of this standard. This Global Issue comprises one Cross Reference to GI-40 addressing one SF1 code review Gap and one SF1 Additional Gap. Safety Significance Level 3 is assigned for this Global Issue based on deterministic and probabilistic defence-in-depth considerations. Completion of the cross referenced proposed Resolution Statement under GI-40 (completion of the installation of Phase 2 Emergency Mitigating Equipment) will support and strengthen defence-in-depth Levels 3 and 4 for the extended operating period.					
Safety Significance Level:	3	**Category:**	Engineering	**Reassessment Beyond 2024:**	N

SECTION 2 - GI-37 ASSOCIATED GAPS	
SF1-13	Per CSA N290.3-11, a Containment Energy Management System (EMS) and Radionuclide Management System (RMS) are required to protect Containment and minimize radiological releases for Beyond Design Basis Accidents (BDBAs). The Pickering EMS and RMS use the Filtered Air Discharge System (FADS) and Reactor Building Air Cooling Units (ACUs). Enhancements to the AC power supplies to these systems and related loads are being provided by Phase 2 Emergency Mitigating Equipment (EME), which is not yet fully implemented. This PSR2 gap has been identified to track the implementation of Phase 2 EME such that it can be used to support the EMS and RMS. ***Code Review N290.3-11*** **Associated Resolutions:** GI-40-RS1
SF1-AG15	An SF1 Gap related to conformance with specific clauses of CSA N290.3 on Containment behaviour during a BDBA, and the installation of a filtered venting system for extended operation to protect containment and prevent uncontrolled large releases following a BDBA, was identified in [P-CORR-00531-05107, e-Doc 5305945, July 26, 2017]. The associated resolution addresses Item (a) of the reference. Item (b) is being addressed as described in [P-CORR-00531-05132, *Pickering Periodic Safety Review 2- Process for Addressing CNSC Identified Additional Gaps*, September 18, 2017]. **Associated Resolutions:** GI-40-RS1

SECTION 3 - GI-37 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

This Global Issue contains one SF1 code review gap and one SF1 Additional Gap related to CSA N290.3-11.

CSA N290.3 is identified in Appendix E.1 of the R05 Pickering Licence Conditions Handbook as “Guidance or Criteria”. Two gaps were identified related to completion of EME Phase 2.

The high-level intents of a Containment Energy Management System (EMS) and a Radionuclide Management System (RMS) are addressed in Pickering 5-8 by the completion of EME Phase 2, which includes supplying cooling water, and power supplies to essential loads via EME generators, to allow for operation of Air Cooling Units and Hydrogen Igniters. For Pickering 1,4, the additional design, operational and/or analytical enhancements to improve Severe Core Damage Frequency and Large Release Frequency resulting from GI-27-RS2 will supplement the enhancements achieved by the completion of the EME Phase 2 project. These enhancements will further complement the existing Pickering design provisions for maintaining containment integrity.

SECTION 4 - GI-37 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								Overall Safety Significance Level
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	3	N/A	**3**	N/A	3	4	N/A	N/A	N/A	N/A	**3**	**3**

Rationale:

This Global Issue is related to CSA N290.3-11, Requirements for the Containment System of Nuclear Power Plants. A number of initiatives related to BDBAs, including completion of Phase 2 of the Emergency Mitigating Equipment project, are proposed.

Regarding deterministic considerations, this Global Issue is assigned Safety Significance Level 3 for Defence in Depth (E1) on the basis that the issue is related to minimizing radiological releases for BDBAs (row 3 of Table E1 in the PSR2 Basis Document). Safety Significance Levels (E2) is not directly applicable since this Global Issue can have a direct impact on nuclear safety, whereas E2 primarily relates to issues that are indirectly related to nuclear safety. Hence, the overall Safety Significance Level of 3 for deterministic considerations is dictated by the E1 categorization.

Regarding probabilistic considerations, the issue has Safety Significance Level 3 for Reactor

SECTION 4 - GI-37 PRIORITY DETERMINATION
Safety - Defence in Depth (F2). This is because the initiatives are related to BDBAs. Therefore, the fourth column of the top row of Table F2 of the PSR2 Basis Document is applicable, and the corresponding Safety Significance Level is 3 because the frequency range for BDBAs is less than 10^{-5}/y. Public Radiation Safety (F3) is assigned Safety Significance Level 4 on the basis that the public individual dose could be reduced by ~0.1 mSv for some events with initiating frequency ~ 10^{-5}/y. This Global Issue has no direct impact on the other probabilistic considerations, i.e., Core Damage Frequency (F1) (because Containment does not affect Core Damage Frequency), Plant Operability (F4), Occupational Radiation Safety (F5), Emergency Preparedness (F6) and Environment (F7). Therefore, these probabilistic considerations are not applicable, and the overall Safety Significance Level of 3 for probabilistic considerations is dictated by the F2 categorization. In summary, the overall Safety Significance Level is 3. OPG is actively progressing Phase 2 of the Emergency Mitigating Equipment project in support of extended operation at Pickering NGS.

SECTION 5 - GI-37 RESOLUTION PLAN	
GI-37-XRF-GI-40-RS1	Complete the planned Phase 2 EME implementation. This includes supplying cooling water, and power to essential loads via EME generators, to allow for operation of Air Cooling Units (ACUs) and Hydrogen Igniters [P-CORR-00531-04945, OPG Correspondence, *Pickering NGS – CNSC Action Item 2016-48-7470 Status Update on Emergency Mitigating Equipment and Telecommunications Projects*, February 16, 2017]. OPG is actively progressing this work in support of extended operation at Pickering NGS. (SF1-13) (SF1-AG15 Item (a))

B.38. GI-38 CSA N290.11 - Requirements for Reactor Heat Sinks

SECTION 1 - GI-38 GLOBAL ISSUE SUMMARY					
The goal of GI-38, CSA N290.11- Requirements for Reactor Heat Sinks, is to demonstrate an appropriate degree of conformance with the requirements of this standard. This Global Issue includes one SF1 code review Gap related to outage heat sink reliability requirements that is assessed as an Acceptable Deviation and one SF1 Additional Gap related to assessment of manual operator actions for outage heat sinks that is assessed as requiring No Further Action. Safety Significance Level 4 is assigned for this Global Issue based on deterministic and probabilistic defence-in-depth considerations and probabilistic core damage frequency considerations. The proposed Resolution Plan comprises an Acceptable Deviation based on the reliability of all outage heat sinks being integrated and assessed as part of the Outage Probabilistic Safety Assessments. Therefore, the reactor safety impact of not having individual heat sink design reliability requirements is not significant. The proposed Resolution Plan also comprises an item which requires No Further Action since the information and main assumptions regarding operator actions for heat sink operation, maintenance and recall are documented.					
Safety Significance Level:	4	**Category:**	Analytical	**Reassessment Beyond 2024:**	N

SECTION 2 - GI-38 ASSOCIATED GAPS	
SF1-17	Clause 5.6.1 of CSA N290.11-13 requires design reliability to be established for outage heat sinks. Although some emergency heat sinks (e.g., Emergency Boiler Water Supply and Emergency Water Supply) have design reliability requirements, design reliability requirements have not been established for all normal and back-up heat sinks used at Pickering. Reliability of all outage heat sinks (including those without explicit targets) is managed under the Risk & Reliability Program (both through unavailability models as well as through Probabilistic Safety Assessment), hence reactor safety impact is assessed and monitored. However, there is a PSR2 gap with respect to establishment of design reliability requirements for Pickering Units 1,4 and 5-8 outage heat sinks. ***Code Review N290.11-13*** **Associated Resolutions:** GI-38-AD1
SF1-AG20	An SF1 Gap related to the main assumptions and assessment methodology used when postulating a delay or error for manual actions to recall a heat sink during outage and maintenance was identified in [P-CORR-00531-05107, e-Doc 5305945, July 26, 2017]. **Associated Resolutions:** GI-38-NFA1

SECTION 3 – GI-38 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

This Global Issue contains two Gaps: one SF1 Gap on the CSA N290.11-13 requirement for specifying reliability targets for outage heat sinks, and one Additional Gap on the assessment methodology when postulating a delay or error for manual actions during outage and maintenance.

SECTION 4 – GI-38 PRIORITY DETERMINATION

	Deterministic Considerations			Probabilistic Considerations								
Safety Significance of Global Issue	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	**Overall Safety Significance Level**
	4	N/A	**4**	4	4	N/A	N/A	N/A	N/A	N/A	**4**	**4**

<u>Rationale:</u>

This Global Issue is related to the CSA N290.11 requirement on specifying reliability targets for outage heat sinks. As discussed in Section 5 of this Global Issue, the reliability of all outage heat sinks is assessed as part of the Outage Probabilistic Safety Assessments. Therefore, the reactor safety impact of not having explicit heat sink design reliability is assessed as not being a significant issue.

Regarding deterministic considerations, this Global Issue is assigned Safety Significance Level 4 for Defence in Depth (E1). This is because this issue is not significant by itself and it has no impact on the level of operational performance. Safety Significance Levels (E2) is not directly applicable since this Global Issue can have a direct impact on nuclear safety, whereas E2 primarily relates to issues that are indirectly related to nuclear safety. Hence, the overall Safety Significance Level of 4 for deterministic considerations is dictated by the E1 categorization.

Safety Significance Level 4 is assigned to probabilistic considerations Core Damage Frequency (F1) and Defence in Depth (F2). This is because, as mentioned above, Pickering takes an integrated approach to outage heat sink management and therefore, the impact on these considerations is not significant. This Global Issue has no direct impact on the other probabilistic considerations, i.e., Public Radiation Safety (F3), Plant Operability (F4), Occupational Radiation Safety (F5), Emergency Preparedness (F6) and Environment (F7). The overall Safety Significance Level of 4 for probabilistic considerations is dictated by the F1 and F2 categorizations.

In summary, the overall Safety Significance Level is 4.

SECTION 5 – GI-38 RESOLUTION PLAN	
GI-38-AD1	Unavailability targets for the shutdown cooling systems and for other systems/components that are part of the various heat sinks (e.g., EBWS and HT P&IC) are established and reported annually in the annual risk and reliability report [P-REP-09051.1-00016-R000, OPG Report, *Pickering NGS – 2016 Annual Risk and Reliability Report*, March 31, 2017]. The reliability of all outage heat sinks is integrated and assessed as part of the Outage Probabilistic Safety Assessments [P-REP-03611-00006-R000, *Pickering NGS PSA Update to Include Enhancements from the Fukushima Integrated Action Plan*, April 30, 2014]. Therefore the reactor safety impact of not having explicit heat sink design reliability is assessed, monitored and is not a significant issue. Due to the very low safety significance (Safety Significance Level 4), this is assessed as an Acceptable Deviation. (SF1-17)
GI-38-NFA1	Governance is in place that defines nuclear safety principles and requirements for management of reactor heat sinks [N-STD-OP-0025-R003, *Heat Sink Management,* October 23, 2014]. It ensures that alternate heat sinks are provided as defence-in-depth against loss of heat sink events by requiring availability of a back-up heat sink, for more probable failures of the primary heat sink, in order to prevent total loss of process heat sinks and emergency heat sinks, to mitigate consequences of a loss of process heat sink accident considered as part of the design or licensing basis (i.e., lower probability events). The outage heat sinks are managed according to the Shutdown Heat Sink operating manuals ([NK30-OM-5-04300, *Shutdown Heat Sinks Unit 5-8],* [NA44-OM-14-04300, *Shutdown Heat Sinks Units 1, 4*]) where each heat sink configuration is specified with the required operator actions, and the time allowed for each configuration is managed throughout the outage with the preparation of the daily Shutdown Heat Sinks Check Sheet ([P-FORM-10216-R018, *Shutdown Heat Sink Check Sheet PNGS-B*, May 1, 2017], [P-FORM-10308-R020, *Shutdown Heat Sinks Checksheet*, May 26, 2017]) which specifies the primary, backup and emergency heat sinks and the available time for operators to take action. The assessment for delay or error in manual actions and risk evaluation of human error involves use of the Outage Probabilistic Safety Assessments [P-REP-03611-00006-R000, *Pickering NGS PSA Update to Include Enhancements from the Fukushima Integrated Action Plan*, April 30, 2014] based on the details of the heat sink operating procedures. The human reliability methodology applied in the Outage PSA is documented in the OPG Outage PSA and At-Power PSA Guides ([N-GUID-03611-10001 Vol 4 R01, OPG Outage Probabilistic Risk Assessment (PRA) Guide – Level 1, July 9, 2010], [N-GUID-03611-10001 Vol 1 R04, OPG Probabilistic Risk Assessment (PRA) Guide – Level 1 (At-Power), October 24, 2014]). In the Outage PSA guide, the section on Human Reliability Analysis discusses how operator error associated with establishing the required mitigating system configuration within specified recall times is also a consideration for Outage PRA Human Interaction modeling, including operator actions for the manual initiation of mitigating systems during shutdown (i.e., operator action to establish back-up and emergency heat sinks). Therefore, the information and main assumptions regarding operator actions for heat sink operation, maintenance and recall are documented. No further action is required. (SF1-AG20)

B.39. GI-39 CSA N290.14 - Qualification of Digital Hardware and Software for Use in I&C Applications

SECTION 1 - GI-39 GLOBAL ISSUE SUMMARY					
The goal of GI-39, CSA N290.14 - Qualification of Digital Hardware and Software for Use in I&C Applications, is to assess and resolve the lack of a review of legacy Real-Time Process Computing (RTPC) software against CSA N290.14. This Global Issue comprises one Acceptable Deviation addressing one SF1 code review Gap [CSA N290.14-15]. GI-39 is Safety Significance Level 4 based on deterministic defence-in-depth considerations. OPG has an RTPC program, Software [N-PROG-MP-0006 R010, *Software*, June 2, 2017], in place that adequately deals with both legacy and new RTPC software installations in a manner that meets applicable standards and the intent of the CSA standard. The program identifies the processes and overall requirements for an effective Software Program that supports safe and efficient plant operation. Evaluation of the legacy software installations with respect to the CSA N290.14-15 requirements is not practicable and would provide very little safety benefit. On this basis, and given its very low Safety Significance, this issue is assessed as an Acceptable Deviation.					
Safety Significance Level:	4	**Category:**	Analytical	**Reassessment Beyond 2024:**	N

SECTION 2 - GI-39 ASSOCIATED GAPS	
SF1-19	Correspondence with the CNSC identifies all of the software application qualifications for software Categories 1, 2 and 3 from January 1, 2007 to the time of the correspondence (June 2016). However, an evaluation of legacy Real-Time Process Computing applications with respect to the requirements of N290.14-15 for Categories 1, 2 and 3 software has not been performed. Therefore, this has been identified as a PSR2 gap. ***Code Review N290.14-15*** **Associated Resolutions:** GI-39-AD1

SECTION 3 - GI-39 BACKGROUND INFORMATION AND RESOLUTION STRATEGY
This Global Issue contains one SF1 Gap related to CSA N290.14-15. Compliance with CSA-N290.14 is not currently a licence requirement for Pickering NGS (in accordance with PROL 48.03/2018) per the R05 Pickering Licence Conditions Handbook. The gap (SF1-19) indicates that the extent or type of qualification of legacy Real-Time Process Computing (RTPC) applications is not identified.

SECTION 4 - GI-39 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								Overall Safety Significance Level
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	4	N/A	**4**	N/A	N/A	N/A	N/A	N/A	N/A	N/A	**N/A**	**4**

Rationale:

This Global Issue is related to evaluation of legacy Real-Time Process Computing applications with respect to the requirements of CSA N290.14-15. As discussed in Section 5 of this Global Issue, all OPG Nuclear software installations are subject to the OPG software program documents. The legacy software applications have decades of successful service, which indicates the adequacy of the current OPG approach and the programmatic implementation.

Regarding deterministic considerations, this Global Issue is assigned Safety Significance Level 4 for Defence in Depth (E1), since this issue is not significant by itself and it has no impact on the level of operational performance. Safety Significance Levels (E2) is not directly applicable since E2 primarily relates to issues that are not directly related to nuclear safety. Hence, the overall Safety Significance Level of 4 for deterministic considerations is dictated by the E1 categorization.

This Global Issue has no direct impact on the probabilistic considerations, i.e., Core Damage Frequency (F1), Defence in Depth (F2), Public Radiation Safety (F3), Plant Operability (F4), Occupational Radiation Safety (F5), Emergency Preparedness (F6) and Environment (F7). Therefore, the probabilistic considerations are not directly applicable to this issue.

In summary, the overall Safety Significance Level is 4. The successful service of legacy software applications has been demonstrated.

SECTION 5 - GI-39 RESOLUTION PLAN

GI-39-AD1	OPG has a RTPC program in place that adequately deals with both legacy and new RTPC software installations in a manner that meets the intent of the CSA standard and other standards (e.g., International Standards Organization (ISO)/International Electrotechnical Commission (IEC)). The OPG program document N-PROG-MP-0006 [N-PROG-MP-0006 R010, *Software*, June 2, 2017] identifies the processes and overall requirements for an effective Software Program that supports safe and efficient plant operation and that meets the intent

SECTION 5 - GI-39 RESOLUTION PLAN	
	of CSA-N290.14 and other standards. This program applies to software classified as Real-Time Process Computing (RTPC) and Scientific, Engineering and Safety Analysis (SESA) Software or Software Engineering Tools in OPG Nuclear. This includes embedded RTPC software installation in equipment and components (see OPG procedures N-PROC-MP-0099 [N-PROC-MP-0099 R004, OPG Nuclear Procedure, *Development of Real-Time Process Computing Systems*, April 2016] and N-PROC-MP-0049 [N-PROC-MP-0049 R009, OPG Nuclear Procedure, *Procurement of Software and Products Containing Software*, June 2016]). These documents identify: Processes and overall requirements for classification of software. Governing standards for each software categorization defining requirements for software development, maintenance, procurement, qualification, use and retirement and includes the security of RTPC critical cyber assets. All OPG Nuclear software installations are subject to the OPG software program documents. The legacy software applications have decades of successful service, which indicates the adequacy of the current OPG approach and the programmatic implementation. Evaluation of the legacy software installations with respect to the N290.14-15 requirements is not practicable and would provide very little safety benefit. On this basis, and given the very low safety significance (Safety Significance Level 4), this issue is assessed as an Acceptable Deviation. (SF1-19) This Acceptable Deviation also includes/addresses GI-44.

B.40. GI-40 Accident Management

SECTION 1 - GI-40 GLOBAL ISSUE SUMMARY					
The goal of GI-40, Accident Management, is to ensure that the Emergency Mitigating Equipment (EME) Phase 2 activities are completed. This Global Issue comprises one proposed Resolution Statement and a Cross Reference to GI-27 that address two Gaps: one SF1 code review Gap [REGDOC-2.3.2] and one SF1 Additional Gap. GI-40 is Safety Significance Level 3 based on deterministic and probabilistic defence-in-depth considerations. The proposed Resolution Plan primarily comprises activities to ensure completion of the EME Phase 2 project. In addition, a Cross Reference to GI-27 is included because further mitigation of this issue is provided through the risk improvement initiatives referred to in GI-27. Completion of the proposed Resolution Plan will support and strengthen Level 3 and Level 4 defence-in-depth for the extended operating period.					
Safety Significance Level:	3	**Category:**	Engineering	**Reassessment Beyond 2024:**	N

SECTION 2 - GI-40 ASSOCIATED GAPS	
SF1-33	Full provision of Complementary Design Features for Containment integrity as required by Clause 4.2.1 of REGDOC-2.3.2 will be addressed with the completion of Phase 2 Emergency Mitigating Equipment. This work is currently scheduled to be fully implemented by the end of 2017. Since this work is still in progress, it has been identified as a PSR2 gap. ***Code Review CNSC REGDOC-2.3.2*** **Associated Resolutions:** GI-27-RS2, GI-40-RS1
SF1-AG4	An SF1 Gap related to the submission of updated information in response to CNSC observations made during a SAMG drill involving Emergency Mitigating Equipment was identified in [P-CORR-00531-05107, e-Doc 5305945, July 26, 2017]. **Associated Resolutions:** GI-40-RS1

SECTION 3 - GI-40 BACKGROUND INFORMATION AND RESOLUTION STRATEGY
This Global Issue contains one SF1 Gap related to accident management and one SF1 Additional Gap related to EME implementation. The Gap is completion of Phase 2 of the Emergency Mitigating Equipment (EME) project. EMS and RMS requirements related to CSA N290.3 under GI-37 are addressed through the implementation of Phase 2 EME for Pickering 1,4 and Pickering 5-8. When credit is taken for Phase 2 EME modifications in the station PSAs, the more stringent Administrative Safety Goals for Large

SECTION 3 - GI-40 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

Release Frequency are not presently achieved for Pickering 1,4. Therefore, to satisfy Administrative Safety Goals for Large Release Frequency for Pickering 1,4, OPG has committed to implement additional design, operational and/or analytical enhancements through GI-27-RS2. Implementation of these modifications focuses on prevention of accident progression, by providing a redundant, independent supply of makeup water to the Heat Transport System, Steam Generators, and Calandria for Pickering 1,4. This will provide additional fuel cooling capability, which increases the likelihood of In-Vessel Retention. Per reference [P-REP-09013-00002 R001, *Pickering NGS – Beyond Design Basis Containment Integrity*, January 2014], In-Vessel Retention supports maintenance of Containment integrity following a Beyond Design Basis Accident, allowing the existing FADS to be used.

SECTION 4 - GI-40 PRIORITY DETERMINATION

Safety Significance of Global Issue	**Deterministic Considerations**			**Probabilistic Considerations**								**Overall Safety Significance Level**
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	3	N/A	**3**	N/A	3	N/A	N/A	N/A	N/A	N/A	**3**	**3**

Rationale:

Similar to GI-37, this Global Issue is related to the completion of a number of initiatives, including Phase 2 of the Emergency Mitigating Equipment project, to minimize radiological releases and enhance Containment protection for BDBAs.

Regarding deterministic considerations, this Global Issue is assigned a Safety Significance Level 3 for Defence in Depth (E1) on the basis that the issue is related to minimizing radiological releases for BDBAs (row 3 of Table E1 in the PSR2 Basis Document). Safety Significance Levels (E2) is not directly applicable since this Global Issue can have a direct impact on nuclear safety, whereas E2 primarily relates to issues that are indirectly related to nuclear safety. Hence, the overall Safety Significance Level of 3 for deterministic considerations is dictated by the E1 categorization.

Regarding probabilistic considerations, the issue has a Safety Significance Level of 3 for Reactor Safety - Defence in Depth (F2). This is because the initiatives are related to BDBAs (frequency less than 10^{-5}/y).

This Global Issue has no direct impact on the other probabilistic considerations, i.e., Core Damage

SECTION 4 - GI-40 PRIORITY DETERMINATION

Frequency (F1), Public Radiation Safety (F3), Plant Operability (F4), Occupational Radiation Safety (F5), Emergency Preparedness (F6) and Environment (F7). Therefore, these probabilistic considerations are not directly applicable, and the overall Safety Significance Level of 3 for probabilistic considerations is dictated by the F2 categorization.

In summary, the overall Safety Significance Level is 3. OPG is actively progressing Phase 2 of the Emergency Mitigating Equipment project in support of extended operation at Pickering NGS.

SECTION 5 - GI-40 RESOLUTION PLAN

GI-40-RS1	Complete the planned Phase 2 EME implementation. This includes supplying cooling water, and power to essential loads via EME generators, to allow for operation of Air Cooling Units (ACUs) and Hydrogen Igniters [P-CORR-00531-04945, OPG Correspondence, *Pickering NGS – CNSC Action Item 2016-48-7470 Status Update on Emergency Mitigating Equipment and Telecommunications Projects*, February 16, 2017]. OPG is actively progressing this work in support of extended operation at Pickering NGS. (SF1-33) (SF1-AG4) This Resolution Statement includes/addresses GI-37 and is related to GI-27.
GI-40-XRF-GI-27-RS2	Further mitigation of the requirement for Complementary Design Features for Containment integrity is provided through the risk improvement initiatives referred to in GI-27. (SF1-33)

B.41. GI-41 REGDOC-2.10.1 - Nuclear Emergency Preparedness and Response

SECTION 1 - GI-41 GLOBAL ISSUE SUMMARY					
The goal of GI-41, REGDOC-2.10.1 - Nuclear Emergency Preparedness and Response, is to confirm that OPG governance is in compliance with CNSC REGDOC-2.10.1, Nuclear Emergency Preparedness and Response. This Global Issue comprises one item assessed as requiring No Further Action addressing one SF13 code review Gap [REGDOC-2.10.1]. GI-41 is Safety Significance Level 4 based on deterministic safety significance levels, probabilistic defence-in-depth and probabilistic emergency preparedness considerations. OPG's Consolidated Nuclear Emergency Plan [N-PROG-RA-0001 R015] was revised to address the two specific issues identified in the gap, which are to ensure that the evacuation time estimates and KI pill programs will be sustained. Because this addresses the code review Gap, the issue is assessed as requiring No Further Action.					
Safety Significance Level:	4	**Category:**	Programmatic	**Reassessment Beyond 2024:**	N

SECTION 2 - GI-41 ASSOCIATED GAPS	
SF13-1	OPG has completed a gap analysis for transition to CNSC REGDOC-2.10.1 and has developed an action plan to achieve compliance. The transition plan that OPG has committed in order to bring Darlington into compliance with REGDOC-2.10.1 applies across the nuclear fleet and will also bring Pickering into compliance. Updating OPG governance to ensure that the Pickering Evacuation Time Estimate study is maintained and to define how the Potassium Iodide (KI) pill program will be sustained is in progress. As these two actions are not yet complete, this is identified as a PSR2 gap. ***Code Review CNSC REGDOC-2.10.1*** • **Note – Per GI-41-NFA1, this gap has been fully addressed.** **Associated Resolutions:** GI-41-NFA1

SECTION 3 - GI-41 BACKGROUND INFORMATION AND RESOLUTION STRATEGY
This Global Issue contains one SF13 gap related to CNSC REGDOC-2.10.1. This gap identifies required updates of OPG governance to bring OPG's Nuclear fleet in compliance with REGDOC-2.10.1 (2014). The specified required actions are to maintain Pickering Evacuation Time Estimates (ETE) and to define how the KI pill program will be sustained. The transition plan for REGDOC-2.10.1 was prepared for Darlington and was subsequently provided to the CNSC in [NK38-CORR-00531-17593, OPG Correspondence, *Darlington NGS – Transition Plan for Regulatory Document Nuclear Emergency Preparedness and Response (REGDOC–2.10.1)*,

SECTION 3 - GI-41 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

September 30, 2015]. Regulatory Management Action Request AR # 28184526 was initiated to track completion of the transition plan.

- AR 28184526-06 was closed with a complete description of the ETE update process followed in 2015/16.
- AR 28184526-09 was closed with a complete description of the KI procurement and distribution process followed in 2015, and the on-going KI availability process developed and maintained.

OPG's nuclear emergency plan was revised [N-PROG-RA-0001 R015, OPG Nuclear Program, *Consolidated Nuclear Emergency Plan*, November 2016] to incorporate the required information, as follows:

> "OPG shall assist the province and designated municipalities in their planning and preparedness for a nuclear emergency, and collaborate with them to:
>
> (a) Develop and maintain public evacuation time estimates based on current census data, and future population growth projections on a per-decade estimation."
>
> "In consultation with the designated municipalities, OPGN EP Department shall procure stable iodine tablets and maintain them within expiry dates on behalf of the nuclear sites. Distribution of iodine tablets is the responsibility of Durham Region and City of Toronto, with the support of OPG.
>
> Initial distribution of stable iodine tablets to residences, businesses and institutions within the Pickering and Darlington Primary Zones was completed in 2015. The program established and maintained by the designated municipalities and OPG ensures continued availability and that information is available to the general public, including online."

SECTION 4 - GI-41 PRIORITY DETERMINATION

Safety Significance of Global Issue	**Deterministic Considerations**			**Probabilistic Considerations**								**Overall Safety Significance Level**
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	N/A	4	**4**	N/A	4	N/A	N/A	N/A	4	N/A	**4**	**4**

SECTION 4 - GI-41 PRIORITY DETERMINATION

Rationale:

This Global Issue is related to required updates of OPG governance to bring Pickering NGS into compliance with REGDOC-2.10.1 (2014). It is related to maintaining Pickering Evacuation Time Estimates (ETE) and to defining how the KI pill program will be sustained.

Regarding deterministic considerations, this Global Issue is not associated with a physical barrier, so Defence in Depth (E1) is not applicable. Safety Significance Levels (E2) is assigned Safety Significance Level 4 since resolution of this issue "may help identify areas that need more attention", in this case the adequacy of governance implementation or its effectiveness. Therefore, a Safety Significance Level of 4 is selected for deterministic considerations.

Safety Significance Level 4 is assigned to probabilistic considerations Defence in Depth (F2) and Emergency Preparedness (F6). This Global Issue pertains to defence–in-depth Level 5 but is expected to have insignificant impact since the issue pertains to sustaining provisions that are already in place. Similarly, for Emergency Preparedness (F6), the Safety Significance Level is 4 because the KI pill program is already in place.

For probabilistic considerations, this Global Issue has no impact on Core Damage Frequency (F1), nor does it impact Public Radiation Safety (F3), Plant Operability (F4), Occupational Radiation Safety (F5), or Environment (F7). Therefore, these other probabilistic considerations are not applicable.

In summary, the overall Safety Significance Level is 4. As indicated in Section 5 of this Global Issue, the OPG nuclear emergency preparedness governance [N-PROG-RA-0001 R015, OPG Nuclear Program, *Consolidated Nuclear Emergency Plan*, November 2016] was revised to ensure that the evacuation time estimates and KI pill programs will be sustained.

SECTION 5 - GI-41 RESOLUTION PLAN

GI-41-NFA1	The OPG nuclear emergency preparedness governance [N-PROG-RA-0001 R015, OPG Nuclear Program, *Consolidated Nuclear Emergency Plan*, November 2016] was revised to ensure that the evacuation time estimates and KI pill programs will be sustained. No further action is required. (SF13-1)

B.42. GI-42 Examination and Testing Requirements for Design of Concrete Containment Structures

SECTION 1 - GI-42 GLOBAL ISSUE SUMMARY					
The goal of GI-42, Examination and Testing Requirements for Design of Concrete Containment Structures, is to ensure that the Concrete Containment Structures (CCS) remain fit for service for the extended operating period. This Global Issue comprises one Acceptable Deviation addressing one SF1 code review Gap [CSA N287.5-11]. GI-42 is Safety Significance Level 3 based on deterministic defence-in-depth considerations. The CCSs at Pickering A and B were built and tested to meet the 1965 and 1970 National Building Code of Canada requirements, respectively. Ongoing confirmation that the Pickering NGS CCSs remain fit for service is demonstrated via periodic and in-service inspections conducted in accordance with the requirements of CSA N285.5 and CSA N287.7. In addition, the Engineering Change Control program [N-PROG-MP-0001] ensures that any design changes made to the Pickering CCSs will comply with CSA N287.5 going forward. The proposed Resolution Plan assesses the Gap in demonstrating that the CCS design meets CSA N287.5-11 as an Acceptable Deviation because of its very low Safety Significance and because it is being addressed outside of PSR2. The controls in place at the time of construction and the ongoing controls in place for inspections, aging management and modifications adequately meet the intent of CSA N287.5-11.					
Safety Significance Level:	3	**Category:**	Programmatic	**Reassessment Beyond 2024:**	N

SECTION 2 - GI-42 ASSOCIATED GAPS	
SF1-6	The Concrete Containment Structures (CCSs) at Pickering A and B were built and tested to meet the 1965 and 1970 National Building Code of Canada requirements, respectively, prior to the initial issuance of CSA N287.5. No assessments exist which demonstrate that the requirements in effect during construction of Pickering NGS CCSs comply with the requirements of CSA N287.5. Ongoing confirmation that the Pickering NGS CCSs remain fit for service is demonstrated via periodic and in-service inspections conducted in accordance with the requirements of CSA N285.5 and N287.7, and the resultant inspection reports attest to the quality of the design. In addition, the Engineering Change Control process ensures that that any design changes made to the Pickering CCSs will comply with N287.5 going forward, as applicable. The original Pickering construction included requirements for tests and quality control procedures which generally meet the intent of N287.5. Furthermore, retroactive application of N287.5 to the as-built design of CCSs cannot be practically achieved without rebuilding them. Nevertheless, there is a PSR2 gap for Pickering NGS given that compliance with the specific requirements of N287.5 has not been demonstrated.

SECTION 2 - GI-42 ASSOCIATED GAPS

Code Review N287.5-11

Associated Resolutions: GI-42-AD1

SECTION 3 - GI-42 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

This Global Issue contains one SF1 Gap related to the examination and testing requirements for the design of Concrete Containment Structures (CCSs).

CSA N287.5 is identified in Appendix E.1 of the R05 Pickering Licence Conditions Handbook as *Guidance or Criteria* and is relevant to Section 6.1, Design Program, of the Licence Conditions Handbook. Gap SF1-6 is related to Containment concrete structure design requirements. GI-19 addresses N287.7, in-service examination and testing requirements for Containment Concrete Structures.

SECTION 4 - GI-42 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								Overall Safety Significance Level
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	3	N/A	**3**	N/A	N/A	N/A	N/A	N/A	N/A	N/A	**N/A**	**3**

<u>Rationale:</u>

This Global Issue is related to Containment Concrete Structure design requirements. Specifically, the identified gap states that no assessments exist which demonstrate that the requirements in effect during construction of Pickering NGS Containment Concrete Structures comply with the requirements of CSA N287.5.

Regarding deterministic considerations, this Global Issue is assigned a Safety Significance Level 3 for Defence in Depth (E1) on the basis that the issue is related to but does not impact Containment capability (row 3 of Table E1 in the PSR2 Basis Document). This is because, as indicated in Section 5 of this Global Issue, the controls in place at the time of construction and the ongoing controls in place

SECTION 4 - GI-42 PRIORITY DETERMINATION

for inspections, aging management and modifications adequately meet the intent of CSA N287.5-11. Safety Significance Levels (E2) is not directly applicable since this Global Issue can have a direct impact on nuclear safety, whereas E2 primarily relates to issues without a direct impact on nuclear safety. Hence, the overall Safety Significance Level of 3 for deterministic considerations is dictated by the E1 categorization.

For probabilistic considerations, this Global Issue has no impact on Core Damage Frequency (F1) or Defence in Depth (F2), nor does it impact Public Radiation Safety (F3), Plant Operability (F4), Occupational Radiation Safety (F5), Emergency Preparedness (F6) or Environment (F7). Therefore, probabilistic considerations are not applicable.

In summary, the overall Safety Significance Level is 3. The issue is indirectly addressed by the fact that continuing compliance with CSA N287.7-08 is provided by the existing Containment Periodic Inspection Program and testing programs.

SECTION 5 - GI-42 RESOLUTION PLAN

GI-42-AD1	The Concrete Containment Structures (CCSs) at Pickering A and B were built and tested to meet the 1965 and 1970 National Building Code of Canada requirements, respectively. The standards that applied during original construction of Pickering NGS included requirements for tests and quality control procedures to ensure that the concrete used in the as-built structures met the original design requirements. For Pickering Units 5-8, the original Pickering Concrete Placing and Workmanship Specification [L-715-80, *Pickering Generating Station B L-715-80 Specification for Concrete Placing and Workmanship*] and Tendering and Contract Document [T-NK30-20541-01, *Pickering Generating Station B, Tendering and Contract Documents NK30-LH-20541-01 for Supply of Pre-Mix Concrete in Ready Mix Trucks*, April 19, 1974], included requirements for quality control and compliance with CSA A23.1, A23.2 and A23.3 (which address concrete materials, methods of concrete construction and test methods and standard practices for concrete). For Pickering Units 1,4, the concrete structures were built and tested to meet the 1965 NBCC requirements and associated CSA A23 Series Standards, supplemented by specific loading requirements and the requirements in the Design Manuals. Ongoing confirmation that the Pickering NGS CCSs remain fit for service is demonstrated via periodic and in-service inspections conducted in accordance with the requirements of CSA N285.5 and N287.7, and the resultant inspection reports attest to the quality of the design. In addition, the Engineering Change Control process ensures that any design changes made to the Pickering CCSs will comply with N287.5 going forward. The original Pickering construction included requirements for tests and quality control procedures which generally meet the intent of N287.5. The controls in place at the time of construction and the ongoing controls in place for inspections, aging management and modifications adequately meet the intent of CSA N287.5-11. This issue is of low safety significance (Safety Significance Level 3) and has been addressed to the extent practicable. Therefore this issue is assessed as an Acceptable Deviation. (SF1-6)

B.43. GI-43 Safety-Related Structures (Non-Containment) for Nuclear Power Plants

SECTION 1 - GI-43 GLOBAL ISSUE SUMMARY					
The goal of GI-43, Safety-Related Structures (Non-Containment) for Nuclear Power Plants, is to ensure fitness for service of non-Containment Safety-Related Structures, and to demonstrate conformance to applicable codes and standards for the extended operation period. This Global Issue comprises three proposed Resolution Statements, one Acceptable Deviation and one Cross-Reference to GI-19, which address seven Gaps: three SF1 code review Gaps [CSA N291-15], one SF2 review task Gap, one SF4 code review Gap [CSA N287.7-08] and two COP Review Gaps. GI-43 is Safety Significance Level 3 based on deterministic and probabilistic defence-in-depth considerations. The proposed Resolution Plan primarily comprises activities to perform inspections, develop a risk-based approach in aging management governance, and prepare condition assessments for the extended operating period, for non-Containment safety-significant civil structures. These activities address two of the three SF1 Gaps, and the SF2 and SF4 Gaps. The two COP Review Gaps, which are related to GI-19, FFS of Containment for the Extended Operating Period, are addressed by the proposed Resolution Statement for GI-19. The remaining SF1 Gap is assessed as an Acceptable Deviation since it is being managed and has low Safety Significance. Completion of the proposed Resolution Plan will support and strengthen Level 1 defence-in-depth for the extended operating period.					
Safety Significance Level:	3	**Category:**	Engineering, Programmatic, Analytical	**Reassessment Beyond 2024:**	Y

SECTION 2 - GI-43 ASSOCIATED GAPS	
SF4-13	Actions #31, #32, and #33 from the Pickering Units 5-8 Continued Operations Plan are related to N287.7 and although complete, need to be reassessed for Pickering operation past 2020. (IIP Action #31 involved submission of Periodic Inspection Plans and Life Cycle Management Plans for a number of safety-significant civil structures. IIP Action #32 involved submission of Aging Management Plans for Concrete Containment Structures to the CNSC for acceptance. IIP Action #33 involved revising the Reactor Building Periodic Inspection Plan and submitting to the CNSC for acceptance.) ***Code Review N287.7-08*** <u>Note:</u> Only Action #31 is covered under this Global Issue. **Associated Resolutions:** GI-43-RS1, GI-43-RS2, GI-43-RS3
SF1-20	Clause 6.5.2.2 of CSA N291-15 imposes new requirements for bolted connections in members that are part of the seismic load resisting system. Pickering NGS structures were not explicitly designed to meet these requirements and this is therefore identified as a PSR2

SECTION 2 - GI-43 ASSOCIATED GAPS	
	gap. ***Code Review N291-15*** **Associated Resolutions:** GI-43-AD1
SF1-21	Clause 9 of CSA N291-15 contains new requirements related to aging management (including design provisions to account for aging) that are not in CSA N291-08 and that may have significance for operation of Pickering beyond 2020. Pickering structures were not explicitly designed to meet these requirements and this is therefore identified as a PSR2 gap. ***Code Review N291-15*** **Associated Resolutions:** GI-43-RS1, GI-43-RS2, GI-43-RS3
SF1-22	Clauses 6.1.1(b) and 6.9.2.1.4 of CSA N291-15 state requirements for aspects of the design that are specifically based on the plant service life. Pickering structures were not explicitly designed or assessed in relation to the requirements of these clauses for operation beyond 2020. This is identified as a PSR2 gap. ***Code Review N291-15*** **Associated Resolutions:** GI-43-RS1, GI-43-RS2, GI-43-RS3
SF2-11	Condition Assessments for civil structures are not complete for station operation to 2028[33]. ***Review Task #1 Actual Condition of SSCs*** **Associated Resolutions:** GI-43-RS1, GI-43-RS2, GI-43-RS3
COP-4	Work has not been completed to demonstrate adequate margin to operate the Pickering High Pressure Emergency Coolant Injection storage tank foundation piles by performing engineering analysis of loss of thickness due to corrosion, for the extended operation period and for the period until the ECIS is no longer required. ***Pickering PSR2 Gap COP-4*** **Associated Resolutions:** GI-19-RS1
COP-25	An assessment of margin to operate all the Pickering Reactor Building foundations has not been completed for the period of extended operation and until Reactor Building integrity can be demonstrated to no longer be required. This issue applies also to the Vacuum Building and Pressure Relief Duct for the extended operation period and for the period until the Negative Pressure Containment System integrity can be demonstrated to no longer be required. ***Pickering PSR2 gap COP-25*** Note 1: Recently CNSC requested OPG (CD# P-CORR-00531-04901 November 28, 2016) to re-consider COP actions F06 and I15-6B with an expanded scope to include the foundation

SECTION 2 - GI-43 ASSOCIATED GAPS

piles supporting the Pickering A Reactor Building, Vacuum Building and Pressure Relief Ducts as part of PSR2, to demonstrate whether the foundation steel H-piles at the Pickering site will withstand their design loads for all civil structures that they support for operation beyond 2020. OPG's path forward was communicated to CNSC staff in [P-CORR-00531-04896 *Pickering NGS: Continued Operations Plan (COP) Actions F06 and I15-6B -Periodic Safety Review Reassessment for Operation Beyond 2020*, January 23, 2017] where it was indicated that PSR2 will consider the expanded scope of COP actions F06 and I15-6B. Additional CNSC feedback was provided in [P-CORR-00531-04973*, Pickering NGS: CNSC Staff Review of OPG's Reassessment of COP Actions for Consideration in the PSR2,* February 24, 2017].

Note 2:

Non-Containment foundation H-piles are addressed in this Global Issue (GI-43).

Associated Resolutions: GI-19-RS1

SECTION 3 - GI-43 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

This Global Issue contains a total of seven Gaps (three SF1 Gaps, one SF2 Gap, one SF4 Gap and two COP Review Gaps) related to non-Containment safety related structures.

Containment structures are addressed in GI-19.

SF4-13 is shared by this Global Issue and GI-19. The relevant portion of SF4-13 to this Global Issue (GI-43) is Action #31 "Include the periodic inspection programs and LCMPs for the Safety-significant civil structures that are under the scope of CSA N291-08, but not covered by the N287.7 standard". Actions #32 and #33 are related only to Containment structures and are addressed in GI-19.

With respect to COP-25, CNSC staff requested OPG [P-CORR-00531-04901, CNSC Correspondence e-Doc 5129748, *Pickering NGS: Closure of Actions F06 and I15-6b of the Continued Operations Plan (COP)*, November 28, 2016] to re-consider COP actions F06 and I15-6B with an expanded scope to demonstrate whether the foundation steel H-piles at the Pickering site will withstand their design loads for all civil structures that they support for operation beyond 2020. OPG's path forward was communicated to the CNSC in [P-CORR-00531-04896, OPG Correspondence, *Pickering NGS: Continued Operations Plan (COP) Actions F06 and I15-6B - Periodic Safety Review Reassessment for Operation Beyond 2020*, January 23, 2017] where it was indicated that PSR2 will address the expanded scope of COP actions F06 and I15-6B. Additional CNSC feedback was provided in [P-CORR-00531-04973, CNSC Correspondence e-Doc 5189874, *Pickering NGS: CNSC Staff Review of OPG's Reassessment of COP Actions for Consideration in the PSR2*, February 24, 2017]. Consideration of the integrity of H-piles supporting Reactor Buildings, Pressure Relief Duct, and the Vacuum Building structures is addressed in GI-19 while that for H-piles supporting the non-Containment civil structures is addressed in this Global Issue (GI-43).

For non-Containment safety-significant Civil Structures, a Preventive Maintenance program (PM 00121151) has been established. The list of structures included is identified in Memorandum [P-CORR-20000-0608706, OPG Memorandum, *Pickering NGS Inspection Criteria for Non-Containment Buildings and Structures (including safety-related structures and components)*, August 17, 2016]. An

SECTION 3 - GI-43 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

initial inspection will be managed and performed by Pickering Field Engineering in order to identify any degradation. After the initial inspection, the structures and components will be inspected a minimum of once every 5 years. Subsequent inspections will be managed by Station Engineering. The initial inspection will cover all accessible areas, make recommendations for inspections of inaccessible areas, and identify areas that should receive additional examination in future inspections. The first round of these inspections for structures is planned to be completed in 2017 and the inspection results will be documented in an Inspection & Test Plan.

Preventive Maintenance and As-Found Condition results are inputs to Condition Assessments.

There is a requirement to develop Condition Assessments for Safety-Significant Civil Structures. OPG is enhancing its methodology to address Aging Management of non-Containment Safety-Significant Civil Structures. Execution of this methodology will identify critical structures, and establish a means for scoping of non-Containment Safety-Significant Civil Structures for Aging Management consideration.

SECTION 4 - GI-43 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								Overall Safety Significance Level
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	3	N/A	**3**	N/A	3	N/A	N/A	N/A	N/A	N/A	**3**	**3**

Rationale:

This Global Issue is related to the fitness for service of safety-related structures (Non-Containment).

Regarding deterministic considerations, this Global Issue is assigned a Safety Significance Level 3 for Defence in Depth (E1) on the basis that the issue is related to civil structures associated with safety functions for DBAs (row 3 of Table E1 in the PSR2 Basis Document). Safety Significance Levels (E2) is not directly applicable, since this Global Issue can have a nuclear safety impact, whereas E2 primarily relates to issues without a direct nuclear safety impact. Hence, the overall Safety Significance Level of 3 for deterministic considerations is dictated by the E1 categorization.

Regarding probabilistic considerations, the issue has Safety Significance Level 3 for Reactor Safety - Defence in Depth (F2). This is because the issue is related to the reliability of structures important to

SECTION 4 - GI-43 PRIORITY DETERMINATION
safety for DBAs (initiating frequency less than 10^{-3}/y) and BDBAs. The other probabilistic considerations, i.e., Core Damage Frequency (F1), Public Radiation Safety (F3), Plant Operability (F4), Occupational Radiation Safety (F5), Emergency Preparedness (F6) and Environment (F7), are not directly applicable. Hence, the overall Safety Significance Level of 3 for probabilistic considerations is dictated by the F2 categorization. In summary, the overall Safety Significance Level is 3. OPG has established a Preventive Maintenance program (PM 00121151) and is also currently progressing this issue by using a risk based approach for aging management of safety-significant civil structures for the extended operating period.

SECTION 5 - GI-43 RESOLUTION PLAN	
GI-43-RS1	Perform the scope of inspections for non-Containment safety-significant civil structures as per the established Preventive Maintenance program (PM 00121151). (SF1-21) (SF1-22) (SF2-11) (SF4-13 Action #31)
GI-43-RS2	Develop program governance using a risk based approach for aging management of safety-significant civil structures for the extended operating period. This applies to non-Containment Safety-Related Civil Structures. (SF1-21) (SF1-22) (SF2-11) (SF4-13 Action #31)
GI-43-RS3	Prepare Condition Assessments as appropriate for safety-significant civil structures for the extended operating period. Recommendations from these Condition Assessments will be tracked and reported along with those related to GI-8. This applies to non-Containment Safety-Related Civil Structures. (SF1-21) (SF1-22) (SF2-11) (SF4-13 Action #31) This Resolution Statement also includes/addresses GI-22.
GI-43-AD1	An assessment of this gap (SF1-20) was completed [P-CORR-03680-0620823, *Re: Resolution for Pickering Periodic Safety Review 2 (PSR2) Global Issue #GI-43 Gap SF1-20*, June 30, 2017]. The original code requirements for PNGS safety related structures [National Building Code of Canada] and the Seismic Margin Assessments (SMA) include requirements of CSA S16 “Design of Steel Structures”, which is the basis for the requirements for bolted connections of steel structures in CSA N291-15. Given that the SMA methodology and the plant design both include requirements of S16/S16.1 (predecessor of S16), the plant meets or has been assessed to these requirements. Structures on seismic success paths have been qualified and/or assessed to meet their performance requirements. In addition, SMA-based Probabilistic Safety Assessments for Pickering Units 1,4 and 5-8 [NA44-REP-03611-00022 R000, *PRA-Based Seismic Margin Assessment of PNGS-A*, January 2014], [NK30-REP-03611-00013 R001, *PRA-Based Seismic Margin Assessment of PNGS-B*, April 2015] demonstrate that risk associated with seismic hazard is sufficiently low. As it would not be practicable to make changes to the bolted steel connections in the existing structural design, and given the low safety significance (Significance Level 3), this

SECTION 5 - GI-43 RESOLUTION PLAN	
	is assessed as an Acceptable Deviation. (SF1-20)
GI-43-XRF-GI-19-RS1	Demonstrate the FFS of the foundation steel H-piles for the Pickering A Reactor Building, Vacuum Building and Pressure Relief Duct at the Pickering site for the extended operating period. (COP-4) (COP-25)

B.44. GI-44 REGDOC-2.5.2 - Design of Reactor Facilities: Nuclear Power Plants

SECTION 1 - GI-44 GLOBAL ISSUE SUMMARY					
The goal of GI-44, REGDOC-2.5.2 – Design of Reactor Facilities: Nuclear Power Plants, is to demonstrate an appropriate degree of conformance with CNSC REGDOC-2.5.2 given that the document applies to new reactors. This Global Issue comprises 10 Acceptable Deviations and one Cross-Reference to GI-39 which address 13 Gaps: ten code review Gaps (eight SF1 Gaps, one SF5 Gap and one SF6 Gap) and three SF1 Additional Gaps. Specifically, the Gaps are related to REGDOC-2.5.2 requirements for deterministic safety analysis, human factors, seismic qualification, Safety Goals, Containment leak tightness, on-demand reliability of safety systems, sharing of systems, operator action time limits, ECI heat exchanger leak detection, safety parameter display systems and their qualification for Design Extension Conditions and use of computer based systems or equipment. GI-44 is Safety Significance Level 3 based on deterministic defence-in-depth and safety significance levels considerations, as well as probabilistic defence-in-depth considerations. Considering the legacy nature of the original Pickering design and construction, and based on the extensive safe operating experience and modifications processes, as well as the low safety significance of the Gaps and considering that practical solutions are not readily evident, the SF1 Gaps and the SF1 Additional Gaps are assessed as Acceptable Deviations. With respect to the SF5 Gap, the REGDOC-2.4.1 implementation plan will be updated. With respect to the SF6 Gap, Pickering NGS meets current safety goals, and, although meeting the safety goals in REGDOC-2.5.2 is not practicable, risk improvement initiatives are being investigated. In summary, the proposed Resolution Plan comprises ten Acceptable Deviations and one Cross-Reference to GI-39, which address the 13 Gaps in this Global Issue.					
Safety Significance Level:	3	**Category:**	Analytical, Programmatic, Engineering	**Reassessment Beyond 2024:**	N

SECTION 2 - GI-44 ASSOCIATED GAPS	
SF5-5	Clauses 4.2.1, 6.4 and 7.3 of REGDOC-2.5.2 introduce new requirements and limits for Anticipated Operational Occurrences (AOOs), Design Basis Accidents (DBAs) and Beyond Design Basis Accidents (BDBAs) and include specific dose limits for AOOs and DBAs. Current Pickering Safety Report analyses do not identify and classify events into these categories. Dose limits currently used in Pickering are aligned with the single failure/dual failure limits in accordance with the Pickering Licence Conditions Handbook. This issue has therefore been identified as a PSR2 gap. It is being addressed as part of REGDOC-2.4.1 implementation. ***Code Review CNSC REGDOC-2.5.2*** **Associated Resolutions:** GI-44-AD1
SF1-30	Human Factors in Design: Clauses 7.21 and 8.10.1 of REGDOC-2.5.2 introduce new requirements for the systematic application of Human Factors Engineering (HFE) principles

SECTION 2 - GI-44 ASSOCIATED GAPS	
	to plant design. Many years of safe and reliable operating experience indicate that the design and processes for integration of human interactions with the plant were and remain robust. However, Pickering plant design predates the current requirements for incorporating HFE into the design and the existing plant has not been systematically demonstrated to meet the requirements for a new plant. Therefore, this has been identified as a PSR2 gap. ***Code Review CNSC REGDOC-2.5.2*** **Associated Resolutions:** GI-44-AD2
SF1-29	Seismic Qualification and Design: Clause 7.13.1 of REGDOC-2.5.2 requires that Beyond Design Basis (BDB) Earthquake seismic margin be a factor of 1.67 beyond that required for the new plant Design Basis Earthquake (DBE). Fragility evaluations were completed for seismic mitigating SSCs, however, based on available information it could not be confirmed that the new plant BDB Earthquake margin of 1.67 would be achieved. Therefore, this has been identified as a PSR2 gap. ***Code Review CNSC REGDOC-2.5.2*** **Associated Resolutions:** GI-44-AD3
SF6-5	Clause 4.2.2 of REGDOC-2.5.2 introduces new requirements and limits for probabilistic analysis risk limits, such as a Core Damage Frequency limit of less than 10^{-5}/y. It has not been demonstrated that these requirements can be achieved. Therefore, this has been identified as a PSR2 gap. ***Code Review CNSC REGDOC-2.5.2*** **Associated Resolutions:** GI-44-AD4
SF1-25	Containment Leak Tightness for Design Extension Conditions (DECs): Clauses 7.3 and 8.6.12 of REGDOC-2.5.2 require Containment to provide a leak tight barrier following DECs with severe core damage for a period sufficient to implement off-site emergency measures. REGDOC-2.5.2 guidance suggests this period be at least 24 hours. Such an explicit requirement does not exist in Beyond Design Basis Accident (BDBA)/ severe accident mitigation, so this represents a PSR2 gap. ***Code Review CNSC REGDOC-2.5.2*** **Associated Resolutions:** GI-44-AD5
SF1-26	On-Demand Reliability of Safety Systems: Clause 7.6 of REGDOC-2.5.2 requires all SSCs important to safety to meet an on-demand failure rate less than 10^{-3}/y. This requirement is not met for several systems including Pickering 1,4 Emergency Coolant Injection (ECI) and is therefore identified as a PSR2 gap. ***Code Review CNSC REGDOC-2.5.2*** **Associated Resolutions:** GI-44-AD6
SF1-27	Sharing of Safety Systems and Turbine Hall: Clause 7.6.5 of REGDOC-2.5.2 has a new requirement that sharing of safety systems and the turbine generator building not be

SECTION 2 - GI-44 ASSOCIATED GAPS	
	permitted. Pickering Units share Emergency Coolant Injection (ECI) and Negative Pressure Containment, as well as the turbine hall; therefore, this has been identified as a PSR2 gap. ***Code Review CNSC REGDOC-2.5.2*** **Associated Resolutions:** GI-44-AD7
SF1-28	Allowable Times for Crediting On-site Operator Actions: Clauses 7.10 and 8.10.4 of REGDOC-2.5.2 establish new time limits for crediting operator actions, i.e., 30 minutes for Main Control Room (MCR) actions and 1 hour for field actions. Pickering NGS has not demonstrated that deterministic safety analysis consequences are acceptable if MCR and field action are not credited for these times respectively. Therefore, this has been identified as a PSR2 gap. ***Code Review CNSC REGDOC-2.5.2*** **Associated Resolutions:** GI-44-AD8
SF1-31	Detection/Isolation of Emergency Coolant Injection (ECI) Heat Exchanger (HX) Tube Leak: Clause 8.5 of REGDOC-2.5.2 requires ECI recovery heat exchanger tube leak detection capability. Pickering Units 5-8 ECI recovery heat exchangers do not have leak detection capability on the cooling water side. Therefore, this has been identified as a PSR2 gap. ***Code Review CNSC REGDOC-2.5.2*** **Associated Resolutions:** GI-44-AD9
SF1-32	Safety Parameter Display System Qualification for Design Extension Conditions (DECs): Clause 8.10.1.1 of REGDOC-2.5.2 requires the Main Control Room (MCR) to contain a Safety Parameter Display System (SPDS) that presents sufficient information on safety-critical parameters for the diagnosis and mitigation of Design Basis Accidents (DBAs) and DECs. The SPDSs are to be qualified for DEC and have parameters available in both the MCR and Secondary Control Areas (SCA), per Clause 8.10.2. Pickering SPDSs are not Review Level Condition (RLC) qualified or available in all locations. As part of the follow-up to the 2011 Fukushima accident, instrumentation to support critical parameters required to function for DECs has been evaluated for survivability. The instrument loops associated with these parameters have been identified for use in Critical Safety Parameter Monitoring (CSPM) and Beyond Design Basis Accident (BDBA) procedures. However, the indications from these loops are not in one central location and, in some cases, require field action (e.g., power) to obtain data. This does not fully satisfy the requirements to have these parameters available from a SPDS in the MCR and SCA. Therefore, this has been identified as a PSR2 gap relating to the new plant requirement to have SPDS that is DEC qualified and with parameters available in the MCR and SCA. ***Code Review CNSC REGDOC-2.5.2*** **Associated Resolutions:** GI-44-AD10
SF1-AG17	An SF1 Gap related to the absence of a Safety Parameter Display System in Pickering A and B, and in emergency response facilities, was identified in [P-CORR-00531-05107, e-Doc 5305945, July 26, 2017].

SECTION 2 - GI-44 ASSOCIATED GAPS

	Associated Resolutions: GI-44-AD10
SF1-AG18	An SF1 Gap related to an assessment of Section 7.9.2 of REGDOC-2.5.2 for the use of computer based systems or equipment, i.e., Digital Controllers, was identified in [P-CORR-00531-05107, e-Doc 5305945, July 26, 2017]. **Associated Resolutions:** GI-39-AD1
SF1-AG19	An SF1 Gap related to the analysis of AOOs, as required by Section 9.1 of REGDOC-2.5.2 was identified in [P-CORR-00531-05107, e-Doc 5305945, July 26, 2017]. **Associated Resolutions:** GI-44-AD1

SECTION 3 - GI-44 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

GI-44 contains 13 Gaps: eight SF1 Gaps, one SF5 Gap, one SF6 Gap and three SF1 Additional Gaps related to REGDOC-2.5.2. Some of these Gaps are similar to issues covered by other Global Issues, as identified in the Resolution Plan.

Compliance with REGDOC-2.5.2 is not currently a licence requirement for Pickering NGS (in accordance with PROL 48.03/2018) per the R05 Pickering Licence Conditions Handbook.

SECTION 4 - GI-44 PRIORITY DETERMINATION

Safety Significance of Global Issue	**Deterministic Considerations**			**Probabilistic Considerations**								**Overall Safety Significance Level**
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	3	3	**3**	N/A	3	N/A	N/A	N/A	N/A	N/A	**3**	**3**

Rationale:

This Global Issue includes 13 gaps that are related to modern design requirements for nuclear power plants. The gaps are related to requirements for deterministic safety analysis, human factors, seismic qualification, Safety Goals, Containment leak tightness, on-demand reliability of safety systems,

SECTION 4 - GI-44 PRIORITY DETERMINATION

sharing of systems, operator action time limits, ECI heat exchanger leak detection, safety parameter display systems and their qualification for Design Extension Conditions, and use of computer based systems or equipment. Some of these gaps are similar to issues covered by other Global Issues.

Regarding deterministic considerations, this Global Issue comprises some gaps that pertain to physical barriers, and other gaps that do not have a direct impact on nuclear safety. Therefore, both considerations E1 and E2 are applicable. Defence in Depth (E1) is assigned Safety Significance Level 3 because the gaps do not identify any impairment in the physical barriers for defence-in-depth. Resolution of the relevant gaps may enhance defence-in-depth barriers, so row 3 of Table E1 of the PSR2 Basis Document is applicable. Safety Significance Levels (E2) is assigned a Safety Significance Level 3. This is consistent with the prioritization of the Global Issues that are related to other REGDOC-2.5.2 gaps (For example, GI-31 addresses issues that are related to REGDOC-2.5.2 gap SF5-5). The remaining issues are not significant by themselves and do not identify adverse conditions for current safe operation (row 3 of Table E2 in the PSR2 Basis Document). Hence, the overall Safety Significance Level of 3 for deterministic considerations is dictated by the E1 and E2 categorizations.

Regarding probabilistic considerations, this Global issue is assigned Safety Significance Level 3 for Reactor Safety - Defence in Depth (F2). This is because resolution of some of the gaps would enhance safety margin on a primary parameter of a System Important to Safety or the reliability of a structure important to safety for DBAs (initiating frequency less than 10^{-3}/y) and BDBAs. The other probabilistic considerations, i.e., Core Damage Frequency (F1), Public Radiation Safety (F3), Plant Operability (F4), Occupational Radiation Safety (F5), Emergency Preparedness (F6) and Environment (F7), are not directly applicable. Hence, the overall Safety Significance Level of 3 for probabilistic considerations is dictated by the F2 categorization.

In summary, the overall Safety Significance Level for this Global Issue is 3. Most of the gaps within this Global Issue are similar to issues covered by other Global Issues, and the remaining gaps are not significant.

SECTION 5 - GI-44 RESOLUTION PLAN

GI-44-AD1	Consideration of deterministic safety analysis covering required potential event sequences and specific dose limits introduced by REGDOC-2.5.2 are addressed under REGDOC-2.4.1 implementation. The REGDOC-2.4.1 Implementation Plan will be updated in accordance with the Licence Conditions Handbook and will identify any changes required to support the continued safe operation of Pickering NGS. (Addressed in GI-31) These changes will be informed by the timeline of the Darlington REGDOC-2.4.1 Implementation Plan and the limited additional years of Pickering NGS operation. With the current deterministic analyses, dose limits for event sequences are as defined in the Licence Conditions Handbook (for single failure and dual failure sequences) and OPG is compliant with those requirements as demonstrated through the accident analyses in the Pickering Safety Reports. The DBA events bound the AOO sequences in terms of system response and consequences. In addition, the Pickering NGS PSAs consider a full range of event sequences – covering AOO-type events, DBAs and BDBAs. Pickering NGS meets its required Risk-based Safety Goals as demonstrated through the PSAs. REGDOC-2.5.2 includes general high level requirements where their intent is met

SECTION 5 - GI-44 RESOLUTION PLAN	
	by Pickering NGS Safety Reports and PSA consideration of PIEs. (SF1-AG19) A practicable solution to addressing the gap against the specific requirements of REGDOC-2.5.2 on event identification and classification into Anticipated Operational Occurrences (AOOs), Design Basis Accidents (DBAs) and Beyond Design Basis Accidents (BDBAs) and the dose limits for AOOs and DBAs, and implementing any enhancements within the time available during extended operation, is not readily evident and would provide limited safety benefit. On this basis, and given the low safety significance (Safety Significance Level 3), this is assessed as an Acceptable Deviation. (SF5-5)
GI-44-AD2	Human Factors in Design. The original design phase of Pickering NGS recognized the need for focus on the operator interfaces in the control centres, and on recognition of the integrated whole of the control centre and the related human-systems interfaces. The Pickering units as a result have decades of safe operation and operating experience with monitoring, operation, testing, maintenance and training (including simulators). With regards to plant modifications, all plant modifications are required to be completed in compliance with N-PROG-MP-0001 [N-PROG-MP-0001 R015, OPG Nuclear Program, *Engineering Change Control* May 12, 2017], which includes the requirements for Human Factors and Ergonomics to be considered in design. If the modification is judged to have an HFE impact, a Human Factors Engineering Specialist must concur with the Human Factors Level of Activity. The modification may require the preparation of the Human Factors Engineering Plan or the Human Factors Worksheet. These instructions and processes ensure that the human-system interface elements for the modification are addressed. The technical, design and operator reviews, during and following the design process and via the Availability for Service (AFS) process, ensure the usability requirements will be achieved. Based on the extensive operating experience and modifications processes, there is no justification for revisiting the overall Main Control Room design from an HF perspective. On this basis, and as per the low safety significance (Safety Significance Level 3), and that the issue is being managed to the extent practicable, it is assessed as an Acceptable Deviation. Human factors issues are also addressed in GI-35. (SF1-30)
GI-44-AD3	Beyond Design Basis Seismic Margin. Pickering safety-related structures were designed and analyzed in accordance with the applicable Standards current at the time. Seismic-related enhancements have been made to the plant through the course of plant operation. Both Pickering 1,4 and Pickering 5-8 performed PSA-based seismic margin assessments (SMA) [P-REP-03611-00006-R000, *Pickering NGS PSA Update to Include Enhancements from the Fukushima Integrated Action Plan*, April 30, 2014] that demonstrated acceptable plant level seismic capacities and acceptable plant risk resulting from seismic events. The SMAs demonstrated that the plant has adequate capability to respond to BDBAs. Beyond Design Basis Nuclear Safety Enhancements are evaluated for seismic robustness using a Review Level Earthquake that exceeds the Design Basis Earthquake as detailed in [N-GUID-01130-10000 R01, *Modifications for Beyond Design Basis Accidents*, February 6, 2015]. Beyond Design Basis seismic capacity has been implemented

SECTION 5 - GI-44 RESOLUTION PLAN	
	at the plant to the extent practicable. This gap relates to the non-mandatory guidance section of REGDOC 2.5.2 in this regard. On this basis, and given the low safety significance (Safety Significance Level 3), this is assessed as an Acceptable Deviation. (SF1-29)
GI-44-AD4	Safety Goals for New Build Reactors: Pickering NGS meets its current Safety Goals. Due to the legacy nature of the Pickering original design and construction, meeting REGDOC-2.5.2 Safety Goals for new nuclear power plants is not practicable. Risk improvement initiatives continue to be developed to further mitigate this and align with all modern Canadian Nuclear Power Plants (see GI-27-RS2). The residual impact is deemed to be a low safety significance issue (Safety Significance Level 3) and is being addressed to the extent practicable. Therefore, it is assessed as an Acceptable Deviation. (SF6-5)
GI-44-AD5	Containment Leak Tightness: The requirement of Containment leak tightness for a period sufficient to implement off-site emergency measures cannot be explicitly demonstrated since an explicit set of DECs is not identified within the licensing basis. The current engineered provisions provide a sufficient Containment barrier to ensure compliance with existing Safety Goals. Beyond Design Basis provisions using Emergency Mitigating Equipment and SAMG provide additional means of protecting Containment integrity against potential accident sequences. Risk improvement initiatives continue to be developed to further mitigate this issue (see GI-27-RS2). Furthermore, due to the nature of the Pickering design and construction, it is not practicable to retrofit a leak tightness barrier for a period sufficient to implement off-site emergency measures following certain BDBAs. This is a low safety significance issue (Safety Significance Level 3) and is being managed to the extent practicable. Therefore, it is assessed as an Acceptable Deviation. (SF1-25)
GI-44-AD6	On-Demand Reliability of Safety Systems. It is not practicable to implement retroactive design modifications to meet the REGDOC-2.5.2 stated unavailability for some safety systems. However, safety system performance is closely scrutinized and monitored, and established unavailability targets and performance of all safety systems are monitored and reported annually in the Annual Risk and Reliability Report [P-REP-09051.1-00016-R000, OPG Report, *Pickering NGS – 2016 Annual Risk and Reliability Report*, March 31, 2017]. The current engineered provisions provide sufficient functionality to ensure compliance with PSA Safety Goals. This is a low safety significance issue (Safety Significance Level 3) and is being managed to the extent practicable. Therefore, it is assessed as an Acceptable Deviation. (SF1-26)
GI-44-AD7	Sharing of Systems. The main impacts of sharing of ECI and Containment are addressed since if either ECI or Containment is unavailable, all affected units are considered impaired and must shutdown within specified time limits, hence reducing the risk of a coincidental DBA. Moreover environmental conditions in the common turbine building have been assessed and credited provisions have been protected to ensure the ability to shutdown/control, cool and monitor remains

SECTION 5 - GI-44 RESOLUTION PLAN	
	available on non-accident units. Required Safety Goals are met. This is a low safety significance issue (Safety Significance Level 3) and the impact of the clause requirement is addressed as shown above. It is impracticable to achieve direct compliance due to the nature of the Pickering design and construction. Thus, this is assessed as an Acceptable Deviation. (SF1-27)
GI-44-AD8	Operator Action Time Credits. REGDOC-2.5.2 requirements for allowable times for operator action from the MCR or the field are more limiting than the corresponding requirements of REGDOC-2.4.1. Pickering A and B Safety Report credits for operator actions from the MCR and in the field are consistent with REGDOC-2.4.1 requirements of 15 and 30 minutes, respectively. The ability to execute required actions within these time limits has been demonstrated through decades of operation, through effective training and testing programs. The gap against the corresponding REGDOC 2.5.2 requirements is a low safety significance issue (Safety Significance Level 3) and has been addressed to the extent practicable. Therefore, it is assessed as an Acceptable Deviation. (SF1-28) This Acceptable Deviation also includes/addresses GI-31.
GI-44-AD9	ECI Heat Exchanger Leak Detection. Pickering 5-8 has ECI recovery piping, pumps and heat exchangers outside of Containment. Components penetrating and outside Containment are all DBE qualified and Nuclear Class 2 [NK30-DM-33350-00002 R003, *Emergency Coolant Injection System Part 1 – General Requirements and Overview*, May 17, 2017], in accordance with the Design Guide [NK30-REF-68000-0379145 R001, Atomic Energy of Canada Limited (AECL) Engineering Design Guide, DG-30-68000-6 R001, *Containment Provisions for Extensions of the Containment Envelope for Pickering Generating Station 'B'*, May 1, 1977]. Since the intent of leakage detection is served by the system leakage collection, recovery and radiation monitoring in the vicinity, the added benefit of implementing a design modification for direct leakage detection is a low safety significance issue (Safety Significance Level 3) and has been addressed to the extent practicable. Therefore, it is assessed as an Acceptable Deviation. (SF1-31)
GI-44-AD10	Pickering NGS has the appropriate indications and alarms in the Main Control Room (and in secondary areas should the Main Control Room become uninhabitable) to inform Operations staff of mitigating system action and changes in key plant parameters such that initiating events can be controlled. The capability to monitor post-accident conditions remotely from the Main Control Room is provided in the SDSE Instrument Rooms for Units 1,4 and in the Unit Emergency Control Centres for Units 5-8. All OPG Nuclear emergency facilities have the capability to access WebEOC, which is a web-based information management software tool used by the Emergency Response Organization (ERO) that allows real-time information posting and multidirectional communication over an Internet connection [OPG Guideline, N-GUID-03491-10000 R000, WebEOC User's Guide, June 30, 2009]. P-MAN-03490-00002, "Pickering ERO Equipment and Facility Manual" provides requirements and direction for Pickering ERO facilities configuration management

SECTION 5 - GI-44 RESOLUTION PLAN	
	and maintenance to ensure readiness, and to provide contingency actions, in accordance with the Consolidated Nuclear Emergency Plan. However, the Safety Parameter Display System cannot be demonstrated to be Design Extension Condition (DEC) qualified since an explicit set of DECs is not identified within the licensing basis. As part of the follow-up to the 2011 Fukushima accident, instrumentation to support critical parameters required to function for BDBAs has been evaluated for survivability in [N-REP-09013-10007 R000, OPG Report, *Ontario Power Generation Severe Accident Management Guidance Instrumentation and Equipment Survivability - Summary Report*, December 13, 2013]. Additionally, as part of FAI 1.8.1 deliverables, OPG has completed the site specific instrumentation and equipment survivability assessments and has prepared a plan and schedule for identified SAMG response enhancements [N-REP-09013-10009 R000, *Information to Support Closure of FAI 1.8.1 - Survivability Assessments for Equipment and Instrumentation for Severe Accident Management*, December 17, 2013]. The instrument loops associated with these critical parameters have been identified for use in CSPM and BDBA accident procedures. The indications from these loops are not in one central location and, in some cases, require field action (e.g., power) to obtain data. Since the intent of this requirement is met as demonstrated above, the added benefit of implementing a design modification addressing this gap is not significant. This is a low safety significance issue (Safety Significance Level 3) and has been addressed to the extent practicable with enhancements identified in N-REP-09013-10009. Therefore, it is assessed as an Acceptable Deviation. (SF1-32) (SF1-AG17)
GI-44-XRF-GI-39-AD1	The PSR2 code review of REGDOC-2.5.2 specifies that the only identified Gap against Clause 7.9.2 is the same Gap identified in the code review of CSA N290.14 (2015) relating to qualification of hardware and software. The Gap is related to absence of categorization and qualification for some legacy real-time process computing applications. This Gap is applicable to Clause 7.9.2 of REGDOC-2.5.2 and it is addressed in GI-39. The review of CSA N290.14 (2015) found that new applications and changes comply with the standard. Since the only identified Gap against clause 7.9.2 of REGDOC-2.5.2 is addressed in GI-39, this Gap is addressed by GI-39. (SF1-AG18)

B.45. GI-45 CRN Concession for Fire Protection Components

SECTION 1 - GI-45 GLOBAL ISSUE SUMMARY					
The goal of GI-45, CRN Concession for Fire Protection Components, is to confirm that the existing exemption from the requirement to have a Canadian Registration Number (CRN) for certain fittings and components associated with Fire Protection Systems is applicable for the extended operating period. This Global Issue comprises one item requiring No Further Action which addresses one SF1 Gap identified from a review of the Pickering Licence Conditions Handbook. GI-45 is Safety Significance Level 4 based on deterministic safety significance levels considerations. The CNSC exemption was an industry-wide initiative and is not time dependent. Therefore, the SF1 Gap requires No Further Action.					
Safety Significance Level:	4	**Category:**	Programmatic	**Reassessment Beyond 2024:**	N

SECTION 2 - GI-45 ASSOCIATED GAPS	
SF1-36	Section 6.2 of the Pickering Licence Conditions Handbook [CNSC Report, LCH-PNGS-R004, *Pickering NGS: Licence Conditions Handbook,* December 23, 2015] outlines a concession related to exemption from requiring a Canadian Registration Number (CRN) for certain fittings and components associated with Fire Protection Systems. This concession will need to be considered in the context of Pickering PSR2 for operation past 2020 and is therefore a gap for Pickering PSR2. ***Additional Review Findings*** • **Note – Per GI-45-NFA1, this gap has been fully addressed.** **Associated Resolutions:** GI-45-NFA1

SECTION 3 - GI-45 BACKGROUND INFORMATION AND RESOLUTION STRATEGY
This Global Issue contains one SF1 gap related to a concession for exemption from requiring a Canadian Registration Number (CRN) for certain fittings and components associated with Fire Protection Systems. The inclusion of this exemption into the Pickering Licence Conditions Handbook [LCH-PNGS-R004, *Pickering NGS Licence Conditions Handbook*, December 2015] was initiated as part of the CNSC's move towards a more risk-informed regulatory approach and also to ensure consistency amongst licencees [P-CORR-00531-02957, *Regulatory Approval of Pressure Retaining Components*, October 24, 2005]. This exemption eliminated the need for CNSC staff to approve non-risk significant registration and classification of modifications to existing registered systems. This was an industry-wide initiative. There is no time-dependence associated with this concession. [P-CORR-03680-

SECTION 3 - GI-45 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

0620817, *Resolution Plan for Pickering Periodic Safety Review 2 (PSR2) Global Issue #GI-45*, May 25, 2017].

SECTION 4 - GI-45 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								Overall Safety Significance Level
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	N/A	4	**4**	N/A	N/A	N/A	N/A	N/A	N/A	N/A	**N/A**	**4**

Rationale:

This Global Issue is related to the Licence Conditions Handbook exemption on the classification of Fire Protection Systems and associated fittings and components. As discussed in Section 5 of this Global Issue, this concession has industry-wide application and is not affected by Pickering NGS operation beyond 2020. Nevertheless, the Safety Significance Level is determined in accordance with the Global Assessment process.

Regarding deterministic considerations, this Global Issue is not associated with a physical barrier, so Defence in Depth (E1) is not applicable. Safety Significance Levels (E2) is assigned Safety Significance Level 4 since the Licence Conditions Handbook exemption on the classification of Fire Protection Systems and associated fittings and components is assessed to be applicable for Pickering NGS operation beyond 2020. Therefore, a Safety Significance Level of 4 is selected for deterministic considerations.

For probabilistic considerations, this Global Issue has no impact on Core Damage Frequency (F1) or Defence in Depth (F2), nor does it impact Public Radiation Safety (F3), Plant Operability (F4), Occupational Radiation Safety (F5), Emergency Preparedness (F6) or Environment (F7). Therefore, probabilistic considerations are not applicable.

In summary, the overall Safety Significance Level is 4. This Licence Conditions Handbook exemption is not affected by Pickering NGS operation beyond 2020.

SECTION 5 - GI-45 RESOLUTION PLAN	
GI-45-NFA1	The concession for exemption from requiring a Canadian Registration Number (CRN) for certain fittings and components associated with Fire Protection Systems, as detailed in the Pickering Licence Conditions Handbook, has industry-wide application and is not affected by Pickering NGS operation beyond 2020. [P-CORR-03680-0620817, *Resolution Plan for Pickering Periodic Safety Review 2 (PSR2) Global Issue #GI-45*, May 25, 2017] No further action is required. (SF1-36)

B.46. GI-46 Requirements of National Fire Code of Canada for Units 1 & 4 Standby Generator Fuel Tanks

SECTION 1 - GI-46 GLOBAL ISSUE SUMMARY					
The goal of GI-46, Requirements of National Fire Code of Canada (NFCC) for Units 1 & 4 Standby Generator Fuel Tanks, is to demonstrate conformance with the NFCC. This Global Issue comprises one item requiring No Further Action which addresses one COP Review Gap for Pickering 1,4 related to NFCC Clause 4.5.6.7. GI-46 is Safety Significance Level 4 based on deterministic safety significance levels considerations. NFCC Clause 4.5.6.7 requires that piping for flammable liquids or combustible liquids be located aboveground where the piping enters a building. Upgrades were made to the Pickering 1,4 fuel oil systems as part of Pickering A Return to Service, resulting in aboveground location for all of the relevant piping. Therefore, the proposed Resolution Plan comprises one No Further Action item which addresses the COP Review Gap.					
Safety Significance Level:	4	**Category:**	Analytical	**Reassessment Beyond 2024:**	N

SECTION 2 - GI-46 ASSOCIATED GAPS	
COP-18	An assessment that shows that Standby Generator fuel tanks supporting units 1,4 comply with NFCC could not be found. ***Pickering PSR2 Gap COP-18*** • **Note – Per GI-46-NFA1, this gap has been fully addressed.** **Associated Resolutions:** GI-46-NFA1

SECTION 3 - GI-46 BACKGROUND INFORMATION AND RESOLUTION STRATEGY
This Global Issue contains one COP Review Gap related to requirements of the National Fire Code of Canada (NFCC) for the Pickering Units 1,4 Standby Generator fuel oil systems, with respect to location of fuel oil piping. The PSR2 review of COP Gaps [P-REP-03680-00024 R000, *Pickering 5-8 Continued Operations Plan Review in Support of PNGS Periodic Safety Review 2*, January 17, 2017] identified this COP Gap as applicable to Pickering Units 1,4 since evidence was not found to demonstrate that the Pickering Units 1,4 Standby Generator fuel oil systems were compliant with the requirement of NFCC Clause 4.5.6.7, which requires that piping for flammable liquids or combustible liquids be located aboveground where the piping enters a building. Further investigation has identified that upgrades made to the Pickering 1,4 fuel oil systems as part of the Pickering A Return to Service work have resulted in aboveground

SECTION 3 - GI-46 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

location for all of the relevant piping [NA44-REP-54600-00002, R000, *Available for Service Report – Installation of Standby Generator Aboveground Fuel Oil Piping System*, June 2002]. [P-CORR-03680-0620818, *Resolution Plan for Pickering Periodic Safety Review 2 (PSR2) Global Issue #GI-46*, May 25, 2017]

SECTION 4 - GI-46 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								Overall Safety Significance Level
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	N/A	4	**4**	N/A	N/A	N/A	N/A	N/A	N/A	N/A	**N/A**	**4**

Rationale:

This Global Issue is related to requirements of the National Fire Code of Canada (NFCC) for the Pickering Units 1,4 Standby Generator fuel oil systems, with respect to location of fuel oil piping. As discussed in Section 5 of this Global Issue, further investigation has identified that upgrades made to the Pickering 1,4 fuel oil systems as part of the Pickering A Return to Service work have resulted in aboveground location for all of the relevant piping. Nevertheless, the Safety Significance Level is determined in accordance with the Global Assessment process.

Regarding deterministic considerations, this Global Issue is not associated with a physical barrier. Therefore, Defence in Depth (E1) is not applicable. Safety Significance Levels (E2) is assigned Safety Significance Level 4 because as indicated in Section 5 of this Global Issue, this issue is addressed by upgrades made to the Pickering 1,4 fuel oil systems as part of the Pickering A Return to Service work. Therefore, a Safety Significance Level of 4 is selected for deterministic considerations.

For probabilistic considerations, this Global Issue has no impact on Core Damage Frequency (F1) or Defence in Depth (F2), nor does it impact Public Radiation Safety (F3), Plant Operability (F4), Occupational Radiation Safety (F5), Emergency Preparedness (F6) or Environment (F7). Therefore, probabilistic considerations are not applicable.

In summary, the overall Safety Significance Level is 4. This issue is already addressed by upgrades meeting NFCC requirements with respect to location of fuel oil piping.

SECTION 5 - GI-46 RESOLUTION PLAN	
GI-46-NFA1	Upgrades made to the Pickering 1,4 fuel oil systems as part of the Pickering A Return to Service work have resulted in aboveground location for all of the relevant piping [NA44-REP-54600-00002, *Available for Service Report – Installation of Standby Generator Aboveground Fuel Oil Piping System*, June 07, 2002], which addresses the issue in the COP-18 Gap. [P-CORR-03680-0620818, *Resolution Plan for Pickering Periodic Safety Review 2 (PSR2) Global Issue #GI-46*, May 25, 2017] No further action is required. (COP-18)

B.47. GI-47 Fire Protection Code NFPA 24

SECTION 1 - GI-47 GLOBAL ISSUE SUMMARY					
The goal of GI-47, Fire Protection Code NFPA 24, is to demonstrate conformance with the 2016 version of the National Fire Protection Association Code NFPA 24. This Global Issue comprises one proposed Resolution Statement and one item requiring No Further Action, which address two SF1 code review Gaps [NFPA 24]. One of the SF1 Gaps is related to a deviation against a particular requirement in the 1970 version of NFPA 24, and the other SF1 Gap is related to Pickering 1,4 and 5-8 not being assessed against the most recent version of NFPA 24. GI-47 is Safety Significance Level 3 based on deterministic defence-in-depth considerations. The proposed Resolution Plan for this Global Issue comprises activities which are already in progress to address the deviation against the 1970 version of NFPA 24. To address the SF1 Gap related to Pickering 1,4 and 5-8 not being assessed against the most recent version of NFPA 24, a high level review of Pickering NGS against NFPA 24 was performed in 2017 and resulted in no additional Gaps being identified. Therefore, this Gap requires No Further Action. Completion of the proposed Resolution Plan will support and strengthen defence-in-depth Level 2 and Level 3.					
Safety Significance Level:	3	**Category:**	Programmatic, Analytical, Engineering	**Reassessment Beyond 2024:**	N

SECTION 2 - GI-47 ASSOCIATED GAPS	
SF1-23	For OPG Report NK30-REP-71400-10001 R001, *Fire Protection Code Compliance Review Pickering Nuclear Generating Station B* there is an outstanding issue (Deviation #13301) which relates to NFPA 1970 Section 3601: “Yard post indicator valves at PNGS B are not secured in the open position as required by code” (and which applies to Pickering Units 1,4 as well as Units 5-8). Work to resolve this deviation is currently in progress with locks installed on the majority of the affected valves. Based on OPG List P-LIST-71400-00001 R000, there are a number of SSCs in the yard which directly support plant operation and which are defined as being “related to nuclear safety”. As a result, fire water supply to these SSCs is a credited safety function. Deviation # 13301 is not yet complete. Therefore, this has been identified as a PSR2 gap. ***Code Review NFPA 24*** **Associated Resolutions:** GI-47-RS1
SF1-24	For Pickering Units 5-8 the baseline for NFPA 24 compliance is the 1970 version of the standard. Pickering Units 1,4 have not been previously assessed against NFPA 24. Although recent changes to the 2013 and 2016 versions of NFPA 24 will be addressed in any firewater system design changes going forward (as a result of Code-over-Code reviews performed for NFPA 24), compliance has not been formally documented for Pickering Units 1,4 or Units 5-8 against the most recent versions of NFPA 24. Furthermore, there have been a large number of significant changes to NFPA 24 since 1970, including the 2002 edition which “represented a complete revision of NFPA 24”. Since Pickering NGS has not demonstrated compliance

SECTION 2 - GI-47 ASSOCIATED GAPS

with the 2016 version of NFPA 24, this has been identified as a PSR2 gap. It is noted that OPG is proactively replacing portions of the firewater piping in accordance with NFPA 24, under the Pickering A Firewater Pipe Replacement Project 13-80069.

Code Review NFPA 24

- **Note – Per GI-47-NFA1, this gap has been fully addressed.**

Associated Resolutions: GI-47-NFA1

SECTION 3 - GI-47 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

This Global Issue contains two SF1 Gaps related to Fire Protection Code NFPA 24.

Following completion of the Safety Factor 1 Report, a code assessment review against the most current version of NFPA 24 (2016) was completed. It is also noted that sections of the buried fire protection piping in the north and south yards in the Pickering A un-zoned area are being replaced under Pickering A Firewater Pipe Replacement Project 13-80069. This proactive modification was undertaken to replace the existing cast iron piping. As identified in OPG Project Charter [NA44-PCH-71450-00001 R000, OPG Project Charter, *Project Charter Buried Fire Pipe Replacement*, March 17, 2014], the main goals of the project are to eliminate the need to isolate major portions of the fire suppression systems due to buried pipe failures and to minimize future failures of the buried cast iron pipe. The new portions of piping will be installed in accordance with NFPA 24.

SECTION 4 - GI-47 PRIORITY DETERMINATION

	Deterministic Considerations			Probabilistic Considerations								
Safety Significance of Global Issue	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	**Overall Safety Significance Level**
	3	N/A	**3**	4	N/A	N/A	N/A	N/A	N/A	N/A	**4**	**3**

<u>Rationale:</u>

This Global Issue is related to Fire Protection Code NFPA 24. Resolution of this Global Issue will complete the installation of wrenches and locks on the Units 5-8 yard Fire Protection System. As

SECTION 4 - GI-47 PRIORITY DETERMINATION
described in Section 5 of this Global Issue, a review against NFPA 24 identified no additional gaps. Regarding deterministic considerations, the relevant safety function is fire protection, but the gap does not impair the capability to terminate a fire event. Therefore, Defence in Depth (E1) is applicable and the Safety Significance Level is 3 based on row 3 of the first column of Table E1 in the PSR2 Basis Document. Safety Significance Levels (E2) is not applicable because the gap relates to a defence-in-depth barrier. Therefore, a Safety Significance Level of 3 is selected for deterministic considerations. For probabilistic considerations, the Probabilistic Safety Assessment includes fire events, but this Global Issue has insignificant impact on Core Damage Frequency (F1). Therefore, Safety Significance Level 4 is assigned. Defence in Depth (F2) is not impacted, nor is Public Radiation Safety (F3), Plant Operability (F4), Occupational Radiation Safety (F5), Emergency Preparedness (F6) or Environment (F7). Therefore, the Safety Significance Level is 4 for probabilistic considerations. In summary, the overall Safety Significance Level is 3. This issue is being addressed by OPG.

SECTION 5 - GI-47 RESOLUTION PLAN	
GI-47-RS1	To resolve deviation #13301 from NK30-REP-71400-10001 R001 [NK30-REP-71400-10001 R001, OPG Report, *Fire Protection Code Compliance Review Pickering Nuclear Generating Station B*, November 23, 2010] the following work orders need to be completed to install wrenches and locks on the 058 Yard Fire Protection System: WO 3259862, 3259894, 3259893. (SF1-23)
GI-47-NFA1	A PSR2 high level review against NFPA 24 (2016) was completed for Pickering NGS [P-CORR-03680-0620821, *Re: Resolution for Pickering Periodic Safety Review 2 (PSR2) Global Issue #47 Gap SF1-24,*, June 9, 2017] (i.e., Gap SF1-23 in GI-47 is the only NFPA 24 gap remaining open). No further action is required. (SF1-24)

B.48. GI-48 CSA N293-12 Fire Protection of Nuclear Power Plants

SECTION 1 - GI-48 GLOBAL ISSUE SUMMARY					
The goal of GI-48, CSA N293-12 Fire Protection of Nuclear Power Plants, is to demonstrate conformance with the CSA nuclear standard on fire protection, CSA N293-12. This Global Issue comprises one proposed Resolution Statement and two Acceptable Deviations, which address three SF1 Gaps which were identified from a code review of CSA N293-12. GI-48 is Safety Significance Level 3 based on deterministic defence-in-depth considerations and probabilistic core damage frequency considerations. The proposed Resolution Plan for this Global Issue comprises activities to interconnect the firewater systems of Units 1,4 and 5-8. These activities address one of the SF1 Gaps. The proposed Resolution Plan assesses the other two SF1 Gaps as Acceptable Deviations, one related to electrical conductors and one related to annunciation and display. Completion of the proposed Resolution Plan will support and strengthen defence-in-depth Level 2 and Level 3.					
Safety Significance Level:	3	**Category:**	Analytical, Engineering	**Reassessment Beyond 2024:**	N

SECTION 2 - GI-48 ASSOCIATED GAPS	
SF1-4	Clause 7.2.1.13 of CSA N293-12 states: “Electrical conductors that are installed in service spaces containing other combustible materials and that are used in connection with fire alarm systems and emergency equipment, including fire alarm cables… shall be capable of performing their intended functions for not less than 1 hour after the start of a fire.” Modifications to the Fire Protection System meet the requirements of CAN/ULC-S524 which mandates a 1 hour fire rating as described in Appendix 1 (Section 2.5) of NA44-DM-71400.2-00001 R001, Section A.2 of NA44-DM-71400-00002 R000 and Section 2 of NK30-DM-71400-00001 R006. This is achieved by the use of Edwards System Technology (EST) that connects the fire alarm control panels via a data communication link with dual redundant circuit wiring paths. However, existing Pyrotronics fire alarm control panels are not similarly connected and, hence, may be susceptible to loss of alarm signal due to spot burning of a cable. While measures such as lack of combustible material in service spaces, combustible transient material control practices, and inherent protection afforded by Pickering NGS cable routing practices used in the Fire Protection Systems mitigate the lack of such a feature, it could not be confirmed based on existing documentation that all essential fire alarm cables are capable of performing their intended functions for not less than 1 hour after the start of a fire to meet the requirement of N293-12 sub-clause 7.2.1.13. As a result, this has been identified as a PSR2 gap. ***Code Review N293-12*** **Associated Resolutions:** GI-48-AD1
SF1-5	Clause 7.3.2.2 (d) of CSA N293-12 states: “At a minimum, the fire protection water pumping system shall consist of at least one diesel-engine-driven fire pump and one electric-motor-

SECTION 2 - GI-48 ASSOCIATED GAPS	
	driven fire pump set, with each pump set being capable of providing, the flow rate and pressure specified in Item (a)". This Clause is met at Pickering Units 1,4 with the provision of diesel-driven firewater pumps, backed up by supplies from the High Pressure Service Water (HPSW) System (as noted in the Pickering A Safety Report NA44-SR-01320-00001 R015, Section 11.5.1.1). It is not met at Pickering Units 5-8, where the Fire Protection System is comprised of the HPSW supplies from the four units only. As a result, Pickering Units 5-8 does not comply with Clause 7.3.2.2 (d) of CSA N293-12 and this has been identified as a PSR2 gap. ***Code Review N293-12*** **Associated Resolutions:** GI-48-RS1
SF1-3	Clause 7.2.1.10.1 of CSA N293-12 states: "A display and control centre shall be located in the MCR (Main Control Room)… capable of providing detailed information on the location and nature of the signal. In addition, the panel operator shall be able to control the fire alarm system without having to leave his or her station." Pickering 014 Display Annunciation Station 014-67140-WS2342 in the Emergency Operating Centre is capable of providing annunciation only, and there is no Display Annunciation Station in the Pickering 014 MCR (although there is limited annunciation). Therefore, this has been identified as a PSR2 gap. ***Code Review N293-12*** **Associated Resolutions:** GI-48-AD1, GI-48-AD2

SECTION 3 - GI-48 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

This Global Issue contains three SF1 Gaps related to CSA N293-12.

The first gap (SF1-4) is related to the ability of electrical conductors of the original Pyrotronics System 3 Fire Alarm Systems to perform their intended functions for not less than 1 hour after the start of a fire.

The second gap (SF1-5) is related to the requirement that the fire protection water pumping system consist of at least one diesel-engine-driven fire pump and one electric-motor-driven fire pump set. At Pickering 1,4 the Fire Protection System water supply is fed by four diesel driven fire pumps installed in the Pickering 1,4 Screenhouse. At Pickering 5-8, the Fire Protection System water supply is fed by the station's High Pressure Service Water System powered by Class III and Class IV Electrical Power System [NA44-OM-014-71400-05 R018, Section 5.1, and NK30-OM-058-71400-03, R022, Section 3.4.1]. While normally isolated from each other, the Fire Protection Ring headers can be interconnected if required.

The third gap (SF1-3) is related to the requirement that a display and control centre be located in the Main Control Room and capable of providing detailed information on the location and nature of the signal.

SECTION 4 - GI-48 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								Overall Safety Significance Level
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	3	N/A	**3**	3	N/A	N/A	N/A	N/A	N/A	N/A	**3**	**3**

Rationale:

This Global Issue relates to the CSA N293-12 standard on fire protection. There is one Acceptable Deviation related to electrical conductors, one Resolution Statement to interconnect the firewater systems of Units 1,4 and 5-8, and an Acceptable Deviation related to annunciation and display.

Regarding deterministic considerations, the relevant safety function is fire protection, but the gaps do not impair the capability to terminate a fire event. Therefore, Defence in Depth (E1) is applicable and the Safety Significance Level is 3 based on row 3 of the first column of Table E1 in the PSR2 Basis Document. Safety Significance Levels (E2) is not applicable because the gaps relate to a defence-in-depth barrier. Therefore, a Safety Significance Level of 3 is selected for deterministic considerations.

For probabilistic considerations, the Probabilistic Safety Assessment includes fire events, and the impact on Core Damage Frequency (F1) is expected to be small, in the range 10^{-7}/y to 10^{-6}/y. Therefore, Safety Significance Level 3 is assigned. Defence in Depth (F2) is not impacted, nor is Public Radiation Safety (F3), Plant Operability (F4), Occupational Radiation Safety (F5), Emergency Preparedness (F6) or Environment (F7). Therefore, the Safety Significance Level is 3 for probabilistic considerations.

In summary, the overall Safety Significance Level is 3. Work is in progress by OPG to address the firewater interconnection.

SECTION 5 - GI-48 RESOLUTION PLAN

GI-48-RS1	Provide, as necessary, design and/or operational changes and commissioning/testing to facilitate required interconnection of Pickering 1,4 and Pickering 5-8 Fire Protection System water supplies to meet the safety intent of CSA N293-12 Clause 7.3.2.2 (d). (SF1-5)
GI-48-AD1	An assessment of this gap (SF1-4) was completed [P-CORR-03680-0620819, *Re: Resolution Plan for Pickering Periodic Safety Review 2 (PSR2) Global Issue #GI-*

SECTION 5 - GI-48 RESOLUTION PLAN	
	48 Gap SF1-4, June 20, 2017]. There is some credit taken in the Fire Safe Shutdown Assessments for fire detection by the Pyrotronics-based automatic detection systems, for some rooms at Pickering NGS. The assessment concluded that the characteristics of the detection/alarm system (detection prior to potential alarm system fire damage, and signal latching function) as well as the physical separation in the equipment/cabling configuration, ensure that lack of a one hour fire rating for cabling in these systems would not cause failure to detect the fire or loss of the fire signal once detected. As such, manual suppression of the fire by Emergency Response Team personnel would not be impeded and there is no adverse nuclear safety impact. This is a low safety significance issue (Safety Significance Level 3) and has been addressed to the extent practicable. Therefore, it is assessed as an Acceptable Deviation. (SF1-3) (SF1-4)
GI-48-AD2	An assessment of this gap (SF1-3) was completed [P-CORR-03680-0620820, *Re: Resolution Plan for Pickering Periodic Safety Review 2 (PSR2) Global Issue #GI-48 Gap SF1-3*, June 2, 2017]. The assessment identified that fire detection and manual fire suppression has some nuclear safety credit at Pickering 1,4. It also confirmed that the existing arrangement of annunciation/display (including a Fire Control Panel located immediately outside the Main Control Room (MCR) and annunciation within the MCR), together with emergency response protocols and staff training, meets the intent of the requirement in providing timely response to fire signals. This is a low safety significance issue (Safety Significance Level 3) and has been addressed to the extent practicable. Therefore, it is assessed as an Acceptable Deviation. (SF1-3)

B.49. GI-49 FFS of Primary Heat Transport Auxiliary Piping Systems, and Primary Heat Transport Valves

SECTION 1 - GI-49 GLOBAL ISSUE SUMMARY

The goal of GI-49, FFS of Primary Heat Transport Auxiliary Piping Systems, and Primary Heat Transport Valves, is to ensure that the Primary Heat Transport auxiliary piping system and pump discharge and boiler inlet and outlet valves remain fit for service for the extended operating period. This Global Issue comprises one Cross-Reference to GI-8 which addresses one COP Review Gap. GI-49 is Safety Significance Level 2 based on deterministic and probabilistic defence-in-depth considerations as well as plant operability considerations.

The proposed Resolution Plan for GI-49 cross-references one of the proposed Resolution Statements for GI-8, Completion/Updating of the Condition Assessments, to complete and update the Condition Assessments for the Primary Heat Transport auxiliary piping system and pump discharge and boiler inlet and outlet valves. Completion of the proposed Resolution Plan will support and strengthen defence-in-depth Level 1.

Safety Significance Level:	2	**Category:**	Analytical	**Reassessment Beyond 2024:**	Y

SECTION 2 - GI-49 ASSOCIATED GAPS

COP-24	A review for extended operation has not been performed of the inspection results for areas susceptible to wall thinning, high stress areas of the Primary Heat Transport auxiliary piping system, and Primary Heat Transport pump discharge and boiler inlet and outlet valves for Pickering Units 1,4 and 5-8. ***Pickering PSR2 gap COP-24*** **Associated Resolutions:** GI-8-RS1

SECTION 3 - GI-49 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

This Global Issue contains one COP Review Gap related to FFS of Primary Heat Transport auxiliary piping systems and Primary Heat Transport pump discharge and boiler inlet and outlet valves.

CNSC staff has accepted the thickness measurement and inspection of pump discharge and boiler inlet/outlet valves [NK30-PLAN-00531-00001 R005, *Pickering 5-8 Continued Operations Plan*, November 24, 2015; Appendix A, Item 67].

The Primary Heat Transport Piping System is a part of the N285.4 inspection scope. Furthermore, OPG is in the process of updating Condition Assessments in support of extended operation.

SECTION 4 - GI-49 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								Overall Safety Significance Level
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	2	N/A	**2**	4	2	4	2	N/A	N/A	N/A	**2**	**2**

Rationale:

Addressing this Global Issue assures the ongoing fitness for service of Primary Heat Transport auxiliary piping systems and Primary Heat Transport valves for the operational life of the station. As discussed in Section 3 of this Global Issue, OPG is completing the Condition Assessments to demonstrate that these components are fit for service for the extended operating period.

Nevertheless, this Global Issue has Safety Significance Level 2 for deterministic considerations with respect to Defence in Depth (E1) because addressing it will assure the effectiveness of the pressure boundary barrier (second row of Table E1 in the PSR2 Basis Document). The E2 Safety Significance Level for this Global Issue is considered not applicable because this Global Issue potentially impacts on a nuclear safety barrier, whereas E2 primarily relates to issues that impact other objectives or that are indirectly related to nuclear safety. Hence, the overall Safety Significance Level of 2 for deterministic considerations is dictated by the E1 categorization.

With respect to probabilistic considerations, the auxiliary piping systems interface with the Primary Heat Transport safety barrier, the effectiveness of which can potentially be impacted by a postulated initiating event of frequency is less than 10^{-3}/y. Consistent with Table F2 in the PSR2 Basis Document, the issue has Safety Significance Level 2 for Defence in Depth (F2). Similarly, the Safety Significance Level with respect to Plant Operability is 2, since an extended period of plant shutdown as a result of fitness for service issues of these components is expected to have a probability less than 0.1.

A Safety Significance Level of 4 is assigned for Reactor Safety – Core Damage Frequency (F1). Potential failure of these components is accounted for in the Probabilistic Safety Assessment, and resolution of this Global Issue will ensure that the assumptions in the Probabilistic Safety Assessment regarding the components failure frequency remain valid. Therefore, the change in the Core Damage Frequency will be less than 10^{-7}/y, which corresponds to the fourth row of Table F1 in the PSR2 Basis Document, for which the Safety Significance Level is 4.

With respect to Public Radiation Safety (F3), the radiological consequences of a postulated failure of these components are already accounted for in the safety analysis and shown to be within regulatory

SECTION 4 - GI-49 PRIORITY DETERMINATION
limits. Therefore, this Global Issue will result in no adverse change in Public Radiation Safety (F3), which is the determining factor for the applicability of this consideration. This Global Issue has no direct impact on the other probabilistic considerations, i.e., Occupational Radiation Safety (F5), Emergency Preparedness (F6) and Environment (F7). Therefore, these probabilistic considerations are not applicable. In summary, both deterministic and probabilistic considerations dictate that this Global Issue has a Safety Significance Level of 2. As noted, OPG is completing the Condition Assessments of these components to ensure their fitness for service for extended operation.

SECTION 5 - GI-49 RESOLUTION PLAN	
GI-49-XRF-GI-8-RS1	Complete Condition Assessments for the Primary Heat Transport auxiliary piping system and Primary Heat Transport pump discharge and boiler inlet and outlet valves for Pickering Units 1,4 and 5-8. (COP-24)

B.50. GI-50 N285.4 PIP / Documentation Revision

SECTION 1 - GI-50 GLOBAL ISSUE SUMMARY					
The goal of GI-50, N285.4 PIP/Documentation Revision, is to demonstrate conformance with CSA N285.4, Periodic Inspection of CANDU Nuclear Power Plant Components. This Global Issue comprises two proposed Resolution Statements and one item requiring No Further Action, which address six SF4 Gaps identified from a code review of CSA N285.4-14, and one SF2 Additional Gap. GI-50 is Safety Significance Level 3 based on deterministic safety significance levels considerations. The proposed Resolution Plan for this Global Issue comprises activities to revise OPG's Periodic Inspection Plans and governing documentation to align with elements of the 2014 version of CSA N285.4. These activities address five of the six SF4 Gaps. The remaining SF4 Gap is assessed as requiring No Further Action as it is already addressed by existing OPG governance. The proposed Resolution Plan also includes an activity to assess the impact of extended operation on concessions against CSA N285.4. This activity addresses the SF2 Additional Gap. Completion of the proposed Resolution Plan will support and strengthen defence-in-depth Level 1.					
Safety Significance Level:	3	**Category:**	Programmatic	**Reassessment Beyond 2024:**	N

SECTION 2 - GI-50 ASSOCIATED GAPS	
SF4-3	N285.4 PIP Governance references N285.4-05, not N285.4-09 including Updates 1 and 2. This (programmatic) Darlington gap is a PSR2 gap against N285.4-09 including Updates 1 and 2. ***Code Review N285.4-14*** **Associated Resolutions:** GI-50-RS1
SF4-4	There has been a significant change in the wording of clause 4.2.7 in CSA N285.4-09 including Updates 1 and 2. I-PROC-AS-0009, *Inspection Qualification of Non-Destructive Examination Processes* does not identify the authorized inspector as a qualifying authority as directed by clause 4.2.7. Instead it establishes the CANDU Inspection Qualification Bureau as the organization that would approve procedures and personnel. This (programmatic) Darlington gap is a PSR2 gap against N285.4-09 including Updates 1 and 2. ***Code Review N285.4-14*** • **Note – Per GI-50-NFA1, this gap has been fully addressed.** **Associated Resolutions:** GI-50-NFA1
SF4-5	New erosion and corrosion inspection requirements in N285.4-09 including Updates 1 and 2 are not reflected in current PIP governance. NK38-REP-03680-10137 R000 states that: "It should be noted specifically that [this ISR Issue] is likely to have a major impact on piping PIPs because sub-clauses 7.4.7.X in CSA N285.4-09 including UPD1 and UPD2 include

SECTION 2 - GI-50 ASSOCIATED GAPS	
	substantive changes. Under the new standard erosion and corrosion inspection exemptions can no longer be justified on the basis of [sic] that conditions are determined to be non-erosive and non-corrosive." This Darlington PIP gap will also need to be addressed in the Pickering PIPs. Therefore, this is a PSR2 gap against N285.4-09 including Updates 1 and 2. ***Code Review N285.4-14*** **Associated Resolutions:** GI-50-RS1
SF4-6	Extended life inspection schedules in N285.4-09 including Updates 1 and 2 are not reflected in PIP governance. This (programmatic) Darlington gap is a PSR2 gap against N285.4-09 including Updates 1 and 2. ***Code Review N285.4-14*** **Associated Resolutions:** GI-50-RS1
SF4-7	An assessment of the prior operating non-conforming state, as required by N285.4-09 including Updates 1 and 2, is required when dispositioning inspection results. This requirement has not been included in the Feeder PIP plan. This Darlington PIP gap will also need to be addressed in the Pickering PIPs. Therefore, this is a PSR2 gap against N285.4-09 including Updates 1 and 2. ***Code Review N285.4-14*** **Associated Resolutions:** GI-50-RS1
SF4-8	There is a PSR2 gap for Pickering NGS against N285.4-14 to address: • Revised requirements for pressure tube volumetric and dimensional inspection (Clause 12.2), pressure tube hydrogen equivalent determination (Clause 12.3) and pressure tube material property testing (Clause 12.4); • Clause 12.5 which specifies minimum annulus spacer surveillance examination and testing requirements; • Selection criteria for identifying candidate tube for pressure tube surveillance examination and testing (Annex E) to include selection criteria for annulus spacer surveillance examination and testing; and • Clause 7.4.8 which specifies requirements for inspection of Environmentally Assisted Cracking, and Clauses 7.5.1/7.5.2 which specify requirements for inspection of identical components. ***Code Review N285.4-14*** **Associated Resolutions:** GI-50-RS1
SF2-AG10	An SF2 Gap related to a review of program implementation concessions for CSA N285.4 for extended operation was identified in [P-CORR-00531-05099, e-Doc 5295534, July 12, 2017]. **Associated Resolutions:** GI-50-RS2

SECTION 3 – GI-50 BACKGROUND INFORMATION AND RESOLUTION STRATEGY

This Global Issue consists of seven Gaps. Six of these Gaps are Darlington programmatic gaps against CSA N285.4-14 which are applicable to PSR2, and the seventh gap is an Additional Gap related to an assessment of the applicability of previous concessions on N285.4-14 with respect to extended operation. OPG has unit-specific Periodic Inspection Plans (PIPs) in place to support CSA N285.4 compliance. The Pickering NGS Licence Conditions Handbook currently requires OPG to comply with the requirements of the 2005 version of the standard.

SECTION 4 – GI-50 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								Overall Safety Significance Level
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	
	N/A	3	**3**	N/A	N/A	N/A	N/A	N/A	N/A	N/A	**N/A**	**3**

Rationale:

This Global Issue consists of gaps related to CSA N285.4-14. OPG has unit-specific Periodic Inspection Plans in place to support CSA N285.4 compliance. Resolution of this Global Issue will result in updates to the N285.4 Periodic Inspection Plan, governance and Fuel Channel inspection and surveillance requirements.

Regarding deterministic considerations, this Global Issue is not directly applicable to Defence in Depth (E1). The issue is assigned Safety Significance Level 3 for E2 on the basis that the issue is not significant. This is because, as discussed in Section 5 of this Global Issue, the N285.4 Periodic Inspection Plans and governance are currently in compliance with the 2005 version of the standard and the revisions are not significant. Hence, the overall Safety Significance Level of 3 for deterministic considerations is dictated by the E2 categorization.

For probabilistic considerations, this Global Issue has no impact on Core Damage Frequency (F1) or Defence in Depth (F2), nor does it impact Public Radiation Safety (F3), Plant Operability (F4), Occupational Radiation Safety (F5), Emergency Preparedness (F6) or Environment (F7). Therefore, probabilistic considerations are not applicable.

In summary, the overall Safety Significance Level for this Global Issue is 3.

SECTION 5 – GI-50 RESOLUTION PLAN	
GI-50-RS1	Revise the CSA N285.4 PIPs and governance to align with elements of N285.4-14, including making reference to CSA N285.4-14, addressing erosion and corrosion inspection requirements, reflecting extended life inspection schedules, and addressing assessment of the prior non-conforming state when dispositioning inspection results. (SF4-3) (SF4-5) (SF4-6) (SF4-7) (SF4-8)
GI-50-RS2	Assess the impact of extended operation on concessions against CSA N285.4. (SF2-AG10)
GI-50-NFA1	Clause 4.2.7 of CSA N285.4-09 Update 2 states, *"Procedures satisfying the requirements of Clause 4.2.2 that are used to detect and characterize in-service degradation or manufacturing flaws shall be demonstrated to the satisfaction of the authorized inspector".* OPG is in compliance with this requirement as Section 1.4.2 of OPG Procedure [I-PROC-AS-0002-R009, *Technical Document Control*, May 19, 2016], requires demonstration to the Authorized Nuclear Inspector (ANI) of all Non-Destructive Examination (NDE) pressure boundary related procedures for those NDE activities involving volumetric methods in support of CSA standards N285.4 and N285.5 or as required by the CNSC. This is required for any revision or change to authorized procedures, or whenever requested by the CNSC and/or the ANI. No further action is required. (SF4-4)

B.51. GI-51 Fuelling with Pressure Tube Sag

SECTION 1 - GI-51 GLOBAL ISSUE SUMMARY					
The goal of GI-51, Fuelling with Pressure Tube Sag, is to address a Gap identified by the Expert Panel related to the impact of Pressure Tube sag on the ability to fuel for the duration of extended operation. This Global Issue comprises one No Further Action item addressing the Expert Panel gap. GI-51 is Safety Significance Level 4 based on deterministic defence-in-depth considerations. The Expert Panel Gap is assessed as requiring no further action based on the Pickering EFPH and the predicted Pressure Tube sag.					
Safety Significance Level:	4	**Category:**	Engineering, Analytical	**Reassessment Beyond 2024:**	Y

SECTION 2 - GI-51 ASSOCIATED GAPS	
EP-1	The Expert Panel review of the Global Issues identified that it was not apparent that the impact of Pressure Tube sag on the ability to fuel the channels for the period of extended operation had been considered. **Associated Resolutions:** GI-51-NFA1

SECTION 3 - GI-51 BACKGROUND INFORMATION AND RESOLUTION STRATEGY
This Global Issue contains one Expert Panel Gap related to the impact of Pressure Tube sag on the ability to fuel for the duration of extended operation. The impact of Pressure Tube sag on the ability to fuel the Fuel Channels for the period of extended operation is assessed by comparing the Pickering EFPH and the associated predicted Pressure Tube sag for the extended operating period to the EFPH values at which the limits for sag on fuelling are predicted to be reached, as documented in the Fuel Channel LCMPs.

SECTION 4 - GI-51 PRIORITY DETERMINATION

Safety Significance of Global Issue	Deterministic Considerations			Probabilistic Considerations								
	E1 – Defence in Depth	E2 – Safety Significance Level	**Overall Deterministic Considerations**	F1 – Reactor Safety – Core Damage Frequency	F2 – Reactor Safety – Defence In Depth	F3 – Public Radiation Safety	F4 – Plant Operability	F5 – Occupational Radiation Safety	F6 – Emergency Preparedness	F7 – Environment	**Overall Probabilistic Considerations**	**Overall Safety Significance Level**
	4	N/A	**4**	N/A	N/A	N/A	4	N/A	N/A	N/A	**4**	**4**

<u>Rationale:</u>

Resolution of this Global Issue will confirm the ability to fuel the Fuel Channels for the period of extended operation. The issue of Pressure Tube sag is included in GI-1, Fitness for Service for Fuel Channels, and GI-4, Fitness for Service for Reactor Components and Structures.

Regarding deterministic considerations, this Global Issue is associated with a physical barrier, so Defence in Depth (E1) is applicable. A Safety Significance Level of 4 is assigned based on the assessment described in the proposed Resolution Plan that Pressure Tube sag will not adversely impact the ability to fuel until after 2024. The E2 Safety Significance Level for this Global Issue is considered not applicable because this Global Issue potentially impacts on a nuclear safety barrier, whereas E2 primarily relates to issues that impact other objectives or that are indirectly related to nuclear safety. Therefore, an overall Safety Significance Level of 4 is selected for deterministic considerations.

With respect to probabilistic considerations, this Global Issue is assigned Safety Significance Level 4 for Plant Operability (F4) since it may to lead to some loss of operating margin if Pressure Tube sag is more than expected. This Global Issue has no direct impact on the other probabilistic factors, i.e., Core Damage Frequency (F1), probabilistic Defence in Depth (F2), Public Radiation Safety (F3), Occupational Radiation Safety (F5), Emergency Preparedness (F6) or Environment (F7). Therefore, these probabilistic considerations are not applicable.

In summary, the overall Safety Significance Level is 4.

SECTION 5 - GI-51 RESOLUTION PLAN

GI-51-NFA1	The extent of and the acceptable limits for Pressure Tube (PT) sag on fuelling are documented in the Pickering Fuel Channel Life Cycle Management Plan (LCMP) references for Pickering 1,4 [NA44-REP-31100-00089-R000, *Status of Degradation Mechanisms for Pickering Units 1&4 as of December 2016 –*

SECTION 5 - GI-51 RESOLUTION PLAN	
	Summary for Fuel Channel Life Cycle Management Plan, May 19, 2017] and Pickering 5-8 [NK30-REP-31100-10188-R000, *Status of Degradation Mechanisms for Pickering 5-8 as of December 2016 – Summary for Fuel Channel Life Cycle Management Plan*, February 28, 2017]. In these reports, linear regression of in-service Pressure Tube sag measurements is used to predict Pressure Tube sag as a function of EFPH. In [P-CORR-31100-00007, *Pickering NGS – Fuel Channel Fitness-for-Service Risk Assessment – Impact of mid-2017 Planning Assumptions*, August 22, 2017], the EFPHs at 2024 are estimated for each unit. Based on the EFPHs, fuel bundle passage is assured for all Pickering Units up to 2024. The acceptability of the degree of Pressure Tube sag will be updated as required, as part of the regular updates of the LCMPs. Therefore no further PSR2 action is required for ensuring the ability to fuel at Pickering with Pressure Tube sag up to 2024. (EP-1)

Appendix C – Ranking of Proposed Resolution Statements

C.1. Introduction

Ranking of Global Issues with identified actions is the seventh element of the Global Assessment process, as listed in Section 3.3.2 of the PSR2 Basis Document [P-REP-03680-00001-R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016]. This appendix describes the method for performing this ranking and presents the ranking results. The method is applied to Global Issue proposed Resolution Statements that have identified actions. Acceptable Deviations and No Further Action statements do not go through the ranking process.

Section C.2 describes the factors that are taken into account when defining the ranking methodology. Section C.3 describes the Value Tree methodology that is used for the ranking process. Section C.4 develops the Value Tree for PSR2. Sections C.5 and C.6 present the results of the ranking methodology steps, and Section C.7 presents the ranked proposed Resolution Statements.

C.2. Ranking Process Description and Approach

The prioritized Global Issues and their proposed Resolution Statements (refer to Appendix B) are the inputs to the ranking process described in this appendix. The purpose of ranking proposed Resolution Statements is to determine the activities that will be most effective in enhancing safety given the limited extended operating period of the plant. The ranking process recognizes that there are many factors that have to be taken into account in determining the rank of a specific proposed Resolution Statement, but a key consideration is that the ranking takes place within the context of the specific time period related to extended operation. Activities that will take a relatively long time to implement or to take effect may have relatively little practical benefit if the subsequent period of operation is short.

The PSR2 ranking process, which determines the relative merit of a number of potentially competing actions based on a variety of considerations, is a multi-attribute decision process. Such processes are typically addressed through a method known as a Value Tree, which is described in the next sub-section.

C.3. The Value Tree Method

The relative merit of implementing the proposed Resolution Statement of a Global Issue depends on the degree to which it contributes to the overall objective of PSR2, which is to determine reasonable and practicable enhancements that may be made to support continued safe operation. To aid in determining the degree to which each proposed Resolution Statement will contribute to the overall objective, relative to the contributions of the other proposed Resolution Statements, the overall objective is divided into a number of sub-objectives against which the merits of the proposed Resolution Statements are assessed.

The Value Tree for PSR2 is a representation of the overall objective and its sub-objectives arranged in a hierarchy. At the highest level or "trunk" of the tree there is a single *overall* objective. The branches of the tree consist of six *fundamental* objectives that contribute directly to the overall objective. Each fundamental objective is characterized in terms of a measurable attribute that is used to determine the degree to which a proposed Resolution Statement accomplishes the fundamental objective.

The Value Tree is used in the following manner:

- Each fundamental objective is assigned a weight that denotes the importance of the associated objective in contributing to the overall objective, relative to the other fundamental objectives. Weights are assigned to each fundamental objective using the Pairwise Comparison method. This method was devised by Thomas L. Saaty for his Analytic Hierarchy Process [Saaty, T. L., *Decision Making with the Analytic Hierarchy Process*, International Journal of Services Sciences, Vol. 1, No. 1, 2008]. This method has been successfully used in previous PSRs to rank multiple, potentially competing actions. The determination of objective weights is described in Section C.5.
- Each proposed Resolution Statement is matched to the most relevant fundamental objective, and a score (referred to as the utility) is assigned to the proposed Resolution Statement to indicate the degree to which the outcome of implementing it will address the fundamental objective. Each score is derived by applying a utility function that takes into account the time it will take to implement a proposed Resolution Statement and the benefit it will have once implemented. This method is based on utility theory as originally developed in the field of micro-economics. This approach is used in the Pickering NGS PSR2 process, as it has been used in previous PSRs and because the nature of the proposed Resolution Statements for PSR2 is similar to enhancement actions identified in previous PSRs. The utility scoring method is described in Section C.6.
- The overall ranking score of each specific proposed Resolution Statement is calculated as the product of the objective weight and the utility. The results are presented in Section C.7.

Proposed Resolution Statements with the highest ranking scores contribute to the fundamental objective to a greater degree than those with lower ranking scores.

C.4. The Pickering PSR2 Value Tree

The overall objective for PSR2 is stated as follows:

> *Enhance confidence in the continued safe operation of Pickering NGS for the period of PSR2.*

Development of the Value Tree starts with consideration of the 15 Safety Factors of PSR2. This is helpful since Global Issues can for the most part be associated with a single Safety Factor. Experience has shown that six to eight fundamental objectives is a good practical number to use in the Value Tree methodology. The 15 Safety Factors are, therefore, grouped to formulate fundamental objectives. The starting point is the five groupings used by IAEA SSG-25 [IAEA Specific Safety Guide SSG-25, *Periodic Safety Review of Nuclear Power Plants*, March 2013] for its 14 Safety Factors plus the additional Safety Factor on Radiation Protection prescribed by CNSC REGDOC-2.3.3 [CNSC Regulatory Document REGDOC-2.3.3, *Periodic Safety Reviews*, April 2015], which is grouped with the fourteenth Safety Factor from SSG-25. The result is presented in Table 22.

Table 22: Safety Factor Groupings According to SSG-25 and CNSC REGDOC-2.3.3

SF#	Title
Safety factors relating to the plant	
1	Plant design
2	Actual condition of structures, systems and components (SSCs) important to safety
3	Equipment qualification
4	Aging
Safety factors relating to safety analysis	
5	Deterministic safety analysis
6	Probabilistic safety assessment
7	Hazard analysis
Safety factors relating to performance and feedback of experience	
8	Safety performance
9	Use of experience from other plants and research findings
Safety factors relating to management	
10	Organization, the management system and safety culture
11	Procedures
12	Human factors
13	Emergency planning
Safety factors relating to the environment and radiation protection	
14	Radiological impact on the environment
15	Radiation protection

Given the stage in the life-cycle of Pickering NGS it is reasonable to assign SF1 on Design to its own separate group but to keep SF2, SF3 and SF4, which all deal with fitness for service, together in their own group. That is because design changes to the plant at this stage in its life cannot be considered as important as maintaining the fitness for service of the existing equipment. The resultant six Safety Factor groups are:

- Safety Factors relating to plant design (SF1);
- Safety Factors relating to fitness for service (SF2, SF3, and SF4);
- Safety Factors relating to safety analysis (SF5, SF6, and SF7);
- Safety Factors relating to safety performance and feedback of experience (SF8 and SF9);
- Safety Factors relating to management (SF10, SF11, SF12, and SF13); and,
- Safety Factors relating to radiation protection and the environment (SF14 and SF15).

The six Safety Factor groups are used as the basis for the fundamental objectives of the Value Tree as illustrated in Figure 3.

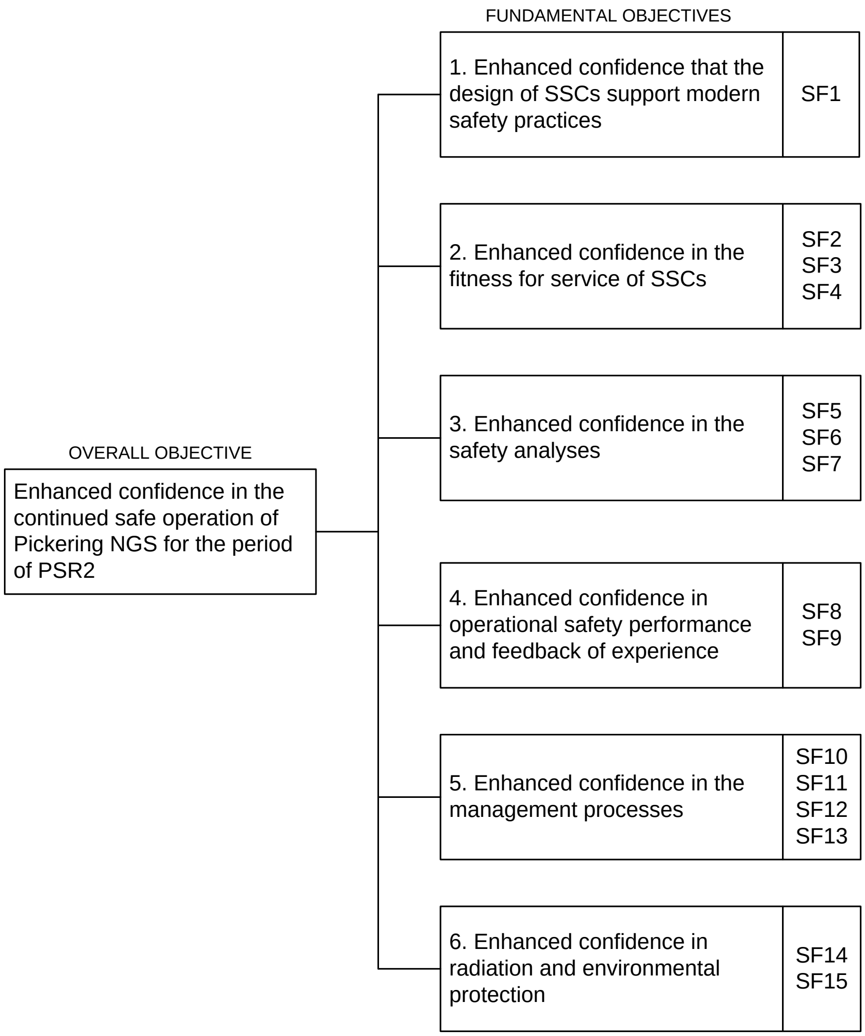

Figure 3: Value Tree for PSR2

C.5. PSR2 Value Tree Objective Weights

The method for assigning the weights to the Value Tree fundamental objectives is the pairwise comparison method used in the Analytical Hierarchy Process as discussed in Section C.3. The process consists of the following steps:

(i) Convene a group of subject matter experts and lead them through a process of using pairwise comparisons to rank all of the fundamental objectives in terms of importance on a scale from 1 to 9, resulting in a 6 x 6 reciprocal matrix;

(ii) Compute the eigenvalues of the matrix and find the eigenvector corresponding to the largest eigenvalue λ_{max};

(iii) Normalize the largest eigenvector and decompose it into its 6 components;

(iv) Assign the components of the normalized eigenvector as weights to the corresponding objectives on the Value Tree;

(v) Compute the consistency index (CI) as CI = $(\lambda_{max} – n) / (n – 1)$ where n=6; and

(vi) Compute the consistency ratio (CR), as the ratio of the CI for a particular set of judgments, to the random index (RI) for a matrix of the same size as published by Saaty. If CR is less than 10%, the judgment results are considered acceptable.

The importance scale used in the first step of the pairwise comparison process is given in Table 23.

Table 23: Pairwise Comparison Intensity of Importance Scale

Intensity of Importance	Definition	Explanation
1	Equal Importance	Two actions contribute equally to the objective
2	Weak or slight	-
3	Moderate importance	Experience and judgement slightly favour one action over another
4	Moderate plus	-
5	Strong importance	Experience and judgement strongly favour one action over another
6	Strong plus	-
7	Very strong or demonstrated importance	An action is favoured very strongly over another; its dominance demonstrated in practice
8	Very, very strong	-
9	Extreme importance	The evidence favouring one action over another is of the highest possible order of affirmation
Reciprocals of above	If action *i* has one of the above non-zero numbers assigned to it when compared with action *j*, then *j* has the reciprocal value when compared with *i*	A reasonable assumption
1.1–1.9	If the actions are very close	May be difficult to assign the best value but when compared with other contrasting actions the size of the small numbers would not be too noticeable, yet they can still indicate the relative importance of the activities.

The subject matter expert group compared all pair combinations of fundamental objectives in terms of their relative importance in contributing to the overall objective using the scale shown in

Table 23. When assigning relative importance to a fundamental objective, the following were taken into account:

- Issues relating to fitness for service and aging management are more important than the other issues;
- Issues involving conformance with modern design standards and nuclear safety analysis standards are of less importance given the limited timeframe of extended operation; and
- Pickering NGS has mature management processes as well as environmental and radiation practices. Issues involving marginal improvements to these practices are therefore less important.

The pairwise comparisons are documented in Table 24 for each pair of objectives (X, Y). If objective X was favoured over Y, the number associated with the strength of relative importance from the table above was inserted in the Favour X column. If Y was favoured over X the number was inserted in the Favour Y column. The Basis column provides the rationale for the assessment.

Table 24: Pairwise Comparison

Favour X	Objective X	Objective Y	Favour Y	Basis
-	1. Enhanced confidence in the design of SSCs	2. Enhanced confidence in the fitness for service of SSCs	2	Both objectives are, in principle, of similar importance for supporting continued safe operation. Given that the proposed continued operation is for a limited period, the practicality of making design changes is generally limited, and therefore taking actions necessary to ensure confidence in the fitness for service (FFS) of SSCs are considered the more important actions for ensuring safe operation. Accordingly, FFS of SSCs is considered to be somewhat more important than design changes that would be implemented for the purpose of enhancing conformance with the requirements of new codes and standards.
-	1. Enhanced confidence in the design of SSCs	3. Enhanced confidence in the safety analyses	2	Both objectives are, in principle, of similar importance for supporting continued safe operation. Enhanced confidence in the safety analyses of the actual aged conditions of the plant is somewhat more important in comparison to introducing design changes for the purpose of enhancing conformance with the requirements of new codes and standards. This is in particular true given that the proposed continued operation is for

Favour X	Objective X	Objective Y	Favour Y	Basis
				a limited period.
2	1. Enhanced confidence in the design of SSCs	4. Enhanced confidence in safety performance and feedback of experience	-	Consideration of lessons learned from CANDU Owners Group (COG) and international Operating Experience in operating plants is important in ensuring safe operation of Pickering NGS. However, introducing practicable design modifications can significantly contribute to enhancing the confidence in continued safe operation by demonstrating conformance with modern codes and standards. Hence, enhanced confidence in design is somewhat more important in comparison to incorporating feedback of lessons learned and/or OPEX.
2	1. Enhanced confidence in the design of SSCs	5. Enhanced confidence in the management processes	-	The management and governance of Condition Assessments, periodic inspections and updates of life cycle management plans is important for ensuring FFS of safety significant SSCs and play an important role in maintaining safe operation. However, given the maturity of the Pickering NGS management processes this is somewhat less important than introducing design changes for the purpose of enhancing conformance with the requirements of new codes and standards.
2	1. Enhanced confidence in the design of SSCs	6. Enhanced confidence in the radiation and environmental protection	-	Regulatory and public expectations with respect to maintaining progressively lower radiological releases and environmental impact continue to be a prominent topic. In addition, the recent OPEX from the 2011 Fukushima accident resulted in the implementation of initiatives that are designed to minimize potential releases and environmental impacts of BDBAs. However, radiation and environmental practices at Pickering NGS are mature and successful. Allocating additional resources to marginally improve the capability to maintain a low environmental impact under normal operating conditions, as well as emergencies, is therefore somewhat less important than enhanced confidence in the design basis of the SSCs.
2	2. Enhanced confidence in the fitness for service of -	3. Enhanced confidence in the safety analyses	-	In considering extended plant operation, FFS of SSCs to ensure their continued functional capability is considered to be important for safe and reliable operation. Similarly, the plant safety case must be demonstrated based on modeling

Favour X	Objective X	Objective Y	Favour Y	Basis
	SSCs			the actual configuration and condition of the plant by considering the planned extended operating period to maintain safety margins. However due to its direct impact, FFS of SSCs is judged to be somewhat more important than safety analysis.
2	2. Enhanced confidence in the fitness for service of SSCs	4. Enhanced confidence in safety performance and feedback of experience	-	In considering extended plant operation, FFS of SSCs to ensure their continued functional capability is considered to be important for safe and reliable operation. Consideration of lessons learned from COG and international Operating Experience in operating plants is also important in ensuring safe operation, however due to its more direct impact, FFS of SSCs is considered somewhat more important in comparison with OPEX feedback.
2	2. Enhanced confidence in the fitness for service of SSCs	5. Enhanced confidence in the management processes	-	In considering extended plant operation, FFS of SSCs to ensure their continued functional capability is considered to be important for safe and reliable operation. The management and governance of Condition Assessments, periodic inspections and updates of Life Cycle Management Plans are also important for ensuring FFS of safety significant SSCs and accordingly play an important role in maintaining safe operation. However due to its more direct impact, FFS of SSCs is considered somewhat more important in comparison with the confidence in management processes.
3	2. Enhanced confidence in the fitness for service of SSCs	6. Enhanced confidence in the radiation and environmental protection	-	In considering extended plant life, FFS of SSCs to ensure their continued functional capability is considered to be important for safe and reliable operation. Improving the capability to maintain a low environmental impact under normal operating conditions, as well as emergencies, is also important. However, the direct impact of FFS is a moderately more important factor because of its role in ensuring event-free operation, which would lead to reducing the risk of radiological releases and any impact on the environment.
2	3. Enhanced confidence in the safety	4. Enhanced confidence in safety performance	-	The plant safety case is demonstrated based on modeling the actual configuration and condition of the plant by considering the planned extended operation to maintain safety margins. It is also

Favour X	Objective X	Objective Y	Favour Y	Basis
	analyses	and feedback of experience		important to consider lessons learned from COG and international operating experience in operating plants, however, the safety analysis of the actual plant conditions is somewhat more important in comparison to consideration of OPEX.
2	3. Enhanced confidence in the safety analyses	5. Enhanced confidence in the management processes	-	The plant safety case is demonstrated based on modeling the actual configuration and condition of the plant by considering the planned extended life to maintain safety margins. The safety analysis of the actual plant conditions is deemed somewhat more important in comparison to marginal improvements in already mature management processes.
2	3. Enhanced confidence in the safety analyses	6. Enhanced confidence in the radiation and environmental protection	-	The plant safety case is demonstrated based on modeling the actual configuration and condition of the plant by considering the planned extended operation to maintain safety margins. Improving the capability to maintain a low environmental impact under normal operating conditions, as well as emergencies, is also important, however, demonstrating that the safety margins are maintained also contributes to enhancing the confidence in the radiation and environmental protection and, overall, is somewhat more important than enhancing the confidence in already mature radiation and environmental protection practices.
2	4. Enhanced confidence in safety performance and feedback of experience	5. Enhanced confidence in the management processes	-	Consideration of lessons learned from COG and international operating experience in operating plants is important in ensuring safe operation of Pickering NGS. This is considered to have a relatively more direct impact on and is somewhat more important for the enhanced confidence in extended operation than enhancing confidence in the management processes.
2	4. Enhanced confidence in safety performance and feedback of experience	6. Enhanced confidence in the radiation and environmental protection	-	Consideration of lessons learned from COG and international operating experience in operating plants is important in ensuring safe operation of Pickering NGS. Demonstrating that such feedback is considered in the safety case would play an important role in ensuring that the risk of radiological releases and environmental impact is not increased due to extended operation. This is

Favour X	Objective X	Objective Y	Favour Y	Basis
				considered to have a somewhat more important impact than enhancing confidence in the already mature radiation and environmental protection practices.
2	5. Enhanced confidence in the management processes	6. Enhanced confidence in the radiation and environmental protection	-	The management and governance of Condition Assessments, periodic inspections and updates of life cycle management plans are important for ensuring FFS of safety significant SSCs and accordingly play an important role in maintaining safe operation. Enhancing the confidence in the management processes, in particular those relevant to FFS of SSCs, has a contribution in enhancing the confidence that Pickering NGS will ensure safety margins during the extended operating period, which also contributes to enhancing the confidence in the radiation and environmental protection practices and, overall, is somewhat more important than enhancing confidence in the already mature radiation and environmental protection practices.

The matrix resulting from the pairwise comparison process is shown in Table 25.

Table 25: Objective Weights Matrix

		1	2	3	4	5	6
		Enhanced confidence that the design of SSCs support modern safety practices	Enhanced confidence in the fitness for service of SSCs	Enhanced confidence in the safety analyses	Enhanced confidence in operational safety performance and feedback of experience	Enhanced confidence in the management processes	Enhanced confidence in the radiation and environmental protection
1	Enhanced confidence that the design of SSCs support modern safety practices	1.00	0.50	0.50	2.00	2.00	2.00
2	Enhanced confidence in the fitness for service of SSCs	2.00	1.00	2.00	2.00	2.00	3.00
3	Enhanced confidence in the safety analyses	2.00	0.50	1.00	2.00	2.00	2.00
4	Enhanced confidence in operational safety performance and feedback of experience	0.50	0.50	0.50	1.00	2.00	2.00
5	Enhanced confidence in the management processes	0.50	0.50	0.50	0.50	1.00	2.00
6	Enhanced confidence in the radiation and environmental protection	0.50	0.33	0.50	0.50	0.50	1.00

The weight of each of the objectives is obtained by solving for the normalized components of the eigenvector associated with the principal eigenvector of the matrix. The results are as follows:

- Principal Eigenvalue = 6.224;
- Normalized Eigenvector components: 0.172, 0.288, 0.217, 0.136, 0.108, 0.079; and
- The Consistency Ratio of this matrix is 3.61% which is within the acceptance limit of 10%.

Table 26 shows the resultant weights for each of the six fundamental objectives. Enhanced confidence in the fitness for service is the highest ranked objective, followed by Enhanced confidence in the safety analyses. These weights reflect the earlier discussion that issues relating to fitness for service and aging management are more important than the other issues.

Table 26: Fundamental Objective Weight

No.	Fundamental Objective	Weight
1	Enhanced confidence that the design of SSCs support modern safety practices	0.17
2	Enhanced confidence in the fitness for service of SSCs	0.29
3	Enhanced confidence in the safety analyses	0.21
4	Enhanced confidence in operational safety performance and feedback of experience	0.14
5	Enhanced confidence in the management processes	0.11
6	Enhanced confidence in the radiation and environmental protection	0.08

C.6. Assigning Scores to Proposed Resolution Statements

Once the weights for the fundamental objectives in the Value Tree have been determined, each proposed Resolution Statement is assigned to one of the fundamental objectives. The next step is to assign a utility score to the proposed Resolution Statement which reflects the degree to which completing the activities in the proposed Resolution Statement will address the associated objective.

The Utility Function method of scoring is used to account for potential uncertainty in how effective a proposed Resolution Statement may be in terms of addressing its associated fundamental objective, and for potential uncertainty in the time it will take for the benefit to occur. Scores are assigned to proposed Resolution Statements using a two-variable utility function that considers time and impact attributes, as follows:

- The **Time** attribute measures the impact of implementing the proposed Resolution Statement by answering the following question: “How long will it take to implement the resolution and to see the associated objective being realized?”
- The **Impact** attribute measures how directly or strongly the issue impacts the fundamental objective by asking: “If one could somehow resolve the Global Issue immediately (i.e., the proposed Resolution Statement happens overnight), how direct or big would the impact be in terms of realizing the associated objective?”

For ease of use, a discrete and not continuous utility function is used with the input variables associated with the two attributes being assigned a value, using a rating system on a scale of 1 to 5 as presented in Table 27 and Table 28.

Table 27: Rating System for the Time Attribute

Time Rating	Definition
1	Resolving the issue and to have it affect the objective will take at least 5 years
2	Resolving the issue and to have it affect the objective will take 4 to 5 years
3	Resolving the issue and to have it affect the objective will take 3 to 4 years
4	Resolving the issue and to have it affect the objective will take 2 to 3 years
5	Resolving the issue and to have it affect the objective will take 0 to 2 years

Table 28: Rating System for the Impact Attribute

Impact Rating	Definition
1	Resolving the issue will have an indirect and negligible impact on the objective
2	Resolving the issue will have an indirect and minor or incremental impact on the objective
3	Resolving the issue will have a direct and minor or incremental impact on the objective
4	Resolving the issue will have an indirect and major impact on the objective
5	Resolving the issue will have a direct and major impact on the objective

To assist in the assignment of Impact ratings, guidance is provided in the matrix presented in Table 29. Generally, these are maximum values; actual values also take into account the specific nature of the proposed Resolution Statement.

Table 29: Impact Guidance Matrix

Type of Global Issue Resolution Statement			Nature of Improvement in Terms of Barriers and Practices			
			1	2	3	4
			New Barriers and Practices	Augmentation of the Current Barriers and Practices	Improvement of the Current Barriers and Practices	Modernization of Current Barriers and Practices
1	Design Change and Implementation		(5)	(5)	(4)	(3)
2	Improve Operational or Safety Performance – (Fitness for Service Assessment, Maintenance, Configuration Management, Prevention of/Response to Events)		(5)	(5)	(4)	(3)
3	Reduce Uncertainty – (Analysis, Safe Operating Envelope)		(4)	(3)	(2)	(1)
4	Improve Managed System and Organizational Effectiveness- (Process, program, procedure)	a. Field Impact (e.g., operating, outage, maintenance, field procedures)	(4)	(3)	(2)	(1)
		b. Impact on managed system and support processes (e.g., training)	(3)	(2)	(2)	(1)

Notes:

1. Values in parentheses represent guidance for Impact Rating defined in Table 28.
2. “New” means there is no barrier or practice currently in place.
3. “Augmentation” means current barrier is not complete or execution gaps in current practices or the modern codes and standards have complementary requirements that are not currently in place (e.g., FFS assessments to address execution gaps for Major Components is considered as direct and major impact on the objective).
4. “Improvement” means current barriers or practices require additional supporting assessment (e.g., other supporting FFS assessments for SSCs is considered here as major and indirect impact on the objective).
5. “Modernization” means current barriers or practices are effective but documentation and/or practices need to be updated to reflect current trends, state of the art approaches, terminology, etc. with no impact on operational performance (e.g., FFS assessments of non-Major Components is considered as direct and minor impact on the objective).
6. Replacement of any original SSCs is considered as a Design Change.

The characteristics of the impact of resolving the activity (i.e., Direct, Indirect, Major, Minor) are considered within the context of: 1-Design Changes and Implementation, 2-Improving Operational or Safety Performance, 3-Reduction of Uncertainty and 4-Improvement of Enablers to maintain or enhance the safe operation of the plant over the extended operating period. These aspects are considered with respect to their contribution to addition of new barriers, and augmentation, improvement or modernization of current barriers and practices.

Once the time and impact ratings are determined, the utility score for each is determined based on the following utility function:

$$U(x) = 1 - e^{-x/R}$$

Where:

- U is the utility score of attribute x, with $0 < U < 1$
- x is the rating of the attribute on some scale (in this case, the 1 to 5 scale used for Impact and Time ratings).
- R is an adjustable parameter that expresses a judgment about the relative importance of Impact and Time. By adjusting R the utility is more strongly weighted towards solutions with high Impact or Time ratings.

The combined utility of the Time and Impact parameters is determined using the following formula which integrates the utility function discussed above:

$$U(i,t) = k_i\, U_I(i) + k_t\, U_T(t) + (1\text{-}k_i\text{-}k_t)\, U_I(i)U_T(t)$$

Where:

- $U(i,t)$ is the utility score of the proposed Resolution Statement, taking into account the Time and Impact ratings each assessed on a 1 to 5 scale
- k_i is the contribution of the Impact attribute
- $U_I\ = 1 - e^{-i/RI}$
- k_t is the contribution of the Time attribute
- $U_T\ = 1 - e^{-t/RT}$

Consistent with previous PSRs the values for each k and R are:

- k_i = 0.3
- R_I = -5.0
- k_t = 0.1
- R_T = -2.0

The result is the utility matrix given in Table 30. To use the matrix, the Time and Impact ratings are determined on a 1 to 5 scale, and the resulting utility (score) associated with resolving the proposed Resolution Statement is given by the number in the cell associated with the Time and Impact rating.

Table 30: Utility Matrix

		Time				
		1	2	3	4	5
Impact	1	0.00	0.01	0.03	0.05	0.10
	2	0.05	0.08	0.11	0.17	0.26
	3	0.12	0.15	0.21	0.31	0.46
	4	0.20	0.25	0.34	0.48	0.70
	5	0.30	0.37	0.49	0.68	1.00

The matrix introduces a preference towards solutions that:

- Have greater impact on the objective in a shorter time, i.e., (4,4) is 6 times more preferable than (2,2).
- Have larger impact versus one that can be done quickly but with little impact, i.e., (1,5) is three times more preferable than (5,1).

C.7. Ranking Results

The Ranking Value of each proposed Resolution Statement is determined as follows:

- Each proposed Resolution Statement is assigned to one of the fundamental objectives in Table 26 and is given the Weight associated with the fundamental objective shown in the table;
- The Time Rating of the proposed Resolution Statement is determined on a scale of 1 to 5 using Table 27;
- The Impact Rating of the proposed Resolution Statement is determined on a scale of 1 to 5 using Table 28 and Table 29;
- The Impact and Time Ratings are used to read off the Utility Score from Table 30;
- The Ranking Value of the proposed Resolution Statement is calculated by multiplying its Utility Score with its Weight; and
- The resulting ranking is subjected to review and rationalization by the PSR2 Expert Panel and OPG PSR2 Project Staff. This allows for the use of engineering judgment and considerations such as the impact of the ranking on multiple objectives to arrive at a final rank order.

The results of applying the ranking methodology are shown in Table 31. For each proposed Resolution Statement, the associated Value Tree objective is shown, along with the Objective Weight and the values of the Time and Impact attributes, along with the rationale for each. Finally, the Utility Score and Ranking Value are shown.

The results in Table 31 are normalized to 100, and the normalized Ranking Values of the proposed Resolution Statements are shown in decreasing order in Table 18.

Table 31: Global Issue Proposed Resolution Statement Ranking

Global Issue/RS Title	Objective	Objective Weight	Objective Justification	Time Attribute	Time Attribute Justification	Impact Attribute	Impact Attribute Justification	Utility Score	GI-RS Ranking Value
GI-1-RS1 Complete CSA N285.8 Compliance Plan activities, including responding to comments.	2. Enhanced Confidence in the fitness for service of SSCs	0.29	The issue is indirectly related to FFS for Fuel Channels.	5	Issue resolution can occur within 2 years, with immediate impact on the objective.	4	Completing the Compliance Plan activities has an indirect and major impact on the confidence in FFS of Fuel Channels.	0.70	0.203
GI-1-RS2 Review and revise if/as required the CSA N285.4 compliant Periodic Inspection Plans for Fuel Channels for Pickering NGS to cover the extended operating period.	2. Enhanced Confidence in the fitness for service of SSCs	0.29	The issue is indirectly related to FFS for Fuel Channels.	5	Issue resolution can occur within 2 years, with immediate impact on the objective.	4	Completing the PIP has an indirect and major impact on the confidence in FFS of Fuel Channels.	0.70	0.203
GI-1-RS3 Update the Fuel Channels LCMP for Pickering 1,4 for the extended operating period.	2. Enhanced Confidence in the fitness for service of SSCs	0.29	The issue is directly related to FFS for Fuel Channels.	5	Issue resolution can occur within 2 years, with immediate impact on the objective.	5	Completing the required LCMP update has a direct and major impact on the confidence in FFS of the Fuel Channels.	1.00	0.290
GI-1-RS4 Update the structure of the Fuel Channels LCMP.	2. Enhanced Confidence in the fitness for service of SSCs	0.29	The issue is directly linked to operating licence conditions.	5	Issue resolution can occur within 2 years with immediate impact on the objective.	4	Updating the LCMP for Fuel Channels will have an indirect and major impact on the objective.	0.70	0.203
GI-2-RS1 Update the Feeders Life Cycle Management Plan for Pickering 1,4 for the extended operating period based on updated fitness for service assessment.	2. Enhanced Confidence in the fitness for service of SSCs	0.29	The issue is directly related to FFS for Feeders.	5	Issue resolution can occur within 2 years, with immediate impact on the objective.	5	Completing the required LCMP update has a direct and major impact on the confidence in FFS of the Feeders.	1.00	0.290
GI-3-RS1 Update the Steam Generators Life Cycle Management Plan for Pickering 1,4 for the extended operating period based on updated fitness for service assessment.	2. Enhanced Confidence in the fitness for service of SSCs	0.29	The issue is directly related to FFS for Steam Generators.	5	Issue resolution can occur within 2 years, with immediate impact on the objective.	5	Completing the required LCMP update has a direct and major impact on the confidence in FFS of the Steam Generators.	1.00	0.290
GI-4-RS1 Update the Reactor Components and Structures Life Cycle Management Plan for Pickering 1,4 for the extended operating period based on updated fitness for service assessment.	2. Enhanced Confidence in the fitness for service of SSCs	0.29	The issue is directly related to FFS for Reactor Components.	5	Issue resolution can occur within 2 years, with immediate impact on the objective.	5	Completing the required LCMP update has a direct and major impact on the confidence in FFS of the Reactor Components.	1.00	0.290
GI-4-RS2	2. Enhanced Confidence in the fitness for service	0.29	The issue is directly related to FFS for Reactor	5	Issue resolution can occur within 2 years, depending	5	Completing the required measurements has a direct	1.00	0.290

Global Issue/RS Title	Objective	Objective Weight	Objective Justification	Time Attribute	Time Attribute Justification	Impact Attribute	Impact Attribute Justification	Utility Score	GI-RS Ranking Value
Perform measurements of Calandria Tube/Liquid Injection Shutdown System nozzle gaps on Units 5-8 to refine the gap closure rates. Using this new measurement data, update analyses as required, to demonstrate Fitness for Service.	of SSCs		Components.		on the outage schedule, with immediate impact on the objective.		and major impact on the confidence in FFS of Calandria Tube/LISS nozzles.		
GI-5-RS1 Confirm the adequacy of the service limits assessments for Nuclear Class 1 Piping (Excluding Major Components) after accounting for any impact of environmental factors	2. Enhanced Confidence in the fitness for service of SSCs	0.29	Related to SF2 and Condition Assessment.	5	Issue resolution can occur within 2 years, with immediate impact on the objective.	4	The impact of environmental factors has an indirect and major impact on the FFS of HTS components.	0.70	0.203
GI-6-RS1 Reassess the impact of the changes in the cable Criticality Coding and update the scope of the cable surveillance plan.	2. Enhanced Confidence in the fitness for service of SSCs	0.29	Related to Cable Surveillance Program risk assessment and Condition Assessments.	4	Implementation of the Cable Surveillance program taking into account the updated equipment Criticality Coding will take place in 2-3 years.	3	The resolution of this issue will have a direct and minor impact on the objective.	0.31	0.090
GI-7-RS1 Update the Buried Piping Program asset management plan and risk ranking for the extended operating period.	2. Enhanced Confidence in the fitness for service of SSCs	0.29	Related to SF2 and Condition Assessment.	4	The update of the Buried Piping Program risk assessment would take less than two years, however, its implementation and accordingly its impact on the FFS of buried piping would take 2-3 years.	3	The resolution of this issue will have a direct and minor impact on the objective.	0.31	0.090
GI-7-RS2 Update governance to reflect a graded approach in the event that leakage in fuel oil piping occurs.	4. Enhanced confidence in safety performance and feedback of experience	0.14	Graded approach supports maintaining Systems Important to Safety in-service.	3	AR# 28175307 currently has corrective actions in place and is expected to be completed by Q1 2020. Resolving the issue will take 3 to 4 years to have its effect on the objective.	2	The resolution of this issue will have an indirect and minor impact on the objective.	0.11	0.015
GI-8-RS1 Complete and update CAs for the piping systems and commodity groups in PSR2 scope for station operation for the extended operating period.	2. Enhanced Confidence in the fitness for service of SSCs	0.29	Condition Assessments are related to fitness for service.	5	Issue resolution can occur within 2 years, with immediate impact on the objective.	3	The resolution of this issue will have a direct and minor impact on the objective. The process established to address Condition Assessment recommendations will have a direct impact on FFS of affected SSCs.	0.46	0.133
GI-8-RS2	2. Enhanced Confidence	0.29	Condition Assessments are	5	Issue resolution can occur	3	The process established to	0.46	0.133

Global Issue/RS Title	Objective	Objective Weight	Objective Justification	Time Attribute	Time Attribute Justification	Impact Attribute	Impact Attribute Justification	Utility Score	GI-RS Ranking Value
Develop and implement a process to track and report aging-management-related actions from the Condition Assessment recommendations.	in the fitness for service of SSCs		related to fitness for service.		within 2 years, with immediate impact on the objective.		address Condition Assessment recommendations will have a direct and minor impact on FFS of affected SSCs.		
GI-9-RS1 Complete the required assessment to support the current fuel basket stacking arrangements in the Pickering IFBs.	3. Enhanced confidence in the safety analyses	0.21	The resolution of this Global Issue is related to the seismic qualification that would impact the safety analyses of the IFBs.	5	Issue resolution can occur within 2 years, with immediate impact on the objective.	3	The resolution would have a direct and minor impact on the objective.	0.46	0.097
GI-10-RS1 Complete the Pickering 5-8 IFB Leakage Mitigation Project to mitigate leaks from IFB-B to the interspace.	2. Enhanced Confidence in the fitness for service of SSCs	0.29	The issue is directly related to FFS of the IFBs.	5	The Pickering 5-8 IFB Leakage Mitigation Project is expected to be complete within 2 years, with immediate impact on the objective.	3	Resolving the issue will have a direct and minor impact on the FFS of IFBs.	0.46	0.133
GI-12-RS1 Complete EQA re-assessments to support the extended operating period.	2. Enhanced Confidence in the fitness for service of SSCs	0.29	Environmental qualifications of the equipment and their Condition Assessments impact the FFS.	5	Issue resolution can occur within 2 years, with immediate impact on the objective.	3	Environmental Qualification of equipment has a direct and minor impact on the FFS.	0.46	0.133
GI-19-RS1 Demonstrate the FFS of the foundation steel H-piles for the Pickering A Reactor Building, Vacuum Building and Pressure Relief Duct at the Pickering site for the extended operating period.	2. Enhanced Confidence in the fitness for service of SSCs	0.29	The demonstration of FFS includes inspections, testing or analysis to confirm the integrity of foundation steel H-piles.	5	Issue resolution can occur within 2 years, with immediate impact on the objective.	3	Resolving the issue will have a direct and minor impact on the objective, given the results to date.	0.46	0.133
GI-24-RS1 Update Heat Transport System aging safety analysis models and perform the required safety analysis of events most impacted by aging (SBLOCA, LOF and Neutron Overpower (NOP)) to support extended operation.	3. Enhanced confidence in the safety analyses	0.21	Aging analysis models need to be developed and the safety analysis impacted by aging is required to be updated to demonstrate the adequacy of safety margins for extended operation.	5	Issue resolution can occur within 2 years, with immediate impact on the objective.	5	The resolution of this issue will have a direct and major impact on the objective. Completing the required safety analysis will directly support the safety case for continuing operation.	1.00	0.210
GI-25-RS1 Complete the re-categorization of the Large Break LOCA (LBLOCA) CANDU Safety Issues to Category 2.	3. Enhanced confidence in the safety analyses	0.21	The Resolution Statements will be the bases for OPG request to re-classify the CSIs on large break LOCA to category 2 to support Pickering safety case.	5	Given the industry progress in addressing the findings of CNSC staff reviews, it is expected that the re-categorization will be approved by CNSC with immediate impact on the objective.	2	Resolving the issue will have an indirect and minor impact on the objective as work has already been performed and CNSC approval is pending.	0.26	0.055

Global Issue/RS Title	Objective	Objective Weight	Objective Justification	Time Attribute	Time Attribute Justification	Impact Attribute	Impact Attribute Justification	Utility Score	GI-RS Ranking Value
GI-25-RS2 Complete the re-categorization of CANDU Safety Issue CSI-IH6 for Pickering to Category 2. (Pickering 1,4 high-energy piping)	3. Enhanced confidence in the safety analyses	0.21	The Resolution Statement will be the bases for OPG request to re-classify the CSI-IH6 to category 2 to support Pickering safety case.	5	Issue resolution is expected soon, with immediate impact on the objective.	3	Resolving the issue will have a direct and minor impact on the objective.	0.46	0.097
GI-26-RS1 Complete the emergency response projection enhancements identified in Action Item 2016-OPG-7469: Implementation of Emergency Response Projection Computer Code Upgrades.	5. Enhanced confidence in the management processes	0.11	The issue is related to the development of an improved emergency response tool to predict emergency response projections.	5	Issue resolution can occur within 2 years, with immediate impact on the objective.	5	Resolving the issue will have direct and major impact on the objective.	1.00	0.110
GI-27-RS1 Complete actions from PSA improvement Plan.	3. Enhanced confidence in the safety analyses	0.21	PSA and analysis refinements will enhance the confidence in plant overall risk.	5	Issue resolution can occur within 2 years, with immediate impact on the objective.	3	The resolution will have a direct and minor impact on the objective, given that the high impact work has been completed.	0.46	0.097
GI-27-RS2 Investigate and implement additional practicable design, operational and/or analytical enhancements to further improve Pickering 1,4 Severe Core Damage Frequency and Large Release Frequency (e.g., alternative emergency cooling water makeup).	3. Enhanced confidence in the safety analyses	0.21	Practicable design and/or operational enhancements, will reduce plant overall risk.	5	Significant progress on issue resolution can occur within 2 years with immediate impact on the objective.	5	The resolution will have a direct and major impact on the objective.	1.00	0.210
GI-31-RS1 Complete the Pickering NGS Implementation Plan for REGDOC-2.4.1.	3. Enhanced confidence in the safety analyses	0.21	This is a requirement in the Licence Conditions Handbook that is directly relevant to the confidence in the safety analysis.	4	Issue resolution may take 2-3 years with immediate impact on the objective.	3	Resolving the issue will have a direct and minor impact on the objective.	0.31	0.065
GI-31-RS2 Prepare Implementation Plan update for REGDOC-2.4.1 including consideration of the impact of the extended operating period.	3. Enhanced confidence in the safety analyses	0.21	The updated plan will identify any changes required to support the continued safe operation of Pickering NGS.	4	Issue resolution may take 2-3 years with immediate impact on the objective.	3	Resolving the issue will have direct and minor impact on the objective	0.31	0.065
GI-32-RS1 Complete the activities in the REGDOC-2.4.2 Implementation Strategy and update the Strategy in the context of the additional operating period.	3. Enhanced confidence in the safety analyses	0.21	The plan will identify scope of the REGDOC-2.4.2 implementation. This will enhance the confidence in PSA in support of the continued safe operation of	4	Issue resolution may take 2-3 years with immediate impact on the objective.	3	Resolving the issue will have a direct and minor impact on the objective.	0.31	0.065

Global Issue/RS Title	Objective	Objective Weight	Objective Justification	Time Attribute	Time Attribute Justification	Impact Attribute	Impact Attribute Justification	Utility Score	GI-RS Ranking Value
			Pickering NGS.						
GI-40-RS1 Ensure the completion of EME Phase 2 activities.	1.Enhanced confidence that the design of SSCs support modern safety practices	0.17	Completion of EME Phase 2 provisions of complementary design features will enhance the confidence in preserving Containment integrity during postulated events.	5	Issue resolution can occur within 2 years, with immediate impact on the objective.	5	Resolving the issue will have a direct and major impact on the objective.	1.00	0.170
GI-43-RS1 Perform the scope of inspections for non-Containment safety-significant civil structures as per the established Preventive Maintenance program (PM 00121151).	2. Enhanced confidence in the fitness for service of SSCs	0.29	Completing the inspections for these structures will confirm the FFS of these structures for extended operation.	5	Issue resolution can occur within 2 years, with immediate impact on the objective.	3	Resolving the issue will have a direct and minor impact on the objective. The Major Components are covered in other Global Issues.	0.46	0.133
GI-43-RS2 Develop program governance using a risk based approach for aging management of safety-significant civil structures for the extended operating period. This applies to non-Containment Safety-Related Civil Structures.	2. Enhanced Confidence in the fitness for service of SSCs	0.29	Completing this task will support fitness for service of the safety-related civil structures.	5	Issue resolution can occur within 2 years, with immediate impact on the objective	2	Resolving the issue will have an indirect and minor impact on the objective.	0.26	0.075
GI-43-RS3 Prepare Condition Assessments as appropriate for safety-significant civil structures for the extended operating period. Recommendations from these Condition Assessments will be tracked and reported along with those related to GI-8. This applies to non-Containment Safety-Related Civil Structures.	2. Enhanced Confidence in the fitness for service of SSCs	0.29	Completing the Condition Assessments for these structures will confirm the FFS of these structures for extended operation.	4	Issue resolution may take 2-3 years with immediate impact on the objective.	3	The resolution of this issue will have a direct and minor impact on the objective. The Major Components are covered in other Global Issues.	0.31	0.090
GI-47-RS1 Complete installation of locks on the 058 Yard Fire Protection System.	1.Enhanced confidence that the design of SSCs support modern safety practices	0.17	Completing these tasks will close a minor deviation from the Fire Protection Code which enhances confidence in the design of the SSCs.	5	Issue resolution can occur within 2 years, with immediate impact on the objective.	3	The resolution of this issue will have a direct and minor impact on the objective.	0.46	0.078
GI-48-RS1 Provide, as necessary, design and/or operational changes and commissioning/testing to facilitate required interconnection of Pickering 1,4 and Pickering 5-8 Fire Protection System water supplies to meet the safety intent of CSA N293-12 Clause 7.3.2.2 (d).	1.Enhanced confidence that the design of SSCs support modern safety practices	0.17	Completing the changes for meeting the safety intent of the CSA N293-12 clause 7.3.2.2 (d) will enhance the confidence in the design of the SSCs.	5	Issue resolution can occur within 2 years, with immediate impact on the objective.	5	Resolving the issue will have a direct and major impact on the objective.	1.00	0.170

Global Issue/RS Title	Objective	Objective Weight	Objective Justification	Time Attribute	Time Attribute Justification	Impact Attribute	Impact Attribute Justification	Utility Score	GI-RS Ranking Value
GI-50-RS1 Revise the N285.4 PIPs and governance to align with elements of N285.4-14	2. Enhanced Confidence in the fitness for service of SSCs	0.29	Alignment with CSA N285.4-14 will enhance the confidence in the PIP program.	3	Issue resolution can occur within 3-4 years with immediate impact on the objective.	3	Resolving the issue will have a direct and minor impact on the objective.	0.21	0.061
GI-50-RS2 Assess the impact of extended operation on concessions against CSA N285.4	2. Enhanced Confidence in the fitness for service of SSCs	0.29	Alignment with CSA N285.4-14 will enhance the confidence in the PIP program.	5	Issue resolution can occur within 2 years, with immediate impact on the objective.	2	Resolving the issue will have an indirect and minor impact on the objective.	0.26	0.075

Appendix D – Review of Safety Principles

D.1. Methodology

As described in Section 18.1, the steps in the assessment of safety principles are:

- Identification of the safety principles from [IAEA SRS-46, *Assessment of Defence in Depth for Nuclear Power Plants*, February 2005] that are applicable to the defence-in-depth review.
- Establishment of the defence-in-depth levels impacted for each applicable safety principle (taken from IAEA SRS-46).
- Mapping of each safety principle to the relevant Safety Factor(s).
- Assessment of the adequacy of the Pickering NGS design and operation with respect to each safety principle.

In the following sub-sections, each of the first three steps is described further. The results of the fourth step, the assessment of each principle, are discussed in Sections D.2 to D.12. Information on the design features of Units 1,4 and 5-8 is taken primarily from the station Safety Reports [NA44-SR-01320-00001-R016, *Pickering A Safety Report*, July 20, 2017], [NK30-SR-01320-00002-R004, *Pickering B Safety Report – Part 2*, October 10, 2012].

D.1.1. Identification of Applicable Safety Principles

[IAEA INSAG-12, *Basic Safety Principles for Nuclear Power Plants*, 75-INSAG-3 Rev. 1, October 1999] lists three safety objectives related to general nuclear safety, radiation protection and technical safety for nuclear power plants, as well as accompanying safety principles to achieve these safety objectives. Fifty-four specific safety principles are defined in the following eight categories:

- **Siting:** Four safety principles
- **Design:** Twenty-five safety principles
- **Manufacturing and Construction:** Two safety principles
- **Commissioning:** Four safety principles
- **Operation:** Twelve safety principles
- **Accident Management:** Three safety principles
- **Decommissioning:** One safety principle
- **Emergency Preparedness:** Three safety principles

The safety principle related to decommissioning was not considered in [IAEA SRS-46, *Assessment of Defence in Depth for Nuclear Power Plants*, February 2005] and is thus not included in the PSR2 Defence-in-Depth Assessment. This is considered acceptable since PSR2 is focused on operation, not decommissioning, as described in the PSR2 Basis Document [P-REP-03680-00001-R002, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, June 2016].

Safety principle D-242, Physical Protection of Plant, pertains to security provisions, which are not in the PSR2 scope and therefore this safety principle is not considered here.

Therefore, 52 of the 54 safety principles in [IAEA INSAG-12, *Basic Safety Principles for Nuclear Power Plants*, 75-INSAG-3 Rev. 1, October 1999] are considered applicable to Pickering NGS and are assessed with respect to defence-in-depth for PSR2.

D.1.2. Establishment of Defence-in-Depth Levels Impacted for Each Safety Principle

Table 2 of [IAEA SRS-46, *Assessment of Defence in Depth for Nuclear Power Plants*, February 2005] assigns the safety principles in [IAEA INSAG-12, *Basic Safety Principles for Nuclear Power Plants*, 75-INSAG-3 Rev. 1, October 1999] to individual levels of defence-in-depth based on the descriptions of the safety principles. This association has been used for the PSR2 Defence-in-Depth Assessment.

In accordance with Table 2 of [IAEA SRS-46, *Assessment of Defence in Depth for Nuclear Power Plants*, February 2005], a number of safety principles are related to more than one level of defence, as follows:

- **Levels 1, 2, 3, 4, 5:** Three safety principles
- **Levels 1, 2, 3, 4:** Twenty safety principles
- **Levels 1, 2, 3:** Four safety principles
- **Levels 3, 4, 5:** One safety principle
- **Levels 1, 2:** Four safety principles
- **Levels 3, 4:** Five safety principles
- **Levels 4, 5:** Two safety principles

Some safety principles are related to only one level of defence, as follows:

- **Level 1:** Four safety principles
- **Level 3:** Five safety principles
- **Level 4:** Three safety principles
- **Level 5:** One safety principle

D.1.3. Mapping of Safety Principles to Relevant Safety Factor Reviews

Based on the description of each safety principle and the associated level or levels of defence, each safety principle is mapped to one or more of the Safety Factors. In the mapping of the safety principles to the Safety Factors, consideration is given to the review tasks of each Safety Factor, as well as the modern Laws, Regulations, Codes and Standards assessed for each Safety Factor.

D.1.4. List of Safety Principles

Table 32, based on Table 2 of [IAEA SRS-46, *Assessment of Defence in Depth for Nuclear Power Plants*, February 2005], lists the safety principles together with applicable levels of defence-in-depth (see Section 18 for a brief description of defence-in-depth levels), as well as the Safety Factors related to them. The table is arranged such that safety principles applicable to the greatest number of defence-in-depth levels are listed at the top. Ascending order of defence-in-depth levels is used as the secondary ordering sequence[34].

In Table 32, each safety principle is assigned a number in accordance with the number provided in Table 2 of [IAEA SRS-46, *Assessment of Defence in Depth for Nuclear Power Plants*, February 2005], preceded by one of the following:

- S – Siting
- D – Design
- M&C – Manufacture and Construction
- C – Commissioning
- O – Operation
- AM – Accident Management
- EP – Emergency Preparedness

Table 32: Relation of Safety Principles to Defence-in-Depth Levels and Safety Factors

No.	Safety Principle (extracted from Table 2 of SRS-46)	Levels	Safety Factor
S-138	Radiological Impact on the Public and the Local Environment	1, 2, 3, 4, 5	1, 6, 14
O-265	Organization, Responsibilities and Staffing	1, 2, 3, 4, 5	2, 10, 12, 13

[34] Ascending order used as the secondary ordering sequence means that the groups of safety principles that are applicable to, e.g., three defence-in-depth levels (lmn) are ordered by ascending numerical value of “lmn”. For example, the hypothetical group applicable to defence-in-depth levels 134 comes before the group applicable to defence-in-depth levels 235, since halting an event at a lower level is preferable.

No.	Safety Principle (extracted from Table 2 of SRS-46)	Levels	Safety Factor
O-296	Engineering and Technical Support of Operations	1, 2, 3, 4, 5	2, 10, 12
S-142	Ultimate Heat Sink Provisions	1, 2, 3, 4	1
D-150	Design Management	1, 2, 3, 4	1, 5, 8, 10, 12
D-154	Proven Technology	1, 2, 3, 4	1, 2, 4, 5, 7
D-158	General Basis for Design	1, 2, 3, 4	1, 3, 4, 5, 6, 7
D-186	Inspectability of Safety Equipment	1, 2, 3, 4	1, 4
D-205	Startup, Shutdown, and Low Power Operation	1, 2, 3, 4	1, 5
D-227	Monitoring of Plant Safety Status	1, 2, 3, 4	1, 3, 5, 12
D-230	Preservation of Control Capability	1, 2, 3, 4	1, 2, 4, 5, 7
M&C-246	Safety Evaluation of Design	1, 2, 3, 4	1, 5, 6, 7
M&C-249	Achievement of Quality	1, 2, 3, 4	1, 2, 3, 4, 10, 11
C-255	Verification of Design and Construction	1, 2, 3, 4	1, 4, 5
C-258	Validation of Operating and Functional Test Procedures	1, 2, 3, 4	1
C-260	Collection of Baseline Data	1, 2, 3, 4	1, 4
C-262	Pre-Operational Adjustment of Plant	1, 2, 3, 4	1, 4
O-269	Safety Review Procedures	1, 2, 3, 4	10, 11
O-290	Emergency Operating Procedures	1, 2, 3, 4	5, 6, 7, 10, 11, 12, 13
O-292	Radiation Protection Procedures	1, 2, 3, 4	1, 8, 10, 11, 12, 14, 15
O-299	Feedback of Operating Experience	1, 2, 3, 4	8, 9, 10, 15
O-305	Maintenance, Testing and Inspection	1, 2, 3, 4	2, 3, 4, 8, 10, 11, 12, 14, 15
O-312	Quality Assurance in Operation	1, 2, 3, 4	10
D-192	Protection Against Power Transient Accidents	1, 2, 3	1, 5
D-195	Reactor Core Integrity	1, 2, 3	1, 4, 5
O-278	Training	1, 2, 3	5, 8, 9, 10, 12, 13, 15
O-284	Operational Limits and Conditions	1, 2, 3	1, 2, 4, 5, 10, 11, 12
EP-339	Assessment of Accident Consequences and Radiological Monitoring	3, 4, 5	5, 13, 15
D-164	Plant Process Control Systems	1, 2	1, 2, 4, 5, 7
D-203	Normal Heat Removal	1, 2	1, 4
D-209	Reactor Coolant System Integrity	1, 2	1, 4, 5, 7
D-240	New and Spent Fuel Storage	1, 2	1, 2, 4
D-200	Automatic Shutdown Systems	3, 4	1, 4, 5, 6
D-207	Emergency Heat Removal	3, 4	1
D-217	Confinement of Radioactive Material	3, 4	1, 4
D-221	Protection of Confinement Structure	3, 4	1, 4, 5, 6

No.	Safety Principle (extracted from Table 2 of SRS-46)	Levels	Safety Factor
D-233	Station Blackout	3, 4	1, 5
EP-333	Emergency Plans	4, 5	7, 10, 11, 12, 13, 15
EP-336	Emergency Response Facilities	4, 5	1, 13
S-136	External Factors Affecting the Plant	1	1, 6, 7, 14
D-188	Radiation Protection in Design	1	1, 5, 15
O-272	Conduct of Operations	1	10, 11, 12
O-288	Normal Operating Procedures	1	10, 11, 12
D-168	Automatic Safety Systems	3	1, 2, 4, 5, 6, 7, 8
D-174	Reliability Targets	3	1, 2, 4, 5, 6, 8, 10
D-177	Dependent Failures	3	1, 3, 5, 7
D-182	Equipment Qualification	3	1, 2, 3, 4, 5, 7
D-237	Control of Accidents Within the Design Basis	3	1, 5, 11, 12
AM-318	Strategy for Accident Management	4	1, 5, 6, 13
AM-323	Training and Procedures for Accident Management	4	1, 5, 10, 11, 12, 13
AM-326	Engineered Features for Accident Management	4	1, 5, 13
S-140	Feasibility of Emergency Plans	5	1, 13

D.2. Safety Principles Related to Levels 1, 2, 3, 4, 5

The following safety principles are related to all five levels of defence-in-depth:

- S-138 – Radiological Impact on the Public and the Local Environment
- O-265 – Organization, Responsibilities and Staffing
- O-296 – Engineering and Technical Support of Operations

As demonstrated below, Pickering NGS design and operation are aligned with all safety principles related to all Levels 1, 2, 3, 4, 5.

S-138 – Radiological Impact on the Public and the Local Environment

Principle: Sites are investigated from the standpoint of the radiological impact of the plant in normal operation and in accident conditions.

The radiological impact on the public and the local environment was considered at the Pickering NGS design stage, when the site characteristics were established, as described in regards to Safety Principle S-136, External Factors Affecting the Plant. The site characteristics are documented in Part 1 of the Pickering NGS Safety Reports. The minimal impact of the operation of Pickering NGS is confirmed through the Pickering environmental monitoring program [N-REP-03443-10014-R000, *2014 Results Of Environmental Monitoring Programs*, April 20, 2015]. The environmental monitoring program has been established to ensure that activities at Pickering are conducted in a manner that minimizes the impact on the public and the environment. The objectives of the environmental monitoring program are:

1. To assess the impact on the public and the environment associated with the operation of OPG Nuclear facilities.
2. To demonstrate compliance with limits associated with releases from Pickering.
3. To demonstrate the effectiveness of the control programs.
4. To verify predictions made by Environmental Risk Assessment [P-REP-07010-10012-R000, *Environmental Risk Assessment Report For Pickering Nuclear*, January 27, 2014].

The Environmental Monitoring Program Sampling Plan [N-REP-03443-10014-R000, *2014 Results Of Environmental Monitoring Programs*, April 20, 2015] outlines the radionuclides monitored, the sampling locations, the sample types and the frequency of collection. To calculate the public dose from radiological emissions, various exposure pathways, such as food ingestion, inhalation and water ingestion, are considered. Samples are collected at station boundary locations, as well as at other specified locations.

The data are collected annually, or more frequently, e.g., milk from local dairy farms is sampled on a monthly basis, and comprehensive assessment and reporting is performed annually.

Site specific surveys allow identification of the various potential critical groups around each nuclear site. The site specific surveys are used for development of the environmental monitoring programs and site derived release limits, and for calculating public collective dose. Site-specific survey instructions are documented in OPG Instruction [N-INS-03481-10000-R000, *Instruction For Performing A Site Specific Survey For Ontario Power Generation Nuclear Sites*, September 23, 2009]. Specific responsibilities of environment operations support related to site-specific surveys are outlined in OPG Procedure [N-PROC-OP-0025-R011, *Management Of The Environmental Monitoring Programs*, January 8, 2016].

The Environmental Risk Assessment [P-REP-07010-10012-R000, *Environmental Risk Assessment Report For Pickering Nuclear*, January 27, 2014] was performed in 2014. As part of this assessment, an assessment of the radiological dose received by terrestrial and aquatic biota from air, surface water and soil was performed.

Section 1.2.4 of OPG Program [N-PROG-OP-0006-R018, *Environmental Management*, April 29, 2015] states that radiological emissions from OPG Nuclear facilities shall not exceed the derived release limits specified in station Power Reactor Operating Licences issued by the CNSC. Section 1.2.4 of OPG Program [N-PROG-OP-0006-R018, *Environmental Management*, April 29, 2015] makes reference to OPG Standard [N-STD-OP-0031-R006, *Monitoring Of Nuclear And Hazardous Substances In Effluents*, October 1, 2014], which establishes minimum requirements for the monitoring of airborne and waterborne effluents from OPG Nuclear facilities operating under normal and abnormal operating conditions.

In the OPG Standard [N-STD-OP-0031-R006, *Monitoring Of Nuclear And Hazardous Substances In Effluents*, October 1, 2014] OPG facilities are required to have an Emission Monitoring Plan that documents the site's emission monitoring program. OPG Plan [P-PLAN-03480-00001-R000, *Pickering Nuclear Radioactive And Hazardous Emissions Monitoring Plan*, January 31, 2017] outlines the Pickering-specific monitoring requirements for the radiological airborne and liquid effluent pathways. This plan also outlines compliance of the monitoring program with CSA N288.5-11, *Effluent Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills*, and therefore demonstrates that a comprehensive program for monitoring effluent releases is in place.

The radiological impact of postulated accidents on the public is assessed in Part 3 of the Pickering NGS Safety Reports [NK30-SR-01320-00003-R004, *Pickering Nuclear 5-8 Safety Report: Part 3 Accident Analysis Vol 1-5*, October 10, 2014], [NA44-SR-01320-00002-R004, *Pickering Nuclear 1-4 Safety Report: Part 3 – Accident Analysis*, October 31, 2013]. Conservative estimates of whole body and thyroid doses to the critical individual at the site boundary and to the population are calculated using the site-specific terrain, meteorological conditions and population data and are compared against the regulatory dose limits.

The impact of BDBAs, including severe accidents, is also assessed in the PSA against the Large Release Frequency Safety Goal. This ensures that the risk of a large radiological impact is understood and is acceptable, and that emergency planning is effective. The results of the PSAs show that the Safety Goals are met for Pickering NGS [P-REP-03611-00006-R000, *Pickering NGS PSA Update to Include Enhancements from the Fukushima Integrated Action Plan,* April 30, 2014].

To summarize, the Pickering environmental monitoring program provides comprehensive monitoring of normal and inadvertent radioactivity releases from Pickering NGS, such that the radiological impact is understood. The radiological impact from postulated accidents within the design basis is assessed and the risk associated with BDBAs is understood and meets established Safety Goals. Therefore, all levels of defence-in-depth are effectively addressed with regard to radiological impact on the public and the local environment.

O-265 – Organization, Responsibilities and Staffing

Principle: The operating organization exerts full responsibility for the safe operation of a nuclear power plant through a strong organizational structure under the line authority of the plant manager. The plant manager ensures that all elements for safe plant operation are in place, including an adequate number of qualified and experienced personnel.

OPG has a well-defined organizational structure and strong lines of authority. Although all functions in the organizational structure support operating units, some portions, such as the Environment unit, are centre-led under OPG as a whole, rather than specifically under OPG Nuclear. OPG Standard [N-STD-AS-0020-R014, *Nuclear Management Systems Organizations*, May 19, 2016] establishes the lines of authority and definition of duties. It provides a summary of the interfacing organizations that own programs supporting the Nuclear Management System. Operation and maintenance of the station as per regulatory requirements and Nuclear standards for public safety are directed by the Director, Operations and Maintenance, who also coordinates with the centre-led organizations to effectively use resources to achieve performance targets. The quality and quantity of services provided by these centre-led support organizations are monitored by the Senior Site Vice President, who holds responsibility for establishing site requirements and priorities. Position-specific role documents in the N-MAN-08131 series describe the duties, authorities and accountabilities of the positions described in the standard. Additional guidance is given in OPG Programs [N-PROG-OP-0001-R008, *Nuclear Operations*, December 4, 2015], [N-PROG-MA-0004-R011, *Conduct Of Maintenance*, May 5, 2015] and [N-PROG-MP-0007-R012, *Conduct Of Engineering*, October 26, 2012]. The training and qualification description document [N-TQD-601-00001-R017, *Leadership And Management Training And Qualification Description*, May 4, 2015] provides qualification and professional development requirements for supervisors and managers, including Safety Culture for Managers.

OPG Charter [N-CHAR-AS-0002-R019, *Nuclear Management System*, November 1, 2016] gives authority to the OPG Nuclear safety processes and defines responsibilities. It specifies that the Chief Nuclear Officer is accountable for:

> “the effectiveness of the overall Nuclear Management System in ensuring our Nuclear facilities are operated and maintained using sound Nuclear safety and defence-in-depth practices to ensure radiological risks to workers, the public, and environment are as low as reasonably achievable, and in keeping with the Nuclear Safety Policy, and the best practices of the international Nuclear community.”

Managers ensure that tasks are executed as defined through OPG Program [N-PROG-AS-0002-R016, *Human Performance*, May 5, 17, 2016]. This program is specifically designed to achieve higher levels of nuclear and industrial safety, higher unit reliability and reduced operating costs through event-free operation. This performance is accomplished through pre-job briefings, post-job debriefings, self-checking programs, communications, self-assessments, and an observation and coaching program.

OPG Program [N-PROG-TR-0005-R016, *Training*, January 5, 2016] describes the training program for regular staff, contractors, temporary personnel and other staff assigned work. It includes the structure, processes and tools for defining, developing, implementing, documenting, assessing and improving the training required to ensure that nuclear staff have the appropriate knowledge, skill and attitudes for safe and efficient plant operation.

As described in Section 1.6.3 of [P-CORR-00531-03719-R000, *Application For Renewal Of Pickering Nuclear Generating Station Power Reactor Operating Licence*, July 4, 2012], OPG identifies qualified and competent individuals for key positions. With career development and succession planning being key elements in the management capability strategy, a Corporate Succession Plan ensures that individuals with high leadership potential are identified.

To conclude, the existing corporate structure supports well-defined lines of responsibility throughout the organization. In particular, the appropriate functions are in place and adequately staffed to support and enhance nuclear safety at all levels of defence-in-depth.

O-296 – Engineering and Technical Support of Operations

Principle: Engineering and technical support, competent in all disciplines important for safety, is available throughout the lifetime of the plant.

The roles and responsibilities of individuals responsible for safe operation are clearly defined and documented in published OPG Nuclear organization procedures [N-STD-AS-0020, *Nuclear Management Systems Organization,* February 2015]. This organization is understood and effective governance is in place to ensure availability of these resources and control organizational changes.

At OPG, positions are filled based on knowledge and skill requirements as identified in Training and Qualification Documents and in Qualification Guides. For example, Table 1 of OPG Instruction, N-INS-03490-10003 [*Minimum Shift Complement Resources, Qualifications and Procedures Required For Responding to Resource Limiting Events*, November 2013] lists the Training and Qualification Documents (TQDs) associated with Minimum Shift Complement staff qualification requirements. Individuals are either recruited with the necessary knowledge and skills documented in the Training and Qualification Documents and Qualification Guides, or are provided training prior to working independently. The Training and Qualification Documents and the Qualification Guides cover technical and engineering support functions, including but not limited to fire protection, radiation protection and maintenance. OPG also has consultants and contractors to support the operating organizations. The review of Safety Principle M&C-249, Achievement of Quality, describes OPG processes for ensuring quality in procured services.

The new hire engineering support training is intended to support the continued availability of trained and competent staff for technical positions within Engineering. The education background requirement for the new hire engineering support as engineering graduates or equivalent is provided in [N-TQD-403-00001, *Nuclear Engineering Support Personnel Training And Qualification Description*, September 21, 2016]. Once recruited, the new hires are required to go through a structured training program per OPG Procedure [N-PROC-TR-0008, *Systematic Approach To Training*, December 5, 2016]. The content is based on what a new graduate engineer initially needs to know to work at a nuclear power plant. In addition, there is training for the engineer once he/she situates in a home department. The new hire engineers are then put through training that includes initial training for core, extended core and duty area qualifications. This training is completed within three years from entering an engineering position.

In conclusion, adequate programs are in place and implemented to ensure that the appropriate engineering and technical capability is available to support nuclear safety at all levels of defence-in-depth.

D.3. Safety Principles Related to Levels 1, 2, 3, 4

The following safety principles are related to defence-in-depth Levels 1, 2, 3, 4:

- S-142 – Ultimate Heat Sink Provisions

- D-150 – Design Management
- D-154 – Proven Technology
- D-158 – General Basis for Design
- D-186 – Inspectability of Safety Equipment
- D-205 – Startup, Shutdown, and Low Power Operation
- D-227 – Monitoring of Plant Safety Status
- D-230 – Preservation of Control Capability
- M&C-246 – Safety Evaluation of Design
- M&C-249 – Achievement of Quality
- C-255 – Verification of Design and Construction
- C-258 – Validation of Operating and Functional Test Procedures
- C-260 – Collection of Baseline Data
- C-262 – Pre-Operational Adjustment of Plant
- O-269 – Safety Review Procedures
- O-290 – Emergency Operating Procedures
- O-292 – Radiation Protection Procedures
- O-299 – Feedback of Operating Experience
- O-305 – Maintenance, Testing and Inspection
- O-312 – Quality Assurance in Operation

As demonstrated below, Pickering NGS design and operation are aligned with all safety principles related to Levels 1, 2, 3, 4.

S-142 – Ultimate Heat Sink Provisions

Principle: The site selected for a nuclear power plant has a reliable long term heat sink that can remove energy generated in the plant after shutdown, both immediately after shutdown and over the longer term.

The ultimate heat sink for energy generated in the plant during normal operation and after shutdown is Lake Ontario. All water for equipment cooling and/or services in Pickering NGS Units 1,4 and 5-8 is drawn from Lake Ontario through an intake channel bounded by two groynes extending into the lake. Lake water is drawn through the station intake to the common screenhouse on the forebay. The water is strained through bar racks and traveling screens in the screenhouse and then flows through a common duct to a common pump suction header under the powerhouse. This water is used for cooling by the Condenser Circulating Water System and station Service Water Systems, and is returned to the lake through the cooling water discharge duct or through the Reactor Building discharge duct.

The Pickering NGS units are provided with the following Cooling/Service Water Systems:

- Condenser Circulating Water System
- Low Pressure Service Water System
- High Pressure Service Water System
- Recirculated Cooling Water System
- Vacuum Building Emergency Storage Water System

When the reactor is shut down for maintenance or to repair equipment, the Shutdown Cooling System removes residual heat. High Pressure Service Water is supplied to the shell side of the shutdown cooling heat exchangers. The High Pressure Service Water System has been upgraded as part of the return of Units 1,4 to service. The upgrades include installation of larger capacity Class III pumps and separation of the Class III High-Pressure Service Water System supply from the Class III Low Pressure Service Water System. Two 100 percent deep well pumps operating on Class III power supply high pressure service water in the event of a loss of Class IV power.

In Units 5-8, two of the high pressure service water pumps are powered by Class IV power, and two pumps are powered by Class III power. High pressure service water is supplied to the shutdown coolers only by Class IV power, and therefore, if Class IV power fails while a shutdown cooling circuit is operating, the Steam Generators become the primary heat sink. Steam is rejected to the atmosphere through the steam reject valves.

The Emergency Boiler Water Supply System is designed to provide emergency water to the boilers of Units 1,4. The emergency water is supplied from the discharge headers of the Units 6 and 7 High Pressure Service Water System. The system is credited as an emergency heat sink for Units 1,4.

In Units 5-8, the Emergency Water System supplies emergency makeup to the boilers for decay heat removal following an irrecoverable failure of the feedwater supply. The Emergency Water System is seismically qualified and is powered from the Emergency Power System.

The Emergency Water Storage Tank in the top of the Vacuum Building serves as a passive emergency supply of water to Pickering Units 5-8 in the event of failure of both Group I and Group II heat sinks. Under certain circumstances, such as loss of shutdown cooling capability, the steam generators become the primary heat sinks by rejecting steam to atmosphere through the steam reject valves.

Emergency heat removal capability, including the provision of Emergency Mitigating Equipment for accidents beyond the design basis, is described and reviewed in Safety Principle D-207, Emergency Heat Removal.

To summarize, multiple systems of adequate capacity, and their support systems, are in place to ensure that an appropriate heat sink for the Fuel exists in operating and shutdown/outage conditions, as well as in response to DBAs and BDBAs.

D-150 – Design Management

Principle: The assignment and subdivision of responsibility for safety are kept well defined throughout the design phase of a nuclear power plant project, and during any subsequent modifications.

OPG Program [N-PROG-MP-0009-R012, *Design Management*, May 12, 2017] provides the framework for the establishment, maintenance and compliance with the design basis for Pickering NGS. The design management program provides assurance that design and procedure changes are prepared, reviewed, approved, documented and implemented in accordance with approved procedures, applicable regulatory requirements, standards and industry practices.

N-PROG-MP-0009-R012 requires that design changes be initiated, implemented and tracked in accordance with OPG Program [N-PROG-MP-0001-R015, *Engineering Change Control*, May 12, 2017]. The engineering change control program defines a systematic process and methodology for controlling design modifications for plant SSCs. A primary input to the modification process is defined in OPG Procedure [N-PROC-MP-0083-R009, *Constructability, Operability, Maintainability, And Safety (COMS)*, April 21, 2016] and OPG Form [N-FORM-10480, *COMS Checklist*, April 22, 2016]. N-PROC-MP-0083-R009 provides direction on:

1. The identification of stakeholders from departments involved with or impacted by the modification.
2. Determination of which stakeholders comprise the Constructability, Operability, Maintainability and Safety team.

N-FORM-10480 is a repository of questions to assist in determining whether all appropriate issues have been identified during the design phase. The use of N-PROC-MP-0083-R009 and N-FORM-10480 ensures that stakeholder and subject matter expert input is considered and that risks impacting the safety of the plant and personnel are adequately identified and addressed.

In summary, the Pickering NGS design management program defines the appropriate processes to ensure that nuclear safety requirements are met in the planning and execution of design modifications.

D-154 – Proven Technology

Principle: Technologies incorporated into design have been proven by experience and testing. Significant new design features or new reactor types are introduced only after thorough research and prototype testing at the component, system or plant level, as appropriate.

Pickering was the first large-scale CANDU power reactor design and was the culmination of a series of CANDU demonstration plants at increasing scale (specifically, Nuclear Power Demonstration and Douglas Point). In addition, the CANDU design has benefitted from decades of experience and knowledge acquired through the design and operation of heavy water research reactors at Chalk River. Experience gained at each stage of the design development process was used to guide and improve the design at the next stage. The excellence and inherent safety features of the CANDU design have been proven through more than 500 reactor-years of safe and reliable operation worldwide since Pickering NGS Units 1-4 were first put into service.

The process for adopting and implementing design changes, including those which can be characterized as significant new design features is governed by OPG Program [N-PROG-MP-0009-R012, *Design Management*, May 12, 2017]. This program requires that design changes are initiated, implemented and tracked in accordance with OPG Program [N-PROG-MP-0001-R015, *Engineering Change Control*, May 12, 2017]. The engineering change control program defines a systematic process and methodology for controlling design modifications for plant SSCs to meet the requirements of CSA N285.0, *General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants* and CSA N286, *Management System Requirements for Nuclear Power Plants*.

OPG actively participates in the research and development activities in CANDU and related technologies to improve plant operation, equipment performance and reliability and analytical capabilities and scientific codes used in engineering and safety analysis.

To conclude, the use of proven technology has always been a driving principle at Pickering NGS, and appropriate processes are in place to address the provenness of design modifications.

D-158 – General Basis for Design

Principle: A nuclear power plant is designed to cope with a set of events including normal conditions, anticipated operational occurrences, extreme external events and accident conditions. For this purpose, conservative rules and criteria incorporating safety margins are used to establish design requirements. Comprehensive analyses are carried out to evaluate the safety performance or capability of the various components and systems in the plant.

The Pickering NGS design was developed using the principles of defence-in-depth. Inherent to this approach is the requirement to postulate a range of process equipment failures which would impair one or more of the barriers and establish that resultant releases of radioactive material will not result in radiation doses above allowable limits. To meet this requirement, a number of safety-related functions (as distinct from process functions associated with routine power production) are provided. The levels of events considered and the alignment of the Pickering NGS design against them can be summarized as:

1. **Level 1 – Prevent deviations from normal operation and prevent failures of SSCs**

   The first level of defence requires a high quality in the design and construction of the plant with barriers to prevent the occurrence of abnormal operating conditions. This is particularly important for the physical barriers surrounding the radioactive material in the Fuel. Safe, conservative operation of the plant by qualified staff and a continued focus on preventive maintenance ensures reliable functionality of plant equipment under normal operation and therefore prevents process upsets and failures.

2. **Level 2 – Detect and intercept deviations from normal operation in order to prevent process upsets from escalating to accident conditions**

   The second level of defence is the provision of barriers to prevent process upsets from progressing to accidents. The Pickering NGS plant design possesses a number of strong features regarding defence-in-depth Level 2. For example:

   - Automatic reactor control features detect and respond to abnormal conditions before these conditions progress to the point that the next level of barriers are required to act.
   - A large number of Safety-Related System tests are completed routinely, based on prescribed schedules to detect problems regarding plant equipment.
   - A well-established framework of operating procedures is in place to respond to equipment malfunctions in a timely manner thereby ensuring that the plant stays within its well-defined SOE.

3. **Level 3 – Minimize the consequences of accidents**

   The third level of defence consists of the barriers to minimize the consequences of accidents should they occur by providing inherent safety features, fail-safe design, additional equipment (including Emergency Mitigating Equipment) and mitigating procedures. The Pickering NGS Units 1,4 and 5-8 Safety Reports demonstrate that the radiological consequences of postulated accidents within the design basis meet established dose limits. The PSA [P-REP-03611-00006-R00, *Pickering NGS PSA Update to Include Enhancements from the Fukushima Integrated Action Plan*, April 30, 2014] demonstrates that the overall plant design has a Core Damage Frequency and Large Release Frequency within the specified Safety Goals, indicating robustness in the design and reliable equipment that is capable of responding effectively to accident scenarios.

4. **Level 4 – Ensure that radioactive releases caused by severe accidents are kept as low as practicable**

   The fourth level of defence includes those barriers to control severe plant conditions. Significant improvement in the Severe Accident Management Guidelines implementation has resulted in Pickering NGS strengthening its capability to respond to low probability severe accidents. Implementation of lessons learned from the 2011 Fukushima accident, and enhancements to power supplies for hydrogen mitigation equipment for BDBAs, will add further capability to this defence-in-depth level (see GI-40-RS1 in Appendix F).

The scope of postulated initiating events addressed by the existing deterministic safety analysis for Pickering NGS Units 1,4 and 5-8 (Part 3 of the Safety Reports), consists of single/dual failure events based on the requirements of the Siting Guide. As outlined in [N-CORR-00531-18239, *Progress Report on OPG Safety Analysis Improvement and REGDOC-2.4.1 Implementation*, October 17, 2016], OPG has developed an implementation plan, which defines the CNSC REGDOC-2.4.1 compliant analyses to be undertaken in the 2014-2017 timeframe.

For protection against seismic events, Units 1,4 were constructed according to the National Building Code of Canada seismic provisions, whereas Units 5-8 included specific provisions to withstand a prescribed Design Basis Earthquake. The Units 1,4 SSCs required to perform functions during and following an earthquake were not originally required to be seismically qualified. However, the common Containment structures (Reactor Building, Pressure Relief Duct and Vacuum Building) were designed to exceed the National Building Code 1965 seismic design provisions and were subsequently confirmed analytically to meet Units 5-8 Design Basis Earthquake seismic design requirements. The OPG Report [NA44-REP-02004-0073-R000 Vol 1-7, *Seismic Assessment of Pickering A Nuclear Generating Station Summary Repor*t, February 25, 1998] evaluated the seismic capacity of those Pickering A SSCs required to perform the above functions and identified necessary seismic upgrades, which have been implemented. Seismic success path SSCs were identified and updated in OPG Report [NA44-REP-02004-00002-R001, *Pickering NGS A Seismic Success Path Addendum Including the Safe Shutdown Equipment List*, August 13, 2013]. The Electric Power Research Institute/Seismic Qualification Utility Group seismic margin assessment methodology was used to evaluate seismic capacity of the seismic success path, as well as to verify the seismic adequacy of supporting structures and services.

Subsequently, a screened list of internal and external events for inclusion in the Level 1/Level 2 PSA for the Pickering NGS Units 1,4 and 5-8 was completed as part of the hazard screening analysis completed in 2012 and includes discussion of pipe whip, missiles, explosions, toxic gas (external), flooding (external), extreme temperatures, and aircraft impacts [P-REP-03611-00006-R000, *Pickering NGS PSA Update to Include Enhancements from the Fukushima Integrated Action Plan,* April 30, 2014].

To summarize, Pickering NGS includes SSCs designed to cope with a set of events, including normal conditions, external events and accident conditions with appropriate safety margins, based on assessments of the events that challenge the barriers to radioactivity release.

D-186 – Inspectability of Safety Equipment

Principle: Safety related components, systems and structures are designed and constructed so that they can be inspected throughout their operating lifetimes to verify their continued acceptability for service with an adequate safety margin.

Periodic inspection is the mandatory non-destructive examination of nuclear equipment. It provides assurance that the likelihood of failure remains acceptably low throughout the operating lifetime. The Periodic Inspection Program (PIP) at Pickering NGS is prepared on the basis of a sampling system which selects components subject to the most severe operating conditions from the nuclear process systems. The program delineates the selection of inspection areas, inspection frequency, procedures, techniques and acceptance criteria, the collection and recording of data and the reporting of results. The complete program and any revisions thereto must be approved by the regulatory authority. To meet these requirements the design and arrangement of components provides for access for inspection and maintenance.

Regarding periodic inspections, the Licence Conditions Handbook states that "OPG shall carry out periodic inspections in accordance with the accepted PIP documents. If a deviation from the accepted PIP program is anticipated during inspection planning activities, OPG shall obtain CNSC acceptance prior to conducting the affected inspections. However, for any findings, discoveries or deviations from the accepted PIP that are identified during an inspection, OPG shall provide justification to CNSC in the inspection report submission following OPG governance, OPEX and best industry practices. For permanently required exemptions to the requirements of CSA PIP standards, OPG shall revise the affected PIP document accordingly prior to issuing the next scheduled revision of the PIP document." The Licence Conditions Handbook wording provides a way to resolve a deviation from the accepted PIP. If a permanent concession is needed as a result of future updates to CSA Standards associated with PIPs, this concession will need to be updated. CNSC acceptance will be obtained at that time if it is needed.

Inspection of key SSCs is facilitated in the design as follows:

- Pressure boundary piping is monitored using non-destructive inspection techniques to assure that the likelihood of a pipe failure is kept low. The scope of periodic inspection includes components, piping and supports. Radiation fields are held to levels that permit personnel access for maintenance and inspection of the Heat Transport System, Shutdown Cooling, ECI and Heat Transport Relief System components.
- Pressure tube leaks can be readily detected by monitoring the moisture content and pressure in the gas space between the pressure tube and Calandria tube. This is done on a continuous basis. In addition, ultrasonic scanning devices can be mounted on the fuelling machines for periodic in-service inspection of the pressure tubes. Fuel Channel inspections and Fuel Channel fitness for service are governed by CSA N285.4, *Periodic Inspection of CANDU Nuclear Power Plant Components*, and compliance to this standard is required by the Power Reactor Operating Licence. As per the Pickering Fuel Channels Fitness for Service submission [P-CORR-00531-04953, *Pickering NGS-Assurance of Fuel Channel Fitness-for-Service for the Assumed Target Service Life of*

the Pickering Units, April 4, 2017], the Life Cycle Management Plan for Fuel Channels [N-PLAN-01060-10002-R017, *Fuel Channel Life Cycle Management Plan*, October 2016] includes inspection scope that exceeds the minimum requirements of CSA N285.4 to demonstrate fitness for service.

- The Feeder Life Cycle Management Plan [N-PLAN-01060-10001-R018, *Feeders Life Cycle Management Plan*, October 2016] specifies the required PIP, in-service inspections and maintenance for Feeders. The Feeders Life Cycle Management Plan is revised on a regular basis to capture changes that may be required in response to issues identified by inspection, industry experience and ongoing research activities. As per the Fitness for Service Memorandum [P-CORR-01060-0632223, *Fitness for Service of Major Components,* February 2017], the latest inspection results demonstrate that the most recent measured wall thickness remained greater than the minimum allowable wall thickness and will be fit for service for the next operating cycle. Each Pickering unit conducts inspections periodically. The thickness inspections for future scopes are focused on monitoring lead Feeders and dispositioned Feeders. Lead Feeders are those that are approaching their minimum design-required thickness. Dispositioned Feeders are those with specific analysis defining their service limits which were accepted by the CNSC. This population has been determined by previous campaigns where 100% of the Feeder inspections were completed. The inspection scope increases as the dispositioned Feeder population increases. As per the Life Cycle Management Plan, these Feeders will continue to be monitored.
- According to the Fitness for Service Memorandum [P-CORR-01060-0632223, *Fitness for Service of Major Components,* February 2017], each Pickering NGS unit has a Steam Generator inspection approximately every two years during planned unit outages. The Steam Generator Life Cycle Management Plan identifies the inspection scope to identify active and plausible tube degradation mechanisms and the extent of the condition in the Steam Generators.

In addition to the Major Components inspection program, OPG conducts regular inspection, testing and maintenance throughout the plant, as per OPG program [N-PROG-MA-0004-R011, *Conduct of Maintenance*, May 2015].

As part of the engineering change control process [N-PROG-MP-0001-R015, *Engineering Change Control*, May 12, 2017], all station modifications are evaluated to ensure maintainability requirements are considered in the design.

In conclusion, OPG has a robust and multi-faceted SSC inspection program, such that the current condition of the plant is well understood. This provides a solid baseline for the intended operation extension as well as confidence that the requisite data will continue to be acquired going forward.

D-205 – Startup, Shutdown, and Low Power Operation

Principle: Components, structures, and systems used during startup, low power and shutdown operations are designed to maintain or restore the reactivity control, decay heat removal, and the integrity of the fission product barriers, so as to prevent the release of radioactive material resulting from accidents initiated during those operations.

Normally, the Reactor Regulating System is used to control reactor power including during reactor startup and shutdown. Reactor shutdown by the Reactor Regulating System can be initiated manually through a keyboard or by a special setback pushbutton on the control panel. The Reactor Regulating System uses light water liquid zone controllers, adjuster rods, control absorbers (Pickering Units 5-8), and moderator boron/gadolinium addition for reactivity control.

In addition to the Reactor Regulating System, the Shutdown Systems can also be used to shut down the reactors with either shutoff rods or the Liquid Injection Shutdown System (Pickering 5-8), or a moderator discharge (dump) capability (Pickering 1,4).

For a Guaranteed Shutdown State (during outages), several means are available for keeping the reactors shut down:

- **Overpoisoning:** One method used for keeping the reactor in a guaranteed shutdown state is by adding neutron absorbing chemicals (poison) to the Moderator.
- **Moderator Drain:** Draining the Moderator into separate storage tanks is another way of putting the reactor in a guaranteed shutdown state.
- **Rod Based:** In the Rod Based guaranteed shutdown state, the shutoff rods and control absorbers are placed in the in-core position, and controls are in place to prevent inadvertent removal. Poison is added to the Moderator and adjuster rods are placed in core for extra subcriticality margin as a conservative measure.

The instrumentation for initial startup comprises two complementary Neutron Detecting Systems, one using neutron proportional counters, the other out-of-core counters. Only the out-of-core counters are required for low-level startup after an extended shutdown period. Neutron poison (boron) solution can be added to the Moderator to suppress excess reactivity and can be removed by ion exchange. It is used to suppress relatively long-term excess reactivity due to fresh Fuel and is used for reactivity control during unit startups.

In Units 1,4, on SDSA each trip channel can be connected to a channel of start-up instrumentation which measures very low levels of neutron flux. Start-up instrumentation is credited as a replacement for the neutron overpower protection and Log N rate trip parameters at very low reactor power levels. Each start-up instrumentation channel consists of a neutron detector, amplifiers and ratemeter that activates an alarm module on high neutron count rate and low neutron count rate. The instrumentation satisfies the requirement for measurement of neutron flux level when in the guaranteed shutdown state and on the approach to critical.

In Units 5-8, the initial start-up instrumentation comprised two complementary Neutron-Detecting Systems. One system had neutron proportional counters located in a horizontal in-core flux detector tube, while the other system used counters located out-of-core in a spare compartment in the SDS2 ion chamber housings. Only the out-of-core counters are required for any subsequent low-level start-up after an extended shutdown period. Protection against reactivity transients is provided by the neutronic trip parameters for SDS1 and SDS2, as described in Safety Principle D-200, Automatic Shutdown Systems, and D-192, Protection Against Power Transient Accidents.

Analysis demonstrating the effectiveness of the Shutdown and Heat Removal Systems in mitigating accidents initiated during operation at all power levels, including low power levels associated with reactor startup and shutdown, is documented in Part 3 of the Pickering NGS Safety Reports. The analyzed accidents include a range of assumed control failures, electrical failures, SBLOCA (including in-core breaks) and large break LOCA, as well as failures in the Feedwater and Steam Supply System and Moderator System. The trip coverage analyses performed for these events demonstrate that there is at least one trip parameter in each of SDSA and SDSE for Pickering 1,4 (SDS1 and SDS2 for Pickering 5-8) that will shut down the reactor before Fuel sheath failure can occur.

The main condensate and feedwater pumping train has three 50 per cent system capacity condensate extraction pumps and three 50 per cent capacity feed pumps. Reactor startup and cooldown requirements are provided by an auxiliary condensate pump sized to supply 5 per cent of full load condensate flow and an auxiliary feed pump sized to supply 3 per cent of full load feedwater flow. When the reactor is shut down for maintenance or to repair equipment, Shutdown Cooling Systems remove residual heat (or decay heat).

For low power operation and reactor shutdown, the Shutdown Cooling System is provided to cool down the Heat Transport System from 177°C and maintain cooling for an indefinite period of time. The Shutdown Cooling System provides cooling for the Heat Transport System during outage operation and is designed to provide core cooling with the Heat Transport System depressurized to permit maintenance. Under emergency conditions, the shutdown coolers may be used to cool down the Heat Transport System from the operating temperature.

To summarize, Pickering NGS has the SSCs in place to control the reactor power and provide fuel cooling in response to events occurring during startup, shutdown and low power operation.

D-227 – Monitoring of Plant Safety Status

Principle: Parameters to be monitored in the control room are selected, and their displays are arranged, to ensure that operators have clear and unambiguous indications of the status of plant conditions important for safety, especially for the purpose of identifying and diagnosing the automatic actuation and operation of a safety system or the degradation of defence in depth.

The Main Control Rooms for Pickering NGS Units 1,4 and Units 5-8 contain the main control panels for each of the generating units. Each unit has its own control panels, and the panels for all units have the same layout. Two additional panels are provided for common equipment and electrical controls. All indications and controls essential for operation are located on the Main

Control Room panels. To provide post-accident monitoring capability, the critical safety parameters for Units 1,4 include:

- Reactor power level
- Reactor inlet header pressure
- Inlet Feeder temperature
- Outlet Feeder temperature
- Reactor Building water level
- Reactor Building/reactor auxiliary bay pressure differential
- Boiler level
- Boiler pressure

For Units 5-8 there is similar monitoring of critical post-accident parameters.

Colour monitors display plant data and alarms which would otherwise have to be displayed on panel indicators. Sufficient conventional display, annunciation and recording of plant variables is included to allow the plant to be maintained safely in the shutdown condition with all computers out of service. The Main Control Room is spacious to provide clear access routes and free movement within operating areas.

Information about the state of the unit is presented to the operator by a number of computer systems. This information includes sequence of events functions, display of process variables and initiation of most alarms. In both Units 1,4 and Units 5-8, the display of process variables provided by the computers complements the existing conventional instrumentation. For the digital control computers, dual channel computers provide redundancy and backup in case of failure. If both computers fail, the operator will lose some of the normal sources of information. However, information important to the safety of the unit, such as the status of all the safety systems and information about the status of the unit, is available even when computers are not available so operators will be able to establish the existence, nature and extent of any safety-related failure and take the appropriate action.

In Units 1,4, additional safety-related process information associated with critical safety parameter monitoring, SDSE and Negative Pressure Containment is provided via the Data Extraction System computer.

The following information is displayed directly on the Main Control Room panels:

- Alarm windows to indicate the tripped state of any parameter in any channel of either Shutdown System, the status of the ECI System and Containment.
- The value of each trip parameter in each channel of either Shutdown System and values of ECI System and Containment parameters.
- Other alarm windows to indicate abnormalities in the Special Safety Systems, e.g., low level in emergency storage tank.
- Alarm windows to indicate the existence of single and dual control computer failures.

- Process indicators to display information on the status of subsystems required for the operation of Special Safety Systems and Safety-Related Systems, e.g., dousing tank and Vacuum Building floor water levels and Reactor Building pressures.

Capability to monitor post-accident conditions remotely from the Main Control Room is provided in the SDSE Instrument Rooms for Units 1,4 and in the Unit Emergency Control Centres for Units 5-8. The Units 1,4 SDSE Instrument Rooms provide monitoring of the critical safety parameters listed above. The instrument rooms house SDSE trip parameters signal processing instrumentation, trip logic and monitoring computers. Buffered/isolated signals are transmitted from the instrument room to the Main Control Room to provide indications of the SDSE parameters. These instrument rooms provide separation from SDSA instrumentation and trip logic.

In Units 5-8, the Unit Emergency Control Centres contain the unit controls and logic panels associated with the following systems:

- SDS2
- Emergency Water Supply
- Emergency Power Supply
- Containment
- Plant monitoring systems
- ECI Recovery System (controls in Units 5 and 7 Unit Emergency Control Centre only)
- Filtered Air Discharge System (controls in Unit 5 Unit Emergency Control Centre only)

When the main control centre is not available for any reason, the reactor can be safely shut down and maintained in that state indefinitely from the Unit Emergency Control Centre.

In conclusion, Pickering NGS has the appropriate indications and alarms in the Main Control Room (and in secondary areas should the Main Control Room become uninhabitable) to inform operations staff of mitigating system action and degradation of key plant parameters such that initiating events can be controlled.

D-230 – Preservation of Control Capability

Principle: The Main Control Room is designed to remain habitable under normal operating conditions, anticipated abnormal occurrences and accidents considered in the design. Independent monitoring and the essential capability for control needed to maintain ultimate cooling, shutdown and confinement are provided remote from the Main Control Room for circumstances in which the Main Control Room may be uninhabitable or damaged.

Provisions for Main Control Room habitability during normal operation and following an accident are in place for both Units 1,4 and Units 5-8.

Precise control of space temperature and humidity is required at all times to provide the necessary environment for computing and other solid state equipment as well as for the comfort of occupants. During normal operation, cooling of the Main Control Rooms and Control Equipment Rooms is maintained at all times to offset heat generation from lighting, equipment and staff.

As part of the Environmental Qualification retrofit to improve mitigation of powerhouse harsh environment events, modifications were made that included a Powerhouse Emergency Venting System that promotes a flue effect to sweep steam out of the building and protect the Main Control Room.

An enclosure for the air-conditioning equipment located on the 294 ft. elevation maintains Main Control Room conditions in the first 10 minutes following a postulated steam line failure in the Units 5-8 powerhouse or reactor auxiliary bay. The enclosure provides assurance of Control Room habitability in the short-term and survival of the Main Control Room operators so that they can act before evacuation to the Unit Emergency Control Centre.

The air-conditioning equipment in Units 5-8 does not need protection for continuous operation; however, equipment that may provide a leak path into the Main Control Room or Control Equipment Rooms is protected by the enclosure to withstand the environmental conditions caused by the steam line failure. The enclosure limits steam ingress to the extent that the Main Control Room and Control Equipment Room temperature does not exceed 50°C at 100% relative humidity in the first 10 minutes following a steam line release, and withstands a sustained pressure up to 0.1 kPa(d).

Design and procedural changes were performed for Pickering NGS Units 1,4 Restart to provide the following safety functions remote from the Main Control Room, in the event that the Units 1,4 Main Control Room becomes uninhabitable.

- Shut down the reactor and maintain it shut down.
- Remove decay heat.
- Maintain Containment integrity.
- Provide post-accident monitoring of critical safety parameters.

To provide monitoring capability independent of the Main Control Room, the critical safety parameters have been duplicated in the SDSE Instrument Rooms as discussed in Safety Principle D-227, Monitoring of Plant Safety Status. By monitoring critical safety parameters and following critical safety parameter restoration procedures via field actions, it is possible to maintain the Control, Cool, Contain and Monitor safety functions externally from the Main Control Room.

The Unit Emergency Control Centre is an additional control room for each generating unit of Units 5-8. As is discussed in Safety Principle D-227, Monitoring of Plant Safety Status, the Unit Emergency Control Centres provide monitoring and control of key systems external to the Main Control Room. When the Main Control Room is not available for any reason, the reactor can be safely shut down and maintained in that state indefinitely from the Unit Emergency Control Centre. Each Unit Emergency Control Centre is required to be accessible, operational and habitable following any event requiring operator action in the Unit Emergency Control Centre. On that basis, the following are the Unit Emergency Control Centre room criteria:

- Self-contained operation
- Immunity from events that may disable the Main Control Room
- Independence from the equipment in the Main Control Room
- Independence from normal plant service systems

A seismically qualified emergency lighting system is provided in the four Unit Emergency Control Centres, the Emergency Water and Power System Building, the four Emergency Water System booster pumphouses, the Filtered Air Discharge Building, the areas in the Reactor Auxiliary Bay beside the ECI System recovery pumps, and the escape route from the Main Control Room to outside the powerhouse below the Pressure Relief Duct.

In conclusion, the Main Control Rooms in Pickering NGS are designed to remain habitable during normal operation and postulated DBAs. Should a Main Control Room become uninhabitable, the SDSE Instrument Rooms in Units 1,4 and the Unit Emergency Control Centres in Units 5-8 are designed to inform operations staff of key plant parameters and allow for critical safety system action to be initiated.

M&C-246 – Safety Evaluation of Design

Principle: Construction of a nuclear power plant is begun only after the operating organization and the regulatory organization have satisfied themselves by appropriate assessments that the main safety issues have been satisfactorily resolved and that the remainder are amenable to solution before operations are scheduled to begin.

Prior to the construction of Pickering NGS, relevant knowledge and experience with respect to licensing and operating the CANDU design had been acquired through the Nuclear Power Demonstration and Douglas Point projects. The construction of both Units 1-4 and Units 5-8 was initiated and completed in accordance with the applicable regulatory requirements. As part of the pre-construction and construction licensing process, assessments were submitted to the regulator demonstrating that safety issues were resolved or amenable to resolution.

Design and operating improvements that have been implemented in Units 1,4 and Units 5-8 in response to OPEX and other emerging issues are documented in the respective Safety Reports and Probabilistic Safety Assessments, together with the safety features of the original plants. Experience subsequent to the initial operation of Units 1-4 and Units 5-8 has shown that, whenever a safety issue has been identified, it has been dealt with effectively. This has occurred due to the infrastructure and processes in place, allowing for sufficient flexibility to address emerging issues. An example is the provision of Emergency Mitigating Equipment in Units 1,4 and Units 5-8 for mitigation of BDBAs.

In conclusion, processes are in place to prevent or address any safety issues that may arise. Note that this safety principle, with its reference to “main safety issues” identified during the plant construction period, is intended for the stage prior to operation of the plant. In the spirit of the PSR2 assessment for the extended operating period, the emphasis for the current review is on safety evaluation relevant to continuing operation.

M&C-249 – Achievement of Quality

Principle: The plant manufacturers and constructors discharge their responsibilities for the provision of equipment and construction of high quality by using well proven and established techniques and procedures supported by quality assurance practices.

Design and quality assurance processes were put in place for design analysis, stress analysis, material control and traceability, fabrication, in-process inspection, installation and welding, control of weld quality, non-destructive examination and inspection. Meeting nuclear industry standards requires that suppliers have a level of quality assurance commensurate with the nature and application of the goods being supplied. These standards formed the basis of the series of Canadian National Standards initially issued in the 1970's under the Nuclear Standards Steering Committee. They were found useful in other industries and were taken over by the Quality Standards Steering Committee. Later, the international community decided that these standards and similar ones in other countries should be harmonized. Using these as the basis, the ISO produced ISO 9000, now used extensively in a wide range of industries around the world. In September 2016, the CSA N299 series of standards was issued to update the quality assurance program requirements for the supply of items and services for nuclear power plants. The CSA N299 standards are outside the scope of the PSR2 since they postdate the PSR2 documentation freeze date of January 15, 2016.

The design management program specifies requirements for procurement engineering processes ensuring implementation and maintenance of the physical nuclear facilities meet the design basis requirements. The design management program complies with both CSA N286, *Management System Requirements for Nuclear Power Plants*, as well as CSA N285.0, *General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants* and OPG Manual [N-MAN-01913.11-10000-R019, *Pressure Boundary Program Manual*, April 5, 2017].

Sections of Pickering NGS governing documents are cross referenced to clauses of CSA N286-12, *Management System Requirements for Nuclear Facilities* in OPG List [N-LIST-08130-10025-R000, *CSA N286-12 to OPGN Governance Cross-Matrix*, September 18, 2015], respectively. The sections of the governance such as OPG Program [OPG-PROG-0009-R002, *Items and Services Management*, May 15, 2015] also contain cross references to the applicable clauses of CSA N286-12. There are many programs and procedures that control purchasing of equipment and services where this affects plant safety.

The processes identified in OPG-PROG-0009-R002, *Items and Services Management*, ensure that items, services and nuclear Fuel are purchased in accordance with stated requirements and controlled through proper identification, handling, storage, issuance and shipping to ensure that the quality of equipment and components is preserved.

The controls for establishing and maintaining the OPG Approved Supplier List are documented in OPG Procedure [N-PROC-MM-0010-R021, *Establishing And Maintaining Ontario Power Generation Approved Suppliers List*, February 1, 2017]. N-PROC-MM-0010-R021 describes the methods used to originate, request, evaluate, qualify and maintain the qualification of suppliers of items and services.

To summarize, the design and construction of Pickering NGS was undertaken under an industry-leading quality assurance regime. Currently, design modifications are performed in compliance with rigorous quality assurance requirements aligned with current regulatory requirements and industry best practices.

C-255 – Verification of Design and Construction

Principle: The commissioning programme is established and followed to demonstrate that the entire plant, especially items important to safety and radiation protection, has been constructed and functions according to the design intent, and to ensure that weaknesses are detected and corrected.

The Pickering NGS commissioning program was conducted to demonstrate that each unit and the overall station functions as designed. The effectiveness of the commissioning program is borne out by the record over several decades of safe operation of both Units 1,4 and Units 5-8.

Currently and in the period of extended operation, verification of effectiveness is, or will be, required for any changes in plant SSCs. OPG Program [N-PROG-MP-0009-R012, *Design Management*, May 12, 2017] requires that design changes are initiated, implemented and tracked in accordance with OPG Program [N-PROG-MP-0001-R015, *Engineering Change Control*, May 12, 2017]. The primary objective of the engineering change control program is to ensure that all modifications to plant SSCs, including software and station engineered tooling, are planned, designed, installed, commissioned, decommissioned, placed into service or removed from service within the SOE, design basis and plant licensing conditions.

It is concluded that design verification of design modifications is ensured by engineering change control.

C-258 – Validation of Operating and Functional Test Procedures

Principle: Procedures for normal plant and systems operation and for functional tests to be performed during the operating phase are validated as part of the commissioning programme.

Validation of the planned operating procedures and for testing to be performed during operation was carried out during commissioning. The effectiveness of the commissioning program is borne out by the record over several decades of safe operation of both Units 1,4 and Units 5-8.

All new procedures and major revisions to existing procedures are validated either in the simulator or in the field per OPG Procedure [N-PROC-AS-0028-R018, *Development Review*

And Approval Of Technical Procedures, February 9, 2017]. The completion of applicable validation activities is consistent with industry best practice. The validation process ensures that documents are correct, meet the intended function and are usable by a qualified individual. The method selected for technical procedure validation depends on various considerations such as the complexity of the document, availability of a suitable validation site and the number of disciplines involved in the performance of the tasks. Validation methods include the following:

- **Field Validation:** A validation method that requires tasks specified in the technical procedure be performed on actual plant equipment.
- **Simulated Performance Validation:** A validation method that requires tasks specified in the technical procedure be performed on simulators, models, mock-ups or on shop equipment that is not considered to be plant equipment.
- **Table-Top Discussion and Walk Through:** A validation method that requires instructions in the technical procedure be talked through step-by-step followed by the steps being walked through in the normal work environment.

Irrespective of the validation method selected for a given procedure, the validation process requires involvement from staff who will be using the procedure, once it is approved. The procedure author addresses feedback from the validation process and obtains concurrence from individual validators on the dispositions to their comments.

In summary, there are robust processes in place to validate any proposed change in operating procedures at Pickering NGS.

C-260 – Collection of Baseline Data

Principle: During commissioning tests, detailed diagnostic data are collected on components having special safety significance and the initial operating parameters of the systems are recorded.

During the initial commissioning of Pickering NGS, data were collected for diagnosing any identified issues and for characterizing the operating conditions of SSCs. Data collection and analysis have been ongoing since operation began. OPG Programs [N-PROG-MA-0017 R009, *Component and Equipment Surveillance*, May 31, 2017] and [N-PROG-MA-0026 R002, *Equipment Reliability*, June 4, 2015] are in place with requirements to continually monitor and trend performance as required.

C-262 – Pre-Operational Adjustment of Plant

Principle: During the commissioning programme, the as-built operating characteristics of safety and process systems are determined and documented. Operating points are adjusted to conform to design values and to safety analyses. Training procedures and limiting conditions for operation are modified to reflect accurately the operating characteristics of the systems as built.

The initial determination of the plant operating characteristics obtained during the commissioning phase allowed for the subsequent safe operation of Units 1,4 and Units 5-8. Operation over several decades has also provided a wealth of data and information that has been used to further optimize how the plant is operated. This is done in accordance with OPG Program [N-PROG-OP-0001-R008, *Nuclear Operations*, December 4, 2015]. The configuration management program is an integrated management process that ensures that:

- Physical and functional characteristics, operation and maintenance conform to the design and licensing basis.
- Operating, training, modification and maintenance processes are consistent with the design and licensing basis conditions.

The limiting conditions for operation are determined and reflected in the SOE, which is reviewed in Safety Principle O-284, Operational Limits and Conditions. As discussed under Safety Principle C-255, Verification of Design and Construction, the commissioning process under [N-PROG-MP-0001-R015, *Engineering Change Control*, May 12, 2017] for new modifications ensures the incorporation of such data going forward.

To summarize, processes are in place to ensure that adjustments of the plant operating conditions are identified, analyzed and implemented if deemed appropriate (e.g., in response to aging of SSCs) to ensure adequate safety margins are maintained.

O-269 – Safety Review Procedures

Principle: Safety review procedures are maintained by the operating organization to provide a continuing surveillance and audit of plant operational safety and to support the plant manager in the overall safety responsibilities.

OPG Program [N-PROG-OP-0001-R008, *Nuclear Operations*, December 4, 2015] establishes safe, uniform and efficient operating practices and processes within OPG Nuclear facilities that provide the operating staff the ability to ensure facilities are operated in such a manner that Power Reactor Operating Licences, Operating Policies and Principles and applicable regulations and standards are followed. The nuclear operations program captures a series of standards and procedures to ensure safety of the public, environment, plant personnel and plant equipment.

OPG Policy [N-POL-0001-R003, *Nuclear Safety Policy*, April 7, 2014] requires that nuclear safety undergoes constant examination. The OPG Charter [N-CHAR-AS-0002-R019, *Nuclear Management System*, November 1, 2016] identifies expectations for the organization to develop priorities based on performance indicators and known challenges.

As described in OPG Procedure [N-PROC-AS-0078-R004, *Nuclear Performance Monitoring And Reporting*, May 22, 2014], the Nuclear Performance Index metric is reported quarterly and is a weighted composite of ten WANO Performance Indicators related to safety and production performance reliability. This metric is one of the many measures used to trend performance and monitor the effectiveness of various improvement programs and allows OPG Nuclear to benchmark against other nuclear plants worldwide.

The Reactivity Management program is routinely assessed and a performance index is used to track performance [N-STD-OP-0009-R011, *Reactivity Management*, January 31, 2017].

Site Event Free Day Resets is reported monthly and reflects the effectiveness of management in reducing human performance events and improving organizational processes. The criteria for application and administration of this metric are conducted in accordance with OPG Instruction [N-INS-09030-10002-R010, *Site And Department Level Event Free Day Resets*, May 1, 2017].

Deficiencies, non-conformances and opportunities to improve a process, document, service or condition are treated as learning opportunities and captured in OPG Program [N-PROG-RA-0003-R010, *Corrective Action*, January 14, 2015]. Individuals are encouraged to identify opportunities to improve and notify appropriate levels of management in order to identify actions to prevent recurrence or improve the process.

External independent reviews provided by a Nuclear Safety Review Board and the Nuclear Oversight Committee of the Board of Directors, as described in OPG Charter [N-CHAR-AS-0002-R019, *Nuclear Management System*, November 1, 2016], provide a review of occurrences and trends that may affect nuclear safety. The Nuclear Safety Review Board, comprised of senior external nuclear experienced individuals, provides the Chief Nuclear Officer with an annual independent assessment of OPG Nuclear activities at each station that may impact nuclear safety and performance. The scope and terms of reference for the operation of the Nuclear Safety Review Board are described in OPG Standard [N-STD-RA-0035-R004, *Nuclear Safety Review Board*, May 25, 2015].

OPG also invites WANO to conduct peer reviews at each OPG Nuclear site every 2 years. An external team of experts reviews all key functional and organizational areas against the WANO Performance Objectives and Criteria. These assessments identify areas for improvement at the host plant and note strengths that could be useful to share with other plants.

In September and October 2016, Pickering NGS hosted an IAEA Operational Safety Review Team (OSART) mission. The OSART review is another example of a safety review conducted by an external peer group.

The Nuclear Executive Committee reviews nuclear safety performance as part of its normal business reviews. This review is fully integrated into the business/operating reports and takes account of related information such as independent evaluations, internal self-assessments and international OPEX.

It is concluded that a variety of safety review procedures are implemented in Pickering NGS to provide a continuing audit and surveillance of plant operational safety.

O-290 – Emergency Operating Procedures

Principle: Emergency operating procedures are established, documented and approved to provide a basis for suitable operator response to abnormal events.

This safety principle is addressed together with Safety Principle O-288, Normal Operating Procedures (Section D.9 of this document) as they constitute a continuum of plant operating procedures.

O-292 – Radiation Protection Procedures

Principle: The radiation protection staff of the operating organization establish written procedures for the control, guidance and protection of personnel, carry out routine monitoring of in-plant radiological conditions, monitor the exposure of plant personnel to radiation, and also monitor releases of radioactive effluents.

Radiation Protection and Regulatory Affairs departments report centrally within the Nuclear Services division. OPG Program [N-PROG-RA-0013-R010, *Radiation Protection*, February 21, 2017] governs the conduct of radiation protection activities and contains cross references to the applicable clauses of CSA N286-05. N-PROG-RA-0013-R010 mandates that "when making engineering changes, engineers maintain or improve upon designs that reduce occupational exposures throughout the lifecycle of the facility". At Pickering NGS, this process is implemented through OPG Standard [N-STD-RA-0018, *Controlling Exposure As Low As Reasonably Achievable*, November 27, 2014].

Per N-PROG-RA-0013-R010, the initial design of the station was created such that the layout and operation of facility SSCs and processes were consistent with the established radiation protection guidelines and contributes to maintaining occupational radiation exposures ALARA. During the initial design of Pickering NGS, emphasis was placed on the reduction of occupational radiation doses. This resulted in design improvements that allowed Pickering NGS and other early CANDU designs to have far less worker exposure to radiation than most other utilities, despite increased staffing needs associated with online fueling, as documented in OPG Report [N-REP-N7250014, *Excellence Through Radiation Protection Practices in Ontario Hydro CANDU-PHW Nuclear-Electric Generating Stations*, September 1, 1987].

Recent improvements in the ALARA strategy include:

- Establishing dose goals for radioactive work to improve individual and station dose performance.
- Use of robotics to perform tasks in radioactive work areas, reducing radiation exposure and therefore dose to workers.

- Use of dynamic learning activities to provide workers an opportunity to practice radiation protection fundamentals in a simulated radioactive work environment using remotely controlled radio technology.
- Implementation of remote reading radiation detection instrumentation and real time data transmission to facilitate improved job planning and awareness of current radiological conditions.
- Implementation of a gamma ray imaging spectrometer to perform enhanced radiation surveys and to identify areas with elevated dose rates, enabling more effective shielding to reduce dose to workers.
- Improved vapour recovery dryer performance.
- Implementation of a reactor face shielding cabinet and other innovative shielding designs to reduce radiation dose rates and, therefore, doses to workers.
- Continued improvements to the provision of remote monitoring and job coverage.

N-PROG-RA-0013-R010 discusses the various procedures and methods used to maintain both exposure control and contamination control at OPG Nuclear facilities, including Pickering NGS. Requirements regarding radiation monitoring instrumentation are primarily discussed in Section 1.5.4, “Hazard Identification and Assessment”, which mandates that all “instruments used for performing surveys are approved by the Health Physics Department to ensure they are appropriate and effective for use in measuring hazards encountered in the station.”

Procedures have been developed for the recording of radiation doses and radioactive effluents. Documentation includes Quarterly Performance Reports. The following are submitted to CNSC per CNSC REGDOC-3.1.1 in accordance with Pickering NGS’s Power Reactor Operating Licence:

- Quarterly Operations Report(s) that include:
  - Worker radiation dose due to events described in S-99 clauses 6.3.1(7) or 6.3.1(8) (superseded by Table A.1 20 b) in REGDOC-3.1.1) and the collective radiation dose of each group of workers.
  - The quantity of radioactive material released in routinely discharged (gaseous and liquid) effluents and hazardous substances.

In conclusion, OPG has implemented a highly effective, well documented and mature Radiation Protection program, based on industry best practices, to keep doses ALARA at Pickering NGS and to monitor releases.

O-299 – Feedback of Operating Experience

Principle: Plant management institutes measures to ensure that events significant for safety are detected and evaluated in depth, and that any necessary corrective measures are taken promptly and information on them is disseminated. The plant management has access to operational experience relevant to plant safety from other nuclear power plants around the world.

OPG Charter [N-CHAR-AS-0002-R019, *Nuclear Management System*, November 1, 2016] establishes the overall requirements for sustaining and improving station performance. In part, this is accomplished by utilizing internal and industry OPEX to improve human, plant, and equipment performance and design, procurement, construction, commissioning, and operating requirements and practices.

Per OPG Program [N-PROG-RA-0003-R010, *Corrective Action*, January 14, 2015], conditions adverse to quality are promptly identified, documented in sufficient detail and reported. Based on the condition report the significance level and resolution category is assigned. The aspect of using OPEX from within OPG Nuclear and the industry is an integral part of this program.

Section 1.4.1 of OPG Procedure [N-PROC-AS-0028-R018, *Development Review And Approval Of Technical Procedures*, February 9, 2017] specifies that for new procedures and major revisions to existing procedures, the preparation stage of the process must involve a review of OPEX lessons learned, event information and just-in-time packages to incorporate applicable information. This ensures that industry best practices are incorporated in the procedure development process.

Section 1.6 of OPG Procedure [N-PROC-OP-0005-R012, *Pre-job Briefing And Post-job Debriefing*, June 6, 2013] provides direction on the use of OPEX during pre-job briefings. The review of OPEX prompts staff to consider any industry best practices which may have emerged since the applicable procedure was last revised or used. Following the execution of the procedure, the post-job debriefing process documented in N-PROC-OP-0005-R012 directs staff to document lessons learned or opportunities for improvement using the appropriate work process (i.e., a Technical Procedure Action Request would be initiated in accordance with N-PROC-AS-0028-R018 to incorporate any procedural improvements/enhancements that were identified during the course of performing the task).

The adequacy of safety-related procedures is also assessed through the ongoing OPEX process described in OPG Procedure [N-PROC-RA-0035-R019, *Operating Experience Process*, September 16, 2016], which monitors events around the world to determine if there is any unforeseen event that may have applicability to Pickering NGS. If/when an event is deemed to be applicable, existing procedures are reviewed to identify any vulnerabilities or weaknesses that could result in similar events or problems with corrective actions initiated as required.

A weekly CANDU Owners Group (COG) OPEX screening meeting, facilitated and administered by COG, serves as an initial screening forum to review event reports from CANDU stations, nuclear industry and non-nuclear sources for applicability and significance to CANDU units. Committee members include representatives from all CANDU facilities (including

Pickering NGS), vendors, research organizations and WANO. Station Condition Records are initiated for items from the screening meeting where a potential vulnerability is identified that is assessed to be applicable to Pickering NGS. Station Condition Records are processed in accordance with OPG Procedure [N-PROC-RA-0022-R034, *Processing Station Condition Records*, March 24, 2017]. An outcome of the Station Condition Record process may include corrective actions to update existing procedures.

Prior to the weekly meeting, the Senior Officer, OPEX screens the recent events at their site, and selects those events that they believe may be of relevance to other sites for review at the COG OPEX Weekly Screening Meeting. On behalf of the utilities, COG provides the initial screening of the international nuclear industry reports and relevant non-industry events.

OPEX representatives at Pickering, Darlington and Nuclear Support each submits relevant OPG significant events to be presented at the COG OPEX Weekly Screening Meeting as per OPG Procedure [N-PROC-RA-0035-R019, *Operating Experience Process*, September 16, 2016]. When potentially significant, these events have been investigated according to OPG Standard [N-STD-RA-0008-R014, *Incident Investigation*, November 16, 2016] and entered into the OPG Station Condition Record process.

Upon completion of the COG OPEX Weekly Screening Meeting, plant OPEX staff in consultation with line staff and/or appropriate OPEX Single Point of Contacts performs a further screening of the Weekly Screening Meeting events from other utilities/non utilities. Items believed to be applicable and actionable at the plant are dispositioned in an Action Request, and if a more significant gap is identified, a Station Condition Record is created and the gap is evaluated according to N-PROC-RA-0022-R034. This process includes not only consideration of potential weaknesses of the plant equipment and operation, but also opportunities for improvement and utilization of research findings.

When an external OPEX item is entered into the electronic Station Condition Record System, it is analyzed by the relevant line department as an OPEX Station Condition Record and processed and stored according to the corrective action program specified in OPG Program [N-PROG-RA-0003-R010, *Corrective Action*, January 14, 2015].

OPG Procedure [N-PROC-RA-0094-R007, *Discovery Issue Resolution Process*, March 21, 2017] is provided to ensure that regulatory limits are maintained by evaluating discovery issues that may impact the SOE. The adequacy of the program for the sending and receiving of experience relevant to safety is monitored and assessed through audits and self-assessments.

In conclusion, Pickering NGS benefits from and effectively implements appropriate modifications arising from comprehensive OPEX feedback programs conducted by the CANDU industry and by OPG as a whole, as well as programs specifically established to identify and assess operational experience within the plant.

O-305 – Maintenance, Testing and Inspection

Principle: Safety related structures, components and systems are the subject of regular preventive and predictive maintenance, inspection, testing and servicing when needed, to ensure that they remain capable of meeting their design requirements throughout the lifetime of the plant. Such activities are carried out in accordance with written procedures supported by quality assurance measures.

As described in [P-CORR-01060-0632223, *Fitness For Service Of Major Components*, February 2017], Major Component aging is managed through a comprehensive program of in-service inspections, maintenance, engineering assessment and confirmatory research and development. These processes provide for the timely detection and mitigation of aging effects in SSCs that impact plant safety, reliability and economics; thereby providing a decision making process to optimize asset management. Regular preventive and predictive maintenance, inspection, testing and servicing of SSCs important to safety and reliability are conducted in accordance with OPG Programs [N-PROG-MA-0026-R002, *Equipment Reliability*, June 4, 2015], [N-PROG-MA-0004 R011, *Conduct of Maintenance*, April 28, 2015] and [N-PROG-MP-0004 R016, *Pressure Boundary*, November 27, 2015]. These programs are supported by a set of detailed implementing procedures.

OPG Program [N-PROG-MP-0008 R006B, *Integrated Aging Management*, May 2, 2016] ensures that the condition of critical nuclear power plant equipment is understood and that required activities are in place to assure the health of these components and systems while the plant ages. The Integrated Aging Management program covers all critical SSCs having nuclear safety, production, cost, conventional safety and environmental significance. OPG Procedure [N-PROC-MP-0060 R005B, *Aging Management Process*, October 1, 2015] is the predominant method for identifying SSCs to be included in the aging management program.

OPG Program [N-PROG-MA-0017 R009, *Component and Equipment Surveillance*, May 31, 2017] defines requirements for establishing programs to ensure the health of select nuclear power plant components and equipment. This program defines the requirements for establishing component programs that manage component and equipment health including inspection, maintenance and testing.

Major components and structures such as Steam Generators, Fuel and Fuel Channels, Feeder piping and Reactor Components are covered by formal Life Cycle Management Plans and are managed in accordance with N-PROG-MA-0025 R002, *Major Components,* March 25, 2015.

In conclusion, OPG has a robust and multi-faceted SSC inspection program, such that the current condition of the plant is well understood. This provides a solid baseline for the intended operation extension as well as confidence that the requisite data will continue to be acquired going forward. Maintenance programs are aligned with industry best practice and are carried out in accordance with written procedures that are part of the overall management system.

O-312 – Quality Assurance in Operation

Principle: An operational quality assurance programme is established by the operating organization to assist in ensuring satisfactory performance in all plant activities important to plant safety.

OPG Charter [N-CHAR-AS-0002-R019, Nuclear Management System, November 1, 2016] and supporting documents referenced in the Charter, establish the Nuclear Management System for OPG Nuclear, which is compliant with CSA N286-05 and CSA N286-12, and assures that systems, equipment and activities are of the required quality throughout the life of the OPG Nuclear facilities. Supporting organizations and contractors who do not have a quality program approved by the OPG Nuclear organization are required to follow Nuclear Management System requirements.

OPG Standard [N-STD-AS-0020 R014, *Nuclear Management Systems Organizations*, May 19, 2016] outlines the implementation of the OPG Nuclear Quality Program described in N-CHAR-AS-0002-R019. The program is implemented through procedures that are developed, approved, modified and documented through a formal process. There are operating procedures that apply comprehensively to normal, abnormal and emergency conditions (including DBA conditions, post-accident conditions and design extension conditions). These are described in the review of Safety Principle O-288, Normal Operating Procedures, which also addresses Safety Principle O-290, Emergency Operating Procedures.

OPG Standard [N-STD-AS-0023-R008, *Nuclear Safety Oversight*, September 15, 2015] describes independent assessment (external and internal) processes used for oversight and assessment of OPG Nuclear safety. Internal independent assessments are performed by Supply Chain, Nuclear Waste Management and Nuclear Oversight. Nuclear Oversight audits are conducted in accordance with OPG Procedure [N-PROC-RA-0048-R018, *Conducting Performance Based Audits and Assessments*, April 28, 2016]. External independent assessments are performed by the Nuclear Safety Review Board in accordance with OPG Standard [N-STD-RA-0035-R004, *Nuclear Safety Review Board*, May 25, 2015]. Audits are carried out at frequencies determined through risk-based assessments, with sufficient frequency to confirm conformance with the quality assurance program and related programs, procedures and instructions. Audit frequency ranges from one to five years, depending on results of the risk based frequency assessment documented in the Nuclear Operations Assurance Map. Objectives of N-PROC-RA-0048-R018 include identification and documentation of conditions adverse to quality in accordance with OPG Program [N-PROG-RA-0003-R029, *Corrective Action*, January 14, 2015], verification of compliance and effectiveness of the pressure boundary quality assurance program, as well as verification of compliance and effectiveness of the quality assurance manual for testing and repairing relief valves.

Quality assurance records are defined in OPG Procedure [OPG-PROC-0179-R001, *Nuclear Quality Assurance Records*, March 14, 2016] as essential records providing evidence of licensing, design, construction, operation, maintenance, testing and modification of OPG Nuclear facilities.

OPG Program [N-PROG-OP-0001-R008, *Nuclear Operations*, November 13, 2015] establishes safe and efficient operating practices and processes within OPG Nuclear facilities that provide operating staff the ability to ensure facilities are operated in such a manner that Power Reactor Operating Licences, Operating Policies and Principles and other applicable regulations and standards are followed.

To summarize, Pickering NGS operation is supported by a comprehensive and effective quality assurance program. Operation of the Pickering NGS units is conducted in compliance with rigorous quality assurance requirements.

D.4. Safety Principles Related to Levels 1, 2, 3

The following safety principles are related to defence-in-depth Levels 1, 2, 3:

- D-192 – Protection Against Power Transient Accidents
- D-195 – Reactor Core Integrity
- O-278 – Training
- O-284 – Operational Limits and Conditions

As demonstrated below, Pickering NGS design and operation are aligned with all safety principles related to Levels 1, 2, 3.

D-192 – Protection Against Power Transient Accidents

Principle: *The reactor is designed so that reactivity induced accidents are protected against, with a conservative margin of safety.*

In Pickering NGS Units 1,4 and 5-8, the neutronic characteristics provide inherent protection against a reactivity induced accident. The use of natural uranium Fuel, on-power refuelling and a heavy water Moderator leads to a design characterized by good neutron economy and low excess reactivity. The reference lattice pitch is optimum in the sense that an increase or decrease of the lattice pitch would reduce reactivity.

In the Pickering NGS design, all reactivity rods for control and shutdown are introduced into guide tubes positioned in the low-pressure Moderator. These guide tubes are located interstitially between rows of Calandria tubes.

Units 1,4 and 5-8 Reactor Regulating Systems automatically control power and reactivity. A setback function within the Reactor Regulating System reduces power upon detection of parameters outside prescribed normal limits.

The Pickering NGS Units 5-8 design includes the stepback feature of the Reactor Regulating System, which can control reactivity induced accidents. Stepback occurs by dropping four control absorbers into the core on detection of high neutron power or high log rate of neutron power. Units 5-8 have two fast-acting Shutdown Systems with neutronic trip parameters that are able to mitigate rapid reactivity transients as well as slow transients. In Units 1,4, the Shutdown Systems SDSA and SDSE both employ the same set of shutoff rods, but have independent neutron overpower and high log neutron rate detection and trip logic. SDSA also has a high linear neutron rate trip parameter. SDSE has a distributed Neutron Overpower trip parameter. For these accidents, as well as for other transients, the Shutdown Systems incorporate diverse trip parameters for each accident scenario.

Safety analyses documented in Part 3, Appendix 3 of the Pickering NGS Safety Reports [NA44-SR-01320-00002-R004, *Pickering Nuclear 1-4 Safety Report: Part 3 – Accident Analysis*, October 31, 2013] and [NK30-SR-01320-00003-R004, Vol. 2, *Pickering Nuclear 5-8 Safety Report Part 3 – Accident Analysis*, October 30, 2014] demonstrate the effectiveness of the Shutdown Systems in mitigating reactivity induced accidents. Further description of the Shutdown Systems is provided in the review of Safety Principle D-200, Automatic Shutdown Systems.

To summarize, Pickering NGS has been designed with a focus on protection against reactivity-induced transients, specifically through highly reliable and effective reactor Shutdown Systems.

D-195 – Reactor Core Integrity

Principle: The core is designed to have mechanical stability. It is designed to tolerate an appropriate range of anticipated variations in operational parameters. The core design is such that the expected core distortion or movement during an accident within the design basis would not impair the effectiveness of the reactivity control or the safety shutdown systems or prevent cooling of the fuel.

The Pickering NGS cores (each comprising the reactor and Fuel Channels) have been designed to high standards. Heavy water is contained in the Calandria which envelops the Fuel Channels. Each pressure tube is insulated from the Moderator by a tube surrounding it, but separated from it, by an annular space filled with dry carbon dioxide. This second tube is called the Calandria tube and it is roll expanded at both ends to the flat Calandria tubesheets, thereby completing the Moderator pressure boundary. This configuration results in a low temperature, low pressure design for the Calandria vessel, with the Moderator System being operated independently of the high temperature high pressure heat transport coolant in the pressure tubes. The Calandria operates near atmospheric pressure.

The design, manufacture, installation and inspection of the Calandria assembly comply with the following sections of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code:

- Section II (Materials Specification)
- Section III (Nuclear Vessels) (1963 edition, unless otherwise indicated)
- Section VIII, Division 1 (Pressure Vessels)
- Section IX (Welding Qualifications)

The Units 1,4 reanalysis of the Calandria assembly, carried out during the large scale Fuel Channel replacement program (Proceedings of CNS Symposium on Advanced Nuclear Services, 1986), was performed in accordance with ASME Code, Section II, Subsection NB, 1983 edition with addenda up to and including winter 1983.

Operating Conditions

The operating conditions have been established for the design purposes of the Calandria assembly. These conditions include:

- Design conditions
- Normal conditions
- Upset conditions
- Emergency conditions
- Maximum allowable stress limit

The allowable stresses are selected on the basis that different types of stresses require different limits.

Cyclic Loading

The design of the Calandria assembly components is based on appropriate combinations of coincident loadings due to temperature, pressure and specific combinations of mechanical loads.

Fuel Channels and Reactor Structures were also designed to meet stringent performance quality requirements. Continued monitoring of these components is performed per the Major Components program [N-PROG-MA-0025-R002, *Major Components*, March 25, 2015]. Pressure Tube design is further discussed under Safety Principle D-209, Reactor Coolant System Integrity.

Postulated initiating events that potentially pose a risk to the structural integrity of the reactor core have been identified and analyzed in the Pickering NGS Safety Reports. Analyses of postulated in-core LOCA documented in Part 3, Appendix 4 of the Safety Reports [NA44-SR-01320-00002-R004, *Pickering Nuclear 1-4 Safety Report: Part 3 – Accident Analysis*, October 31, 2013] and [NK30-SR-01320-00003-R004, Vol. 2, *Pickering Nuclear 5-8 Safety Report Part 3 – Accident Analysis*, October 30, 2014] have demonstrated that the damage to in-core components will not result in failure of any other Fuel Channel or of the Calandria vessel. In addition, the reactivity depth provided by the shutoff rods sufficiently delays re-criticality of the reactor for the most limiting operating state.

In summary, the integrity of the Pickering NGS core for the range of normal operation and accidents within the design basis is assured through high quality design and operation, and is demonstrated by comprehensive assessments and analyses.

O-278 – Training

Principle: Programmes are established for training and retraining operations and maintenance, technical support, chemistry and radiation protection personnel to enable them to perform their duties safely and efficiently. Training is particularly intensive for control room staff, and includes the use of plant simulators.

OPG has adequate staff training facilities, training staff and training programs, and has the oversight to confirm this. The training program requirements apply to all disciplines, including operations and maintenance, technical support, chemistry and radiation protection. [N-TQD-601-00001-R017, *Leadership and Management Training and Qualification Description*, April 4, 2015] provides qualification and professional development requirements for supervisors and managers which includes Safety Culture for Managers. OPG Program [N-PROG-TR-0005-R016, *Training*, January 5, 2016] describes the training program for regular staff, contractors, temporary personnel and other staff assigned work at OPG Nuclear. It includes the structure, processes and tools for defining, developing, implementing, documenting, assessing and improving the training required to ensure OPG Nuclear staff have the appropriate knowledge, skill and attitudes for safe and efficient plant operation.

As described in Section 1.6.3 of [P-CORR-00531-03719, *Application for Renewal of Pickering Nuclear Generating Station Power Reactor Operating Licence*, July 4, 2012], OPG identifies qualified and competent individuals for key positions. With career development and succession planning being key elements in the management capability strategy, a Corporate Succession Plan ensures that individuals with high leadership potential are identified.

OPG Procedure [N-PROC-TR-0002-R008, *Control of Vendor-Supplied Training*, October 14, 2016] establishes requirements, process and accountabilities for control of training developed by vendors and training delivered by vendors within OPG Nuclear premises and externally. This procedure addresses the following:

- Vendor instructor capabilities and qualifications to instruct OPG Nuclear staff.
- Requirements for review, verification and acceptance of vendor-supplied training prior to and after training delivery.

Similarly, training and qualification description documents are also available for operations and maintenance staff. OPG List [N-LIST-08920-10001-R008, *Trained Performance Areas*, March 2, 2016] identifies the list for trained performance jobs (i.e., performance areas in the training program), and supporting Qualification Guides reside within the framework of OPG Procedure [N-PROC-TR-0021-R011, *Training and Qualification Description Development and Approval Process*, June 24, 2016].

N-PROC-TR-0021-R011 provides requirements for developing, approving, revising and implementing the Training and Qualification Documents and Qualification Guides. The Training and Qualification Documents and the Qualification Guides document entry-level, initial and continuing training requirements. Training and Qualification Documents contain details of training stages and requirements for each stage as follows:

- **Initial Training:** Introduce and develop job related knowledge, skills and performance standards, preparing personnel to independently perform assigned duties and tasks.
- **Continuing Training:** Used to maintain and enhance knowledge and skills, and address areas such as plant equipment and procedure changes, infrequently used and difficult skills, knowledge and skills weaknesses, and lessons learned from OPEX. Continuing training may also provide refresher type training at a specified frequency, and may be required to maintain qualified status.

Training and Qualification Documents and Qualification Guides include but are not limited to:

- Authorized Nuclear Operator
- Control Room Shift Supervisor
- Health Physicist
- Maintenance

Training specifically for radiation protection staff is governed by several documents, including [N-TQD-502-00001-R018, *Nuclear Radiation Protection Training and Qualification Description*, October 27, 2014].

Continuing training is developed in accordance with OPG Instruction [N-INS-08920-10021-R001, *Continuing and Requalification Training – Curriculum Development and Implementation Process*, September 13, 2016]. Annual continuing training requirements, including impact on qualification status, are presented in the corresponding Training and Qualification Document. These requirements include reference to any existing Continuing Training Plan. The continuing training meets the needs of both refresher and upgrade training.

Training, qualification for, and certification for Main Control Room staff and for field positions ensure that the staff are competent to perform the functions assigned to them. The simulator is used extensively for initial training and qualification, as well as for refresher/requalification training. Per OPG Instruction [N-INS-08920-10002 R007, *Simulator – Based Initial Certification Examinations for Shift Personnel*, August 30, 2016], Simulator Exercise Guides are used as part of training for certified staff. Trainees are assessed per OPG Instruction [N-INS-09110-10059-R004, *Simulator Performance Observation and Crew Critiques*, April 21, 2015].

OPG Instruction [N-INS-08920-10002-R007, *Simulator-Based Initial Certification Examinations for Shift Personnel*, July 19, 2016] provides specific instructions on the application of CNSC-EG2, *Requirements and Guidelines for Simulator-based Certification Examinations for Shift Personnel at Nuclear Power Plants*, for initial CNSC certification, which includes:

- Process for planning, developing, conducting and grading simulator-based certification examinations.
- Examination follow-up activities including process for dealing with passes and conditional passes.
- Requirements, criteria and guidelines for administering simulator-based initial certification examinations.

N-INS-08920-10001, *Requalification Testing of Certified Shift Personnel* provides requirements for requalification tests that OPG Certified Shift Personnel must successfully complete when seeking renewal of certification. The instruction also identifies processes that OPG follows in developing, conducting and grading written and simulator-based requalification tests which demonstrate that Certified Shift Personnel have retained knowledge and skills required to work competently in their assigned positions.

Training for engineering and technical support staff is discussed in detail under Safety Principle O-296, Engineering and Technical Support of Operations.

It is concluded that the training program for Pickering NGS operations, maintenance and technical support staff is comprehensive and ensures that staff are properly trained for their assigned duties.

O-284 – Operational Limits and Conditions

Principle: A set of operational limits and conditions is defined to identify safe boundaries for plant operation. Minimum requirements are also set for the availability of staff and equipment.

The SOE is the set of limits and conditions within which the plant shall be operated to ensure conformance with the Safety Reports and that the Safety Report conclusions remain valid. OPG Standard [N-STD-MP-0016-R002, *Safe Operating Envelope*, June 21, 2012] provides requirements for defining, implementing and maintaining the SOE. The specific objectives of the SOE are to establish the following:

- Thorough and current record of safety credits and operating limits in the form of operational safety requirements. Safe Operating Limits and Conditions of Operability (SOE Limits) are captured in station operating documentation, which provide plant operators with the information required to ensure safe operation of the plant in conformance with the requirements of the Safety Analysis.
- A compliance framework whereby plant operation within the requirements established as part of the SOE is verified on a regular basis and appropriate corrective actions are implemented.
- Infrastructure by which the SOE is integrated with other relevant business processes and maintained current over the life of the station.

SOE systems are listed in OPG Instruction [N-INS-03602-10001-R001, *Preparation of Safe Operating Envelope Compliance Tables*, February 11, 2015] and are identified in N-STD-MP-0016-R002. These SSCs include systems and their associated critical components and structures, for which operational safety requirements are specified to conform to the Pickering NGS Safety Reports and hence, the safety analysis. OPG has prepared formal operational safety requirement documents in support of SOE systems. The SOE systems and associated operational safety requirements are listed in OPG instruction N-INS-03602-10001-R001 and are also listed in the Licence Conditions Handbook [P-CORR-00531-04886, *Pickering NGS: Licence Conditions Handbook, LCH-PNGS-R005*, November 10, 2016].

The methodology for the preparation and revision of operational safety requirement reports is described in [N-ST-08131.02-10000-R02, *Preparation of Operational Safety Requirements*, December 22, 2014]. The operational safety requirement reports contain the following information (as outlined in Section 1.2.4 of N-STD-MP-0016-R002):

- A brief overview of the safety functions of the system in relation to the DBA for which it is credited.
- Safety limits defining the acceptable standards with respect to component or parameter performance.
- Surveillance requirements identifying specific objectives of tests or checks to ensure plant operation within the defined Safety Limits.
- Conditions of Operability defining the impact on overall availability of the system or safety function in the event of operation outside of the defined Safety Limits.

OPG Standard [N-STI-03602-10000-R001, *SOE Instrument Uncertainty and Allowable Value Calculations*, October 07, 2011] describes the methodology for the preparation and revision of Instrument Uncertainty Calculation Reports. The Instrument Uncertainty Calculation Reports contain calculations for applicable instrumentation loops (as outlined in Section 1.2.7 of N-STD-MP-0016-R002).

As per Section 1.3.2 of N-STD-MP-0016-R002, information contained in the operational safety requirement and Instrument Uncertainty Calculation Reports are incorporated into affected station operating documentation (e.g., Operating Manuals, Abnormal Incident Manuals, Safety Related System Tests and the preventive maintenance program) in accordance with OPG Instruction [N-INS-03602-10001-R001, *Preparation of Safe Operating Envelope Compliance Tables*, February 11, 2015].

To ensure effective monitoring (e.g., ensuring SSC compliance with design requirements and specifications), maintenance and enhancement of system performance and reliability, OPG Procedure [N-PROC-MA-0024-R016, *System Performance Monitoring*, December 09, 2016] provides a consistent and comprehensive process for system engineers.

The process for identifying and evaluating degraded station conditions when the ability of SSCs to carry out their defined safety-related functions is questioned is described in the technical operability evaluation process in OPG Procedure [N-PROC-MP-0045 R008, *Technical Operability Evaluation*, September 15, 2015]. The process provides for engineering verification that a SSC is capable of fulfilling its minimum credited safety function(s). It is required when uncertainty arises with respect to operability of equipment to meet the functional requirements of the defined SOE, and is initiated concurrently with a Station Condition Record.

OPG Instruction [P-INS-09100-00003-R010, *Pickering Minimum Shift Complement*, April 13, 2017] defines the minimum shift complement staffing requirements to ensure that safe conditions are maintained during normal operations along with the capability to be able to respond to all station emergencies. Per OPG Instruction [P-INS-09260-00008-R008, *Duty Crew Minimum Complement Assurance*, April 10, 2017], minimum shift complement is the minimum number of qualified workers who shall be present at all times to ensure the safe operation of the Pickering units, respond to all credible events and ensure adequate emergency response capability for the most resource intensive conditions. P-INS-09100-00003-R010 is written to comply with CNSC G-323*, Ensuring the Presence of Sufficient Qualified Staff at Class I Nuclear Facilities – Minimum Staff Complement* and CNSC G-278, *Human Factors Verification and Validation Plans*.

P-INS-09260-00008-R008 defines the responsibilities and processes to ensure that the minimum shift complement according to P-INS-09100-00003-R010 is met, as well as the correct use and updating of the local area network based minimum complement compliance program, which is used to ensure minimum shift complement is met. The instruction defines: actions required when below complement, duty crew accounting, absence reporting, “step-up” (i.e., designating a temporary change in work group or Emergency Response Organization assignment), minimum availability requirement, emergency role qualification and position assignments.

To summarize, operational limits and conditions for Pickering NGS are clearly defined and justified based on appropriate safety analyses. Procedures are in place to ensure that the plant is operated within the applicable limits. Minimum requirements are also set for the availability of staff and equipment.

D.5. Safety Principles Related to Levels 3, 4, 5

There is one safety principle related to defence-in-depth Levels 3, 4, 5:

- EP-339 – Assessment of Accident Consequences and Radiological Monitoring

As demonstrated below, Pickering NGS design and operation are aligned with the safety principle related to Levels 3, 4, 5.

EP-339 – Assessment of Accident Consequences and Radiological Monitoring

Principle: Means are available to the responsible site staff to be used in early prediction of the extent and significance of any release of radioactive materials if an accident were to occur, for rapid and continuous assessment of the radiological situation, and for determining the need for protective measures.

Response and mitigation strategies for accidents that release significant radioactivity to the environment are developed in accordance with the deterministic safety analysis and with insights from the PSA. In addition, the Emergency Response Projection computer code is a tool used to assist in decision-making for off-site emergency response protective actions. Emergency Response Projection is an analysis-like code that uses deterministic methods to evaluate potential offsite consequences and predict the timing of Containment re-pressurization in the event of an accident. OPG and Bruce Power are currently performing a project to enhance their emergency response projection tools. The project is adopting the Unified RASCAL Interface, which is widely used by U.S. utilities.

For continuous assessment of the radiological situation following an accident, OPG Nuclear's emergency facilities are equipped with area radiation monitoring equipment, radiation survey kits, off-site monitoring vehicles and meteorological monitoring data readout equipment, as appropriate to the facility. OPG has also recently installed automated solar-powered Near Boundary radiation monitors around the perimeter of the Pickering site to support radiological monitoring. OPG emergency response facilities are linked to the Provincial Emergency Operations Centre, Municipal Emergency Operations Centre and Regional Emergency Operations Centre through landline phones and other systems to allow information transfer.

It is concluded that prediction tools and monitoring capability have been provided to support emergency response. The ongoing Unified RASCAL Interface project will further enhance the capability to project off-site consequences while an accident is in progress.

D.6. Safety Principles Related to Levels 1, 2

The following safety principles are related to defence-in-depth Levels 1, 2:

- D-164 – Plant Process Control Systems
- D-203 – Normal Heat Removal
- D-209 – Reactor Coolant System Integrity
- D-240 – New and Spent Fuel Storage

As demonstrated below, Pickering NGS design and operation are aligned with all safety principles related to defence-in-depth Levels 1, 2.

D-164 – Plant Process Control Systems

Principle: Normal operation and anticipated operational occurrences are controlled so that plant and system variables remain within their operating ranges. This reduces the frequency of demands on the safety systems.

The major control loops and their basic functions in Units 1,4 (Units 5-8 equivalent in parentheses, if different) are:

1. The Reactor Control System monitors and operates the reactivity control devices to maintain a desired power level and neutron flux shape. The reactor power can be specified by the operator or by Steam Generator pressure control depending on the mode of unit control. There is a reactor power setback facility to keep plant parameters within pre-determined values during abnormal conditions.
2. The Steam Generator pressure control loop manipulates the turbine load setpoint or steam reject valves to maintain the steam pressure setpoint.
3. The turbine-generator control (unit power regulator) loop performs loading of the turbine/generator and provides monitoring of the turbine and generator.

4. The Steam Generator level control loop manipulates the feedwater valves to maintain the water level in the Steam Generator at a level setpoint dependent on reactor power.
5. The heat transport pressure control loop controls the heavy water feed and bleed valves to maintain a constant heat transport pressure.

The Reactor Control System provides a high degree of immunity to process upsets, measurement failures, etc., due to a high degree of redundancy in control devices and process measurements. Extensive checks are performed in the programs to ensure that faulty signals are discarded. In case of loss of an entire set of signals, control is transferred to the standby computer. The ability to maintain control in the presence of partial system failures, combined with high reliability of the Dual Computer Control System, leads to a very high availability of the Reactor Regulating System.

The normal method of shutting down the reactor is by means of the Reactor Regulating System. During plant upsets or potentially undesirable operating conditions, the reactor is shut down or derated automatically by the reactor setback function of Reactor Regulating System (Units 5-8 includes setback and an additional stepback function).

The computer system reduces reactor power setpoint in a ramp fashion when a setback signal is received from triplicated setback relay logic. A setback signal is generated by certain plant conditions. Reactor power is reduced at a controlled rate until either the setback signal clears or the power is reduced to a specified low value.

In the event that the turbine cannot accept all the steam generated, the Boiler Pressure Control system operates the steam reject valves. When the large steam reject valves are opened the reactor will setback. This setback will terminate if the large steam reject valves close or if the operator intervenes.

It is concluded that the control systems in Pickering NGS have been designed to maintain the reactor operating conditions within the normal operating range, and to effectively respond to anticipated transients to avoid the need for safety system action.

D-203 – Normal Heat Removal

Principle: Heat transport systems are designed for highly reliable heat removal in normal operation. They would also provide means for the removal of heat from the reactor core during anticipated operational occurrences and during most types of accidents that might occur.

In normal operation, heat is removed from the Fuel to the Steam Generators by the forced flow of coolant in the Heat Transport System. Steam generated in the Steam Generators is condensed in the condenser by the Condenser Cooling Water System, which rejects the heat to the ultimate heat sink, Lake Ontario (see Safety Principle S-142, Ultimate Heat Sink Provisions).

Reactor startup after a maintenance outage requires that verification of flow in individual Fuel Channels be performed. During normal operation, resistance temperature detectors in outlet Feeders provide a continuous means of monitoring for major flow blockages in any channel.

During fuelling, monitoring of the channel outlet temperature provides the operator with an indication of a flow blockage in the Fuel Channel being refuelled. As a result of these design, operational and instrumented alarm features, the occurrence of an undetected flow blockage is very unlikely.

In the event of a partial or total loss of forced flow in the Heat Transport System, heat removal from the Fuel to the Steam Generators is maintained.

In Units 1,4, two small and six large Steam Reject Valves discharge steam from the main steam line if the turbine becomes unavailable. The two small valves discharge steam to the forebay when the reactor is warmed up and the six large valves discharge to atmosphere. Rapid shutdown of the turbine causes all the steam reject valves to open. The steam reject valves are also sized to pass the quantity of steam produced by the reactor. The Units 5-8 design is similar, except that there are 10 large Steam Reject Valves in these units.

For events involving a loss of Steam Generator inventory (steam main break or loss of feedwater), qualified backup boiler water supplies are available to remove the decay heat generated in the Fuel. In the short term, the Boiler Emergency Cooling System provides water to Units 1,4 and Units 5-8 for reactor decay heat removal by the boilers for accident situations initiated by or leading to failure of the normal feedwater supply. The Boiler Emergency Cooling System is qualified to Design Basis Earthquake Category B. For long-term cooling, the Emergency Boiler Water Supply system is designed to provide emergency water to the boilers of Units 1,4 following a postulated main steamline failure within the Units 1,4 powerhouse. It is also capable of providing water to the boilers for a total loss of feedwater event. The Emergency Boiler Water Supply System is supplied from the discharge headers of the Units 6 and 7 High Pressure Service Water System. In Units 5-8 in the longer term, the shutdown cooling system is initiated manually to provide the heat sink for the Heat Transport System. In addition, the Emergency Water System is a seismically qualified system supplying long-term emergency makeup to the boilers, Heat Transport System, ECI recovery heat exchangers, Containment air coolers and the Moderator.

The ECI System is designed to refill the Heat Transport System following a LOCA, and to provide coolant makeup and the heat sink in the long term via the ECI heat exchangers.

For additional information on emergency heat removal see Safety Principle D-207, Emergency Heat Removal.

In summary, Pickering NGS Units 1,4 and Units 5-8 are equipped with heat removal systems that effectively and efficiently remove heat from the Fuel under normal operating conditions. These systems are also effective in response to anticipated transients and accidents within the design basis, supplemented by the emergency makeup water systems included in the design.

D-209 – Reactor Coolant System Integrity

Principle: Codes and standards for nuclear vessels and piping are supplemented by additional measures to prevent conditions arising that could lead to a rupture of the primary coolant system boundary at any time during the operational lifetime of the plant.

The CSA N285 series of Standards specifies requirements for nuclear vessels and piping applicable to nuclear power plants in Canada and references the applicable requirements of the ASME Boiler and Pressure Vessel Code. Compliance with CSA N285.0-08, *General Requirements For Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants*, (including Updates No. 1 and No. 2), is currently a licence requirement for Pickering NGS (per PROL 48.03/2018 [PROL 48.03/2018, *Nuclear Power Reactor Operating Licence, Pickering Nuclear Generating Station*]) as indicated in Section 6.2 and Appendix C.1 of the R05 Pickering Licence Conditions Handbook [P-CORR-00531-04886, *Pickering NGS: Licence Conditions Handbook, LCH-PNGS-R005*, November 10, 2016].

CSA N285.0 provides general requirements for pressure-retaining systems, components and supports in CANDU nuclear power plants. It specifies the technical requirements for the design, procurement, fabrication, installation, modification, repair, replacement, testing, examination and inspection of, and other work related to, pressure-retaining systems, components and supports over the service life of a CANDU nuclear power plant.

In Units 1,4, the Feeder piping design has been registered as being in accordance with the ASME B31.1 Power Piping Code. The actual fabrication, testing, material and inspection requirements were in accordance with the historical standard B31.7, *Nuclear Piping for Class 1*. Unit 1 Feeder pipes were subsequently reanalyzed for the large scale Fuel Channel replacement program. The stress analysis was carried out according to the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB-3600 (Class 1 Piping), 1983 edition including winter 1983 addenda.

The Feeder supports that were replaced for the large scale Fuel Channel replacement program were analyzed and redesigned to the requirements of ASME Code Section III, Subsection NF, 1983 edition including winter 1983 addenda. These replacement Feeder supports received an Ontario Special Ruling Registration with the Technical Standards and Safety Authority.

The reactor headers are designed and manufactured to the requirements for Class A pressure vessels of Section III of the ASME Code.

In Units 5-8, the heat transport piping, including the headers and Feeders is exclusively seamless carbon steel ASME SA I06 Grade B. The pipe was designed and constructed to the requirements for Class 1 components in Section Ill of the ASME Code, 1974 Edition without Addenda.

The primary side of the Steam Generators is designed and manufactured to the requirements for Class 1 components in Section Ill of the ASME Code.

In Units 1,4 and Units 5-8, the pressure tubes are classified as Class 1 nuclear components. The zirconium-niobium alloy is not an ASME Code material but the design stresses are based on the factors of safety specified by ASME Section Ill. The wall thickness calculations were conducted using the methods set out in ASME Section Ill and a corrosion and wear allowance of 0.20 mm (0.008 in.) was provided. Zirconium alloys are covered by CSA standard CAN/CSA N285.6. The cold-worked Zr-2.5 Nb alloy was therefore fabricated and tested to meet the requirements of AECL Specification [TS-XX-31110-5, Rev. 2, *Cold Worked Zirconium – 2.5 wt% Niobium Extruded and Drawn Pressure Tubes*, May 1976] and surface conditioned and stress relieved to AECL Specifications [TS-XX-31110-4, Rev. 05, *Cleaning and Stress Relief of Cold Worked Zirconium – 2.5 wt% Niobium Pressure Tubes*, March 29, 1979] or TS-XX-31110-8.

Knowledge and experience gained over many reactor-years of operation and research and development have demonstrated that pressure tube failure will be preceded by a period of leakage (leak-before-break). The Annulus Gas System has provisions for monitoring the moisture content of the annulus gas, which can provide an early indication of a pressure tube leak and allow for reactor shutdown. In Units 1,4, the reliability of the Annulus Gas System beetles was improved by relocating them to an accessible area to facilitate increased testing frequency. Also, a drains tank was added to monitor the collection rate of water condensed by the cooling coil and to alarm at a collection rate of 2 kg/h. Units 5-8 also have accessible Annulus Gas System beetles and a drains tank that is manually monitored.

Protection against overpressure for the Heat Transport System circuit is provided by the combined effects of pressure relief valves and the reactor Shutdown System. Two instrumented control valves in each loop function as pressure relief valves. Overpressure protection for the Heat Transport System is provided by four instrumented pressure relief valves (two connected to the north loop and two to the south loop) which relieve to the bleed condenser. While two valves are sufficient to relieve the pressure, 200 per cent relief capacity has been provided.

Life Cycle Management Plans described in the reviews of Safety Principles D-186, Inspectability of Safety Equipment, and O-305, Maintenance, Testing and Inspection, ensure ongoing compliance of the components relevant to Reactor Coolant System integrity with the design codes and standards.

In conclusion, the Pickering NGS Heat Transport System is designed to avoid pressure boundary failure through high-quality design and maintenance, provision of adequate pressure relief capability and comprehensive leak detection capability. Confidence in the integrity of the pressure boundary is supported through appropriate assessments and research and development.

D-240 – New and Spent Fuel Storage

Principle: Plant designs provide for the handling and storage of new and spent fuel in such a way as to ensure protection of workers and to prevent the release of radioactive material.

A lattice of natural uranium and light water cannot be made critical in any configuration. Hence, no criticality problem exists in handling new Fuel or in the IFB of CANDU reactors.

The degree of automation of the fuel handling design in Pickering NGS allows fuel handling to be performed remotely from the operators and contributes significantly to the protection of workers. The reactors are fuelled on-power. Each reactor is serviced by two remotely controlled fuelling machines, one at each reactor face, which, operating at opposite ends of the same Fuel Channel, match Primary Heat Transport System pressure, open the channel and insert fresh Fuel or remove irradiated Fuel, without interrupting reactor operation. Irradiated Fuel is transferred by an elevator and conveyor to the primary IFB where it remains for a minimum of four years before transfer to the Auxiliary IFB. The combined storage capacity of the two bays is designed to accommodate the accumulation of irradiated Fuel.

After an adequate cooling period in the Auxiliary IFB, irradiated Fuel is loaded into dry storage containers, which are transferred for processing and storage to the Pickering Waste Management Facility. The Pickering Waste Management Facility provides sufficient capacity to store the used Fuel generated from the Pickering reactors until the end of the station's service life. The Pickering Waste Management Facility is used for processing and storage of dry storage containers and storage of retube components from Units 1-4 in dry storage modules.

Storage and handling facilities are provided to accommodate bulk storage of new Fuel in the east annex, safe transfer of Fuel to the Reactor Building, inspection of the Fuel in the new Fuel loading areas and easy manual loading of new Fuel bundles into the new Fuel magazines for transfer to the fuelling machines. Suitable handling facilities are provided to avoid damage to the Fuel bundles during manual loading. The new fuel loading area is shielded from those areas likely to contain irradiated Fuel. Personnel are adequately protected from radiation exposure by remote handling of irradiated Fuel in the transfer room and by underwater transport and handling of the irradiated Fuel on the conveyor and in the storage bays. Ventilation and filtration are provided to remove fission products should Fuel be inadvertently damaged during handling.

The storage arrangement with irradiated natural uranium Fuel in light water is well subcritical.

Reliable cooling of the irradiated Fuel in the transfer mechanism magazine is provided by two circulating pumps, one on standby plus an emergency supply from the bay water system. In the event that the elevator carriage should stop in air between its end positions while carrying irradiated Fuel, a cooling water spray system can be activated to prevent overheating of the Fuel. The water spray is supplied from the storage bay systems.

To summarize, Pickering NGS fuel handling has been conducted safely over several decades. Provisions are in place for safe handling of the Fuel from its arrival at the plant through to the IFBs and dry storage.

D.7. Safety Principles Related to Levels 3, 4

The following safety principles are related to defence-in-depth Levels 3, 4:

- D-200 – Automatic Shutdown Systems
- D-207 – Emergency Heat Removal
- D-217 – Confinement of Radioactive Material
- D-221 – Protection of Confinement Structure
- D-233 – Station Blackout

As demonstrated below, Pickering NGS design and operation are aligned with all safety principles related to Levels 3, 4.

D-200 – Automatic Shutdown Systems

Principle: Rapidly responding and highly reliable reactivity reduction for safety purposes is designed to be independent of the equipment and processes used to control the reactor power. Safety shutdown action is available at all times when steps to achieve a self-sustaining chain reaction are being intentionally taken or whenever a chain reaction might be initiated accidentally.

The Shutdown Systems are described in Section 6.1.8 of Part 2 of the Units 1-4 Safety Report [NA44-SR-01320-00001-R016, *Pickering A Safety Report*, July 20, 2017] and in Sections 6.2 and 6.3 of Part 2 of the Units 5-8 Safety Report [NK30-SR-01320-00002 R004, *Pickering B Safety Report – Part 2*, October 15, 2012]. Each system has its own initiation sensors, detectors and logic to ensure functional and physical diversity. The Shutdown Systems are independent of the process systems, such that a process system impairment will have minimal (if any) deleterious impact upon the effective functioning of a Special Safety System.

When Units 1,4 began operation in the early 1970s, its design included a single Shutdown System, which employed two diverse means of inserting negative reactivity into the core:

1. Dropping shutoff rods
2. Dumping the Moderator from the Calandria

To reduce the risk of an uncontrolled reactivity increase event to an extremely low level, CANDU designs subsequent to Pickering NGS Units 1-4 (including Pickering NGS Units 5-8) have incorporated two independent, physically separated and diverse Shutdown Systems. The original Units 1,4 design Shutdown System reliability has been enhanced by retrofitting a Shutdown System Enhancement (SDSE) that provides sensors and trip logic independent from those of the original Shutdown System (SDSA). The additional SDSE trip parameters are heat transport high/low pressure, neutron overpower, high neutron log rate and manual trip. In addition to initiating the shutoff rod drop, SDSA and SDSE are both augmented by the Moderator dump feature (see above). The design of SDSA and SDSE is such that they are independent of each other from trip sensing to the final relay contacts in the shut-off rod drop logic and the Moderator dump logic (i.e., they do not include the shut-off rod clutch mechanisms or the Moderator dump valves). The adequacy of the Shutdown System design in limiting the risk of an uncontrolled reactivity increase is demonstrated by the Level 1 at-power PSA results in OPG Report [P-REP-03611-00006-R00, *Pickering NGS PSA Update to Include Enhancements from the Fukushima Integrated Action Plan*, April 30, 2014] which show that event sequences involving the failure to shut down are a very small contributor to the severe Core Damage Frequency.

In Units 5-8, the independent Shutdown Systems are SDS1 (shutoff rods) and SDS2 (poison injection into the Moderator). Both Shutdown Systems are capable of shutting the reactor down fast enough for all AOOs and DBAs such that specified limits are not exceeded. There is no post-trip recriticality following accidents, including the limiting postulated event of an in-core LOCA if the ECI System operates as designed. For SDS1, operator action can be credited 15 minutes after an in-core LOCA to augment the depth of shutdown, for example to ensure that re-criticality does not occur in the unlikely event that ECI does not operate. For SDS2, the shutdown depth is sufficient to keep the reactor shut down indefinitely for even the most reactive conditions of the core.

For Units 1,4, on SDSA and for Units 5-8 on SDS1, each trip channel can be connected to a channel of start-up instrumentation which measures neutron flux at very low power levels. The Pickering NGS startup instrumentation is described in the review of Safety Principle D-205, Startup, Shutdown, and Low Power Operation.

The automatic Shutdown Systems are also described in Safety Principle D-192, Protection Against Power Transient Accidents.

In conclusion, the automatic Shutdown Systems in Pickering NGS are independent of the Reactor Control Systems and are available during the approach to critical, when the reactor is critical at low power, and when the reactor is operating at power.

D-207 – Emergency Heat Removal

Principle: Provision is made for alternative means to restore and maintain fuel cooling under accident conditions, even if normal heat removal fails or the integrity of the primary cooling system boundary is lost.

The ECI System is designed to refill the Heat Transport System following a LOCA and to provide coolant makeup and the heat sink in the long term via the ECI heat exchangers. In both Units 1,4 and Units 5-8, the ECI System has two principal operating modes:

(a) The high pressure injection mode is used immediately after the accident up to the time the ECI water storage tank level is low. Water flows from the ECI storage tank through the high pressure injection pumps to the ECI header. It then enters the reactor headers via the injection lines and the shutdown cooling loops and eventually discharges from the break and flows to the fuelling machine service room sumps. If all high pressure injection pumps are unavailable, the storage tank will feed the injection header by gravity.

(b) The long term recovery mode is used after the initial injection mode for as long as recovery is required. During ECI recovery, water from the fueling machine service room(s) sump(s) (recovery sump in Units 5-8) is used to refill the Heat Transport System using the Moderator pumps for Units 1,4 and ECI Recovery Pumps for Units 5-8.

For events involving a loss of Steam Generator inventory (steam line break or loss of feedwater), qualified backup boiler water supplies are available to remove the decay heat generated in the Fuel. In Units 1,4, the Boiler Emergency Cooling System is designed to provide a short term heat sink for reactor decay heat removal via Steam Generators in order to allow the operator sufficient time to bring in a long term heat sink for accident situations initiated by or leading to failure of the normal feedwater supply. The Boiler Emergency Cooling System is a non-nuclear system qualified to Design Basis Earthquake Category B. The Boiler Emergency Cooling System injection valves will open automatically. The water from the system's water storage tanks is supplied to the boilers via the reheater drains return piping. Normally the water level in TK1/TK2 is maintained at approximately 2.2 m, which should provide sufficient water inventory for at least 15 minutes of heat sink [Section 5.1.8 of NA44-SR-01320-00001-R016, *Pickering A Safety Report*, July 20, 2017].

The Emergency Boiler Water Supply System is designed to provide a long-term supply of emergency water to the boilers of Units 1,4 following a postulated main steamline failure within the Units 1,4 powerhouse. It is also capable of providing water to the boilers for a total loss of feedwater event, following operator action to depressurize the boilers. A steam line failure could result in a loss of service water via failures of non-environmentally qualified Class III and IV electrical distribution equipment exposed to the resultant powerhouse environment. The emergency water is supplied from the discharge headers of the Units 6 and 7 High Pressure Service Water System.

In Units 5-8, the Boiler Emergency Cooling System provides water for reactor decay heat removal by the boilers for accident situations initiated by or leading to failure of the normal feedwater supply. The Boiler Emergency Cooling System injection valves are signaled to open automatically and injection will continue until the inventory stored in the water tanks has been depleted. Boiler Emergency Cooling System operation ensures that at least an additional 15 minutes of heat sink capability is available. In addition, the Emergency Water System is a seismically qualified system for supplying long-term emergency makeup to the boilers, Heat Transport System, ECI recovery heat exchangers, Containment air coolers and the Calandria. The main Emergency Water System pumps are supplied by lake water. The water is distributed through an independent, seismically qualified piping system. The Emergency Water System is manually operated and is common to Units 5, 6, 7 and 8. Each unit has a manually operated valve station located next to the Unit Emergency Control Centre. The Emergency Water System is powered from the seismically qualified Emergency Power System.

In Pickering NGS Units 1,4 and 5-8, Emergency Mitigating Equipment has been provided for an additional makeup water supply. Portable diesel-powered pumps can be deployed to provide make-up to the secondary side of the boilers, to the Heat Transport System and to the Moderator.

In summary, both Units 1,4 and Units 5-8 are equipped with systems that are designed to effectively remove heat from the Fuel and transfer it to the ultimate heat sink under accident conditions. The ECI System is capable of restoring and maintaining fuel cooling following a LOCA.

D-217 – Confinement of Radioactive Material

Principle: The plant is designed to be capable of retaining the bulk of the radioactive material that might be released from fuel, for the entire range of accidents considered in the design.

The Containment System is basically an envelope around the nuclear components of the reactor Primary Heat Transport System with provision for controlling the pressure within the envelope. The envelope is designed to limit any release of radioactivity to the environment which might result from a process or system failure. Because of the large amounts of energy stored in the reactor coolant, an energy absorbing system is provided which allows the Containment structure to withstand the initial pressure rise resulting from escaping fluid flashing to steam.

The principle of Negative Pressure Containment is employed in the Pickering NGS units. A Vacuum Building, maintained at a very low subatmospheric pressure, is linked to the Reactor Buildings by a pressure relief system. This arrangement ensures that the pressure within the Containment boundary will be brought below the surrounding atmospheric pressure within a short time following a pressure rise within a Reactor Building. Long-term control of Containment pressure is accomplished through controlled venting via the Filtered Air Discharge System.

The principal components of the Containment envelope are the Reactor Buildings, the Pressure Relief Duct, the vacuum ducts, the Pressure Relief Valves and the Vacuum Building. The Vacuum Building contains a water spray system consisting of an elevated water storage tank and spray headers designed to provide a dousing spray of water through the main chamber of the Vacuum Building to cool the air and condense any steam present. A Pressure Relief Duct with pressure relief valves interconnects all the Reactor Buildings and the Vacuum Building via the vacuum ducts. Pressure relief panels on Units 1 to 8 are provided to allow for flow of air and steam from individual Reactor Buildings to the Pressure Relief Duct. Pressure relief valves, located in the Pressure Relief Duct, normally isolate the duct from the Vacuum Building and relieve to the Vacuum Building via vacuum ducts when activated by high pressure.

The Vacuum Building, the pressure duct and valves are common to Units 1,4 and Units 5-8, with the Pressure Relief Duct interconnecting all eight Reactor Buildings. Isolation of Containment is initiated automatically if the differential pressure between any Reactor Building and the reactor auxiliary bay reaches the isolation setpoint, or if high activity is detected in a Reactor Building ventilation system exhaust, since these conditions indicate possibility of a radioactive release inside that Reactor Building.

The Emergency Water Storage Tank provides water for the dousing spray in the Vacuum Building, cooling to the bleed cooler and backup fire water to Units 1, 2, 3 and 4.

The structural integrity of the reactor buildings and Pressure Relief Duct is verified by operational tests in which an internal pressure is applied. The Containment operational leakage rate target is based on 1 per cent of the contained air mass per hour at an internal pressure equal to the design value of 41.4 kPa(g) (6.0 psig).

The purpose of the Filtered Air Discharge System is to maintain Containment sufficiently sub-atmospheric following an accident and to provide a filtered and monitored pathway to the environment. The system serves both Units 1,4 and Units 5-8. Using the Filtered Air Discharge System, Containment can be maintained at an adequate negative pressure with respect to the atmosphere after depletion of the vacuum in the Vacuum Building.

The purpose of the post-LOCA Hydrogen Ignition System is to safely combine any hydrogen (or deuterium) gases generated in Containment following accidents. The system is capable of addressing very low concentrations of hydrogen in air/steam mixtures, thus preventing the occurrence of high concentrations of hydrogen in Containment. In addition, Passive Auto-catalytic Recombiners have been installed to control Containment hydrogen levels, providing a backup to the Hydrogen Ignition System.

The effectiveness of the Containment System in limiting the release of radioactivity following an accident (single process failure or dual failure of the ECI System) such that the applicable single and dual failure dose limits are met is demonstrated by the analyses documented in Part 3 of the Pickering NGS Safety Reports.

In conclusion, the Pickering NGS Containment System is designed to retain the radioactive material that might be released from the Fuel during an accident, and Containment effectiveness is demonstrated by safety analyses showing that public dose meets the applicable regulatory limits for the full range of accidents considered in the design.

D-221 – Protection of Confinement Structure

Principle: If specific and inherent features of a nuclear power plant would not prevent detrimental effects on the confinement structure in a severe accident, special protection against the effects of such accidents is provided, to the extent needed to meet the general safety objective.

Specific features of the Pickering NGS design that contribute significantly to preventing Containment damage in a severe accident include:

- The very low frequency of uncontrolled reactivity events due to the highly reliable fast shutdown capability.
- The capability of the passive Moderator heat sink together with the Moderator Cooling System to prevent severe core damage.

The possibility that hydrogen generated during certain accidents could potentially lead to a challenge to Containment integrity was recognized in the CANDU industry well before the Three Mile Island accident. There are two systems for hydrogen control in Pickering NGS:

- Hydrogen Ignition System
- Passive Auto-catalytic Recombiners

The Hydrogen Ignition System in Pickering NGS is designed to safely combine any hydrogen (or deuterium) gases generated in Containment following accidents. Passive Auto-catalytic Recombiners have also been installed to control Containment hydrogen levels, providing a backup to the Hydrogen Ignition System.

Four coolers are provided for each of the two fuelling machine vaults. The cooling water flow through the coolers is increased automatically on detection of high ambient pressure. The licensing requirement is to have at least one fan available per vault to ensure adequate mixing of any hydrogen (or deuterium) gases generated within Containment following an accident, thereby avoiding the buildup of potentially high concentration pockets of gases.

Emergency Mitigating Equipment for design extension conditions has been installed as part of the Fukushima Action Plan, as described in Sections 18.2.2.1 and 18.2.2.2.

As is described in the review of Safety Principle AM-326, Engineered Features for Accident Management, implementation of Phase 2 Emergency Mitigating Equipment is in progress and will provide additional enhancements for the control of Containment pressure following BDBAs (see GI-40-RS1 in Appendix F).

The review of defence-in-depth relating to confinement of radioactive material for accidents within the design basis is covered in Safety Principle D-217, Confinement of Radioactive Material.

To summarize, provisions to minimize the severity of the challenge to Containment integrity resulting from a severe accident have been deployed in Pickering NGS. The Pickering Large Release Frequency Safety Goal is met as is discussed in Safety Principle AM-326, Engineered Features for Accident Management.

D-233 – Station Blackout

Principle: Nuclear plants are so designed that the simultaneous loss of onsite and offsite AC electrical power (a station blackout) will not soon lead to fuel damage. The use of ‘simultaneous’ is not intended to imply that the loss of onsite and offsite power necessarily occurs at the same time.

Features of the Pickering NGS contribute to significant delays in Fuel damage following a station blackout, without crediting operator action. Following a station blackout, a number of reactor trip signals would be generated within seconds, resulting in an immediate reduction of reactor power to decay power level. With forced cooling unavailable, the safety relief valves would open to discharge steam from the boilers and act as the heat sink for the reactor. Heat transfer from the core to the boilers occurs by buoyancy-driven flow. The boilers are designed with sufficient inventory to provide fuel cooling and prevent fuel overheating for a minimum of 15 minutes at decay power levels.

In reality, significant fuel overheating would be delayed well beyond 15 minutes due to the large inventory of water in the boilers. It has been estimated that the secondary side inventory would provide a minimum of 1 hour of cooling [N-CORR-03611-0348939-LOF, *Fukushima – Daiichi Follow – up: Summary of Timing Requirements for Maintaining and/or Restoring Fuel Cooling after Total Loss of Heat Sink Events*, April 6, 2012]. In addition, gravity feed from the Boiler Emergency Cooling System tanks (see Safety Principle D-207, Emergency Heat Removal) is estimated to provide an additional 2.5 hours of cooling. Finally, gravity feed from the elevated deaerator storage tank would provide an additional 5.5 hours of cooling. Therefore, at least

9 hours are available before the need for operator action to connect temporary on-site services (e.g., Emergency Mitigating Equipment). In the absence of such operator action, subsequent pressurization of the Heat Transport System would result in discharge of steam through the liquid relief valves. As Fuel Channels void and heat up, heat begins to be transferred to the Moderator. Pressure tube overheating at high Heat Transport System pressure eventually results in failure of one or a few Fuel Channels, which depressurizes the Heat Transport System. If fuel cooing is not restored, continued fuel heatup at low Heat Transport System pressure will result in pressure tubes sagging into contact with their Calandria tube, creating a path for heat removal to the Moderator for Fuel Channels below the Moderator level. The Moderator continues to remove decay heat and will heat up and boil if Moderator cooling is not provided. Eventually, after the top few rows of channels are exposed to steam cooling in the Calandria, failure of these channels will commence. However, this is delayed for several hours after the deaerator storage tank becomes drained.

As summarized in the Risk Assessment Summary Report [P-REP-03611-00006-R00, *Pickering NGS PSA Update to Include Enhancements from the Fukushima Integrated Action Plan*, April 30, 2014] station blackout is included in the events covered by the Level 2 PSA performed for Pickering NGS. In Units 5-8, a main steam line break, combined with failures causing station blackout, leading to a loss of heat sink and failure of ECI and Moderator cooling at four units simultaneously is selected as one of the representative accident sequences. Similarly, in Units 1,4, a total loss of heat sinks in all six operating Pickering NGS units is selected as a representative accident sequence.

Emergency Mitigating Equipment provided as part of the implementation of lessons learned from the 2011 Fukushima accident can be deployed to prevent significant Fuel damage in the longer term (> 8 hours) following a station blackout.

In conclusion, Fuel damage following a postulated station blackout is delayed for several hours in the Pickering NGS design, due to the large volume of coolant available for heat removal. In addition, deployment of Emergency Mitigating Equipment will significantly extend the time to Fuel damage or prevent it altogether.

D.8. Safety Principles Related to Levels 4, 5

The following safety principles are related to defence-in-depth levels 4, 5:

- EP-333 – Emergency Plans
- EP-336 – Emergency Response Facilities

As demonstrated below, Pickering NGS design and operation are aligned with all safety principles related to Levels 4, 5.

EP-333 – Emergency Plans

Principle: Emergency plans are prepared before the startup of the plant, and are exercised periodically to ensure that protection measures can be implemented in the event of an accident which results in, or has the potential for, significant releases of radioactive materials within and beyond the site boundary. Emergency planning zones defined around the plant allow for the use of a graded response.

OPG Program [N-PROG-RA-0001-R015, *Consolidated Nuclear Emergency Plan*, December 22, 2016] provides a written basis to document concepts, roles and resources required by OPG Nuclear to implement and maintain its emergency response capability to protect the public, employees and the environment in the event of a nuclear emergency. N-PROG-RA-0001-R015 defines a nuclear emergency as an emergency which poses an actual or potential hazard to public health and property or the environment from ionizing radiation, whose source is a major nuclear installation.

The emergency response capability established in accordance with N-PROG-RA-0001-R015 must be sufficiently flexible to be used for a broad range of events and disasters both within and beyond the design basis. That is, the full range of accidents and radiation emergencies spans a wide range of events from an impairment of a plant system to the occurrence of a BDBA that progresses into a severe accident.

The basis for emergency planning is described in Section 1.1 of the N-PROG-RA-0001-R015. Design basis documents for both on-site and off-site planning are listed in Section 4.2.2 of N-PROG-RA-0001-R015; this list represents the primary source from which the Emergency Plan is developed. Two of the documents listed in this section are the Pickering NGS Units 1-4 and Pickering NGS Units 5-8 Safety Reports, which identify and study the full range of accidents and radiation emergencies that are a part of the station design basis. N-PROG-RA-0001-R015 also recognizes the need to address BDBAs, including design extension conditions and severe accidents. Requirements for OPG's approach to BDBA management are documented in OPG Standard [N-STD-MP-0019-R002, *Beyond Design Basis Accident Management*, August 19, 2016], with BDBA strategies implemented through operating procedures, Emergency Mitigating Equipment Guidelines and Severe Accident Management Guidelines.

Elements and procedures include:

- Review issues that could impact emergency response, including local development which may modify/increase risks.
- The OPEX process monitors events around the world to determine if there is any unforeseen event that may have applicability to Pickering NGS. If an event is deemed to be applicable, the emergency response process is reviewed to ensure the event can be adequately dealt with.

OPG Procedure [N-PROC-RA-0040-R006, *Maintenance and Testing of Emergency Preparedness Facilities and Equipment*, April 30, 2015] defines the process used to monitor, periodically test and maintain the emergency response facility and equipment to ensure operability 24 hours a day, 7 days a week. This includes testing and facility walk-through frequencies and covers the different types of equipment, such as faxes, computers, radiation instruments, communication, meteorological and data transmitting equipment [N-PROC-RA-0133-R000, *Management of Equipment Important to Emergency Response*, December 18, 2014].

As noted in Section 1.3.4 of the Consolidated Nuclear Emergency Plan, the Provincial Nuclear Emergency Response Plan requires Pickering NGS to procure adequate quantities of stable iodine tablets for their Primary Zone population. Designated Primary Zone municipalities are also required to establish and maintain a public alerting system in accordance with the Provincial Nuclear Emergency Response Plan. In consultation with the designated municipalities, the OPG Emergency Preparedness Department procures stable iodine tablets and maintains them within expiry dates. Iodine tablets were pre-distributed to the Primary Zone population for Pickering NGS in 2015. OPG also supports the province and municipalities that provide coordinated emergency communications under the jurisdiction of the Provincial Nuclear Emergency Response Plan. Per Section 1.3.5 of the Consolidated Nuclear Emergency Plan, OPG provides resources and assistance to the designated Primary Zone municipalities to enable them to establish and maintain a public alerting system as required by the Provincial Nuclear Emergency Response Plan.

The adequacy of the response and mitigation strategies that have been developed is demonstrated primarily through drills and exercises. The Emergency Preparedness Department assesses the Emergency Response Organization performance to the established objectives identified in OPG Instruction [N-INS-03490-10002-R005, *Conduct of Emergency Preparedness Drills and Exercises*, December 31, 2014] and reviews all drill and exercise related corrective actions to monitor status and ensure completeness. Integrated and partial emergency exercises have been conducted at Pickering NGS to check satisfactory function of the emergency organization and its equipment.

It is concluded that emergency plans have been prepared under the Consolidated Nuclear Emergency Plan, and emergency preparedness drills are mandated and carried out for Pickering NGS. The emergency plans provide for enhanced response in the Primary Zone.

EP-336 – Emergency Response Facilities

Principle: A permanently equipped emergency centre is available off the site for emergency response. On the site, a similar centre is provided for directing emergency activities within the plant and communicating with the off-site emergency.

In the event of an emergency, there are permanent on-site and off-site facilities appropriately equipped for effective emergency response. During the initial emergency phase, Main Control Room staff perform the assessment of plant status. Where possible, the Main Control Room staff also identify damage to plant equipment. Appropriate plant procedures are used and Main Control Room staff initiate an immediate operations response to strive towards taking the plant to a safe and stable configuration.

The Main Control Room team utilizes resources of the on-site Emergency Operations Centre to mobilize and deploy the necessary emergency teams. The Emergency Operations Centre personnel may, if necessary, mobilize, brief and deploy Emergency Response Teams to address specific emergencies, such as a fire, medical emergency, or a search and rescue operation. Appropriate off-site support groups (i.e., non-OPG resources) such as fire and/or ambulance may be activated to respond to the site. In addition, the Emergency Operations Centre personnel may mobilize and deploy both in-plant and off-site radiation survey teams who provide additional data to assess the plant status. If the Main Control Room emergency mitigation strategy requires repair of damaged equipment, then the Emergency Operations Centre personnel assemble, brief and deploy the appropriate emergency repair teams. Main Control Room staff seek, as appropriate, consultative, technical and resource assistance from the Site Management Centre who in turn seek, as appropriate, assistance from the Corporate Emergency Operations Facility.

A permanently equipped Site Management Centre is maintained for directing emergency response. The Corporate Emergency Operations Facility, located in Whitby, is also permanently equipped to support and direct emergency response. The response is managed as detailed in [N-PROG-RA-0001 R015, *Consolidated Nuclear Emergency Plan*, December 22, 2016].

The Provincial Emergency Operations Centre, located at the Ministry of Community Safety and Correctional Services in Toronto, is the provincial facility and organization that directs off-site emergency response operations. OPG provides call-in staff to fill an official liaison officer position in the Provincial Emergency Operations Centre Operations Section, and provides and maintains software codes for dose projection, dedicated telecommunications links, and training and drills as requested to support the Provincial Emergency Operations Centre. During the response stage, site shift and management staff shall make emergency notification to the Provincial Emergency Operations Centre as appropriate. Technical data and situation updates, including off-site survey data, are provided to the Provincial Emergency Operations Centre from the incident site.

The Durham Regional Emergency Operations Centre is comprised of elected officials, regional department staff (e.g., roads and works, emergency planning, social services), school board staff, police, fire and medical representatives. The primary responsibility of this organization is to implement the public protective action directives in the locally affected area to protect the public. Only the Provincial Emergency Operations Centre shall provide direction and information on off-site response to the Regional Emergency Operations Centre. In the event of an emergency, OPG will provide a local site representative to fulfill the OPG liaison officer position at the Regional Emergency Operations Centre. The liaison officer will provide coordination with the site for resources and off-site response activities in the local area. The liaison officer will also provide radiation level interpretation and technical background information to the regional staff. Official emergency communication with the Regional Emergency Operations Centre is through the liaison officer and the Site Management Centre.

Additional off-site emergency response facilities include:

- Durham Region Reception Centres and Emergency Worker Centres. Reception Centres are established to provide, among other functions, a Monitoring and Decontamination

Unit for evacuees. These centres are under the control and direction of the Regional Emergency Operations Centre. OPG Nuclear site organization and call-in personnel staff and equip the Monitoring and Decontamination Unit function of these centres. Monitoring and Decontamination Unit staff are trained to carry out radiation monitoring and decontamination. OPG Monitoring and Decontamination Unit supervisors maintain communication with the Site Management Centre to keep the Emergency Response Organization apprised of the Monitoring and Decontamination Unit status and any need for further resources. Emergency Worker Centres are established during nuclear emergencies to monitor and control radiation exposure of external emergency workers who may be required to enter areas affected by radiation. Similar to Reception Centres, Emergency Worker Centres are under the authority of the Regional Emergency Operations Centre. Emergency Worker Centres are staffed by OPG call-in personnel with equivalent qualifications as Reception Centre Monitoring and Decontamination Unit staff. Communication is regularly maintained between the Emergency Worker Centre OPG supervisors and the Site Management Centre and Regional Emergency Operations Centre.

- City of Toronto, Municipal Emergency Operations Centre. A Municipal Emergency Operations Centre is equivalent to a Regional Emergency Operations Centre with regards to its role in the emergency response. OPG provides the same level of support to the City of Toronto Municipal Emergency Operations Centre as it does to the Durham Regional Emergency Operations Centre.
- Reception Centres and Emergency Worker Centres associated with the City of Toronto have the same functionality as corresponding centres for the Durham Region. OPG provides the same level of support to the City of Toronto Reception Centres and Emergency Worker Centres as it does for corresponding centres in the Durham Region.

It is concluded that suitably equipped on-site and off-site emergency centres are available to carry out emergency response activities.

D.9. Safety Principles Related to Level 1

The following safety principles are related to defence-in-depth Level 1:

- S-136 – External Factors Affecting the Plant
- D-188 – Radiation Protection in Design
- O-272 – Conduct of Operations
- O-288 – Normal Operating Procedures

As demonstrated below, Pickering NGS design and operation are aligned with all safety principles related to Level 1.

S-136 – External Factors Affecting the Plant

Principle: The choice of site takes into account the results of investigations of local factors that could adversely affect the safety of the plant.

The site characteristics are described in the Pickering NGS Safety Reports Part 1. Descriptions and data are provided for:

- Geography and demography (site location, site description, access, population)
- Land use (agriculture, industry, fishing, recreation, transportation)
- Geology (regional geology, site geology)
- Seismology (regional seismicity, seismic ground motion, seismic design)
- Hydrology (water treatment plants, lake currents, water levels, water temperatures, thermal plumes, ice, aquatic weeds, ground water)
- Meteorology (temperature, precipitation, wind, atmospheric stability).

A screened list of external hazards for inclusion in the Level 1/Level 2 PSA for Units 1,4 and Units 5-8 was completed as part of the Hazard Screening Analysis for the Pickering PSA update [P-REP-03611-00006-R00, *Pickering NGS PSA Update to Include Enhancements from the Fukushima Integrated Action Plan*, April 30, 2014]. This list is compliant with both CNSC Regulatory Standard S-294 and international practice (such as US Nuclear Regulatory Commission NUREG/CR-2300). The External Hazards Screenings contained in these reports rely on the concept of Review Level Conditions for external hazards against which the station's current design is reviewed. Originally defined and analyzed for various external events in the hazard screening analyses, the Review Level Conditions were further analyzed and refined for tornado, seismic and external flooding hazards. Any changes in site characteristics with an impact on nuclear safety are captured in Part 1 of the Safety Reports during the Safety Report updates. The impacts of changes in the site characteristics are then captured in the development of models and methodology for internal and external hazard analyses.

To summarize, the local characteristics were considered in the original design of Pickering NGS, a review of potential external hazards has recently been completed and appropriate hazard assessments have been conducted.

D-188 – Radiation Protection in Design

Principle: At the design stage, radiation protection features are incorporated to protect plant personnel from radiation exposure and to keep emissions of radioactive effluents within prescribed limits.

The initial design of Pickering NGS was created to ensure that the layout and operation of facility SSCs and processes were consistent with the established radiation protection guidelines and contribute to maintaining occupational radiation exposures ALARA. During the initial design stage, emphasis was placed on the reduction of occupational radiation doses, which resulted in design improvements that allowed Pickering NGS and other early CANDU designs to have far fewer workers exposed to measurable radiation doses than most other utilities, despite increased staffing needs associated with online fueling, as documented in OPG Report [N-REP-N7250014, *Excellence Through Radiation Protection Practices in Ontario Hydro CANDU-PHW Nuclear Electric Generating Stations*, September 1987].

As indicated in Section 12 of the Pickering 1,4 and 5-8 Safety Reports [NA44-SR-01320-00001-R015, *Pickering A Safety Report*, July 24, 2012, NK30-SR-01320-00002-R004, *Pickering B Safety Report – Part 2*, October 10, 2012] and N-PROG-RA-0013-R010, *Radiation Protection*, February 21, 2017, specific design features at Pickering NGS to control radiation dose include the use of shielding, ventilation and emissions control, radiological zoning and the provision of area radiation monitoring equipment. The use of shielding and ventilation control reduces exposure to external radiation and airborne radioactive material, respectively. The use of emissions control reduces the doses to members of the public. The use of radiological zones limits the spread of contamination by requiring personnel and materials crossing zone boundaries to be monitored for radioactivity, depending on the direction of travel. The use of area radiation monitoring equipment warns personnel of any sudden or unexpected changes in radiological hazards.

N-PROG-RA-0013-R010, *Radiation Protection* further mandates that “when making engineering changes, engineers maintain or improve upon designs that reduce occupational exposures”. At Pickering NGS this process is implemented through OPG Standard [N-STD-RA-0018-R007, *Controlling Exposure As Low As Reasonably Achievable*, November 27, 2014].

At Pickering NGS, facilities are provided for safe temporary storage of liquid and solid wastes, as well as for safe handling of all radioactive gaseous, liquid and solid wastes. All equipment, tanks and facilities for handling liquid and solid wastes are flexible enough to cope with the anticipated increase in waste volume and activity during periods of major maintenance work or adverse reactor operation. Gaseous wastes are monitored, managed and filtered as appropriate to maintain operation within acceptable limits.

The Radioactive Liquid Waste Management System is common to Units 1,4 and Units 5-8. This system receives and selectively treats the waste streams directed to it by the normally inactive and normally active drainage systems, as described in the Safety Report, Part 2.

The system is designed such that on an annual basis the sum of the fractional contributions from all radionuclide groups from Units 1,4 and Units 5-8 will not exceed the station Action Levels. The Action Levels are set at 10% of the Derived Release Limit for a particular radionuclide/radionuclide group for both airborne and liquid emissions. Additionally, for liquid effluents, the derived liquid concentration is used to limit the concentration in the outfall.

Release of tritium from the station is kept within limits by containment of heavy water within the Heat Transport and Moderator Systems where the tritium is produced, by containment of heavy water vapour within the Reactor Building, and by recovery of any leakage from the heavy water circuits.

In conclusion, design features have been included in the Pickering NGS design for minimizing radiation exposure to the plant personnel and limiting releases of radioactivity to the environment.

O-272 – Conduct of Operations

Principle: Operation of the plant is conducted by authorized personnel, according to strict administrative controls and observing procedural discipline.

OPG has a well-defined organizational structure and strong lines of authority. OPG Program [N-PROG-OP-0001-R008, *Nuclear Operations*, December 4, 2015] establishes safe, uniform and efficient operating practices and processes within OPG Nuclear facilities.

The OPG Charter [N-CHAR-AS-0002-R019, *Nuclear Management System*, November 1, 2016] describes the OPG Nuclear Quality Program, while OPG Standard [N-STD-AS-0020-R014, *Nuclear Management Systems Organizations*, May 19, 2016] outlines its implementation. N-STD-AS-0020-R014 establishes the lines of authority and definition of duties. It provides a summary of the interfacing organizations that own programs supporting the Nuclear Management System. Operation and maintenance of the station as per regulatory requirements and Nuclear standards for public safety are directed by the Director, Operations and Maintenance, who also coordinates with the centre-led organizations (see Safety Principle O-265, Organization, Responsibilities and Staffing) to effectively use resources to achieve performance targets. The quality and quantity of services provided by these centre-led support organizations is monitored by the site Senior Vice President, who holds responsibility for establishing site requirements and priorities. The position specific role document, N-MAN-08131 series, describes the duties, authorities and accountabilities of the positions described in the standard.

Nuclear safety oversight is established to ensure that the requirements of OPG Policy [N-POL-0001-R003, *Nuclear Safety Policy*, April 7, 2014] and N-CHAR-AS-0002-R014 are implemented throughout OPG Nuclear. The framework, accountabilities for nuclear safety oversight, as well as the external and internal processes used for oversight and assessment of nuclear safety are summarized in OPG Standard [N-STD-AS-0023-R008, *Nuclear Safety Oversight*, September 15, 2015].

Significant operational decision making is conducted as per [N-STD-OP-0036 R009, *Operational Decision Making*, April 17, 2015].

[N-GUID-03611-10005-R004, *Integrated Risk Management Guidelines*, May 4, 2015] establishes the administrative controls, responsibilities, duties for direction, control, conduct and oversight of risk-significant activities (Nuclear Safety, Conventional Safety, Environmental, Generation and Radiological), infrequently performed tests or evolutions and infrequent maintenance activities at OPG Nuclear.

Pickering NGS processes for managing, implementing and adhering to operating and working procedures and for maintaining compliance with operational limits and conditions (see Safety

Principle O-284, Operational Limits and Conditions) and regulatory requirements are effective at ensuring plant safety. Compliance with procedural adherence requirements is evaluated in part through the use of self-assessment and benchmarking activities. OPG Procedure [N-PROC-RA-0097-R009, *Self – Assessment and Benchmarking*, May 1, 2017] defines the elements required to plan, execute, report and monitor self-assessments and benchmarking activities for the functional and line organizations of OPG Nuclear. Self-assessment and benchmarking are performance improvement tools that provide a structured method to compare performance with management expectations, industry standards of excellence and regulatory requirements to identify areas needing improvement. Self-assessments include evaluations of business programs, processes and performance.

Issues with regard to specific procedures are reviewed under their respective applicable safety principles.

To summarize, administrative controls are in place to ensure that plant operations are conducted by authorized personnel, according to strict administrative controls and observing procedural discipline.

O-288 – Normal Operating Procedures

In this section, both O-288 and O-290 are addressed, as they constitute a continuum of plant operational states.

O-288 – Normal Operating Procedures
Principle: Normal plant operation is controlled by detailed, validated and formally approved procedures.

O-290 – Emergency Operating Procedures
Principle: Emergency operating procedures are established, documented and approved to provide a basis for suitable operator response to abnormal events.

OPG Program [N-PROG-OP-0001-R008, *Nuclear Operations*, December 4, 2015] establishes safe, uniform and efficient operating practices and processes within OPG Nuclear facilities that provide nuclear professionals the ability to ensure facilities are operated in such a manner that Power Reactor Operating Licences, Operating Policies and Principles and other applicable regulations and standards are followed.

There are operating procedures that apply comprehensively to normal, abnormal and emergency conditions (including AOOs, DBAs, post-accident conditions and design extension conditions).

Pickering NGS operations staff normally operate the plant through the use of approved Operating Manuals, Operating Procedures, Alarm Response Manuals and Safety-Related System Tests. Operating Manuals are produced for individual plant systems and provide instructions on equipment operation and controls.

Operations staff will initially attempt to respond to an abnormal condition via the system-based Operating Manuals or Alarm Response Manuals, as applicable. Specifically, the following sections of Operating Manuals are applicable to the initial response to abnormal conditions:

- Section 5.0, Non-Standard Operating Conditions
- Section 6.0, Actions Following Trips and Alarms
- Section 8.0, Auxiliary Services Failures

If it is not possible to address the abnormal condition through the system Operating Manuals and/or Alarm Response Manuals, staff will respond per the applicable Abnormal Incident Manual.

The Abnormal Incident Manuals are used to respond to both abnormal and emergency conditions. Of particular relevance to abnormal conditions are Sections 3 and 4 of the Abnormal Incident Manuals, which address faults (commonly referred to as impairments) that may reduce the effectiveness of safety systems or other Safety-Related Systems.

Upon confirmation or diagnosis of emergency conditions, including DBAs, staff respond per the event-based Abnormal Incident Manuals and continue to monitor conditions as per the Critical Safety Parameter Abnormal Incident Manual. Continued use of the Critical Safety Parameter Abnormal Incident Manual ensures that operating staff perform independent checks of key parameters indicative of overall unit health, to determine if any further actions are required to complement the event-based Abnormal Incident Manuals that are in use. Once Abnormal Incident Manual actions to achieve these objectives have been successfully implemented, there is subsequent direction on longer-term actions that need to be completed in order to effectively manage post-accident conditions resulting from the event.

BDBA management strategies are described in OPG Standard [N-STD-MP-0019-R002, *Beyond Design Basis Accident Management*, August 19, 2016] and implemented through operating procedures, Emergency Mitigating Equipment Guidelines and Severe Accident Management Guidelines. Emergency Mitigating Equipment Guidelines and Severe Accident Management Guidelines are used to respond to all BDBAs, including design extension conditions and severe accidents.

Emergency Mitigating Equipment Guidelines are entered when entry conditions specified in the event-based Abnormal Incident Manuals are met. Emergency Mitigating Equipment Guidelines have a primary focus on fuel cooling and are used to mitigate accident progression when design basis equipment is unable to provide adequate core cooling. The intent of Emergency Mitigating Equipment Guideline use is to prevent a BDBA sequence from progressing to a severe accident.

Severe Accident Management Guidelines is a set of written guidance to implement strategies should a BDBA progress to a severe accident. Severe Accident Management Guidelines are entered based on entry conditions in the Emergency Mitigating Equipment Guidelines, Critical Safety Parameter or event-based Abnormal Incident Manuals being met. Entry to Severe Accident Management Guidelines implies that adequate fuel cooling has been lost and severe core damage is either imminent or has occurred. The goals of Severe Accident Management Guidelines are to terminate progression of core damage, if possible, by restoring cooling, and to maintain Containment integrity and minimize radioactive releases. The Severe Accident Management Guidelines allow flexibility in application and the Severe Accident Management Guidelines document set is referred to as “guidance”. In addition, the Severe Accident Management Guidelines aid in identifying longer-term actions that will be required to address post-accident conditions once the station has been returned to a controlled, stable state.

It is concluded that the normal and emergency operating procedures for Pickering NGS are structured and comprehensive, and provide a sound basis for operation of the plant and for operator response to abnormal occurrences.

D.10. Safety Principles Related to Level 3

The following safety principles are related to defence-in-depth Level 3:

- D-168 – Automatic Safety Systems
- D-174 – Reliability Targets
- D-177 – Dependent Failures
- D-182 – Equipment Qualification
- D-237 – Control of Accidents Within the Design Basis

As demonstrated below, Pickering NGS design and operation are aligned with all safety principles related to Level 3.

D-168 – Automatic Safety Systems

Principle: Automatic systems are provided that would safely shut down the reactor, maintain it in a shut down and cooled state, and limit any release of fission products that might possibly ensue, if operating conditions were to exceed predetermined set points.

Descriptions and review of the automatic Safety Systems are provided under the following safety principles:

- D-200 – Automatic Shutdown Systems (for the Shutdown Systems)
- D-207 – Emergency Heat Removal (for the ECI System)
- D-217 – Confinement of Radioactive Material (for the Containment System)

The effectiveness of these systems in performing their safety functions and ensuring that the applicable acceptance criteria are met during a transient or accident is substantiated by safety analyses documented in Part 3 of the Pickering NGS Safety Reports.

D-174 – Reliability Targets

Principle: Reliability targets are assigned to safety systems or functions. The targets are established on the basis of the safety objectives and are consistent with the roles of the systems or functions in different accident sequences. Provision is made for testing and inspection of components and systems for which reliability targets have been set.

Reliability targets and the need for inspection and testing to support them were identified early in the development of the CANDU design. In 1972, the unavailability criteria were made more stringent with the unavailability requirement becoming 1 x 10^{-3} for each safety system, for reactors built after Pickering NGS Units 1-4. The reliability of the Shutdown System in Units 1-4 was improved for units being returned to service (Unit 1 and Unit 4). A second separate system was not feasible because of limited penetrations into the Calandria and high radiation fields in some areas. Instead, an enhancement (SDSE) of the existing shutdown features was installed to provide a new separated and diverse set of triplicated trip sensors and trip logic. In addition, two more shutoff rods were added in Units 1,4 to bring the total to 23. This enhancement has increased shutdown reliability to provide additional assurance that any anticipated transient will be terminated with little or no consequence.

In 1995, the unavailability target for the Shutdown System in Pickering NGS Units 1,4 was changed from 3 x 10^{-3} years/year to 1 x 10^{-3} years/year. During the return to service activities the unavailability target for Units 1,4 was revised to 1 x 10^{-3} years/year for the Negative Pressure Containment System and 2 x 10^{-3} years/year for the ECI System.

In Units 5-8, each of the Special Safety Systems (SDS1, SDS2, ECI and Negative Pressure Containment) has a reliability target of 1 x 10^{-3} years/year.

The Special Safety Systems and standby safety support systems are tested on a regular basis to ensure they will be available to operate if called on. The systems are designed to facilitate testing of all components, either as a system or in a series of overlapping component tests. Test frequencies are established to ensure that the systems meet defined reliability requirements. By testing the components of these systems at known frequencies, the actual availability can be monitored and compared against the expectation. Unavailability and PSA models are used during the design of modifications to the plant to confirm that the systems meet their system reliability requirements. The models use component failure rates and test frequencies to arrive at predicted system unavailability. During operation, component fault data are collected, and predicted future unavailability is recalculated and reported to the CNSC on a yearly basis for Systems Important to Safety, using this actual component experience.

Standby safety support systems, such as the standby emergency generators, are also tested regularly so that the system reliability can be tracked.

The SDS1 and SDS2 tripping detectors and logic are tested at power without initiating the system since 2 out of 3 channels are required for initiation. Partial drop tests of the SDS1 shutoff rods are done regularly, and a full test of the system is done during unit shutdowns. Shutdown System reliability is monitored and reported regularly and has always been substantially better than the 10^{-3} years/year unavailability requirement.

Periodic testing of the ECI System components is done to demonstrate its availability. All necessary testing of ECI System valves can be conducted when the reactors are at power. Since the whole system cannot be tested without interfering with normal reactor operation, the design allows for a series of overlapping tests that cover all functions of the system. If the unit-specific portions of the ECI System are unavailable, there is a limited time to correct the problem before the reactor must be shut down. If the common portion of the ECI System is unavailable, there is again a limited time to correct the problem before the entire station must be shut down.

Periodic functional testing is done on all automatic Containment closures and on the pressure relief valves during operation. The present practice is to test the valves yearly (one valve per month). Visual inspection of the equipment air lock door and the emergency air lock door seals is done during operation. The leakage rates of air locks with both doors closed is checked periodically.

As required by CNSC Regulatory Document RD/GD-98, *Reliability Program*, OPG has developed lists of Systems Important to Safety for Pickering Units 1,4 and 5-8. The Systems Important to Safety, along with their unavailability targets are documented in OPG Report [P-REP-09051.1-00016-R000, *Pickering NGS – 2016 Annual Risk and Reliability Report*, March 31, 2017]. The Systems Important to Safety for Pickering NGS are:

- Shutdown Systems
- Emergency Coolant Injection System
- Negative Pressure Containment System
- Standby Class III Power System
- Auxiliary Boiler Feedwater System
- Standby Class III Service Water System
- Heat Transport Shutdown Cooling System
- Powerhouse Emergency Venting System
- Emergency Water System (Units 5-8) and Emergency Boiler Water System (Units 1,4)
- Class III Inter-Station Transfer Bus and Class III Motor Control Centres (Units 1,4)
- Emergency Power System and Class II Power (Units 5-8)
- Heat Transport D_2O Recovery System (Units 1,4)
- Heat Transport Pressure and Inventory Control System (Units 1,4)

It is concluded that suitable unavailability targets are established for the Special Safety Systems. Periodic testing of the system components verify the targets are met. The safety system unavailability is assessed annually in the Annual Reliability Report. Also, standby safety and support systems are tested regularly so that the system reliability can be tracked.

D-177 – Dependent Failures

Principle: Design provisions seek to prevent the loss of safety functions due to damage to several components, systems or structures resulting from a common cause.

The Pickering NGS Units 1,4 and 5-8 designs include protection against common cause events. The design includes provisions to prevent the loss of safety functions due to damage to several components, systems or structures resulting from a common cause. For Units 5-8, redundant equipment and circuits are separated and grouped (Group 1 and Group 2) to ensure the safety of the station following a common mode event such as a turbine missile. Each group is capable, independently of the other group, of safely shutting down the reactor, cooling the Fuel and providing the operator with indications of system conditions. Units 1,4 were not designed with Group 1 and Group 2 systems. However, systems have been either qualified or retrofitted to function as required for a given common mode event by ensuring effective separation and diversity. Specific common cause events are addressed in the design as follows:

Seismic Qualification

Seismic qualification of Units 5-8 was established during the design and construction phases. Nuclear Safety Design Guide [DG-30-68000-2 R01, Pickering G. S. 'B', *Engineering Design Guide, Seismic Qualification of Safety Related Systems*, December 3, 1979] establishes the basis of Seismic Qualification and identifies those systems that are required to be qualified to permit execution of the basic safety functions following the occurrence of a low probability severe earthquake at the station site. This Design Guide also describes the two seismic categories (Design Basis Earthquake and site design earthquake[35]) that define functional requirements during or following an earthquake, as well as the levels of seismic excitation that structures or equipment must withstand. Acceptable methods for demonstrating qualification are also listed. The requirements of the Nuclear Safety Design Guides were implemented in design output documents such as Design Manuals, System Design Requirements, Technical Specifications and Specification Data Sheets for equipment and System Flow Diagrams. The information related to the Pickering site seismic hazard has been included in the Pickering 5-8 PSA based Seismic Margin Assessment [P-REP-03611-00006-R00, *Pickering NGS PSA Update to Include Enhancements from the Fukushima Integrated Action Plan*, April 30, 2014].

For protection against seismic events, Pickering NGS Units 1,4 was constructed according to the National Building Code of Canada. The Units 1,4 SSCs required to perform functions during and following an earthquake were not originally required to be seismically qualified; however, a static load of 5% was specified by the National Building Code to address seismic events. Seismic Qualification for Units 1,4 has subsequently been established by analysis [NA44-REP-02004-0073-R000 Vol 1-7, *Seismic Assessment of Pickering A Nuclear Generating Station Summary Report*, February 25, 1998]. Although the Units 1,4 systems were not originally required or designed to be Seismically Qualified, the equipment was subsequently evaluated and a Safe Shutdown Equipment List has been defined by the 1998 Seismic Margin Assessment. The Seismic Equipment List identifies the SSCs required for the seismic success

[35] The site design earthquake is defined as an earthquake occurring within 24 hours after a LOCA.

path [NA44-REP-02004-00002-R001, *Pickering NGS A Seismic Success Path Addendum Including the Safe Shutdown Equipment List*, August 13, 2013]. A PSA based Seismic Margin Assessment was completed in 2013 [P-REP-03611-00006-R00, *Pickering NGS PSA Update to Include Enhancements from the Fukushima Integrated Action Plan*, April 30, 2014].

Tornado

A tornado was not considered a design basis event during the design of either Units 1,4 or Units 5-8 but was addressed as part of the building code. Tornado and hurricane winds were assessed in Hazard Screening Analyses that confirmed that the reactor's ability to control and contain will remain intact following a tornado. A detailed assessment of margin available in the structures of both Units 1,4 and Units 5-8 to cope with a probable maximum tornado event was therefore recommended by these reports.

Subsequently, high wind related hazards were included in the scope of both the Units 1,4 and Units 5-8 PSAs. The most current high wind PSA analyses for both stations are documented in [P-REP-03611-00006-R00, *Pickering NGS PSA Update to Include Enhancements from the Fukushima Integrated Action Plan*, April 30, 2014].

External Flooding

The overall flood protection system for the Pickering site consists of the shoreline breakwater works, catch basins and storm sewers. The ground surface elevation of the Pickering site exceeds the 1 in 200 year lake water level. In the event that overtopping of flood protection occurs due to a high-magnitude storm coupled with high Lake Ontario level, lake water will be collected in the catch basins, while the storm sewers will discharge this water either directly back to the lake or to the intake channel.

External Flooding was assessed as per the methodology [N-GUID-03611-10001 Vol 8 R004, *OPG Probabilistic Safety Assessment (PSA) Guide External Hazard Screening*, September 16, 2016]. The consequences of external flooding events were assessed in the Hazard Screening Analyses for both Units 1,4 and Units 5-8. The majority of identified flooding events are screened out of both Units 1,4 and Units 5-8 as a result of these assessments.

Missile Protection

As described above, grouping and separation of equipment is used to protect against the loss of safety functions resulting from missiles. The consequences of internally generated missiles have been assessed in Hazard Screening Analyses as per methodology [N-GUID-03611-10001 Vol 9 R002, *OPG Probabilistic Safety Assessment (PSA) Guide Internal Hazard Screening*, November 10, 2016]. Possible missiles originating from turbine disintegration, the failure of various valves and pumps and the explosion of acetylene bottles used for cutting and welding were assessed. The failure of turbines, pumps and valves were screened out due to low impact after considering the separation of independent systems and the inclusion of protective barriers in the Pickering NGS design, while missiles due to acetylene bottle explosions were screened out due to low frequency.

Protection Against Dynamic Effects Associated with the Rupture of Piping

Pipe ruptures in Pickering NGS have been assessed for their potential for consequential damage to other components which could jeopardize the safe shutdown and continued cooling of the reactor. Critical SSC pipework, particularly for the Primary Heat Transport, Shutdown System (SDS) and ECI Systems have been analysed for pipe whip potential at varying temperatures. As described in Part 2 of the Pickering NGS Units 5-8 Safety Report, earlier studies considered the effect of whipping pipes and jet effects. The layout was reviewed to ensure that loop-to-loop break propagation would not occur. This required the addition of pipe restraints on all the 350 mm (14 in.) primary heat transport pipes between the reactor outlet headers and the Steam Generators. In addition to the primary heat transport piping restraints, a pair of pipe restraints was added to each of the reactor headers to prevent propagation or worsening of consequences from certain postulated pipe breaks. Also, deflector plates have been installed to protect instrumentation and electrical cable runs that are required for safe shutdown and continued cooling of the reactor.

An assessment of the dynamic effects associated with the rupture of high-energy piping for Units 5-8 was completed in 2016 [P-REP-04960-00014 R01, *Pipe – Whip and Jet – Impingement Assessment of Piping Inside Reactor Building - Units 5-8*, May 26, 2016] and confirmed that the layout of the high-energy piping and Safety-Related Systems inside of the Reactor Buildings of Units 5-8 are in compliance with the practices, expectations and guidelines in modern standards. Furthermore it indicated that this finding can be used for the reclassification of CSI-IH6 in Units 5-8 from Category 3 to Category 2, such that appropriate measures are in place to maintain the safety margins for this hazard. A similar assessment for Units 1,4 is underway, and this is reflected in the planned resolution of GI-25, Category 3 CANDU Safety Issues.

In addition, Part 3 of the Pickering NGS Safety Reports address consequential damage resulting from rupture of a Fuel Channel. The assessments confirm that these hazards do not result in unacceptable consequences despite potential damage to in-core structures and Calandria tube collapse for neighbouring Fuel Channels, and partial impairment of the insertion of the shutoff rods.

Environmental Qualification of Safety-Related Equipment.

The Environmental Qualification (EQ) program [N-PROG-RA-0006 R008, *Environmental Qualification*, June 3, 2015] provides the documented assurance that essential Safety-Related Systems, components and structures are capable of performing their functions when subjected to the environmentally harsh conditions that could result from postulated DBAs (see Safety Principle D-182, Equipment Qualification).

As described above, the Pickering NGS design also includes additional provisions to prevent the loss of safety functions due to damage to several components, systems or structures resulting from a common cause. For example, the Inter-Station Transfer Bus and Emergency Boiler Water Supply systems have specifically been installed to mitigate high energy pipe failures in the powerhouse. Steam protected rooms and barriers have been installed to protect critical equipment and staff. For seismic events, systems including power (Class I/II/III), water and control have been hardened to provide assurance that they will remain operational. Specific considerations for the reactor Shutdown Systems include:

- In Units 5-8 the two Shutdown Systems, SDS1 and SDS2, are functionally and physically independent of each other and functionally independent of the Reactor Regulating System. In Units 1,4, the Shutdown System reliability has been improved by retrofitting a Shutdown System Enhancement (SDSE) that provides sensors and trip logic independent from those of the original Shutdown System (SDSA).
- In Units 5-8, independence is achieved by employing diverse shutdown principles, i.e., SDS1 uses solid shutoff rods (gravity driven with spring assistance), and SDS2 directly injects poison into the Moderator (pressurized injection). The systems, including ancillary equipment, are also physically separated. The shutoff rods are inserted vertically into the top of the reactor. The poison injection tubes are inserted horizontally into the side of the reactor.
- In Units 1,4, independence and separation is implemented for the sensors and trip logic, but SDSA and SDSE employ the same gravity-driven shutoff rods with spring assistance.
- The Moderator Dump System in Units 1,4 augments the shut-off rods. Automatic operation of the dump system occurs when reactor power has been measured as not decreasing fast enough, or not reaching a sufficiently low level, following operation of the Shutdown System.

To summarize, Pickering NGS design addresses dependent failures, using separation of redundant SSCs for responding to common mode events. Hazard assessments have been performed that confirm the capability of the design to perform the required safety functions following postulated common mode events.

D-182 – Equipment Qualification

Principle: Safety components and systems are chosen that are qualified for the environmental conditions that would prevail if they were required to function. The effects of ageing on normal and abnormal functioning are considered in design and qualification.

The Pickering NGS EQ program [N-PROG-RA-0006 R008, *Environmental Qualification*, June 3, 2015] establishes an integrated and comprehensive set of requirements that provide assurance that essential equipment can perform as required when exposed to harsh conditions during an accident, and that this capability is preserved over the life of the plant. Implementation of these program requirements provides the methodology, programmatic controls and interfaces for establishing and maintaining EQ of equipment and components.

The scope of the EQ program covers all components that are essential to provide a safety function consistent with the assumptions and requirements documented in the current accident analysis summarized in the Safety Reports. All single and dual failure accidents within the design basis are covered; these are termed DBAs for the purpose of the EQ program. For each accident defined, the EQ program ensures that a reliable and qualified line of defence is provided to achieve and maintain reactor shutdown, fuel heat removal, Containment and post-accident monitoring.

A suite of engineering programs and processes exists to ensure that EQ requirements are met and documented. Qualification is established through testing, analysis, OPEX, ongoing qualification or a combination of these methods. EQ Assessments identify equipment specific design, configuration, maintenance and procurement requirements necessary to establish and maintain the qualified status of equipment.

Environments where equipment and components are installed are monitored to ensure conditions used in EQ are valid. Surveillance is completed to ensure the installed configuration of electrical equipment conforms to EQ related design and configuration requirements. Periodic inspection and maintenance of EQ barriers are completed to ensure their integrity throughout the life of the plant.

The EQ process consists of the following general steps:

- Listing the DBAs and identifying those that result in post-accident harsh environments with the potential to cause common mode failures.
- Identifying in an EQ Design Guide the systems with EQ requirements due to their role in mitigating the harsh environment DBAs, based on compiling a list of structures, systems, equipment and components that are credited in the safety analysis.
- Defining, for each of the harsh environment DBAs, all rooms of the plant as either harsh or mild. For rooms that are classified as harsh, the environmental conditions are derived for the bounding accident.
- Preparing an EQ List Development Package that lists equipment by parent devices as extracted by the EQ Design Guide. Equipment interfaces, functional requirements and failure modes are established and equipment is classified as either fail-safe or requiring an EQ Assessment.
- Placing equipment that does not fail-safe and must operate in a harsh DBA environment on the EQ List. Equipment on the EQ List must be qualified and documented with an EQ List Development Package. An EQ Assessment document is produced to document the design basis assurance that demonstrates the equipment’s capability to perform its safety function, under harsh environment resulting from applicable DBAs.

Equipment identified in the EQ List must maintain its qualification status as prescribed in the EQ Assessment documents. Changes can take place that may affect the EQ configuration of equipment as a result of design modifications, changes in safety analysis, storage, equipment maintenance, etc. The configuration management and ongoing equipment performance monitoring must ensure that EQ configuration is maintained.

In conclusion, there is a structured and comprehensive EQ program in Pickering NGS. The EQ process is maintained on an ongoing basis, such that the effects of equipment aging are accounted for.

D-237 – Control of Accidents Within the Design Basis

Principle: Provisions are made at the design stage for the control of accidents within the design basis, including the specification of information and instrumentation needed by the plant staff for following and intervening in the course of accidents.

For Pickering NGS, DBAs and the plant response under DBA conditions, are detailed in Part 3 of the Safety Reports [NK30-SR-01320-00003-R004, *Pickering Nuclear 5-8 Safety Report: Part 3 Accident Analysis Vol 1-5*, October 10, 2014], [NA44-SR-01320-00002-R004, *Pickering Nuclear 1-4 Safety Report: Part 3 – Accident Analysis*, October 31, 2013]. For Pickering NGS, accidents within the design basis are classified as single failure (i.e., a single process failure) and dual failure (i.e., a process failure combined with a coincidental impairment of a Special Safety System). Safety system impairments considered include a failure of the system to perform its safety function.

Accidents within the design basis include events that have a potential of occurring during the lifetime of the plant, events that would be classified as DBAs, and some dual failures that are BDBAs in the CNSC REGDOC-2.4.1 terminology.

Control systems are provided in the design to prevent the more frequent events from requiring Special Safety System action, in particular, the setback/stepback feature of Reactor Regulating System. Similarly, the Special Safety Systems together with their support systems are capable of controlling the more challenging accidents within the design basis, and preventing them from progressing further.

The control systems and Special Safety Systems are designed to perform their function automatically in the short term for most postulated accidents. Analyses demonstrating the effectiveness of the automatic response of the Special Safety Systems are documented in Part 3 of the Pickering NGS Safety Reports. By their nature, certain postulated accidents would require operator action in the short term. A required operator action is credited in the safety analysis no sooner than 15 minutes following the first unambiguous indication of the event if the action can be performed from the Main Control Room, or 30 minutes if field actions are required. A required operator action is one that is credited in the analysis and which if not credited, would result in accident consequences that could not be shown to be predicted (or bounded) by other analyses documented in Part 3 of the Safety Report.

Required operator actions are tabulated in Part 3, Section 1 of the Safety Reports, including the required timing and definition of the clear, unambiguous signals to the operator to identify the given accident.

The clear, unambiguous signals designed to allow the operator to respond to the event are identified in the Operating Manuals and Abnormal Incident Manuals (see Safety Principle O-288, Normal Operating Procedures, which also addresses Safety Principle O-290, Emergency Operating Procedures). To provide post-accident monitoring capability, the critical safety parameters are displayed in the Main Control Room. For Units 1,4, the SDSE Instrument Rooms also provide for monitoring of the critical safety parameters (a list of critical safety parameters for Units 1,4 is provided in Safety Principle D-227, Monitoring of Plant Safety

Status). The SDSE Instrument Rooms house SDSE trip parameters signal processing instrumentation, trip logic and monitoring computers. Buffered/isolated signals are transmitted from the SDSE Instrument Room to the Main Control Room to provide indications of the SDSE parameters. These instrument rooms provide separation from SDSA instrumentation and trip logic. In Units 5-8, the Unit Emergency Control Centre rooms contain unit controls and logic panels and provision for monitoring of critical parameters (see Safety Principle D-227, Monitoring of Plant Safety Status).

It is concluded that adequate provisions for the control of accidents within the design basis are provided at Pickering NGS. The operator has indications and alarms as well as the capability to perform actions from the Main Control Room for this purpose.

D.11. Safety Principles Related to Level 4

The following safety principles are related to defence-in-depth Level 4:

- AM-318 – Strategy for Accident Management
- AM-323 – Training and Procedures for Accident Management
- AM-326 – Engineered Features for Accident Management

As demonstrated below, Pickering NGS design and operation are aligned with all safety principles related to Level 4.

AM-318 – Strategy for Accident Management

> *Principle: The results of an analysis of the response of the plant to potential accidents beyond the design basis are used in preparing guidance on an accident management strategy.*

As per [N-PROG-MP-0014-R006, *Reactor Safety Program*, July 18, 2016], the Reactor Safety Program provides the framework for major aspects of safe operation. One element of the reactor safety program relates to the management of BDBAs and severe accidents. In this context, a BDBA refers to a relatively low frequency event sequence that is not included in the plant design basis (due to the low frequency of occurrence) and is not necessarily bounded by the analyses of the station design basis. If the consequences of such events are significant core degradation, these BDBAs are referred to as severe accidents. The term design extension condition is introduced (as per CNSC REGDOC-2.5.2, *Design of Reactor Facilities: Nuclear Power Plants*) to describe a sub-set of BDBA conditions for which specific SSCs, referred to as complementary design features, are provided for mitigation.

OPG's response to BDBAs is detailed in [N-STD-MP-0019 R002, *Beyond Design Basis Accident Management*, August 19, 2016]. The written guidance that implements the strategies for managing BDBAs is referred to as Emergency Mitigating Equipment Guidelines and Severe Accident Management Guidelines. Application of Emergency Mitigating Equipment Guidelines and Severe Accident Management Guidelines may require temporary changes to permit operation of specific SSCs or Emergency Mitigating Equipment to implement Emergency Mitigating Equipment Guidelines/Severe Accident Management Guidelines objectives. For permanent changes to SSCs to facilitate Emergency Mitigating Equipment Guidelines/Severe Accident Management Guidelines, [N-GUID-01130-10000-R001, *Modifications for Beyond Design Basis Accidents*, March 17, 2015] provides guidance related to the design, modification, procurement, operation and testing of SSCs, for responding to BDBAs.

The physical processes that govern severe accident phenomena are complex and, consequently, Severe Accident Management Guidelines are designed to be flexible, accommodating a wide range of severe accident causes and progression. Supporting analysis for BDBAs has been prepared as part of the PSAs [P-REP-03611-00006-R00, *Pickering NGS PSA Update to Include Enhancements from the Fukushima Integrated Action Plan*, April 30, 2014]. Reasonable strategies for coping with severe accident progression have been identified and developed using “state of the art” reviews, PSAs and insights on accident behaviours from accident analyses that are presented in the COG Severe Accident Management Guidance Technical Basis Documents.

Following the March 11, 2011 accident at the Fukushima Daiichi nuclear power plant, the CNSC requested all Canadian utilities to complete an assessment to review the impact of a similar event (i.e., an event resulting in a total loss of power, subsequently resulting in a total loss of heat sinks to cool the fuel post-shutdown) at their respective stations. OPG Report [P-REP-03490-10012-R01, *Fukushima Daiichi - Total Loss of Heat Sink Assessment for Pickering A and Pickering B*, June 14, 2013] consolidates the results from various evaluations performed to date and establishes the time line for progression of a total loss of heat sink event. P-REP-03490-10012-R01 also identifies mitigating provisions which could be put in place to prevent progression to a severe accident. Provisions to maintain or re-establish the Control, Cool, Contain and Monitoring safety functions were examined to determine those that are most practical to implement and also meet specified requirements.

From a PSA perspective, the Pickering 1,4 and 5-8 Risk Assessment has been updated as part of the Fukushima Action Item update [P-REP-03611-00006-R00, *Pickering NGS PSA Update to Include Enhancements from the Fukushima Integrated Action Plan,* April 30, 2014]. The PSA for Pickering NGS Units 1,4 and 5-8 has incorporated the risk benefits gained from the Fukushima Action Item enhancements (e.g., BDBA procedures/guides and Emergency Mitigating Equipment).

To summarize, BDBAs have been analyzed for Pickering NGS. The station operating procedures, Emergency Mitigating Equipment Guidelines and Severe Accident Management Guidelines are supported by analysis of BDBA progression.

AM-323 – Training and Procedures for Accident Management

Principle: Nuclear plant staff are trained and retrained in the procedures to follow if an accident occurs that exceeds the design basis of the plant.

BDBA management strategies are described in OPG Standard [N-STD-MP-0019-R002, *Beyond Design Basis Accident Management*, August 19, 2016] and implemented through operating procedures, Emergency Mitigating Equipment Guidelines and Severe Accident Management Guidelines, which are used to respond to all BDBAs, including design extension conditions and severe accidents.

The adequacy of the response and mitigation strategies that have been developed is demonstrated primarily through drills and exercises. The Emergency Preparedness Department assesses the Emergency Response Organization performance to the established objectives identified in OPG Instruction [N-INS-03490-10002-R005, *Conduct of Emergency Preparedness Drills and Exercises*, December 31, 2014] and reviews all drill and exercise related Station Condition Records and Action Requests to monitor status and ensure completeness. These assessments are documented in the Emergency Preparedness Drill and Exercise Performance Objectives Reports. Deficiencies found during drills and exercises are documented using the Station Condition Record System in accordance with OPG Procedure [N-PROC-RA-0022-R034, *Processing Station Condition Records*, March 24, 2017]. Corrective Action plans are developed and corrective actions are initiated and tracked to completion.

The drills and exercises outlined above are a component of the overall training for accident management. As described in Safety Principle O-278, Training, OPG Program [N-PROG-TR-0005-R016, *Training*, January 5, 2016] summarizes how training needs are addressed at OPG. All tasks to be performed are subjected to an analysis to determine the training requirements, documented in documents designated as Job Training Analysis. Through this process, of the order of fifty separate tasks have been identified for Emergency Mitigating Equipment Guidelines/Severe Accident Management Guidelines. Following the training analysis, a training package designated as “OVH” has been generated for each task to ensure the identified tasks are properly executed.

[N-TQD-503-00001 R017, *Nuclear Emergency Response Organization Training and Qualification Description*, January 12, 2016] establishes training and qualification requirements for Emergency Response Organization individuals assigned to response positions, including Emergency Response Organization staff in accordance with OPG Program [N-PROG-RA-0001-R015, *Consolidated Nuclear Emergency Plan*, December 22, 2016]. Initial training is designed to ensure that Emergency Response Organization personnel have the knowledge and skills needed to independently perform tasks associated with the identified Emergency Response Organization position. Continuing training maintains and enhances knowledge, skills and performance standards required to perform tasks of Emergency Response Organization positions. The program also facilitates confirmation that incumbents retain the knowledge and skills required for correct execution of tasks associated with assigned emergency response roles. Continuing training includes:

- Important lessons learned from drills and exercises
- Real events
- Industry experience
- Position-specific procedures and changes

- Associated equipment changes
- Performance standards changes

In conclusion, training is in place at Pickering NGS to support Beyond Design Basis response. This includes training in the use of operating procedures, Emergency Mitigating Equipment Guidelines and Severe Accident Management Guidelines. Drills and exercises are important elements of the training conducted for accident management.

AM-326 – Engineered Features for Accident Management

Principle: Equipment, instrumentation and diagnostic aids are available to operators, who may at some time be faced with the need to control the course and consequences of an accident beyond the design basis.

Pickering NGS Units 1,4 and 5-8 has complementary design features for BDBAs. If actions identified in Operating Manuals and Abnormal Incident Manuals are unsuccessful in terminating the accident progression, actions will be taken per the Emergency Mitigating Equipment Guidelines to prevent the accident from progressing to a severe accident. The Emergency Mitigating Equipment Technical Basis Document [N-BDB-03600-00002-R000, *OPG Emergency Mitigating Equipment for Beyond Design Basis Accidents – Technical Basis Document*, October 2015] summarizes the technical basis for Emergency Mitigating Equipment, including:

- The bounding BDBA event sequence and associated analyses.
- The overall functional requirements for the Emergency Mitigating Equipment.
- Other information relevant to Emergency Mitigating Equipment specification, design and procurement.

Critical safety parameters and associated instrumentation for monitoring plant response in BDBAs have also been identified as discussed in the Emergency Mitigating Equipment Technical Basis Document.

A key line of defence for BDBAs is Emergency Boiler Makeup using Emergency Mitigating Equipment. An additional line of defence is Emergency Mitigating Equipment to provide makeup to the Heat Transport System and to the Moderator. These features are capable of limiting core damage and allowing for Containment pressure control for most BDBAs.

In addition, implementation of Phase 2 Emergency Mitigating Equipment modifications is in progress. Phase 2 Emergency Mitigating Equipment includes restoration of power to critical station equipment for core cooling and monitoring, and for restoring operation of Containment coolers to improve control of Containment pressure following BDBAs.

Additional major Fukushima Project modifications include:

- Passive Auto-catalytic Recombiners have been installed on all units to supplement the existing Hydrogen Igniters for control of hydrogen in Containment.
- Enhancements to water makeup/cooling capability for the IFBs have been installed.

Instrumentation and diagnostic aids to support deployment of Emergency Mitigating Equipment are available to the operator as described in the review of Safety Principle D-227, Monitoring of Plant Safety Status.

In conclusion, the modifications completed prior to and in response to the 2011 Fukushima accident, together with additional Emergency Mitigating Equipment, provide enhanced capability for BDBA Management. Ongoing addition of Phase 2 Emergency Mitigating Equipment will further enhance safety margins.

D.12. Safety Principles Related to Level 5

The following safety principle is related to defence-in-depth Level 5:

- S-140 – Feasibility of Emergency Plans

As demonstrated below, Pickering NGS design and operation are aligned with the safety principle related to Level 5.

S-140 – Feasibility of Emergency Plans

Principle: The site selected for a nuclear power plant is compatible with the offsite countermeasures that may be necessary to limit the effects of accidental releases of radioactive substances, and is expected to remain compatible with such measures.

The original plant did not explicitly consider the feasibility of emergency plans as part of the site selection criteria. The Consolidated Nuclear Emergency Plan [N-PROG-RA-0001-R014, *Consolidated Nuclear Emergency Plan*, May 2015] ensures compatibility with the offsite countermeasures that are necessary to limit the effects of accidental releases of radioactive substances on a continuing basis. The Consolidated Nuclear Emergency Plan provides a basis for OPG Nuclear to implement and maintain its emergency response capability to protect the public, employees and the environment in the event of a nuclear emergency.

The Main Control Room team utilizes resources of the on-site Emergency Operations Centre to mobilize and deploy the necessary emergency teams. The Emergency Operations Centre personnel may, if necessary, mobilize, brief and deploy Emergency Response Teams to address specific emergencies, such as a fire, medical emergency, or a search and rescue operation. Appropriate off-site support groups (i.e., non-OPG resources) such as fire and/or ambulance may be activated to respond to the site. The Main Control Room and the Emergency Operations Centre are equipped with the necessary communications and other equipment as described in Section 1.5 of the Consolidated Nuclear Emergency Plan. On-site and off-site Emergency Response Facilities are discussed further under Safety Principle EP-336, Emergency Response Facilities.

The adequacy of the response and mitigation strategies that have been developed is demonstrated primarily through drills and exercises, which are conducted in accordance with OPG Procedure [N-PROC-RA-0045 R010, *Emergency Preparedness Drills and Exercises*, December 2015]. The Emergency Preparedness Department assesses the Emergency Response Organization performance to the established objectives identified in OPG Instruction [N-INS-03490-10002-R005, *Conduct of Emergency Preparedness Drills and Exercises*, December 31, 2014] and reviews all drill and exercise related Station Condition Records and Action Requests to monitor status and ensure completeness. These assessments are documented in the Emergency Preparedness Drill and Exercise Performance Objectives Reports.

To take into account major changes at site and around the site, OPG Approved Roles Document [N-MAN-08131-10000 Sht. S4-0245 R002, *Manager, Emergency Preparedness*, November 3, 2014] provides specific accountability to the Department Manager of Emergency Preparedness to interface with the regulator and other external stakeholders as necessary to ensure that the relevant regulatory and licensing requirements are built into the emergency preparedness program and stakeholder interfaces are effectively managed.

The Department Manager of Emergency Preparedness is a member of the Toronto and Durham Regional Nuclear Emergency Management Coordinating Committees as well as the Provincial Nuclear Emergency Management Coordinating Committee. These committees meet on a regular basis and review issues that could impact emergency response, including local development which may modify/increase risks. In addition, OPG's OPEX process monitors events around the world to determine if there is any unforeseen event that may have applicability to Pickering NGS. If/when an event is deemed to be applicable, the emergency response process is reviewed to ensure the event can be adequately dealt with.

To summarize, the Pickering NGS emergency plans are tailored for the characteristics of the surrounding area where countermeasures would be taken. Changes in these characteristics are factored into the emergency planning on an ongoing basis.

Appendix E – Strengths Used in the Defence-in-Depth Assessment

Strength ID	Strength Title and Description	Justification	Level
S-01	**Management** The Pickering Station Leadership Team has effectively aligned the organization to significantly improve performance in several focus areas.	At Pickering, the leadership team has effectively aligned the organization to improve performance in several focus areas. Actions in support of these focus areas have been incorporated into department human performance plans and include: • Coaching to Enhance Nuclear Professionalism and Performance • Opportunities to leverage good practices • Analysis, Corrective Action Planning and Feedback of Performance Information • Critical Engaged Thinking & Human Performance Tools • Radiological Worker Practices The Safety Factor 10 Report states that OPG has a well-defined organizational structure and strong lines of authority. Operation and maintenance of the station are conducted as per regulatory requirements and Nuclear standards for public safety are directed by the Director, Operations and Maintenance, who also coordinates with the centre-led organizations to effectively use resources to achieve performance targets. The quality and quantity of services provided by these centre-led support organizations is monitored by the Pickering Senior Site Vice President, who holds responsibility for establishing site requirements and priorities. OPG identifies qualified and competent individuals for key positions with career development and succession planning being key elements in the management capability strategy. The corporate succession plan ensures that individuals with high leadership potential are identified to help continue excellence in nuclear safety. Managers at OPG ensure that tasks are executed as defined through	1, 2, 3, 4, 5

Strength ID	Strength Title and Description	Justification	Level
		programs that are specifically designed to achieve higher levels of Nuclear and industrial safety, higher unit reliability, and reduced operating costs through event-free operation. This performance is accomplished through pre-job briefings, post-job debriefings, self-checking programs, communications, self-assessments, and an observation and coaching program. The Human Performance Program at Pickering has been implemented to continually reduce human performance events and errors by managing defences, in pursuit of zero events. The number of challenges to safe, reliable and effective operation of the Pickering station has been reduced, and the significance of these challenges has also been reduced due to Pickering NGS leadership team's commitment to, and focus on, risk awareness, risk mitigation and worker training. The resultant improved station performance is reflected in improving trends in our established metrics on both Forced Loss Rate and Site Event Free Day Resets. Other utilities have benchmarked Pickering NGS Operations Management and Leadership successes for implementation at their facilities. **Sources:** Section 4.1.1 and 4.1.2 of PSR2 Safety Factor 10 Report [P-REP-03680-00014 R000, *Pickering NGS PSR2 Safety Factor 10 Report: Organization, Management System, and Safety Culture*, December 2016] 2013 Public Hearing on Pickering Licence Renewal	
S-02	**Effective Equipment Reliability Program** Pickering NGS' Equipment Reliability Program implementation and execution of the program is a station priority.	The Equipment Reliability program [N-PROG-MA-0026-R002, *Equipment Reliability*, May 26, 2015] ensures that there are defined activities ensuring that equipment aging issues are identified, understood and effectively managed for equipment important to nuclear safety and equipment reliability. Any Aging Management station actions are captured in the system health reports and managed through the Work Management program. The Equipment Reliability	1, 2, 3, 4

Strength ID	Strength Title and Description	Justification	Level
	The station management team monitors implementation, and leaders enforce accountability. Obsolescence Management and Performance Monitoring, Condition Assessments, and Predictive Maintenance are also noted as taking into consideration the long-term aging management assessments.	Program contains the following elements which ensure ongoing high levels of reliable performance of critical components: • Identifying critical components that require focused attention. • Specifying the required maintenance strategies to maintain high levels of reliability. • Executing Predictive Maintenance (PdM), and Preventive Maintenance (PM) programs. • Monitoring system and component condition and implementing plans to restore and maintain system and component health. • Taking prompt and effective action, when critical equipment fails, to understand the technical and organizational causes and to prevent a recurrence. • Identifying and predicting aging and obsolescence issues on important components and embedding mitigating strategies and actions into the business plan. Multidisciplinary System Health and/or Component Health Teams are established for important systems to provide input to and support for system health activities. These teams are comprised of cross functional individuals from Engineering, Maintenance, Operations (Licensed and Non-Licensed), Work Control and Supply Chain. In addition, as stated in the Safety Factor 2 Report, detailed Condition Assessments are being completed as part of the ongoing Aging Management Program to include full power operation of Pickering NGS to 2028. In recent years, operating performance at Pickering has significantly improved as illustrated by the decreasing Forced Loss Rate. The key contributors to this improved performance are the improvements in equipment reliability and improvements in Human Performance, leading to fewer events and errors. This is a result of improvements in Operations Management and Leadership which has led to a strongly focused organization.	

Strength ID	Strength Title and Description	Justification	Level
		The effectiveness assessment in Safety Factor 4 concludes that the Equipment Reliability Program has been effectively implemented. This conclusion is supported by the presence of comprehensive and effective programs developed and implemented to ensure that the condition of components meets design requirements. Operational improvements are implemented continuously based on national and international OPEX, in addition to those driven by the evolving regulatory requirements (e.g., CNSC RD-334 and CNSC REGDOC-2.6.3.) Also, Pickering NGS is assessed to be in conformance with safety-significant requirements of CNSC RD/GD-210. **Source(s):** PSR2 Safety Factor 2 Report [P-REP-03680-00005 R001, *Pickering NGS PSR2 Safety Factor 2 Report – Actual Condition of Structures, Systems, and Components Important to Safety,* March 2017] PSR2 Safety Factor 4 Report [P-REP-03680-00007 R000, *Pickering NGS PSR2 Safety Factor 4 Report: Aging*, July 2016]	
S-03	**Major Components Program** Strong governance framework with comprehensive programs is in place for Major Component Life Cycle Management Plans, addressing the full spectrum of disciplines, including administrative, engineering, inspection, maintenance, training, quality assurance, engineering change control in conjunction with	The program [N-PROG-MA-0025-R002, *Major Components*, March 25, 2015] is fully developed and effectively implemented for continued validation of fitness for service of the Major Components (Fuel Channels, Feeders, Steam Generators and reactor structures). The program incorporates the reporting requirements associated with demonstrating compliance with design basis documentation relevant to each of the Major Components Program areas. A mature and effective Steam Generator inspection and maintenance program, as part of the Major Components Program, is successful in managing Fitness for Service of the Steam Generators to end of planned service life. The most recent Fleetview Program Health and Performance Report for 2015 [N-REP-08130-0635343 R000*, Fleetview Program Health and Performance*	1, 2, 3, 4

Strength ID	Strength Title and Description	Justification	Level
	application of OPEX and research findings.	*Report (Q1/2016 to Q4/2016)*, January 2017] reported the following significant program accomplishments: • The training program aligned with industry best practices. • The Steam Generator life cycle management plan was recognized as a very robust and well-managed program with a strong team of competent staff by internal and external reviewers. • Major Components Program staff continues to maintain a strong working relationship with peer teams and external working groups and governing bodies, which supports effective OPEX sharing, overall program improvement and program execution efficiency. In the period of 2015, Major Components program completed benchmarking of pressure tube, Feeder, and boiler inspection programs with Bruce Power, as well as CANDU utilities in China, Korea, India, and Romania. Based on this benchmarking opportunity, OPG units were found to be in alignment with industry practices. • Major Components working groups are well established and are fully effective and working together to continuously improve/sustain industry best program performance. In 2016, OPG's Major Component staff members participated in numerous peer team meetings during conferences, workshops, and working group activities. **Source(s):** Section 4.1.10 of Safety Factor 4 Report [P-REP-03680-00007-R000, *Pickering NGS PSR2 Safety Factor 4 Report: Aging*, July 2016] Section B.7 of Code and Standard Reviews Associated with Safety Factors 2, 3, and 4 [P-REP-03680-00004-R000, *Pickering NGS Periodic Safety Review 2: Code and Standard Reviews for Safety Factors 2 (Actual Condition of SSCs), 3 (Equipment Qualification) and 4 (Aging),* July 2016]	

Strength ID	Strength Title and Description	Justification	Level
		Section 4.1.1.2.49 of PSR2 Safety Factor 2 Report [P-REP-03680-00005-R001, *Pickering NGS PSR2 Safety Factor 2 Report – Actual Condition of Structures, Systems, and Components Important to Safety,* March 2017]	
S-04	**System and Component Health Reporting** The conditions of the Pickering NGS SSCs are tracked in System and Component Health Reports that are aligned with industry best practices.	The Equipment Reliability Program [N-PROG-MA-0026-R002, *Equipment Reliability*, May 26, 2015] at OPG is supported by system health reporting [N-PROC-MA-0024, *System Performance Monitoring*, December 9, 2016] and component program health reporting [N-PROG-MA-0017 R009, *Component and Equipment Surveillance*, May 31, 2017] governance. These processes are fully developed and effectively implemented ensuring that equipment reliability issues are identified, understood and effectively managed for critical equipment. Aging Management actions from Condition Assessments are captured in the System and Component Program Health Reports, and managed through the Work Management program. The System Health Reports are managed through a structured, standardized reporting program (SystemIQ) for system monitoring and performance to ensure that systems important to safety will perform their intended functions under the design basis. SystemIQ is a software application that is designed to assist System Engineers and Managers in optimizing the operational performance and condition of plant systems. SystemIQ centralizes, standardizes, and automates System Health Reporting, Performance and Condition Monitoring, and the generation and organization of System Notebooks. Similarly, the ProgramIQ application is used to document and report on Component Program health. **Source(s):** PSR2 Safety Factor 4 Report [P-REP-03680-00007-R000*, Pickering NGS PSR2 Safety Factor 4 Report: Aging*, July 2016]	1

Strength ID	Strength Title and Description	Justification	Level
S-05	**Implementation of Fukushima Action Items** Pickering NGS has implemented and is continuing to enhance significant provisions to prevent and mitigate severe accident progression and protect Containment integrity and enhance defence-in-depth against BDBAs.	Emergency Mitigating Equipment has been provided for an additional makeup water supply to cool the reactor Fuel for BDBAs. Additional major Fukushima Project design modifications that have been installed include enhancements for emergency electrical power supplies for critical parameter monitoring and for water makeup/cooling capability for the IFBs. In addition, implementation of Phase 2 Emergency Mitigating Equipment modifications is in progress. Phase 2 Emergency Mitigating Equipment includes restoration of power to station systems for continued fuel cooling and monitoring and restoration of Containment coolers to enhance control of Containment pressure following BDBAs. All Fukushima Action Items assigned to OPG for Emergency Mitigating Equipment are now closed (per final FAI status report). OPG has been recognized internationally for work done in response to the 2011 Fukushima accident. **Source(s):** Section 4.1.4 and 4.1.7 of PSR2 Safety Factor 1 Report [P-REP-03680-00008-R000, *Pickering NGS PSR2 Safety Factor 1 Report: Plant Design,* March 2017] Fukushima Action Item Status Report [N-REP-03600-10003-R007, *Fukushima Action Item Status Report,* November 2015]	4, 5
S-06	**Deterministic Safety Analysis** The deterministic safety analysis is robust and effectively implemented.	The OPG deterministic safety analysis complied with the regulatory requirements of S-310, one of the predecessors to REGDOC-2.4.1. With the evolution from S-310 to RD-310, and then to REGDOC-2.4.1, OPG prepared an implementation plan to address the evolving changes, applying a graded approach. The programmatic elements that comply with REGDOC-2.4.1 are in place. The Deterministic Safety Analysis in the Safety Reports is comprehensive, covering a full range of DBAs. The CNSC has also concluded	1, 2, 3, 4

Strength ID	Strength Title and Description	Justification	Level
		in a review of OPG's Safety Analysis Program Implementation that "OPG has a robust and effectively implemented program in place". The conclusion of the Safety Factor 5 Report is that the deterministic safety analysis program and supporting programs/procedures at OPG are comprehensive, resulting in a systematic and disciplined approach to identifying, prioritizing and addressing any safety analysis related issues. The CNSC also stated that for Safety Analysis Program Implementation, OPG staff and their service providers demonstrated a clear commitment to safety. Furthermore, the Regulatory Oversight Report for Canadian Nuclear Power Plants: 2015 states that CNSC staff concluded that the "Safety Analysis" Safety and Control Area at Pickering met or exceeded performance objectives and all applicable regulatory requirements, and rated this Safety and Control Area as Fully Satisfactory. **Source(s):** Section B.3 of Code and Standard Reviews Associated with Safety Factors 1, 5, 6, and 7 [P-REP-03680-00029-R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 1, 5, 6, and 7,* March 2017] Section 5 of PSR2 Safety Factor 5 Report [P-REP-03680-00009-R000, *Pickering NGS PSR2 Safety Factor 5 Report: Deterministic Safety Analysis,* March 2017] CNSC Correspondence on Results of the CNSC Review of OPG's Safety Analysis Program Implementation [N-CORR-00531-18039, *Darlington and Pickering NGS: Results of the CNSC Review of OPG's Safety Analysis Program Implementation,* April 15, 2016] CNSC Regulatory Oversight Report for 2015	

Strength ID	Strength Title and Description	Justification	Level
S-07	**Probabilistic Safety Assessment** The PSA program meets or exceeds performance objectives. OPG has developed and implemented a process of maintenance of the probabilistic risk assessment model to ensure that the model is representative of the actual plant configuration and operation and testing at the station.	PSR2 shows that the PSA programs and procedures at OPG are comprehensive, resulting in a systematic and disciplined approach to identifying, prioritizing and addressing any PSA-related issues. The conclusion of the Safety Factor 6 Report is that there are processes to assess the impact of changes in plant design, operation, and plant-specific failure data and to update the PSA to reflect the current plant status. The intent of Safety Factor 6 Report Review Task 2 assessment, which requires confirmation of the existence of processes to assess the impact of changes in plant design, operation and failure data, is met and therefore Pickering NGS is compliant. In addition, a self-assessment concluded that Pickering NGS has good alignment with current governance and best industry practices; recent procedural changes are reflected in the work program; and the Pickering Risk and Reliability Program execution satisfies the intent of the governance. PSA models and results for Pickering have also played a significant role in deriving the Emergency Mitigating Equipment Guidelines and Severe Accident Management Guidelines that are in place to manage accident response for a potential BDBA. The Regulatory Oversight Report for Canadian Nuclear Power Plants: 2015 states that CNSC staff concluded that the “Safety Analysis” Safety and Control Area, including PSA, at Pickering met or exceeded performance objectives and all applicable regulatory requirements, and rated this Safety and Control Area as Fully Satisfactory. **Source(s):** Sections 4.1.1, 4.1.2, and 4.1.3 of PSR2 Safety Factor 6 Report [P-REP-03680-00010-R000, *Pickering NGS PSR2 Safety Factor 6 Report: Probabilistic Safety Assessment*, March 2017] Section B.4 of Code and Standard Reviews Associated with Safety Factors 1, 5, 6, and 7 [P-REP-03680-00029-R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 1, 5, 6, and 7,*	1, 2, 3, 4

Strength ID	Strength Title and Description	Justification	Level
		March 2017] CNSC Regulatory Oversight Report for 2015 CNSC Correspondence on Results of the CNSC Review of OPG's Safety Analysis Program Implementation [N-CORR-00531-18039, *Darlington and Pickering NGS: Results of the CNSC Review of OPG's Safety Analysis Program Implementation,* April 15, 2016]	
S-08	**Operationalization of Probabilistic Safety Assessment** Probabilistic Safety Assessment (PSA) is used to support conduct of engineering, maintenance and operation.	PSA is used to support the conduct of engineering, maintenance and operation at Pickering NGS as follows: • Proposed modifications to plant operation, configuration or procedures that may change the operation of the plant are reviewed to quantify impact on risk and to assess its acceptability. For any modifications that may reduce risk, a PSA is performed to quantify the benefits in terms of impact on risk as in input to decision-making. • PSA assumptions important to safety regarding inspection, testing and maintenance are identified and incorporated into operating and maintenance procedures. • PSA is used to identify accident scenarios. • PSA is used to support in-plant and ex-plant consequence analyses for event sequences beyond the design basis for use in understanding accident progression and management, as allowed by the scope and limitations of the PSA. • Risk information is used in safety decision-making based on the PSA data, models and results. • PSA is used for outage risk assessment and for on-line maintenance risk assessment as required.	1, 2, 3, 4

Strength ID	Strength Title and Description	Justification	Level
		• PSA is used for managing instantaneous risk from discovery issues related to a change in design bases, analysis results, experimental findings or knowledge interpretations. • PSA is used to assess the risk of Equipment Out Of Service and abnormal plant configurations. A Risk Management Matrix is established for Pickering NGS for discussion during daily Station Alignment Meetings. Furthermore, the Regulatory Oversight Report for Canadian Nuclear Power Plants: 2015 states that CNSC staff concluded that the “Safety Analysis” Safety and Control Area, including PSA, at Pickering met or exceeded performance objectives and all applicable regulatory requirements, and rated this Safety and Control Area as Fully Satisfactory. **Source(s):** Section 4.1.2 of PSR2 Safety Factor 6 Report [P-REP-03680-00010-R000, *Pickering NGS PSR2 Safety Factor 6 Report: Probabilistic Safety Assessment,* March 2017] OPG Correspondence, Pickering NGS: Risk Improvement Plan Update [P-CORR-00531-04946, *Pickering NGS: Risk Improvement Plan Update*, February 28, 2017] CNSC Regulatory Oversight Report for 2015	
S-09	**Implementation of Safe Operating Envelope Program** Comprehensive program in place for systematic application of Safe Operating Envelope (SOE) definition, implementation and maintenance (to end of station	SOE is defined by the Operational Safety Requirements, Instrument Uncertainty Calculations and other safety-related limits and system credits that ensure operation within the Safety Analysis Basis. Safe Operating Limits and Conditions of Operability (SOE Limits) bound administrative limits in station operating documentation, which provide plant operators with the information required to ensure safe operation of the plant in conformance with the requirements of the Safety Analysis.	1,2,3

Strength ID	Strength Title and Description	Justification	Level
	life).	Rigorous programs and processes are in place to control and verify design compliance to ensure that the plant design remains within the bounds of the SOE. The SOE program is fully developed and effectively implemented. A compliance framework whereby plant operation within the requirements established as part of the SOE is verified on a regular basis and appropriate corrective actions are initiated upon discovery of plant operation outside of the SOE. While operational limits have always been part of the Operating Policies and Principles, a formal SOE program was not part of the original design or licensing basis for Pickering NGS. Pickering NGS Licence Conditions Handbook (states, “SOE is a new requirements introduced in the current PROL. OPG has had an implemented SOE program since 2012”. PSR2 assessment of CSA N290.15, *Requirements for the Safe Operating Envelope of Nuclear Power Plants*, confirms conformance with the safety-significant requirements of the code. This code is part of on-going CNSC compliance monitoring activities. Furthermore, the Regulatory Oversight Report for Canadian Nuclear Power Plants: 2015 states that CNSC staff concluded that the “Safety Analysis” Safety and Control Area and “Operating Performance” Safety and Control Area at Pickering met or exceeded performance objectives and all applicable regulatory requirements, and rated these Safety and Control Areas as Fully Satisfactory. **Source(s):** Section 4.1.2 of PSR2 Safety Factor 5 Report [P-REP-03680-00009-R000, *Pickering NGS PSR2 Safety Factor 5 Report: Deterministic Safety Analysis*, March 2017] CNSC Regulatory Oversight Report for 2015	

Strength ID	Strength Title and Description	Justification	Level
S-10	**Healthy Safety Culture** Pickering has a healthy Nuclear Safety Culture and a respect for nuclear safety and there is a strong bias to put safety over production. Pickering has formally implemented Nuclear Safety Culture Monitoring Panels for regular monitoring of safety behaviour to promote use of error-free performance tools, risk-informed decision making and questioning attitude in day-to-day activities throughout the organization.	Pickering management is committed to continuously strengthen the safety culture within the Pickering organization. The Safety Factor 10 Report paraphrases a nuclear safety culture assessment as follows: “This assessment consisted of a survey, interviews and field observations…65 on-site interviews and 18 field observations were completed by a team of 17 individuals composed of both internal and external team members. In conclusion, the assessment team determined that Pickering has a healthy nuclear safety culture and a respect for nuclear safety not compromised by production priorities.” This is corroborated by the CNSC noting that OPG has a strong commitment to safety. Pickering’s healthy safety culture is maintained through effective leadership, management and communication of expectations in areas including: • A high level of human performance and nuclear, conventional, radiological and environmental safety performance • A high level of equipment availability • Adequate staff numbers and staff knowledge and capability The Event Free Day Reset (EFDR) indicator is one of the flags OPG uses to identify human performance events. The indicator reflects the effectiveness of management in reducing errors and improving organizational processes and activities to reduce the significance and frequency of human performance events. Performance improvement has been accomplished by effectively identifying human performance events, investigating the causes to determine corrective actions, performing trending and analysis to identify reoccurring and common issue areas and communicating the results throughout all levels of the organization. The associated indicators have continually improved at Pickering NGS. These improvements provide evidence that attention to Safety	1, 2, 3, 4

Strength ID	Strength Title and Description	Justification	Level
		Culture is having tangible benefits, supporting that Safety Culture is a Strength. **Source(s):** Section 4.1.8 of PSR2 Safety Factor 10 Report [P-REP-03680-00014-R000, *Pickering NGS PSR2 Safety Factor 10 Report: Organization, Management System, and Safety Culture,* December 2016]	
S-11	**Relationship with Stakeholders and Public** OPG has fostered a strong relationship with stakeholders and interested public. A longstanding positive relationship with the community is in place to promote communication, education and awareness regarding the role of nuclear power and the overriding priority being placed on safety and environment.	OPG demonstrates open and transparent communication in a timely manner to maintain positive and supportive relationships, as well as the confidence of key stakeholders. OPG is ethical and credible in its relationships with employees, suppliers, customers and the public with whom it does business and in the communities in which it operates. Corporate Stakeholder Relations develops, maintains and implements a public information program that is conducted in accordance with the ethical principles of integrity, excellence and citizenship outlined in the OPG Code of Business Conduct. A highly active stakeholder engagement system is in place in terms of active community engagement, communication and responding to the public on environmental matters and making information on environmental monitoring and emissions data easily accessible for the public, all at a level beyond regulatory requirements. The Pickering Community Advisory Council assists Pickering NGS in identifying and responding effectively to the concerns of the community. Communication activities to support the Pickering Waste Management Facility are undertaken periodically. The Pickering Neighbours Newsletter is issued to the community. In community publications, print ads have been run to promote OPG's Biodiversity and tree planting program. The Pickering Nuclear Information Centre continues to provide public access for further information. OPG/Pickering actively uses social media (e.g., Twitter, Instagram) to connect	4, 5

Strength ID	Strength Title and Description	Justification	Level
		with a wide range of audiences. Other information and updates continue to be made available on www.opg.com. **Source(s):** Pickering Nuclear Community Advisory Council Terms of Reference www.opg.com 2013 Public Hearing on Pickering Licence Renewal	
S-12	**Dose to Public** The dose to the off-site public resulting from operation of the station is very much less than the dose from background radiation.	Safety Factor 14 Report notes this conclusion based on similar text in the externally-prepared Pickering Environmental Risk Assessment. Specifically, the Environmental Risk Assessment states: “Since the dose estimates are a small fraction of the regulatory public dose limit and natural background exposure, no discernible health effects are anticipated due to exposure of potential groups to radioactive releases from the PNGS.” Furthermore, the Regulatory Oversight Report for Canadian Nuclear Power Plants: 2015 states that CNSC staff concluded that the “Radiation Protection” Safety and Control Area at Pickering met or exceeded performance objectives and all applicable regulatory requirements, and rated this Safety and Control Area as Fully Satisfactory. **Source(s):** Section 4.1.7 of PSR2 Safety Factor 14 Report [P-REP-03680-00018-R000, *Pickering NGS PSR2 Safety Factor 14 Report: Radiological Impact on the Environment*, December 2016] Environmental Risk Assessment Report for Pickering Nuclear [P-REP-07010-10012-R000, *Environmental Risk Assessment Report for Pickering Nuclear,* January 2014] CNSC Regulatory Oversight Report for 2015	1, 2, 3, 4, 5

Strength ID	Strength Title and Description	Justification	Level
S-13	**Radiation Exposure Performance** Radiation protection performance exceeds objectives and application of As Low as Reasonably Achievable (ALARA) meets or exceeds regulatory expectations.	Programs and principles are fully developed and effectively implemented. The Safety Factor 9 Report states that "Improvements in radiation exposure performance were realized through the implementation of increased line accountability for the control of radiation exposure, including Departmental Dose Reduction Plans." In addition, the review in the Safety Factor 15 Report has confirmed that Radiation Protection has been adequately accounted for in the design and operation of Pickering NGS, and that Radiation Protection provisions (including design and equipment) provide protection of persons from the harmful effects of radiation and ensure that contamination and radiation exposures and doses to persons are monitored and controlled and maintained ALARA. Assessments of CNSC G-129 and Class I Nuclear Facilities Regulations SOR/2000-203 also confirmed that Pickering NGS conforms with the safety-significant requirements relevant to Radiation Protection. The CNSC Regulatory Oversight Report for Canadian Nuclear Power Plants: 2015 states that OPG continued to implement a highly effective, well documented and mature program, based on industry best practices, to keep doses ALARA at Pickering. Furthermore, the Regulatory Oversight Report for Canadian Nuclear Power Plants: 2015 states that CNSC staff concluded that the "Radiation Protection" and "Operating Performance" Safety and Control Areas at Pickering met or exceeded performance objectives and all applicable regulatory requirements, and rated these Safety and Control Areas as Fully Satisfactory. **Source(s):** Section 4.1.6 of PSR2 Safety Factor 9 Report [P-REP-03680-00013-R000, *Pickering NGS PSR2 Safety Factor 9 Report: Use of Experience from Other Nuclear Power Plants and Research Findings*, October 2016] Section 4.1.4 of PSR2 Safety Factor 15 Report [P-REP-03680-00019-R001,	1, 2, 3, 4

Strength ID	Strength Title and Description	Justification	Level
		Pickering NGS PSR2 Safety Factor 15 Report: Radiation Protection, April 2017] CNSC Regulatory Oversight Report for 2015	
S-14	**Heat Removal Systems** Pickering NGS has made modifications to achieve multiple overlapping heat removal provisions for normal operation and design basis events, and for Beyond Design Basis conditions.	In recognition of modern standards and incorporating OPEX from events such as the 2011 Fukushima accident, Pickering has made modifications to achieve multiple, overlapping heat removal systems of adequate capacity. These heat removal systems, together with their support systems, ensure that heat generated in the Fuel is transferred to the atmosphere or the lake under normal operating and shutdown/outage conditions, as well as in response to DBAs and Beyond Design Basis conditions, which is a requirement for new nuclear power plants. These systems are: • Steam Generators (boilers) supplied by inventory from normal or auxiliary feed water or the Pickering 1,4 Emergency Boiler Water System (EBWS) or Pickering 5-8 Emergency Water System (EWS). • Shutdown Cooling Heat Exchangers supplied by normal or Emergency High Pressure Service Water. • Emergency heat removal with High Pressure ECI and ECI recovery, with Reactor Building Air Coolers. • For events involving a loss of Steam Generator inventory, in the short term, the Boiler Emergency Cooling System (BECS) provides water to both Pickering 1,4 and Pickering 5-8 for reactor decay heat removal by the boilers for accident situations initiated by a failure of the normal feedwater supply. For long-term cooling, the EBWS is designed to provide emergency water to the boilers of Pickering 1,4. In Pickering 5-8, the Shutdown Cooling System is initiated manually to provide the heat sink for the Heat Transport System (HTS). In addition, the EWS is a seismically qualified system supplying long-term emergency makeup to the boilers, HTS, ECI recovery	1, 2, 3, 4

Strength ID	Strength Title and Description	Justification	Level
		heat exchangers, Containment air coolers and the Moderator, for Pickering Units 5-8. • For Beyond Design Basis conditions, which are a requirement for new nuclear power plants, Pickering 5-8 units have emergency heat removal available including EWS to the HTS and Emergency Mitigating Equipment water supply to maintain cooling. Pickering Units 1,4 have Firewater supply or Emergency Mitigating Equipment (EME) water supply to maintain fuel cooling. In Pickering 1,4 and Pickering 5-8, EME provides an additional makeup water supply through the use of portable diesel-powered pumps that can provide make-up to the secondary side of the boilers, to the HTS and to the Moderator. **Sources:** Pickering B Safety Report [NK30-SR-01320-00002-R004, *Pickering B Safety Report – Part 2*, October 10, 2012] Pickering A Safety Report [NA44-SR-01320-00001-R016, *Pickering A Safety Report*, July 20, 2017]	
S-15	**Electrical Power System** Pickering NGS has implemented design modifications to provide standby and Emergency Power Systems to provide the necessary electrical power to maintain the plant in a safe shutdown state and ensure nuclear safety in Design Basis Accidents and Beyond Design Basis conditions,	The originally designed Electrical Power System at Pickering NGS has been enhanced through design modifications such that units are equipped with multiple sources of backup electrical power to ensure that controls and equipment important to safe operation are available during normal and abnormal conditions. These are: • Site Electrical System. This distribution system can be used for transferring backup auxiliary power to any of the unit Class IV systems High Pressure ECI pump motors, ECI associated motorized valves for each of the Pickering 1,4 units, and the electrical services of any single unit that is	1, 2, 3, 4, 5

Strength ID	Strength Title and Description	Justification	Level
	and support Severe Accident Management actions.	deprived of supply from its normal sources). • Standby Generators (SG). This power comes from six independent gas turbine driven generators for Pickering Units 1,4 and six for Pickering Units 5-8. The standby power is dedicated to those loads that are required for the safe shutdown of the reactor and heat sink/cooling of the Fuel core. Additionally, the SG1 of each unit pair can be selected to run a High Pressure ECI pump in an islanded mode. • In addition, the Emergency Power System (EPS) for Pickering 5-8 supplies power to a specific portion of the Safety-Related Systems in the station. The purpose of EPS is to independently perform the critical reactor safety functions, i.e., Control, Cool and Contain on total loss of Group 1 distribution systems. Also, provisions for mitigating a complete loss of onsite and offsite AC power are provided. The purpose of this is to address 'Station Blackout'. At Pickering, this is a BDBA involving total loss of all AC power. Station blackout imposes requirements on stored water systems for heatsinks with make-up from water to boilers using the Auxiliary Power System (APS). The APS is a back-up power supply system that supplies power to selected Class IV loads following a total loss of Class IV power across the Pickering NGS following a failure of the Bulk Electrical System. In addition, Emergency Mitigating Equipment (EME) also supports the response in such events. **Sources:** Pickering B Safety Report [NK30-SR-01320-00002-R004, *Pickering B Safety Report – Part 2*, October 10, 2012] Pickering A Safety Report [NA44-SR-01320-00001-R016, *Pickering A Safety Report*, July 20, 2017]	
S-16	**Human Factors Engineering**	Pickering NGS was originally designed to the standards of the day and designers utilized best design practices in addition to incorporation of	1, 2, 3, 4

Strength ID	Strength Title and Description	Justification	Level
	Program A robust human factors engineering program has been established.	operations and maintenance experience (e.g., for Pickering Units 1-4, experience was incorporated from Douglas Point and the Nuclear Power Demonstration plant, while Pickering Units 5-8 incorporated experience from Pickering Units 1-4 and Bruce NGS A). Operating experience (OPEX) and improvements have been incorporated into the processes and design to improve the human-machine interfaces in many areas (e.g., Control Room annunciation upgrades as a result of changes to computer hardware and operator interface). The Modification Process [N-PROC-MP-0090-R014, *Modification Process*, October 14, 2106] includes a requirement for the preparation of a Design Scoping Checklist [N-FORM-10959-R016, *Design Scoping Checklist*, June 27, 2016], which contains a listing of high-level Human Factors Engineering questions that are designed to identify whether the proposed modification has Human Factors Engineering impact. Since 2000, Human Factors Engineering has been explicitly considered for all design changes at Pickering NGS, resulting in continuous improvements to the human-machine interfaces throughout the plant. In terms of design changes to the Main Control Room and other work stations, when required, Human Factors Engineering Program Plans are prepared in accordance with Guide for OPG Human Factors Engineering Process [N-MAN-06700-10002-R004, *Guide for OPG Human Factors Engineering Process,* December 18, 2015], which describes OPG's Human Factors Engineering processes and approach to the conduct of Human Factors Engineering activities and OPG's expectations for performing Human Factors Engineering activities. In order to improve the human machine interface of the Pickering Units 1,4 Digital Control Computer interface, the Pickering A Control Room Enhancement was initiated. This enhancement is part of the Digital Control Computer system and forms an interface between Main Control Room operators and the Digital Control Computers.	

Strength ID	Strength Title and Description	Justification	Level
		The Human Factors Engineering Program Plan prepared by OPG meets the safety-significant requirements of CNSC G-276 and CNSC G-278, as well as meeting the intent of applicable elements from NUREG-0711. These guides and standards are relatively new, and OPG programs have been updated to meet the requirements of these newer standards. These guides are not mandatory requirements in the Licence Conditions Handbook and hence compliance with these guides is considered a Strength. **Source(s):** Section 4.1.5 of PSR2 Safety Factor 1 Report [P-REP-03680-00008-R000, *Pickering NGS PSR2 Safety Factor 1 Report: Plant Design*, March 2017] Sections B.11, B.12, and B.21 of Code and Standard Reviews Associated with Safety Factors 8, 10, 12, 13, and 14 [P-REP-03680-00021-R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 8, 10, 12, 13, and 14*, December 2016]	
S-17	**Environmental Qualification Program** The Environmental Qualification Program at OPG is fully developed.	The program is fully developed, and interfaces with other station procedures are well identified. This program was not part of the original design or licensing basis for Pickering NGS. The review of Safety Factor 3 has confirmed that the required Pickering NGS equipment important to safety has been properly environmentally qualified. Also, Pickering NGS is assessed as part of PSR2 to be in conformance with the safety-significant requirements of CSA N290.13-05, which, although now a mandatory licence document, was not part of the original Pickering NGS design basis. **Source(s):** PSR2 Safety Factor 3 Report [P-REP-03680-00006-R000, *Pickering NGS PSR2 Safety Factor 3 Report: Equipment Qualification (Seismic and*	1, 2, 3

Strength ID	Strength Title and Description	Justification	Level
		Environmental), July 2016] Section B.1 of Code and Standard Reviews Associated with Safety Factors 2, 3, and 4 [P-REP-03680-00004-R000, *Pickering NGS Periodic Safety Review 2: Code and Standard Reviews for Safety Factors 2 (Actual Condition of SSCs), 3 (Equipment Qualification) and 4 (Aging),* July 2016]	
S-18	**Comprehensive Set of Performance Indicators** A comprehensive set of station performance indicators is in place to monitor operations.	Within OPG, a tiered approach for performance indicators has been implemented based on industry best practice. OPG Nuclear has implemented a top-down and bottom-up approach to business planning where its leaders establish clear performance targets. The leaders then, in subsequent business plans, identify actions and accountabilities required to achieve these targets over a specified period of time. In accordance with PROL 48.03/2018, REGDOC-3.1.1 (identifies specific performance indicators that must be reported to the CNSC on a quarterly basis. OPG records and reports these in the Quarterly Report on Safety Performance Indicators. These indicators are: 1) Collective Radiation Exposure 2) Personnel Contamination Events 3) Unplanned Dose/Unplanned Exposure 4) Loose Contamination Events 5) Environmental Releases – Radiological 6) Spills 7) Mispositioning Index	1, 2, 3, 4, 5

Strength ID	Strength Title and Description	Justification	Level
		8) Number of Unplanned Transients 9) Reactivity Management Index 10) Unit Capability Factor 11) Unplanned Capability Loss Factor 12) Forced Loss Rate 13) Reactor Trip Rate 14) Corrective Maintenance Backlog 15) Deficient Maintenance Backlog 16) Deferral of Preventive Maintenance 17) Safety System Test Performance 18) Preventive Maintenance Completion Ratio 19) Chemistry Index 20) Chemistry Compliance Index (non-Guaranteed Shutdown State and Guaranteed Shutdown State) 21) Conventional Health and Safety 22) Radiological Emergencies Performance Index 23) Emergency Response Organization Drill Participation Index 24) Emergency Response Resources Completion Index 25) Low- and Intermediate-Level Radioactive Solid Waste Generated OPG monitors performance indicators using the Electronic Performance Reporting system to measure the performance of the plant against the business planning goals and targets. Business Plan targets are set based on	

Strength ID	Strength Title and Description	Justification	Level
		extensive industry benchmarking with the objective of achieving top quartile performance. The Electronic Performance Reporting system standardizes data collection, analysis, and the performance indicator reporting process. It tracks performance measure targets and results in a single repository. Monthly results are rated and colour coded based on performance results versus target. The system allows trending and development of corrective action plans and promotes ownership and accountability of staff in improving performance. Furthermore, the Regulatory Oversight Report for Canadian Nuclear Power Plants: 2015 states that CNSC staff concluded that the “Operating Performance” Safety and Control Area at Pickering met or exceeded performance objectives and all applicable regulatory requirements, and rated this Safety and Control Area as Fully Satisfactory. **Source(s):** Section 4.1.4 and Section 4.1.5 of PSR2 Safety Factor 8 Report [P-REP-03680-00012-R000, *Pickering NGS PSR2 Safety Factor 8 Report: Safety Performance,* December 2016] CNSC Regulatory Oversight Report for 2015	
S-19	**Use of Operating Experience and Research Findings** Rigour in the identification and application of OPEX and of research findings, and support of a senior advisory team for Research & Development issue identification and action implementation.	OPG has established procedures for sending and receiving experience relevant to safety from other nuclear power plants and relevant non-nuclear plants Operating Experience Process [N-PROC-RA-0035 R019*, Operating Experience Process,* September 14, 2016]. A weekly CANDU Owners Group (COG) OPEX screening meeting, facilitated and administered by COG, serves as an initial screening forum to review event reports from CANDU stations, nuclear industry and non-nuclear sources for applicability and significance to CANDU units. Committee members include representatives from all CANDU facilities (including OPG Pickering), vendors, research organizations and WANO.	1, 2, 3, 4, 5

Strength ID	Strength Title and Description	Justification	Level
		The PSR2 has confirmed for Pickering NGS that there is good feedback of relevant experience from other nuclear power plants and from findings of research, and that this is used to introduce reasonable and practicable safety improvements at the plant or in the operating organization. Moreover, research activities are pursued and results are used to enhance nuclear safety and equipment performance and reliability. Pickering NGS shares its programs, procedures, Compliance Reporting and Corrective Action processes as commonly maintained programs with Darlington NGS, and thus observations and lessons learned at Darlington NGS are used at Pickering NGS and vice versa. Furthermore, the PSR2 compliance assessment of regulatory mandated standard CSA N286-12, an element of which is OPEX, confirmed conformance with the safety-significant requirements. **Source(s):** CNSC Correspondence on PSR2 - CNSC Staff Assessment of OPG SFR9, SFR11, and SFR15 [P-CORR-00531-04950, *Pickering NGS: PSR2 - CNSC Staff Assessment of OPG SFR9, SFR11, and SFR15*, January 25, 2017] Section A.1 of Code and Standard Reviews Associated with Safety Factors 9, 11, and 15 [P-REP-03680-0586480-R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 9, 11, and 15*, September 2016]	
S-20	**Minimum Staff Complement Management** An online computer program for monitoring staffing level for minimum complement roles, as well as forecasting future staffing	The use of tools available on the local area network to ensure current and future staffing needs demonstrates maturity of this process. This is supported by compliance with the safety-significant requirements of CNSC G-323, Ensuring the Presence of Sufficient Qualified Staff at Class I Nuclear Facilities – Minimum Staff Complement, which is referenced in the Pickering NGS Licence Conditions Handbook (as guidance.	1, 2, 3, 4

Strength ID	Strength Title and Description	Justification	Level
	needs, thereby ensuring minimum complement coverage, is effectively managed on a continuous basis.	Furthermore, the Regulatory Oversight Report for Canadian Nuclear Power Plants: 2015 states that CNSC staff concluded that the “Operating Performance” Safety and Control Area at Pickering met or exceeded performance objectives and all applicable regulatory requirements, and rated this Safety and Control Area as Fully Satisfactory. **Source(s):** Section 4.1.1 of PSR2 Safety Factor 12 Report [P-REP-03680-00016 R000, *Pickering NGS PSR2 Safety Factor 12 Report: Human Factors,* December 2016] Section B.10 of Code and Standard Reviews Associated with Safety Factors 8, 10, 12, 13, and 14 [P-REP-03680-00021 R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 8, 10, 12, 13, and 14*, December 2016] CNSC Regulatory Oversight Report for 2015	
S-21	**Training** A mature training and certification program is in place that includes a rigorous and comprehensive training program for selecting and training candidates.	According to the compliance assessment of regulatory requirements of REGDOC-2.2.2, Personnel Training, training of personnel is a critical and well-established aspect of the OPG Nuclear program. The discussion of training in the Safety Factor 12 Report expresses the comprehensiveness of the Systematic Approach to Training program. Pickering NGS is compliant with RD-204, Certification of Persons Working at Nuclear Power Plants. The most recent Fleetview Program Health and Performance Report for 2015 [N-REP-08130-0635343 R000, *Fleetview Program Health and Performance Report (Q1/2016 to Q4/2016)*, January 2017] reported with respect to Human Performance and Training that the training program aligned with industry best practices.	1, 2, 3, 4, 5

Strength ID	Strength Title and Description	Justification	Level
		Source(s): Section 4.1.3 of PSR2 Safety Factor 12 Report [P-REP-03680-00016-R000, *Pickering NGS PSR2 Safety Factor 12 Report: Human Factors*, December 2016] Sections B.6 and B.17 of Code and Standard Reviews Associated with Safety Factors 8, 10, 12, 13, and 14 [P-REP-03680-00021-R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 8, 10, 12, 13, and 14,* December 2016]	
S-22	**Emergency Management** A mature emergency response infrastructure is in place, and the requisite qualified manpower and expertise are maintained. As well, a mutual aid emergency support agreement among the Canadian nuclear operators has been established for inter-utility emergency support. Pickering has a well-established Severe Accident Management Program.	OPG has comprehensive operating procedures that apply to normal, abnormal and emergency conditions (including Transients, DBAs, post-accident conditions and BDBAs). Pickering NGS operations staff operates the plant through the use of approved Operating Manuals, Operating Procedures, Alarm Response Manuals and Safety-Related System Tests, and uses these manuals to respond to any abnormal conditions. If it is not possible to address an abnormal condition through the system Operating Manuals and/or Alarm Response Manuals, upon confirmation or diagnosis of emergency conditions, including DBAs, staff respond per the event-based Abnormal Incident Manuals. Emergency Mitigating Equipment Guidelines and Severe Accident Management Guidelines are used to respond to BDBAs, including severe accidents. Severe Accident Management Guidelines are written guidance to implement strategies should a BDBA progress to a severe accident. In addition, Pickering has implemented an Automated Source Term Gamma Monitoring System that meets provincial regulations for timely notification of environmental hazards presented following an event that has the risk of	3, 4, 5

Strength ID	Strength Title and Description	Justification	Level
		disseminating radioactive contamination surrounding the plant. Through the evolution of earlier guides to the present version of REGDOC-2.3.2, Accident Management, OPG has developed and validated an effective suite of procedures and guidance as discussed above. In particular, the guidance for BDBAs, coupled with the advanced installation of Emergency Mitigating Equipment, demonstrates a commitment to fulfilling the requirements of defence-in-depth. Further, given the conformance with the safety-significant requirements of REGDOC-2.3.2, this is a Strength. The PSR2 has confirmed that Pickering NGS has adequate plans, staff, facilities and equipment in place for dealing with emergencies, and that there are adequate arrangements in place for regular emergency training and exercises, and interaction and coordination with local and national authorities. The REGDOC-2.10.1, Nuclear Emergency Preparedness and Response compliance assessment noted that there is a transition plan in place and that findings are already being addressed. REGDOC-2.10.1, a new standard currently not in the Licence Conditions Handbook, includes recommendations made by the CNSC Fukushima Task Force and the External Advisory Committee to strengthen Canadian Licensees' emergency preparedness programs. Also, the PSR2 assessment confirmed that Pickering NGS is in conformance with the safety significant requirements in CSA N1600-2014, General Requirements for Nuclear Emergency Management Programs, which is a new standard. **Source(s):** Section 4.1.11 of PSR2 Safety Factor 13 Report [P-REP-03680-00017-R000, *Pickering NGS PSR2 Safety Factor 13 Report: Emergency Planning*, December 2016] Table 1 in Fukushima Action Item Status Report [N-REP-03600-10003-R007, *Fukushima Action Item Status Report*, November 2015]	

Strength ID	Strength Title and Description	Justification	Level
		Section B.18 of Code and Standard Reviews Associated with Safety Factors 8, 10, 12, 13, and 14 [P-REP-03680-00021-R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 8, 10, 12, 13, and 14,* December 2016]	
S-23	**Environmental Protection Program** The Environmental Protection Program is robust.	The PSR2 has confirmed that Pickering NGS has an effective program for monitoring the radiological impact of the plant on the environment; the program ensures that emissions are properly controlled and are ALARA. PSR2 assessments against codes and standards show that OPG is compliant with the safety-significant requirements of the following modern codes and standards related to environmental protection, and this is considered a Strength: • CNSC REGDOC-2.9.1, Environmental Protection Policies, Programs and Procedures – supersedes S-296, which is a mandatory standard in the Pickering NGS Licence Conditions Handbook. • CSA N288.1-14, Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities – a mandatory standard in the Pickering NGS Licence Conditions Handbook. • CSA N288.4-10, Environmental Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills – a mandatory standard in the Pickering NGS Licence Conditions Handbook. • CSA N288.5-11, Effluent Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills – a mandatory standard in the Pickering NGS Licence Conditions Handbook CSA N288.6-12, Environmental Risk Assessments at Class I Nuclear Facilities and Uranium Mines and Mills – not a mandatory standard in the Pickering NGS Licence Conditions Handbook.	1, 2, 3, 4, 5

Strength ID	Strength Title and Description	Justification	Level
		• CSA N288.7-15, Groundwater Protection Programs at Class I Nuclear Facilities and Uranium Mines and Mills – the standard is not identified in the Pickering NGS Licence Conditions Handbook. In addition, the Environmental Management Systems program at Pickering NGS has been developed and implemented in accordance with the elements of the ISO 14001 Standard, *Environmental Management Systems – Requirements with Guidance for Use*. Also the program is registered under the ISO 14001 Standard, and is therefore required to be continually improved in accordance with this Standard. **Source(s):** Section 4.2, Table 3 of Safety Factor 14 Report [P-REP-03680-00018-R000, *Pickering NGS PSR2 Safety Factor 14 Report: Radiological Impact on the Environment*, December 2016] Code and Standard Reviews Associated with Safety Factors 8, 10, 12, 13, and 14 [P-REP-03680-00021-R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 8, 10, 12, 13, and 14*, December 2016] Code and Standard Reviews Associated with Safety Factors 9, 11, and 15 [P-REP-03680-0586480-R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 9, 11, and 15*, September 2016]	
S-24	**Advanced Technology to Support Radiation Protection** Remotely controlled technology in place to simulate radiation environments and hazards is providing a safe learning	Pickering NGS has implemented technological improvements in support of radiological protection that include: • Use of robotics to perform tasks in radioactive work areas, reducing radiation exposure and therefore dose to workers; • Use of dynamic learning activities to provide workers an opportunity to	1, 2, 3, 4

Strength ID	Strength Title and Description	Justification	Level
	environment for training and practicing radiation protection procedures.	practice radiation protection fundamentals in a simulated radioactive work environment using remotely controlled technology; • Implementation of remote reading radiation detection instrumentation and real time data transmission to facilitate improved job planning and awareness of current radiological conditions; and • Implementation of a gamma ray imaging spectrometer to perform enhanced radiation surveys, to identify areas with elevated dose rates, enabling more effective shielding to reduce dose to workers. The Regulatory Oversight Report for Canadian Nuclear Power Plants: 2015 states that OPG continued to implement a highly effective, well documented and mature program, based on industry best practices, to keep doses ALARA at Pickering. **Source(s):** PSR2 Safety Factor 15 Report [P-REP-03680-00019-R001, *Pickering NGS PSR2 Safety Factor 15 Report: Radiation Protection*, April 2017] CNSC Regulatory Oversight Report for 2015	

Appendix F – Proposed Global Issue Resolution Statement Summaries

Resolution Statement #	Resolution Statement Summary	Defence-in-Depth Level
GI-1-RS1	Complete CSA N285.8 Compliance Plan activities, including responding to comments.	1,2,3
GI-1-RS2	Review and revise if/as required the CSA N285.4 compliant Periodic Inspection Plans for Fuel Channels for Pickering NGS to cover the extended operating period.	1,2,3
GI-1-RS3	Update the Fuel Channels Life Cycle Management Plan for Pickering 1,4 for the extended operating period.	1,2,3
GI-1-RS4	Update the structure of the Fuel Channels LCMP.	1,2,3
GI-2-RS1	Update the Feeders Life Cycle Management Plan for Pickering 1,4 for the extended operating period based on updated fitness-for-service assessment.	1,2,3
GI-3-RS1	Update the Steam Generators Life Cycle Management Plan for Pickering 1,4 for the extended operating period based on updated fitness-for-service assessment.	1,2,3
GI-4-RS1	Update the Reactor Components and Structures Life Cycle Management Plan for Pickering 1,4 for the extended operating period based on updated fitness-for-service assessment.	1,2,3
GI-4-RS2	Perform measurements of Calandria Tube/Liquid Injection Shutdown System nozzle gaps on Units 5-8 to refine the gap closure rates. Using this new measurement data, update analyses as required, to demonstrate Fitness for Service.	1,2,3
GI-5-RS1	Confirm the adequacy of the service limits assessments for Nuclear Class 1 piping (excluding Major Components) after accounting for impact of environmental factors.	1,2,3

Resolution Statement #	Resolution Statement Summary	Defence-in-Depth Level
GI-6-RS1	Reassess the impact of the changes in the cable criticality coding and update the scope of the cable surveillance plan.	1, 2, 3, 4
GI-7-RS1	Update the buried piping program asset management plan and risk ranking for the extended operating period.	3, 4
GI-7-RS2	Update governance to reflect a graded approach in the event that leakage in fuel oil piping occurs.	1
GI-8-RS1	Complete and update Condition Assessments for the piping systems and commodity groups in PSR2 scope for station operation for the extended operating period.	1, 2, 3
GI-8-RS2	Develop and implement a process to track and report aging-management-related actions from the Condition Assessment recommendations.	1
GI-9-RS1	Complete the required assessment to support the current fuel basket stacking arrangements in the Pickering NGS IFBs.	1
GI-10-RS1	Complete the Pickering 5-8 IFB Leakage Mitigation Project to mitigate leaks from IFB-B to the interspace.	1
GI-12-RS1	Complete EQ Assessment re-assessments to support the extended operating period.	3
GI-19-RS1	Demonstrate the fitness for service of the foundation steel H-piles for the Pickering A Reactor Building, Vacuum Building and Pressure Relief Duct at the Pickering site for the extended operating period.	1, 2, 3, 4
GI-24-RS1	Update Heat Transport System aging safety analysis models and perform the required safety analysis of events most impacted by aging (small-break LOCA, loss of flow and neutron overpower) to support extended operation.	2, 3
GI-25-RS1	Complete the re-categorization of the large-break LOCA CANDU Safety Issues to Category 2.	3

Resolution Statement #	Resolution Statement Summary	Defence-in-Depth Level
GI-25-RS2	Complete the re-categorization of CANDU Safety Issue CSI-IH6 for Pickering to Category 2. (Pickering 1,4 high-energy piping)	3
GI-26-RS1	Complete the emergency response projection enhancements identified in Action Item 2016-OPG-7469: Implementation of Emergency Response Projection Computer Code Upgrades.	5
GI-27-RS1	Complete actions from PSA improvement plan.	3, 4
GI-27-RS2	Investigate and implement additional practicable design, operational and/or analytical enhancements to further improve Pickering 1,4 Severe Core Damage Frequency and Large Release Frequency.	4
GI-31-RS1	Complete the Pickering NGS Implementation Plan for CNSC REGDOC-2.4.1.	2, 3
GI-31-RS2	Prepare Implementation Plan update for CNSC REGDOC-2.4.1, including consideration of the impact of the extended operating period.	2, 3
GI-32-RS1	Complete the activities in the CNSC REGDOC-2.4.2 Implementation Strategy and update the Strategy in the context of the extended operating period.	2, 3
GI-40-RS1	Ensure the completion of Emergency Mitigating Equipment Phase 2 activities.	3, 4
GI-43-RS1	Perform the scope of inspections for non-Containment safety-significant civil structures as per the established preventive maintenance program (PM 00121151).	1, 2, 3
GI-43-RS2	Develop program governance using a risk-based approach for aging management of safety-significant civil structures for the extended operating period. This applies to non-Containment safety-related civil structures.	1, 2, 3
GI-43-RS3	Prepare Condition Assessments as appropriate for safety-significant civil structures for the extended operating period. Recommendations from these Condition Assessments will be tracked and reported along with those related to GI-8. This applies to non-Containment safety-related civil structures.	1, 2, 3

Resolution Statement #	Resolution Statement Summary	Defence-in-Depth Level
GI-47-RS1	Complete installation of locks on the 058 Yard Fire Protection System.	2, 3
GI-48-RS1	Provide, as necessary, design and/or operational changes and commissioning/testing to facilitate required interconnection of Pickering 1,4 and Pickering 5-8 Fire Protection System water supplies to meet the safety intent of CSA N293-12 Clause 7.3.2.2 (d).	2, 3
GI-50-RS1	Revise the N285.4 PIPs and governance to align with elements of N285.4-14	1, 2, 3
GI-50-RS2	Assess the impact of extended operation on concessions against CSA N285.4	1

Appendix G – Grouping of PSR1 Acceptable Deviations

This appendix groups related and similar Acceptable Deviations (ADs) from PSR1 that have been identified as also being ADs for the extended operation period [P-CORR-03680-0620816-R000, *Re: PSR1 Acceptable Deviations Reassessed for Pickering PSR2*, March 31, 2017] for impact on PSR2. Acceptable Deviations grouped in this appendix are assessed using a combination of common cause and safety principles to determine which engineered and/or administrative safety-related barriers could potentially be impacted. The barriers considered are SSCs or programs that were specifically designed to minimize the likelihood and/or minimize the consequences of events. The PSR1 AD groups in this appendix are carried forward to the assessment in Appendix H.

PSR1 ADs that are assessed to not have an impact, such as legacy or minor document revisions, are grouped in this appendix under PSR1-AD-DOC and are not assessed for aggregate effects in Appendix H as their aggregate impact is deemed not significant.

PSR1 ADs that have been addressed or are not applicable to Pickering NGS are grouped in PSR1-AD-NFA (No Further Action) in this appendix and are not assessed further for aggregate effect in Appendix H.

AD #	PSR1 AD Title	Group #	PSR1 Grouped AD Title, Description and Rationale
46	CSA N289.2-M81 (R2003) PB ISR Review, General	PSR1-AD1	**Seismic Qualification of SSCs - Gaps against N289 Requirements** These PSR1 ADs involve issues relating to assessing seismicity at the Pickering NGS. These ADs are related to accounting for soil-structure interaction effects including those due to differences between modelling techniques and analysis approaches for the evaluation of site response analysis for the free-field case and complete interaction versus substructure technique for soil/foundation/structure systems. However, the Seismic PSAs [P-REP-03611-00006-R00, *Pickering NGS PSA Update to Include Enhancements from the Fukushima Integrated Action Plan*, April 30, 2014] confirm that the Pickering NGS structures meet the intent of the clause.
47	CSA N289.2-M81 (R2008) DNGS ISR Review, Clause 4.2.1.1, Issue D415, “Soil-Structure Interactions”		
48	CSA N289.3-M81 DNGS ISR Review, Issue D414, “Time Histories”		
49	CSA N289.3-M81 DNGS ISR Review, Issue D415, “Soil-Structure Interactions”		

AD #	PSR1 AD Title	Group #	PSR1 Grouped AD Title, Description and Rationale
			The ADs are associated with Safety Principles S-136, External Factors Affecting the Plant, D-177, Dependent Failures, and D-182, Equipment Qualification.
59	CNSC G-278 (2003) PB ISR Review, Clause 6.3, “Validation of Design”	PSR1-AD2	**Human Factors Verification and Validation Related Issues** Pickering NGS B ISR raised a Gap on the basis that for smaller modifications, explicitly documenting each validation element was not judged necessary. These PSR1 ADs involve Human Factors and Human Machine Interaction requirements in CNSC Guidance documents. They are associated with Safety Principles O-284, Operational Limits and Conditions, and O-278, Training.
60	CNSC G-278 (2003) PB ISR Review, Clause 6.3.1, “Approach”		
61	CNSC G-278 (2003) PB ISR Review, Clause 6.3.2, “Location”		
62	CNSC G-278 (2003) PB ISR Review, Clause 6.3.3, “Technique and Tools”		
63	CNSC G-278 (2003) PB ISR Review, Clause 6.3.4, “Participants”		
64	CNSC G-278 (2003) PB ISR Review, Clause 6.3.5, “Participant Training”		
65	CNSC G-278 (2003) PB ISR Review, Clause 6.3.6, “Performance Measurement in Validation”		
66	CNSC G-278 (2003) PB ISR Review, Clause 6.3.7, “Data Collection and Analysis”		
158	N290.1-M80 PARTS Review, Clause 4.4.8.1, “System-Operator Interface”		
226	IAEA NS-R-1 PB ISR Review, Clause I.7, “Internal Events”		

AD #	PSR1 AD Title	Group #	PSR1 Grouped AD Title, Description and Rationale
95	N287.1-93 (R2004) PB ISR Review: Clause 3.1.1, "Classification of Class Containment"	PSR1-AD3	**Requirements of N287.1-93 (R2004) – General Requirements for Concrete Containment Structures** These ADs involve general requirements for the design of Concrete Containment Structures. The original and subsequent revisions of the CSA N287 series of standards were issued after completion of design and construction of Pickering NGS. Therefore, the details related to general requirements of Containment structure, parts, materials, design, fabrications, construction, inspection examination, testing and commissioning in accordance with the series of standards of CSA N287 were not available. At that time, the Concrete Containment Structures were designed to meet the requirements of the National Building Code of Canada (NBCC). These ADs are associated with Safety Principles D-217, Confinement of Radioactive Material, and C-255, Verification of Design and Construction.
96	N287.1-93 (R2004) PB ISR Review: Clause 3.3, "Jurisdictional Boundaries"		
97	N287.1-93 (R2004) PB ISR Review: Clause 4.1.2.1, "Design Specifications"		
98	N287.1-93 (R2004) PB ISR Review: Clause 4.1.2.4, "Site Seismicity"		
99	N287.1-93 (R2004) PB ISR Review: Clause 4.1.3, "Quality Assurance"		
100	N287.1-93 (R2004) PB ISR Review: Clause 4.2.1, "Designer's Responsibility"		
101	N287.1-93 (R2004) PB ISR Review: Clause 5.1, "Design Specifications"		
102	N287.1-93 (R2004) PB ISR Review: Clause 5.3, "Drawings"		
103	N287.1-93 (R2004) PB ISR Review: Clause 5.4, "Design/Stress Report"		
104	N287.1-93 (R2004) PB ISR Review: Clause 7.1.1, "General"		
105	N287.1-93 (R2004) PB ISR Review: Clause 7.1.3, "General"		

AD #	PSR1 AD Title	Group #	PSR1 Grouped AD Title, Description and Rationale
106	N287.1-93 (R2004) PB ISR Review: Clause 7.2, "Quality Verification"		
107	N287.1-93 (R2004) PB ISR Review: Clause 7.3, "Qualification of Inspection Personnel"		
108	N287.1-93 (R2004) PB ISR Review: Clause 7.4, "Quality Assurance Records"		
109	N287.1-93 (R2004) DNGS ISR Review, Issue D358, "Lack of Certified Mill Test Report"		
44	CSA N287.2-08 DNGS ISR Review: Issue D299, "Anchorage Requirements for Safety-Related Structures"	PSR1-AD4	**Requirements of N287.3-93 (R2004) – Design Requirements for Concrete Containment Structures** These ADs involve specific requirements for the design of Concrete Containment Structures. This group of ADs applies to initial construction, installation, and fabrication of mechanical splices for Concrete Containment Structures. The original and subsequent revisions of the CSA N287 series of standards were issued after completion of design and construction of Pickering NGS. The PSR2 assessment confirms that the rationale for classifying these issues as AD in PSR1 remains valid for PSR2 for Pickering 1,4 and Pickering 5-8. These ADs are associated with Safety Principle D-217, Confinement of Radioactive Material.
110	N287.3-93 (R2004) PB ISR Review: Clause 3.2.1, "Requirements"		
111	N287.3-93 (R2004) PB ISR Review: Clause 3.2.2, "Requirements"		
112	N287.3-93 (R2004) PB ISR Review: Clause 4.1.1, "Details of Reinforcement, General"		
113	N287.3-93 (R2004) PB ISR Review: Clause 4.1.2, "Details of Reinforcement, General"		
114	N287.3-93 (R2004) PB ISR Review: Clause 4.2, "Concrete Protection for Reinforcement"		

AD #	PSR1 AD Title	Group #	PSR1 Grouped AD Title, Description and Rationale
115	N287.3-93 (R2004) PB ISR Review: Clause 4.3, “Continuity of Reinforcement”		
116	N287.3-93 (R2004) PB ISR Review: Clause 4.4.1, “Reinforcement for Crack Control”		
117	N287.3-93 (R2004) PB ISR Review: Clause 4.4.4, “Reinforcement for Crack Control”		
118	N287.3-93 (R2004) PB ISR Review: Clause 5.1.1, “Analysis and Design, General”		
119	N287.3-93 (R2004) PB ISR Review: Clause 5.1.2, “Analysis and Design, General”		
120	N287.3-93 (R2004) PB ISR Review: Clause 5.2, “Loading”		
121	N287.3-93 (R2004) PB ISR Review: Clause 5.3.1, “Foundations”		
122	N287.3-93 (R2004) PB ISR Review: Clause 5.4.1, “Elements and Components”		
123	N287.3-93 (R2004) PB ISR Review: Clause 5.4.3, “Elements and Components”		
124	N287.3-93 (R2004) PB ISR Review: Clause 5.4.4, “Elements and Components”		

AD #	PSR1 AD Title	Group #	PSR1 Grouped AD Title, Description and Rationale
125	N287.3-93 (R2004) PB ISR Review: Clause 5.4.5, “Elements and Components”		
126	N287.3-93 (R2004) PB ISR Review: Clause 6, “Strength and Serviceability”		
127	N287.3-93 (R2004) PB ISR Review: Clause 7.1, “General”		
128	N287.3-93 (R2004) PB ISR Review: Clause 7.2, “Extreme Fibre Strains”		
129	N287.3-93 (R2004) PB ISR Review: Clause 8.1.1, “Shear and Torsion, General”		
130	N287.3-93 (R2004) PB ISR Review: Clause 8.1.2, “Shear and Torsion, General”		
131	N287.3-93 (R2004) PB ISR Review: Clause 9.1, “General”		
132	N287.3-93 (R2004) PB ISR Review: Clause 9.2.2, “Welded Splices and Welded Rebar Connections”		
133	N287.3-93 (R2004) PB ISR Review: Clause 11, “Seismic Design, General”		
134	N287.3-93 (R2004) PB ISR Review: Clause 12, “Nonmetallic Liners, General”		

AD #	PSR1 AD Title	Group #	PSR1 Grouped AD Title, Description and Rationale
135	N287.3-93 (R2004) PB ISR Review: Clause 13(b), (c), “Metallic Parts”		
136	N287.3-93 (R2004) PB ISR Review: Clause 13.3.5.1, “Welding”		
137	N287.3-93 (R2004) PB ISR Review: Clause 14, “Anchorage Systems”		
138	N287.3-93 (R2004) DNGS ISR Review, Issue D014, “Anchorage System Requirements in Concrete Containment Structures”		
139	N287.3-93 (R2004) DNGS ISR Review, Issue D017, “Redundancy of Mechanical Splices in Concrete Containment Structures”		
140	N287.3-93 (R2004) DNGS ISR Review, Issue D018, “Maximum Concrete Tensile Stresses”		
151	N290.1-M80 PARTS Review, Clause 4.1.1, “Independence Between Two Shutdown Systems, General”	PSR1-AD5	**CSA N290.X-M80 - Requirements for the Shutdown Systems and Regulating System** These ADs involve requirements for separation, independence and diversity in the design of the Shutdown Systems and are relevant only to Units 1,4. SDSA and SDSE in Pickering 1,4 are independent trip logic components of the reactor protective system. The two systems are physically and operationally independent of each other and process and control systems. They are diverse in design to the extent that their logic components are supplied by different manufacturers. These ADs are associated with Safety Principle D-200, Automatic
152	N290.1-M80 PARTS Review, Clause 4.1.2.1, “Physical Independence”		
153	N290.1-M80 PARTS Review, Clause 4.1.2.3, “Physical Independence”		
154	N290.1-M80 PARTS Review, Clause 4.1.3.1, “Functional and Conceptual Independence”		

AD #	PSR1 AD Title	Group #	PSR1 Grouped AD Title, Description and Rationale
155	N290.1-M80 PARTS Review, Clause 4.1.3.4, "Functional and Conceptual Independence"		Shutdown Systems, and are therefore grouped together.
156	N290.1-M80 PARTS Review, Clause 4.4.4.1, "Selection of Trip Sensors"		
157	N290.1-M80 PARTS Review, Clause 4.4.3.3, "Selection of Trip Sensors"		
185	CSA N290.4-M82 PARTS Review, Clause 4.1, "Specific Requirements"		
275	CNSC R-9 PARTS, Clause 3.5.2, "Independence From Process Systems"	PSR1-AD6	**CSA N290.2 – Requirements for Emergency Core Cooling Systems** These ADs involve requirements for independence and redundancy in the design of the Emergency Core Cooling System. The issues associated with these ADs are specific to Pickering Units 1,4. The Pickering 1,4 ECI System utilizes the Moderator System in its low pressure mode, and a number of singletons existed in the design of the ECI System. The issues were assessed as ADs based on the improvements made to the design to reduce the commonality between the Moderator System and ECI recovery, and to address some of the singletons. These design changes provided the greatest benefit. The ADs are associated with Safety Principle D-207, Emergency Heat Removal.
276	CNSC R-9 PARTS, Clause 3.4.3, "Redundancy"		
25	CSA N287.7-96 (R2005) PB ISR Review: Clause 4.1.5, "General Requirements"	PSR1-AD7	**CSA N287.5 and N287.7 – Examination and Testing Requirements for Design of Concrete Containment Structures** These PSR1 ADs involve requirements for Containment capability in response to accidents. Clause 3.4.2 of R-7 identifies that the negative design pressure of Containment must not be greater than that predicted in the Safety Report for a set of postulated events with failure of dousing. It was not
141	N287.5-93 DNGS ISR Review, Issue D010, "Changes to Examination and Testing Requirements for Concrete Containment Structures"		

AD #	PSR1 AD Title	Group #	PSR1 Grouped AD Title, Description and Rationale
177	CNSC R-7 PARTS Review, Clause 5.1.1, “Pressure Proof Tests”		considered necessary to evaluate event combinations involving failure of dousing for negative Containment design pressure impacts, because these event combinations result in higher rather than lower Containment pressure. These ADs are associated with Safety Principle D-217, Confinement of Radioactive Material.
168	CNSC R-7 PB ISR Review, Clause 3.4.2, “Structural Integrity”		
176	CNSC R-7 PARTS Review, Clause 3.10.3, “Containment Atmosphere Control”	PSR1-AD8	**Requirements for Containment System Leakage Performance** These PSR1 ADs are related to Containment leakage testing. A positive pressure proof test was performed for Pickering NGS A, but a negative pressure test was not performed during commissioning. These ADs are associated with Safety Principle D-217, Confinement of Radioactive Material.
178	CNSC R-7 PARTS Review, Clause 2 of Appendix A, “Isolation”		
179	CNSC R-7 DNGS ISR Review, Issue D015, “Requirements for Penetrations of the Containment Structure”		
181	CSA N290.3-11 DNGS ISR Review, Issue D613, “Containment Boundary Leakage”		
182	CSA N290.3-11 DNGS ISR Review, Issue D607, “Severe Accident and Beyond Design Basis Accident (BDBA) Design/SAMG)”, Gap 02294 only	PSR1-AD9	**N290.3 - Hydrogen Monitoring Requirements** This PSR1 AD involves hydrogen monitoring requirements related to severe accidents. Clause 10.2.3 of N290.3-11 states: “New builds shall monitor the conditions identified according to the requirements of Clause 10.2.2 [... (e.g., pressure, temperature, and hydrogen concentration inside Containment) that need to be monitored for BDBAs ...]”. The absence of hydrogen monitoring instrumentation is acceptable, given installed hydrogen mitigation equipment together with SAMGs provide means to safely manage any hydrogen in Containment. This AD is associated with Safety Principle D-221, Protection of Confinement Structure.

AD #	PSR1 AD Title	Group #	PSR1 Grouped AD Title, Description and Rationale
187	CSA N290.4-M82 PARTS Review, Clause 4.3.6.1, “Flux Distribution”	PSR1-AD10	**N290.4 - Requirements for Reactor Regulating System (RRS)** These PSR1 ADs involve requirements for RRS related to Start-Up Instrumentation. These issues were identified during the PARTS review and are related to the requirement for the operator to initiate monitoring of the flux tilt and initiate power reduction as needed. Collectively, these ADs are associated with Safety Principle D-205, Startup, Shutdown and Low Power Operation, and D-230 Preservation of Control Capability.
191	CSA N290.5-M90 PARTS Review, Clause 5.4.4.2, "Number of Rectifiers"	PSR1-AD11	**N290.5 - Requirements for Electrical Power and Instrument Air Systems** These ADs involve reliability requirements for the power and other support systems design. Per Regulatory Document RD-98, reliability targets are set for Systems Important to Safety. The Systems Important to Safety are monitored to meet their reliability target. Since the reliability of each system has been demonstrated to be acceptable with its current design, the Acceptable Deviation remains applicable. These ADs are associated with Safety Principle D-174, Reliability Targets.
192	CSA N290.6-M82 PB ISR Review, Clause 5.7.1, "Maintenance and Calibration Capability"	PSR1-AD12	**N290.6 - Requirements for Post-Accident Monitoring** These PSR1 ADs involve requirements for post-accident monitoring design. Gaps were identified on the basis that should a component in the post-accident monitoring system fail, it may not be immediately repairable, and there may not be capability to perform post-accident calibration, depending on the component’s location and the accident scenario. These gaps were assessed to be ADs based on the sufficiency of instrument loops that would be accessible post-accident, because there is sufficient redundancy, and because calibrations during
193	CSA N290.6-M82 PB ISR Review, Clause 5.7.3, "Maintenance and Calibration Capability"		
194	CSA N290.6-M82 PB ISR Review, Clause 5.12.1, "Standards for		

AD #	PSR1 AD Title	Group #	PSR1 Grouped AD Title, Description and Rationale
	Mechanical and Pressure-Retaining Piping Components"		post-accident mission are not required. Also, regarding (AD #194), instrument tubes coming out of Feeder cabinets have less than 18” separation. However, the instrument tubes are protected by barriers where the separation is less than 18" [DGS-30-68000-002, *Design Guide Supplement, GP1 Inst Tubing Separation in F/M VAULT/5/63335/63720*, February 1980], and therefore, this issue is assessed as an AD in Pickering B ISR. These ADs are associated with AM-326 Engineered Features of Accident Management.
196	CSA N291-08 DNGS ISR Review: Issue D283, "Design Requirements for Steel Safety-Related Structures"	PSR1-AD13	**N291-08 - Safety-Related Structures (Non-Containment) for Nuclear Power Plants** These PSR1 ADs involve requirements for Non-Containment Safety-Related Structures from CSA N291-08. The ADs relate to issues that design documents are not fully consistent with requirements listed in CAN/CSA-N291-08. In PSR2, these issues are assessed to be ADs based on the fact that safety-related structures were designed and analyzed in accordance with the applicable standards at the time, which meet the intent of N291-08. These ADs are associated with D-150 Design Management.
197	CSA N291-08 DNGS ISR Review: Issue D284, "Load Factors for Metallic Embedded Parts in Safety-Related Building Structures"		
198	CSA N291-08 DNGS ISR Review: Issue D285, "Seismic Design Provisions for Concrete Safety-Related Structures"		
199	CSA N291-08 DNGS ISR Review: Issue D299, "Anchorage Requirements for Safety-Related Structures"		
200	CSA N291-08 DNGS ISR Review: Issue D401, "Load Factors for the Design of Nuclear Safety Related Structures"		

AD #	PSR1 AD Title	Group #	PSR1 Grouped AD Title, Description and Rationale
201	CSA N291-08 DNGS ISR Review: Issue D411		
202	CSA N285.6.2-05 PB ISR Review, Clause 3.2, "Manufacturing Requirements"	PSR1-AD14	**N285.6 Requirements for Material Standards for Reactor Components** These PSR1 ADs involve requirements for the zirconium-alloy material used in Pressure Tubes, Calandria Tubes, Reactivity Control Unit tubing, Liquid Injection Shutdown System Nozzles, Guide Tubes, Garter Springs and zirconium alloy wire. The ADs were identified, for example, on the basis that design documents do not reference the applicable codes and standards or the specific requirements from the codes and standards, but include specifications that align with the required standards. These ADs are associated with D-150 Design Management and D-195 Reactor Core Integrity.
203	CSA N285.6.2-05 PB ISR Review, Clause 3.4, "Manufacturing Requirements"		
204	CSA N285.6.2-05 PB ISR Review, Clause 3.5, "Manufacturing Requirements"		
205	CSA N285.6.2-05 PB ISR Review, Clause 4.2, "Composition Requirements"		
206	CSA N285.6.2-05 PB ISR Review, Clause 5.3, "Inspections"		
207	CSA N285.6.2-05 PB ISR Review, Clause 5.5, "Inspections"		
208	CSA N285.6.4-05 PB ISR Review, Clause 3.2, "General Requirements"		
209	CSA N285.6.4-05 PB ISR Review, Clause 3.4, "General Requirements"		
210	CSA N285.6.4-05 PB ISR Review, Clause 3.5, "General Requirements”.		
211	CSA N285.6.4-05 PB ISR Review, Clause 4.2, "Composition Requirements"		

AD #	PSR1 AD Title	Group #	PSR1 Grouped AD Title, Description and Rationale
212	CSA N285.6.4-05 PB ISR Review, Clause 5.2.2, "Processing of Strip Material"		
213	CSA N285.6.4-05 PB ISR Review, Clause 7.2, "Inspections"		
214	CSA N285.6.4-05 PB ISR Review, Clause 7.3, "Inspections"		
215	CSA N285.6.2 PARTS Review, Clause 5.3, "Hydrostatic Test"		
216	CSA N285.6.2 PARTS Review, Clause 5.7.3, "Surface Quality"		
217	CSA N285.6.2 PARTS Review, Clause 6.8, "Hydrostatic Testing"		
218	CSA N285.6.4 PARTS Review, Clause 5.1.2, "Surface Quality"		
219	CSA N285.6.4 PARTS Review, Clause 6.6, "Ultrasonic Examination"		
220	CSA N285.6 DNGS ISR Review: Issue D021, "Cleaning Solution Composition Limits"		
221	CSA N285.6 DNGS ISR Review: Issue D022, "Non-destructive Testing of Reactor Components"		
222	CSA N285.6 DNGS ISR Review: Issue D023, "Material Properties of Reactor Components"		

AD #	PSR1 AD Title	Group #	PSR1 Grouped AD Title, Description and Rationale
229	IAEA NS-R-1 PB ISR Review, Clause 5.34, "Single Failure Criterion"	PSR1-AD15	**IAEA NS-R-1 and CNSC RD-337 Single Failure Criterion** These PSR1 ADs involve safety-related requirements for application of the Single Failure Criterion in plant design. This AD group relates to issues that were raised against IAEA NS-R-1 and CNSC RD-337 and assessed as ADs. REGDOC-2.5.2, which adopts principles from NS-R-1 and supersedes CNSC RD-337, was reviewed in PSR2, including a detailed assessment of PSR1 ADs. The PSR2 review confirmed that the dispositions provided for Pickering 5-8 during the Pickering NGS B ISR remain valid and are also applicable to Pickering 1,4, and as a result, a new PSR2 Gap was not raised for these ADs. These ADs are all associated with Safety Principle D-150, Design Management, and M&C-246 Safety Evaluation of Design.
230	IAEA NS-R-1 PB ISR Review, Clause 5.35, "Single Failure Criterion"		
231	IAEA NS-R-1 PB ISR Review, Clause 5.37, "Single Failure Criterion"		
232	IAEA NS-R-1 PB ISR Review, Clause 5.38, "Single Failure Criterion"		
233	IAEA NS-R-1 PB ISR Review, Clause 5.41, "Auxiliary Services"		
240	CNSC RD-337 DNGS ISR Review: Issue D072, "Single Failure Criterion"		
180	CSA N290.3-11 DNGS ISR Review, Issue D611, "Coatings and Coverings Within Containment Systems"	PSR1-AD16	**Requirements for Containment Coatings and Coverings** These PSR1 ADs involve safety-related requirements for plant design. The ADs included in this group concern non-metallic coatings and liners that perform a safety function, such as preserving the concrete Containment envelope. The basic requirement is that not only should these materials be carefully selected and applied, but their deterioration (especially under post-accident conditions) should not impair the performance of other safety functions, such as flow through ECI strainers. The ECI System recovery strainer design considers potential materials that could result in a pressure drop across the strainers and affect pump
235	IAEA NS-R-1 PB ISR Review, Clause 6.67, "Coverings and Coatings"		
239	CNSC RD-337 DNGS ISR Review: Issue D071, "Coatings and Coverings Within Containment System"		

AD #	PSR1 AD Title	Group #	PSR1 Grouped AD Title, Description and Rationale
195	CSA N291-08 DNGS ISR Review: Issue D071, "Coatings and Coverings Within Containment System"		operation. As a result, a new PSR2 Gap was not raised for these ADs. These ADs are all associated with Safety Principle D-182, Equipment Qualification, and D-207 Emergency Heat Removal, and therefore they are grouped together.
237	IAEA NS-R-1 PB ISR Review, Clause 6.100, "Radiation Protection"	PSR1-AD17	**IAEA Requirements for Radiation Protection** This PSR1 AD involves requirements for radiation protection. Consideration of the buildup of radiation levels with time in areas of personnel occupancy is required in the design of the plant and in housekeeping practices during the conduct of operating and maintenance activities. Although the specific requirements and design documentation may not be fully consistent with the requirements of IAEA NS-R-1, good engineering practice was followed during the initial design of the station such that the layout and operation of facility SSCs, and processes are consistent with the established guidelines and contribute to maintaining occupational radiation exposures ALARA. An important principle of the Radiation Protection program is the control of exposures. This AD is associated with Safety Principle D-188, Radiation Protection in Design.
243	CNSC RD-337 DNGS ISR Review: Issue D278, "Turbine Orientation"	PSR1-AD18	**CNSC Requirements for Hazards** This PSR1 AD involves safety-related requirements for hazards. While the orientation of the turbine generators does not comply with the requirement of CNSC RD-337, the effect of turbine missiles has been considered in the supporting design analysis. This AD is associated with Safety Principle D-177, Dependent Failures.
236	IAEA NS-R-1 PB ISR Review, Clause 6.9, "Fuel Elements and Assemblies"	PSR1-AD19	**Deterministic Safety Analysis** These PSR1 ADs involve deterministic safety analysis requirements.

AD #	PSR1 AD Title	Group #	PSR1 Grouped AD Title, Description and Rationale
246	CNSC G-144 PB ISR Review, Clause 4.1, "Potential Consequences"		The PSR2 review of CNSC G-144, CNSC R-77 and CSA N288.2 confirmed that the dispositions provided for Pickering 5-8 during the Pickering NGS B ISR are also applicable to Pickering 1,4, and as a result, a PSR2 Gap was not raised for these ADs. These ADs are associated with Safety Principle D-150, Design Management, and M&C-246, Safety Evaluation of Design.
247	CNSC G-144 PB ISR Review, Clause 4.2, "Fuel Sheath Dryout"		
248	CNSC G-144 PB ISR Review, Clause 5.0, "Trip Parameter Acceptance Criteria"		
249	CNSC G-144 PB ISR Review, Clause 6.0, "Conditions for Postulated Reactor Accidents"		
250	CNSC G-144 DNGS ISR Review: Issue D032, "Trip Parameter Acceptance Criteria - CNSC G-144"		
252	CNSC R-77 PB ISR Review, Clause 3.1, "Allowable Service Conditions"		
253	CSA N288.2-M91 (R2008) DNGS ISR Review, Issue D026, "Decay of Parent Nuclides in Hypothetical Accidents"		
254	NBCC 2005 PB ISR Review, Clause 4.1.5.15, "Loads on Guards"	PSR1-AD20	**Requirements of NBCC for Design to Withstand Loads** These ADs involve building code requirements to consider various loads in the plant design. Gaps against the NBC were assessed as ADs mostly based on the fact that the NBC 2005 loads were not specified in NBC1970. This did not impact the design of Systems Important to Safety or the Safe Operating Envelope. The original designs accounted for the uncertainty of the loading and variability of the material properties with established and
255	NBCC 2005 PB ISR Review, Clause 4.1.5.16, "Loads on Vehicle Guardrails"		
256	NBCC 2005 PB ISR Review, Clause 4.1.5.17, "Loads on Walls Acting As Guards"		

AD #	PSR1 AD Title	Group #	PSR1 Grouped AD Title, Description and Rationale
257	NBCC 2005 PB ISR Review, Clause 4.1.6, "Loads Due to Snow and Rain"		accepted factors that would result in an acceptable margin of safety. They are associated with Safety Principle D-150, Design Management, and are therefore grouped together.
258	NBCC 2005 PB ISR Review, Clause 4.2.3.6, "Protection Against Chemical Attack"		
259	NBCC 2005 PB ISR Review, Clause 4.3.2.1, "Design Basis for Plain and Reinforced Masonry"		
260	NBCC 2005 PB ISR Review, Clause 4.3.3.1, "Design Basis for Plain, Reinforced and Pre-stressed Concrete"		
261	NBCC 2005 PB ISR Review, Clause 4.3.4.1, "Design Basis for Structural Steel"		
262	NBCC 2005 PB ISR Review, Clause 4.3.4.2, "Design Basis for Cold-Formed Steel"		
263	NBCC 2005 DNGS ISR Review, Issue D450, "Building Height Determination and Fire Resistance"	PSR1-AD21	**Fire-Related Requirements of NBCC** These ADs arose from the Darlington ISR and involve measures to prevent the spread of fire as well as to ensure staff safety and capability to respond. The scope of this group includes deviations between the existing design of the Darlington NGS and requirements of the NBCC – 2005. For example, Clause 2.1.3.8 is a new Article in NBCC-2005 which calls for integrated testing of fire protection and life safety systems to ensure that they will operate together, as intended. However, documentation could not be found to confirm that integrated systems testing is
264	NBCC 2010 DNGS ISR Review, Issue D519, "Fire Protection and Life Safety System Commissioning"		
265	NBCC 2010 DNGS ISR Review, Issue D524, "Fire and Smoke Characteristics of Electrical Wires and Cables"		

AD #	PSR1 AD Title	Group #	PSR1 Grouped AD Title, Description and Rationale
266	NBCC 2010 DNGS ISR Review, Issue D525, "Elevator Requirements"		performed for the life safety and fire protection systems installed at Darlington NGS. The PSR2 assessment of these ADs confirms that the rationale for these gaps being classified as ADs at Darlington NGS is also generally applicable to Pickering NGS, based on: • Fire Hazard Analysis and the Fire Safe Shutdown Analysis demonstrate that the safety objectives of the Station can be met under postulated fire scenarios. These analyses have also been completed for Pickering 1,4 and 5-8. • In the event of a fire at the Station, personnel, as per training and as instructed via the public address system, are to avoid the incident unit and area as well as to refrain from using the elevators. This also applies to Pickering NGS. • Access to the Station is limited to trained personnel who are familiar with the hazards and safety features of site buildings. Multiple means of egress are available and existing signage is clearly recognizable and understood by the personnel accessing the Station buildings. This also applies to Pickering NGS. • With respect to electrical conductors, there are no high-rise buildings, areas of refuge, or contained use areas located at Darlington NGS. This also applies to Pickering NGS. These ADs are associated with Safety Principle D-177, Dependent Failures, and are therefore grouped together.
268	NBCC 2010 DNGS ISR Review, Issue D527, "Entrance Walkway Smoke Detector"		
269	NBCC 2010 DNGS ISR Review, Issue D528, "Voice Communication Systems"		
270	NBCC 2010 DNGS ISR Review, Issue D529, "Electrical Conductors for Fire Protection Systems"		
271	NBCC 2010 DNGS ISR Review, Issue D530, "Exit Signage"		
228	IAEA NS-R-1 PB ISR Review, Clause 5.31, "Severe Accidents"	PSR1-AD22	**Software Verification and Validation** These ADs are related to requirements for development, verification and review of computer programs used for Safety Analysis, as well as requirements for software validation. OPG governance establishes a quality assurance program that is robust and compliant, and thus ensures the development of Scientific, Engineering and Safety Analysis software is aligned with the requirements of CNSC G-149 and CAN/CSA- N286.7-99. OPG has
251	CNSC G-149 DNGS ISR Review: Issue D146, "Specifications for Developing Computer Programs not Given in Software QA"		

AD #	PSR1 AD Title	Group #	PSR1 Grouped AD Title, Description and Rationale
			been participating in an industry-wide COG program addressing software validation. Therefore, this issue continues to be assessed as an AD. These ADs are associated with Safety Principles M&C-246, Safety Evaluation of Design, and M&C-249, Achievement of Quality.

AD #	PSR1 AD Title	Group #	PSR1 Grouped AD Title, Description and Rationale
The ADs listed below are administrative issues generally related to legacy documents or minor differences between OPG governance and code requirements. They have been assessed to have no significant safety impact, individually or in aggregate.			
1	CSA N290.13-05, PB ISR Review: Clause 5.2.1, "General"	PSR1-AD-DOC1	These PSR1 ADs involve CSA N290.13-05 requirements for qualification test plans, whose intent is met by N-PROC-RA-0044, *Environmental Qualification Assessment*. The Pickering NGS B ISR raised a Gap on the basis that the existing EQ documentation did not include a specific requirement for a qualification test plan and that it be accepted before starting the qualification process. However, it was noted that even with this Gap in the governance, all test reports were first passed through a QA check for suitability as design input documents before being used in EQ Assessments. These items do not impact any specific barriers, or objectives of any of the safety principles for operation during extended period and therefore, are not carried forward for Aggregation. These ADs do not affect the qualification of the equipment because the test reports are required to go through a QA check.
2	CSA N290.13-05, PB ISR Review: Clause 5.2.3, "Qualification Plan"		
12	CSA N285.5-M90, PB ISR Review: Clause 4.1.2 (a), "Responsibility"	PSR1-AD-DOC2	These PSR1 ADs involve high-level requirements for periodic inspection. These ADs identify administrative issues in OPG governance related to N285.5-M90. For example, statements in OPG governance use SHOULD instead of SHALL as specified in the standard. For these cases, all requirements in the standard must be performed unless an exception is granted by the CNSC. Based on this, the use of SHOULD is overridden by the use of SHALL in the PIPs, which have both been submitted to the CNSC for approval. These ADs do not impact any specific barriers, or objectives of any of
13	CSA N285.5-M90, PB ISR Review: Clause 4.5.2, "Accessibility"		
14	CSA N285.5-M90, PB ISR Review: Clause 5.3 (b), "Procedures"		
15	CSA N285.5-M90, PB ISR Review: Clauses 6.1.2.1 through 6.1.2.5, "Duties"		

AD #	PSR1 AD Title	Group #	PSR1 Grouped AD Title, Description and Rationale
16	CSA N285.5-M90, PB ISR Review: Clause 7.1.1, "Inaugural Inspection"		the safety principles for extended operation and therefore, are not carried forward for aggregation.
17	CSA N285.5-M90, PB ISR Review: Clause 8.6.1, "Inspection Interval"		
23	CSA N285.5-M90, PB ISR Review: Clause 8.6.5, "Inspection Interval"		
24	CSA N285.5-M90, PB ISR Review: Clause 8.6.6, "Inspection Interval"		
70	CSA N285.0-06 PB ISR Review, Clause 5.2.8, "Adjoining Systems or Sections"	PSR1-AD-DOC3	These PSR1 ADs involve issues relating to registration and classification of legacy pressure-retaining components. As discussed in the PSR2 N285.0-12 review, [P-REP-03680-00029-R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 1, 5, 6 and 7*, March 2017] the completed configuration management restoration project established the documentation that is required to safely operate and maintain the station. Therefore, this AD is of very low safety significance. These ADs do not impact any specific barriers, or objectives of any of the safety principles for extended operation and therefore, are not carried forward for aggregation.
71	CSA N285.0-06 PB ISR Review, Clause 6.1.1.3, "Designs to be Registered, General Requirements"		
72	CSA N285.0-06 PB ISR Review, Clause 6.1.4, "Supports"		
73	CSA N285.0-06 PB ISR Review, Clause 7.6.2.4, "Supports for Class 1, 2, and 3 Systems and Class 1C, 2C, 3C and 4 Items, Specific Requirements"		
75	CSA N285.0-06 PB ISR Review, Clause 12.1, "Documentation, General"		
76	CSA N285.0-06 PB ISR Review, Clause 12.3.9, "Nameplates and Identification"		

AD #	PSR1 AD Title	Group #	PSR1 Grouped AD Title, Description and Rationale
77	CSA N285.0-06 PB ISR Review, Clause 12.4.7.2, "Pumps and Line Valves"		
78	CSA N285.0-06 PB ISR Review, Clause A.1.6.3.3, "Classification Exemptions"		
82	CSA N285.0-95 PARTS Review, Clause 14.0, "Supports"		
223	CSA B51-03 PB ISR Review, Clause 4.2.1, "Registration of Fittings"		
144	N290.1-M80 PB ISR Review, Clause 4.4.1.3, "Reliability"	PSR1-AD-DOC4	These ADs are related to documentation of detailed design and requirements for reliability and redundancy in the design of the Shutdown Systems. There are no safety impacts associated with the ADs since the reliability requirements are demonstrated by performing unavailability calculations for the Pickering 1,4 and Pickering 5-8 Shutdown Systems per the regulatory requirements in RD-98. These ADs do not impact objectives of any of the safety principles for extended operation and therefore, are not carried forward for aggregation.
145	N290.1-M80 PB ISR Review, Clause 4.4.1.4, "Reliability"		
146	N290.1-M80 PB ISR Review, Clause 4.4.1.5, "Reliability"		
147	N290.1-M80 PB ISR Review, Clause 4.4.2.5, "Redundant Instrumentation Channels"		
148	N290.1-M80 PB ISR Review, Clause 4.4.14.1, "Quality of Equipment"		
149	N290.1-M80 PB ISR Review, Clause 6.1.1, "Documentation, General"		
150	N290.1-M80 PB ISR Review, Clause 6.2.2, "Compliance with Requirements"		

AD #	PSR1 AD Title	Group #	PSR1 Grouped AD Title, Description and Rationale
161	N290.1-M80 PARTS Review, Clause 4.4.14.2, "Quality of Equipment"		
162	N290.1-M80 PARTS Review, Clause 4.4.16.4, "Identification of System Hardware"		
163	N290.1-M80 PARTS Review, Clause 5.1, "Quality Assurance Program"		
172	CNSC R-7 PB ISR Review, Clause 3.13.1, "Codes and Standards"	PSR1-AD-DOC5	This PSR1 AD is associated with the requirement to obtain CNSC approval of “any aspects of the design which fail to comply with the applicable requirements of CSA N287 and CAN3-N285.0” during construction. This AD does not impact any specific barriers or objectives of any of the safety principles for extended operation and therefore, is not carried forward for aggregation.
190	CSA N290.5-M90 PB ISR Review, Clause 2, "Reference Publications"	PSR1-AD-DOC6	This PSR1 AD relates to Clause 2 of the standard, which lists applicable references. The references only affect the support power system design if they appear in a clause stating a requirement. Discussions on the applicability and assessment of the reference publications on the support power system design are documented in the compliance evaluations for clauses where the publications are referenced. This AD does not impact any specific barriers or objectives of any of the safety principles for extended operation and therefore, is not carried forward for aggregation.

AD #	PSR1 AD Title	Group #	PSR1 Grouped AD Title, Description and Rationale
227	IAEA NS-R-1 PB ISR Review, Clause 3.8, "Proven Engineering Practices"	PSR1-AD-DOC7	This PSR1 AD relates to the requirement to specify the process used in the selection of equipment, which some of the original design documents did not. However, OPG standard design practice ensures that equipment that supports a safety function is carefully selected for its functional reliability with due account taken of its probable failure modes. The practice is considered effective. This AD does not impact any specific barriers or objectives of any of the safety principles for extended operation and therefore, is not carried forward for aggregation.

AD #	PSR1 AD Title	Group #	PSR1 Grouped AD Title, Description and Rationale
The following ADs have either been addressed or are not applicable to Pickering NGS. They are deemed to have no significant aggregate impact.			
3	CSA N285.4-05, PB ISR Review: Clause 3.7.1, "Access for Inspection"	PSR1-AD-NFA1	These PSR1 ADs are assessed in [P-CORR-03680-0620816-R000, *Re: PSR1 Acceptable Deviations Reassessed for Pickering PSR2*, March 31, 2017] as no longer applicable because Pickering NGS is now in compliance with N285.4-05. They are retained for completeness, but are not carried further in the AD aggregation.
4	CSA N285.4-05, PB ISR Review: Clause 7.4.2.2 (f), "Supports"		
5	CSA N285.4-05, PB ISR Review: Clause 7.4.3 (a), "Piping"		
6	CSA N285.4-05, PB ISR Review: Clause 7.4.3 (c), "Mechanical Couplings"		
7	CSA N285.4-05, PB ISR Review: Clause 7.4.3 (f), "Supports"		
8	CSA N285.4-05, PB ISR Review: Clause 7.4.4.3 (a), "Piping"		
9	CSA N285.4-05, PB ISR Review: Clause 7.4.4.3 (c), "Mechanical Couplings"		
10	CSA N285.4-05, PB ISR Review: Clause 7.4.4.3 (f), "Supports"		
11	CSA N285.4-05, PB ISR Review: Clause 7.6.1.2, "Subsequent Periodic Inspections…"		
45	CSA N289.1-08 DNGS ISR Review: Issue D281, Clauses 6.5.6.1,	PSR1-AD-	As documented in [P-CORR-03680-0620816-R000, *Re: PSR1 Acceptable Deviations Reassessed for Pickering PSR2*, March 31,

AD #	PSR1 AD Title	Group #	PSR1 Grouped AD Title, Description and Rationale
	6.5.6.2.2, 6.5.6.4 and 6.5.7.3, "Seismic Qualification - Post-Event Response and Actions"	NFA2	2017] additional actions to support compliance have been performed for AD #45 and 46, and actions for AD #50, 51 and 52 demonstrate that they are no longer applicable for Pickering NGS. These ADs are retained for completeness and not carried forward for the aggregation assessment.
50	CSA N289.4-12 DNGS ISR Review, Clause 4.2.6, Issue D427, "Seismic Qualification - General"		
51	CSA N289.5-M91 (R2008) DNGS ISR Review, Issue D064, Clauses 6.2.2 and 6.2.5, "Seismic Monitoring System Testing"		
52	CSA N289.5-12 DNGS ISR Review, Clause 5.1.5, Issue D621, "Continuous Recording Seismic Devices"		
55	CNSC RD-204 (2008) DNGS ISR Review, Issue D272, "Reinstatement of a Person to the Duties of a Position Following Absence or Removal from those Duties"	PSR1-AD-NFA3	These PSR1 ADs involve Training and Human Performance requirements in CNSC Guidance documents. AD # 55, 56, 57 and 58 have subsequently been addressed [P-CORR-03680-0620816-R000, *Re: PSR1 Acceptable Deviations Reassessed for Pickering PSR2*, March 31, 2017]. These ADs are retained for completeness and not carried forward for the aggregation assessment.
56	CNSC RD-204 (2008) DNGS ISR Review, Issue D274 "Recertification Requirements after Decertification"		
57	CNSC G-225 (2001) DNGS ISR Review, Issue D067, "Provisions for Post-Accident Sampling"		
58	CNSC G-323 (2007) DNGS ISR Review, Issue D270, "Analysis and Validation of Minimum Staff Complement Requirement"		

AD #	PSR1 AD Title	Group #	PSR1 Grouped AD Title, Description and Rationale
67	CNSC SOR/2000-203 (May 2000) PB ISR Review, Clause 20 (1)(a), "Labelling of Containers and Devices"	PSR1-AD-NFA4	These PSR1 ADs involve issues relating to radiation protection identified by CNSC. All actions to track the regulatory commitments associated with these ADs have been completed [P-CORR-03680-0620816-R000, *Re: PSR1 Acceptable Deviations Reassessed for Pickering PSR2*, March 31, 2017]. These ADs are retained for completeness and are not carried forward for the aggregation assessment.
68	CNSC SOR/2000-203 (May 2000) PB ISR Review, Clause 21 (1), "Posting of Signs at Boundaries and Points of Access"		
69	CNSC SOR/2000-203 (May 2000) Darlington ISR Review, Issue D157, "Labelling and Posting of Radiation Hazard"		
74	CSA N285.0-06 PB ISR Review, Clause 7.7.2.3, "Heat Transport System"	PSR1-AD-NFA5	Clause 7.7.2.3 of CSA N285.0-06 identifies considerations for the qualification of the Heat Transport System for overpressure events. In the updated version of the standard, N285.0-12, the equivalent Clause 7.6.2.3 (c) has been modified to provide relaxation consistent with Regulatory Document R-77, so there is no longer a gap associated with this clause. This AD is retained for completeness and is not carried forward for the aggregation assessment.
79	CSA N285.0-06 PB ISR Review, Clauses 14.5.3, 14.5.3.1, and 14.5.3.2, "Reconciliation Statements for Modifications"	PSR1-AD-NFA6	This AD relates to the backlog of reconciliation statements for Pickering NGS B. This issue has since been closed [NK30-CORR-00531-06815-R000, *Pickering Units 5 to 8: System Registration Update Recovery Plan Project, Action Item 2014-8-16 (RIB #2408)*, May 6, 2014]. Related ADs against Clauses 14.5.3 and 14.5.3.2, were addressed with the same disposition as for Clause 14.5.3.1. Further, this AD is not impacted by operation beyond 2020. This AD is retained for completeness and is not carried forward for the aggregation assessment.

AD #	PSR1 AD Title	Group #	PSR1 Grouped AD Title, Description and Rationale
83	CSA N293-05 PB ISR Review, Clause 2.1, "Reference Publications"	PSR1-AD-NFA7	These ADs are related to legacy design and construction activities for which, in some cases, documented evidence is not found to confirm compliance with modern codes and standards. Also, there are ADs in this group related to the original design of the Fire Water System in Pickering NGS. Several of these clauses are related to the use of High Pressure Service Water (HPSW) for firewater at Pickering 1,4, which has since been replaced by diesel firewater pumps. The dispositions to these clauses [P-CORR-03680-0620816-R000, *Re: PSR1 Acceptable Deviations Reassessed for Pickering PSR2*, March 31, 2017] also reference the Fire Hazard Assessment and Fire Safe Shutdown Analysis that had been performed. The PSR1 review further stated that, as part of the PARTS Fire Protection Project at the time, Pickering A had committed to installing upgraded fire detection and suppressions systems. These activities have been completed. These ADs are retained for completeness and are not carried forward for the aggregation assessment.
84	CSA N293-05 PB ISR Review, Clause 5.2.2 (d), "Buildings"		
85	CSA N293-05 PB ISR Review, Clause 5.2.4 (a), "Electrical Equipment"		
86	CSA N293-05 PB ISR Review, Clause 7.2.1.1, "Fire Separation Concept"		
87	CSA N293-05 PB ISR Review, Clause 7.2.1.4, "Penetration Seals"		
88	CSA N293-05 PB ISR Review, Clause 7.2.2.4, "Separation Within Safety Groups"		
89	CSA N293-05 PARTS Review, Clause 6.4.5, "Fire Pumps"		
90	CSA N293-05 PARTS Review, Clause 6.4.7, "Fire Water Used for Other Purposes"		
91	CSA N293-05 PARTS Review, Clause 7.2.2.2, "Separation Between Safety Groups"		
92	CSA N293-05 PARTS Review, Clause 7.2.2.3, "Separating Safety Systems from Fire Hazards"		

AD #	PSR1 AD Title	Group #	PSR1 Grouped AD Title, Description and Rationale
93	CSA N293-05 PARTS Review, Clause 7.2.2.4, "Separation Within Safety Groups"		
94	CSA N293-05 PARTS Review, Clause 7.2.3.2, "Separation of the Turbine Building"		
143	N290.1-M80 PB ISR Review, Clause 4.4.1.2, "Reliability"	PSR1-AD-NFA8	Clause 4.4.1.2 in N290.1-M80 identifies a requirement that during the design phase, a target unavailability of $1x10^{-4}$/r-yr be used. This is no longer a requirement in N290.1-13. This AD is retained for completeness and is not carried forward for the aggregation assessment.
159	N290.1-M80 PARTS Review, Clause 4.4.9.2, "Testing"	PSR1-AD-NFA9	These ADs involve requirements for on-line testing of the Shutdown Systems components. The updated version of N290.1-13 does not contain the requirement that response times be measured on-line, so both Pickering 1,4 and Pickering 5-8 Shutdown Systems comply with the requirements and, therefore, this is not a PSR2 Gap, and therefore these ADs are not carried forward for aggregation assessment. Also, with respect to AD #165, requirement for diversity, this clause is no longer contained in
160	N290.1-M80 PARTS Review, Clause 4.4.9.6, "Testing"		

AD #	PSR1 AD Title	Group #	PSR1 Grouped AD Title, Description and Rationale
165	CSA N290.1-80 DNGS ISR Review, Issue D060, "Shutdown System Diversity"		N290.1-13.
169	CNSC R-7 PB ISR Review, Clause 3.8.3, "Separation and Independence Requirements"	PSR1-AD-NFA10	These PSR1 ADs involve separation and independence requirements for Containment. They are associated with Safety Principle D-217, Confinement of Radioactive Material. As documented in [P-CORR-03680-0620816-R000, *Re: PSR1 Acceptable Deviations Reassessed for Pickering PSR2*, March 31, 2017] there is no Gap relating to this issue in PSR2, but it is retained for completeness.
170	CNSC R-7 PB ISR Review, Clause 3.8.4, "Separation and Independence Requirements"		
171	CNSC R-7 PB ISR Review, Clause 3.11.2, "Shielding Requirements"	PSR1-AD-NFA11	This PSR1 AD involves the requirement that shielding assessments be referenced in the Containment design documentation. This issue is a documentation issue only and is not safety significant. The PSR2 assessment is that Clause 11.1 in the updated N290.3-11 requires adequate shielding provisions be present, but does not contain a requirement for the information to be referenced in the design documentation, and therefore this is not considered a PSR2 Gap, but it is retained for completeness.
173	CNSC R-7 PB ISR Review, Clause 3.13.2, "Codes and Standards"	PSR1-AD-NFA12	Clause 3.13.2 of R-7 requires that a list of codes and standards to be applied to the Containment system be prepared and approved by the CNSC prior to construction approval. Codes and standards applicable to Containment are referenced in the design documentation; however,

AD #	PSR1 AD Title	Group #	PSR1 Grouped AD Title, Description and Rationale
			there is no longer an explicit requirement for regulatory approval in the standard. This gap is no longer applicable to Pickering NGS, and is retained for completeness.
175	CNSC R-7 PARTS Review, Clause 3.7.1, "Availability Requirements"	PSR1-AD-NFA13	This PSR1 AD #175 involves availability requirements for Containment. AD #175 is no longer applicable [P-CORR-03680-0620816-R000, *Re: PSR1 Acceptable Deviations Reassessed for Pickering PSR2*, March 31, 2017], but it is retained for completeness under the PSR1-AD-NFA group.
166	CNSC R-9 PB ISR Review, Clause 3.4.9, "Availability Requirements"	PSR1-AD-NFA14	Clause 3.4.9 of CNSC R-9: The gap was raised due to a documentation discrepancy related to the ECI Heat Transport low pressure signal setpoint between the Design Requirements and the OSR. The gap has since been addressed by revising the Design Requirements documentation.
183	CSA N290.4-M82 PB ISR Review, Clause 4.3.11, "Reactor Start-Up"	PSR1-AD-NFA15	Clause 4.3.11 of N290.4-M82 requires that if special equipment is used for reactor start-up after long periods of shutdown, then that equipment may also have requirements imposed on it by N290.1. Related clauses 5.7 and 5.10.1 in N290.4-11 do not contain the requirement for Start-up Instrumentation to meet these requirements. Therefore, this is not a gap but is retained for completeness.
184	CSA N290.4-M82 PB ISR Review, Clause 6.2.1, "Functional and Performance Requirements"	PSR1-AD-NFA16	Clause 6.2.1 of N290.4-M82 requires that a specific set of functional and performance requirements (contained in sub-clauses (a) though (j)) be documented in the RRS design manuals. There is no corresponding requirement in the updated N290.4-11 version. Therefore, this is not a gap but is retained for completeness.
186	CSA N290.4-M82 PARTS Review, Clause 4.3.1.2, "Specific Requirements"	PSR1-AD-NFA17	Clause 4.3.1.2 of N290.4-M82 is superseded by N290.4-11 clauses 5.4.2 and 5.4.3, which require that the design target reliability of RRS be established in the Probabilistic Safety Assessment. The Pickering 1,4 Probabilistic Safety Assessment demonstrates that the RRS reliability including any supporting systems is acceptable. Therefore, this is not a gap but is retained for completeness.

AD #	PSR1 AD Title	Group #	PSR1 Grouped AD Title, Description and Rationale
188	CSA N290.4-M82 PARTS Review, Clause 4.3.25.4, "Identification of System Hardware"	PSR1-AD-NFA18	Clause 4.3.25.4 of N290.4-M82: This requirement refers to practices for human-system interface design. The gap was raised because colour coding for different channels is not used. The Main Control Room panel components are located functionally with the associated system and organized into groupings based on the associated zone. Labels clearly identify the instrument and the associated zone. There are only a limited number of switches. The Main Control Room panels are clearly organized and labeled, such that colour coding to differentiate between channels is not required. There is no longer a specific requirement for colour coding. Therefore, this is not a gap but is retained for completeness.
189	CSA N290.4-11 DNGS ISR Review, Issue D245, "Shutdown System Requirements for Start-up Instrumentation"	PSR1-AD-NFA19	Start-up instrumentation is considered part of Shutdown System functionality. However, related clauses 5.7 and 5.10.1 in N290.4-11 do not contain the requirement for start-up instrumentation to meet N290.1 requirements. Therefore, this is not a gap but is retained for completeness.
224	CSA B51-03 PB ISR Review, Clause 7.5.1.3, "Design Requirements"	PSR1-AD-NFA20	#224: This is not relevant to the present review as the Pickering B SOE/SIS systems do not contain coil-tube boiler blowoff vessels. #225: This is not relevant to the design of SOE/SIS systems at Pickering B. Therefore, these are not gaps but are retained for completeness.
225	CSA B51-03 PB ISR Review, Clause 7.5.2.1, "Cleaning and Inspection Facilities"		
234	IAEA NS-R-1 PB ISR Review, Clause 6.64, "Control and Cleanup of the Containment Atmosphere"	PSR1-AD-NFA21	This PSR1 AD involves requirements for hydrogen cleanup in Containment FADS provides a controlled and monitored release path. Short-term hydrogen control is provided by the Post-LOCA Hydrogen Ignition System (PLHIS). However, the PLHIS was not originally intended to clean up hydrogen in Containment in the longer term. This issue concerning long-term control of hydrogen has been addressed for Pickering with the installation of Passive Auto-catalytic Recombiners and completion of the Fukushima Action Items. Therefore, this is not a gap but is retained for completeness.

AD #	PSR1 AD Title	Group #	PSR1 Grouped AD Title, Description and Rationale
238	CNSC RD-337 DNGS ISR Review: Issue D070, "Consideration of Decommissioning During Design Phase"	PSR1-AD-NFA22	The scope of this ISR Issue covers future decommissioning considerations of the plant during the design phase. Requirements for the decommissioning phase are not in PSR2 scope. Therefore, this is not a gap but is retained for completeness.
242	CNSC RD-337 DNGS ISR Review: Issue D277, "Primary Heat Transport Coolant Supply for Multi Unit Shutdown"	PSR1-AD-NFA23	The scope of this Darlington ISR Issue covers the HTS coolant inventory for multi-unit events. Clause 8.2.2 of CNSC RD-337 states that the inventory in the reactor coolant system and its associated systems shall be sufficient to support cool down from hot operating conditions to zero power cold conditions without the need for transfer from any other systems. This is not an issue for Pickering, as there are no credits for non-unit D2O identified per the Pickering Units 1,4 and Pickering Units 5-8 Heat Transport System (HTS) Operational Safety Requirements (OSRs). Therefore, this is not a gap but is retained for completeness.
244	CNSC RD-337 DNGS ISR Review: Issue D225, "Fire Protection Water Supply"	PSR1-AD-NFA24	Clause 8.8 of CNSC RD-337 requires that if the fire water supply is interconnected to the emergency heat removal system, operation of one does not impair the operation of the other. This is not an issue for Pickering. Pickering 5-8 have an Emergency Water Supply (EWS) supply to the Heat Transport System. Pickering 1,4 have a manual flowpath that supplies water to the HTS from firewater or Emergency Mitigating Equipment (EME) water. The Darlington ISR issue was also related to fire coincident with LOCA relying on a common Emergency Service Water System. This is not an issue for Pickering where fire water is supplied by Diesel pumps with sufficient redundancy (Pickering 1,4) or High Pressure Service Water (Pickering 5-8). Therefore, this is not a gap but is retained for completeness.
245	CNSC RD-337 DNGS ISR Review: Issue D280, "Off-Gas Management System"	PSR1-AD-NFA25	The Off-Gas Management System is no longer in use. Hence, this is not a PSR2 Gap but is retained for completeness.

AD #	PSR1 AD Title	Group #	PSR1 Grouped AD Title, Description and Rationale
273	NFCC 2005 PB ISR Review: Clause 4.3.7.2, "Construction"	PSR1-AD-NFA26	This AD is related to the requirement for installation of impermeable dyke liners for Standby Generators and the Emergency Power Generator tanks. These dyke liners are now installed. Therefore, this is not a gap but is retained for completeness.
274	PB ISR Emergency Planning SF Review Task #9, "Assess OPG compliance with the interim Part I and Part II of the Provincial Nuclear Emergency Plan (PNEP)"	PSR1-AD-NFA27	**Provincial Nuclear Emergency Plan (PNEP)** This AD involves meeting Provincial Nuclear Emergency Response Plan requirements for emergency planning and response. The Pickering NGS B ISR raised a number of Gaps against the requirements of the then current Provincial Nuclear Emergency Response Plan. Considerable work has been done by OPG since PSR1 on the Consolidated Nuclear Emergency Plan [N-PROG-RA-0001 R014, *Consolidated Nuclear Emergency Plan*, May 2015] to align with the Provincial Nuclear Emergency Response Plan. As concluded in PSR2 Safety Factor 13 OPG Nuclear has: a) adequate plans, staff, facilities and equipment in place for dealing with emergencies, and b) there are adequate arrangements in place for regular emergency training and exercises, and interaction and coordination with local and national authorities. Separate PSR2 reviews were performed for REGDOC-2.10.1, *Nuclear Emergency Preparedness and Response* and CSA N1600-14, *General Requirements for Nuclear Emergency Management Programs*. The OPG nuclear emergency preparedness governance [N-PROG-RA-0001 R015, OPG Nuclear Program, *Consolidated Nuclear Emergency Plan*, November 2016] was revised to ensure that the evacuation time estimates and KI pill programs will be sustained. No further action is required.

Appendix H – Aggregation of Acceptable Deviations by Defence-in-Depth Level

This Appendix evaluates the aggregate impact of PSR1 AD groups as described in Appendix G, together with ADs identified in PSR2. For each level of defence-in-depth, ADs associated with each relevant safety principle are listed, and an evaluation of the impact of the collective set of ADs on the safety principle is provided. This is followed by an overall assessment of the impact of all relevant ADs on the level of defence-in-depth.

This appendix includes PSR1 AD groups and PSR2 ADs that are first assessed for impact against the applicable safety principles for each level of defence. The safety principles provide a structured and comprehensive framework for combining ADs with similar attributes to facilitate assessing their aggregate impact on the barriers provided by each level of defence.

The impact of some ADs was associated with a single safety principle but in other cases a conservative approach was taken to include multiple safety principles that are potentially affected, as follows:

- In the most straightforward cases where there is only one AD associated with a safety principle, determining the aggregate impact of the ADs is simple and straightforward. That is, there is no aggregated effect, by definition.
- In other cases, there is a relatively small number of ADs that can impact on a safety principle and the issues are very minor and totally unrelated (e.g., radiation protection, training programs). In such cases, the assessment of aggregation is again simple and straightforward, and it is concluded that there is no aggregated impact.
- In some cases, some of the safety principles can potentially be impacted by several ADs and determining the combined impact involves assessing the cause of the deviation, the mitigating provisions in place and the extent to which each of the safety principles could potentially be affected. In such cases, a rationale is provided for the aggregation assessment outcome.

For a given defence-in-depth level, the AD groups for each safety principle are cross-compared to confirm that there is negligible aggregate impact on defence-in-depth.

AD Aggregation for Defence-in-Depth Level 1

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
D-150 – Design Management	GI-15-AD1 PSR1-AD13 PSR1-AD14 PSR1-AD15 PSR1-AD19 PSR1-AD20	The following AD and AD groups are assigned to Safety Principle D-150, Design Management: • GI-15-AD1 involves supplier identification of digital items. A requirement for a supplier to self-identify whether their product contains any digital items is not reflected in OPG governing documents. OPG Program [N-PROG-MP-0009-R012, *Design Management,* April 24, 2017] provides the framework for establishment, maintenance and compliance with the design basis for Pickering NGS. The Design Management program specifies requirements for procurement engineering processes ensuring that implementation and maintenance of the physical nuclear facilities meet the design basis requirements. In the event that the use of digital items is identified by OPG in advance of issuing a Request for Proposal or Request for Quotation, existing OPG procedures are adequate for ensuring that requirements related to digital items are documented in the technical specification, including checks and confirmation of the form of equipment supplied. This issue is assessed to be very low safety significance. A Documentation Change Request to update specification preparation governance to require suppliers to identify and describe digital items in equipment provided will be managed outside of the PSR2. • PSR1-AD13 involves CSA N291-08 requirements for Non-Containment Safety-Related Structures. The ADs in this group relate to issues that design documents do not specify requirements that are consistent with those listed in CAN/CSA-N291-08. In PSR2, these issues are assessed to be ADs based on the fact that safety-related structures were designed and analyzed in accordance with the applicable standards at the time, which meet the intent of N291-08. • PSR1-AD14 involves requirements for the zirconium-alloy material used in Pressure Tubes, Calandria Tubes, Reactivity Control Unit tubing, Liquid Injection Shutdown System Nozzles, Guide Tubes, Garter Springs and zirconium alloy wire. Items associated with PSR1-AD14 are assessed to be ADs, on the basis that design documents do not reference the applicable codes and standards or the specific requirements from the codes and standards, but include specifications that align with the required standards. • PSR1-AD15 involves safety-related requirements for plant design. This AD group relates to Single Failure Criterion issues that were raised against IAEA NS-R-1 and

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		CNSC RD-337 and assessed as ADs. The PSR2 review of REGDOC-2.5.2, which adopts principles from NS-R-1 and supersedes CNSC RD-337, included a detailed assessment of PSR1 ADs. Also, the PSR2 assessment of CSA N290.2-11, *Requirements for Emergency Core Cooling Systems of Nuclear Power Plants*, identified an AD for the Pickering 1,4 ECI system, which contains singleton components required to operate for the system to operate successfully, on the basis that all practical modifications were made to the Pickering 1,4 ECI system during Pickering A Return to Service and the system meets its unavailability target. The Pickering NGS design permits testing, maintenance and other necessary activities to be completed while keeping the system available. As stated in the PSR2 code review for REGDOC-2.5.2 [P-REP-03680-00029-R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 1, 5, 6, and 7*, March 2, 2017], this issue was transferred to the Pickering B Continued Operations Plan, where it was closed on the basis that the Level 1 and Level 2 PSA demonstrated that the existing plant design satisfied all safety requirements. The applicability of the gap resolution and subsequent disposition is equally applicable to Pickering 1,4, where the PSAs have similarly been updated. Therefore, this issue is of low safety significance. • PSR1-AD19 involves deterministic safety analysis requirements. The ADs in this group are related to dual trip parameter coverage, the criterion for no dry-out for the first trip, the allowable duration of post dry-out operation for the second trip, and interpretation of the acceptance criterion relating to the acceptable duration of post dry-out operation. However, the derived acceptance criteria outlined in the CANDU Owners Group report [COG-13-9035*, Derived Acceptance Criteria For Deterministic Safety Analysis*, November 2014], developed by the industry taking into account feedback from the CNSC, are less restrictive than those outlined in the current Pickering 1,4 and Pickering 5-8 Safety Reports, providing confidence that the derived acceptance criteria documented in COG-13-9035 will not adversely impact on existing margins as demonstrated in the Safety Reports. As part of the PSR2 review, modern versions of the codes and standards were assessed to determine the extent to which Pickering NGS meets the new standards, namely REGDOC-2.5.2 and REGDOC-2.4.1, including a review of PSR1 ADs. The review confirmed that PSR1 ADs in this group have no impact on nuclear safety. The AD rationale continues to remain applicable and is not impacted by the planned extended operation. • PSR1-AD20 involves National Building Code of Canada requirements to consider various loads in the

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		plant design. The original designs accounted for the uncertainty of the loading and variability of the material properties with established and accepted factors that would result in an acceptable margin of safety. The PSR2 review confirmed that although the recent version of the National Building Code of Canada may introduce new requirements related to these topics, the intent is met by the design. The ADs in this group that relate to barriers in Safety Principle D-150, Design Management, have no significant interaction because the issues are associated with unique SSCs, and individual groups have low or very low safety significance or safety impact. Generally, these ADs are related to legacy design issues which have since been addressed by the on-going maintenance, testing and inspection programs at OPG. Furthermore, OPG Program, N-PROG-MP-0009, *Design Management*, provides the framework for establishment, maintenance and compliance with the design basis for Pickering NGS. The Design Management program provides assurance that design and procedure changes are prepared, reviewed, approved, documented and implemented in accordance with approved procedures, applicable regulatory requirements, standards and industry practices. Therefore, the overall conclusion for D-150 is that the aggregate impact on defence-in-depth of these ADs is negligible and the Pickering NGS Design Management program defines the appropriate processes to ensure nuclear safety requirements are met in the planning and execution of design modifications.
C-255 – Verification of Design and Construction	PSR1-AD3	PSR1-AD3 involves general requirements for Containment Concrete Structures. The original and subsequent revisions of the CSA N287 series of standards were issued after completion of design and construction of Pickering NGS. Therefore, the details related to general requirements of Containment structure, parts, materials, design, fabrication, construction, inspection, examination, testing and commissioning in accordance with the series of standards of CSA N287 are not available. At that time, the Concrete Containment Structures were designed to meet the requirements of the National Building Code of Canada (NBCC) The findings during the PSR1 were assessed to be ADs given robust design practices at the time of the construction, together with ongoing periodic inspections and in-service testing. The standards that applied during original construction of Pickering NGS included requirements for tests and quality control procedures to ensure that the concrete used in the as-built structures met the original design requirements. The PSR2 review of N287.1-14 [P-REP-03680-00029-R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 1, 5, 6, and 7*, March 2, 2017] also outlines Programs,

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		Procedures and Standards which are credited with the ability to detect and monitor any safety significant degradation mechanisms and thus to provide assurance of continued fitness for service of the Pickering NGS Containment Concrete Structures. The AD rationale continues to remain applicable and is not impacted by the planned extended operation. There is only one AD group associated with this safety principle, and therefore there is no aggregate impact for Safety Principle C-255, Verification of Design and Construction.
D-195 – Reactor Core Integrity	PSR1-AD14	PSR1-AD14 involves requirements for material standards for Reactor Components. The technical specifications do not reference the applicable standard but include specifications for mechanical properties, grain size, corrosion properties and inspection requirements that align with the standard. Therefore, the deviation does not pose a safety risk. OPG Program, N-PROG-MP-0009, *Design Management*, provides the framework for establishment, maintenance and compliance with the design basis for Pickering NGS. The Design Management program provides assurance that design and procedure changes are prepared, reviewed, approved, documented and implemented in accordance with approved procedures, applicable regulatory requirements, standards and industry practices. OPG Program [N-PROG-MA-0017 R009, *Component and Equipment Surveillance*, May 31, 2017] defines requirements for establishing programs to ensure the health of selected plant components and equipment. Steam Generators, Fuel and Fuel Channels, Feeder piping, and Reactor Components are covered by formal Life Cycle Management Plans. This program defines the requirements for establishing component programs that manage component and equipment health including inspection, maintenance and testing. OPG program [N-PROG-MA-0025, *Major Components*, March 25, 2015] manages aging of the Major Components (Fuel Channels, Feeders, Steam Generators and Reactor Structures) through a comprehensive program of in-service inspections, maintenance, engineering assessment and confirmatory research and development (R&D). The program is fully developed and effectively implemented for continued validation of fitness for service of the Major Components. The program also incorporates the reporting requirements associated with demonstrating compliance with design basis documentation relevant to each of the Major Components Program areas. This program is identified as a Strength for Pickering NGS. OPG has robust and multi-faceted SSC inspection, monitoring and assessment programs, such that the current condition of the plant is well understood. This provides a sound baseline for extended operation

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
		as well as confidence that the requisite data will continue to be acquired. Maintenance programs are aligned with industry best practice and are carried out in accordance with written procedures that are part of the overall Management System. OPG program [N-PROG-MA-0026, *Equipment Reliability*, May 26, 2015] ensures that there are defined activities ensuring that the current condition of the equipment is understood, and that equipment aging issues are identified, understood and effectively managed for equipment important to nuclear safety and equipment reliability. This program and the Major Components program are identified as a Strength for Pickering NGS. System Health reporting, which is an integral element of the Equipment Reliability Program, is also identified as a Strength. There is only one AD group associated with this safety principle, and therefore there is no aggregate impact for Safety Principle D-195, Reactor Core Integrity.
D-186 – Inspectability of Safety Equipment	GI-16-AD1	GI-16-AD1 is also covered under O-305 Maintenance, Testing and Inspection. This AD group is related to concessions in the periodic inspection plans for components that are inaccessible. The CNSC has accepted the reasonableness of the solution to this issue and has determined that the revised PIPs satisfactorily meet the requirements of CSA N285.5-08 Update No. 1 standard. Furthermore, as indicated in the resolution for GI-16-AD1 in Appendix B of this report, where the inspection of a system or component would necessitate the dismantling of equipment, the required inspection should be performed when the equipment is dismantled for other reasons (i.e., maintenance). There is only one AD associated with this safety principle, and therefore there is no aggregate impact for Safety Principle D-186, Inspectability of Safety Equipment.
D-205 – Startup, Shutdown and Low Power Operation	PSR1-AD10	PSR1-AD10 involves requirements for RRS related to Start-up Instrumentation, documentation, reliability, and human-system interface. These issues were identified during the PARTS review and are related to the requirement for the operator to initiate monitoring of the flux tilt and initiate power reduction as needed. The control systems in Pickering NGS maintain the reactor operating conditions within the normal operating range, and effectively respond to anticipated transients to avoid the need for safety system action. Reactor control in Pickering NGS has a high degree of immunity to process upsets, measurement failures, etc., due to extensive redundancy in control devices and process measurements. The ability to maintain control in the presence of partial system failures, combined with high reliability of the Dual Computer Control System, leads to a very high availability of the Reactor Regulating System, which

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
		controls reactor power. The Reactor Regulating System also controls operational reactivity transients. As detailed in Appendix D for Safety Principle D-205, Pickering NGS has the SSCs in place to control the reactor power and provide core cooling in response to events occurring during startup, shutdown and low power operation, confirming that this AD group is an AD for PSR2. There is only one AD group associated with this safety principle, and therefore there is no aggregate impact for Safety Principle D-205, Startup, Shutdown and Low Power Operation.
O-305 – Maintenance, Testing and Inspection	GI-16-AD1 GI-20-AD1 GI-20-AD2	The following ADs are assigned to Safety Principle O-305. • GI-16-AD1 is related to the extent of inspection of Containment components deemed inaccessible for inspection. • GI-20-AD1 involves developing an implementation plan for reviewing and updating Condition Assessments. • GI-20-AD2 concerns oversight and implementation of the Integrated Aging Management program [N-PROG-MP-0008, *Integrated Aging Management*, May 2, 2016] All of these ADs involve refinements or improvements to existing effective OPG processes for maintenance, testing and inspection, and none indicates a concern with the actual condition of the Pickering NGS SSCs. For aggregation, the ADs are grouped into 2 categories: • An AD that focuses on a specific aspect (GI-16-AD1). The CSA N285.5 issues in this AD involve governance for removal of insulation and the inspection interval. The Compliance Matrices in the relevant Pickering NGS Periodic Inspection Plans [NK30-PIP-03642.2-00001 R003, OPG Plan, *Pickering Nuclear Generating Station "B" Periodic Inspection Program for Containment Components*, July 31, 2012], [NA44-PIP-03642.2-00001 R002, OPG Plan, *Pickering Nuclear Generating Station "A" Periodic Inspection Program for Containment Components*, July 31, 2012] and [P-PIP-03642.2-00001 R003, OPG Plan, *Pickering Nuclear Generating Station Periodic Inspection Program for Unit 0 Containment Components*, July 31, 2012] state that full or partial disassembly of components for the purpose of inspection will not be undertaken specifically for periodic inspection, as this may result in component damage. These Periodic Inspection Plans have been accepted by the CNSC. Furthermore, as indicated in the resolution for GI-16-AD1 in Appendix B of this report, where the inspection of a system or component would necessitate the dismantling of equipment, the required inspection should be performed when the equipment is dismantled for other reasons (i.e., maintenance).

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		The issue is assessed as a very low safety significance AD with no further action. • ADs that are of broader impact, and therefore overlap the more specific ADs (GI-20-AD1 and GI-20-AD2) OPG is actively progressing the tasks covered by both GI-20-AD1 and GI-20-AD2 for the extended operating period. OPG has robust and multi-faceted SSC inspection, monitoring and assessment programs, such that the current condition of the plant is well understood. This provides a sound baseline for extended operation as well as confidence that the requisite data will continue to be acquired. Maintenance programs are aligned with industry best practice and are carried out in accordance with written procedures that are part of the overall management system. Therefore, the overall conclusion for O-305 is that the aggregate impact on defence-in-depth of these ADs is very low, and mitigated by the maintenance, testing and inspection programs in place.
O-284 – Operating Limits and Conditions	GI-11-AD1 PSR1-AD2	GI-11-AD1: This AD is related to providing additional information on the Fuel Management and Surveillance Software and updating Operational Safety Requirements documents (SOE) for Fuel. OPG is actively progressing completion of the OPG commitment to provide additional information to address the related Action Item 2016-OPG-8250. Since the issue associated with the AD will be closed in the near future, this AD does not pose any risk to the plant for extended operation and there is no aggregation impact. PSR1-AD2 involves issues relating to Staffing, the Human Factors Engineering (HFE) Program [N-MAN-06700-10002, *Guide for OPG Human Factors Engineering Process*, December 18, 2015] and Verification and Validation during the design process. Some of the issues in this AD group deal with the requirement in CNSC Guide G-323 that the minimum staff complement be determined by the licensee through a systematic analysis and practical validation. Changes in the minimum shift complement as a result of these actions are identified in an OPG letter to the CNSC, [P-CORR-00531-03710, *Pickering A and B Request for Licence Amendments – Minimum Shift Complement*], wherein it is noted that: "The validation methodology has been documented and validation exercises have been observed by CNSC Staff." Therefore, there is no longer a gap in this area. The Pickering B ISR raised a gap on the basis that for smaller modifications, explicitly documenting each validation element was not judged necessary. HFE evaluates the role of humans in human-machine systems and how systems can be designed to work

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		well with people, particularly in terms of safety and efficiency. Operation of the Pickering NGS design over the years has demonstrated that the plant layout and facilities provide a safe working environment. Operating experience and improvements have been incorporated into the processes and design to improve the human-machine interfaces in many areas (e.g., Control Room annunciation upgrades as a result of changes to computer hardware and operator interface). Additionally, training and qualification processes (and certification processes for Control Room staff) for Operations positions ensure that the staff are competent to carry out functions assigned to them. The simulator is used extensively for initial training and qualification, as well as for refresher/requalification training. For example, per N-INS-08920-10002, *Simulator-Based Initial Certification Examinations for Shift Personnel*, Simulator Exercise Guides are used as part of training for certified staff. Since 2000, HFE has been explicitly considered for all design changes at Pickering NGS, resulting in continuous improvements to the human-machine interfaces throughout the plant. The PSR2 review confirmed that the Human Factors Engineering Program Plans [N-MAN-06700-10002-R004, *Guide for OPG Human Factors Engineering Process*, December 18, 2015] are prepared by OPG meet the requirements of CNSC G-276 and CNSC G-278, as well as applicable elements from NUREG-0711. This program is identified as a Strength for Pickering NGS. The assessment presented above shows that both areas have been fully addressed at Pickering NGS and therefore, these ADs do not contribute to the aggregation for Level 1.
O-278 – Training	PSR1-AD2	PSR1-AD2 involves issues relating to the Human Factors Engineering Program and Verification and Validation during the design process. The ADs were identified for smaller modifications where explicitly documenting each validation element was not judged necessary. Pickering NGS was originally designed to the standards of the day and designers relied on best design practices in addition to incorporation of operations and maintenance experience. The PSR2 assessment in the Safety Factor 1 Report confirmed that Pickering NGS has demonstrated that the plant layout and facilities provide a safe working environment. The training and qualification processes (and certification processes for Control Room staff) in place for Operations positions ensure that the staff are competent to carry out functions assigned to them. In addition, the PSR2 review confirmed that Human Factors Engineering Program Plans meet the requirements of CNSC G-276 and CNSC G-278, as well as applicable elements from NUREG-0711. This program is identified as a Strength for Pickering NGS. There is only one AD group associated with this safety principle, and therefore there is no aggregate

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
		impact for Safety Principle O-278, Training.
D-188 – Radiation Protection in Design	PSR1-AD17	PSR1-AD17 is assigned to Safety Principle D-188. This AD group involves IAEA requirements for radiation protection. IAEA NS-R-1 requires that consideration of the buildup of radiation levels with time in areas of personnel occupancy is to be provided for in the design of the plant and in housekeeping practices during the conduct of operating and maintenance activities. At OPG, this is done with the objective of minimizing exposures of station personnel and the generation of radioactive materials as waste. In addition, the review in the Safety Factor 15 Report has confirmed that Radiation Protection has been adequately accounted for in the design and operation of Pickering NGS, and that Radiation Protection provisions (including design and equipment) provide adequate protection of persons from the harmful effects of radiation and ensure that contamination and radiation exposures and doses to persons are monitored and controlled and maintained ALARA. Furthermore, the Regulatory Oversight Report for Canadian Nuclear Power Plants: 2015 states that CNSC staff concluded that the “Radiation Protection” and “Operating Performance” Safety and Control Areas at Pickering met or exceeded performance objectives and all applicable regulatory requirements, and rated these Safety and Control Areas as Fully Satisfactory. There is only one AD group associated with this safety principle, and therefore there is no aggregate impact for Safety Principle D-188, Radiation Protection in Design.
S-136 – External Factors Affecting the Plant	GI-13-AD1 GI-13-AD2 GI-13-AD3 GI-13-AD4 GI-43-AD1 PSR1-AD1	The following ADs and AD groups are assigned to Safety Principle S-136. GI-13-AD1: This AD is related to the CSA N289.3 requirements to confirm that a) the generated time history used within seismic analyses of safety-related systems correctly represents the design ground response spectrum for the Pickering site in compliance with CSA N289.3-10, and b) the power spectral density (PSD) function of each time-history has been calculated and shown to not have any significant gaps in energy over the frequency intervals. The rationale for considering this gap as an AD is the time-history method is a typical method for seismic qualification that has been employed at Pickering NGS. Time-history ground motion inputs at Pickering NGS were established based on the requirements of CSA N289.3-M81 and that the seismic analysis method is conservative and adequate, which is confirmed by the Pickering Seismic PRA that concludes the seismic design of systems and structures are sound. Consequently these gaps are considered to have no impact on the seismic resistance of qualified structures. GI-13-AD2: This AD is related to the CSA N289.4-12 requirements for confirmation that aging effects that

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		may impair seismic functionality shall be accounted for in seismic qualification by testing. Routine monitoring measures ensure that degradation effects will not be permitted to advance to the extent that seismic functionality is impaired. OPG's commitment [N-CORR-00531-05661, Design Codes and Standards Effective Dates for OPG Nuclear Fleet, April 2012] to using CSA N289.4-12 for testing procedures for seismic qualification of nuclear power plant SSCs will ensure that the requirements will be followed. Also, administrative controls [N-STD-MP-0025, *General Requirements For Seismic Qualification of OPG Nuclear Facilities*, October 27, 2016] ensure that seismic qualification of SSCs performing safety-related functions during and following an earthquake is maintained for the life of the facility. Consequently this gap is considered to have no impact on the seismic resistance of equipment qualified by test. GI-13-AD3: This AD is related to the CSA N289.5 requirement for seismic monitoring/recording. OPG has installed in-plant seismic instrumentation to monitor seismic activity at Pickering NGS (and at Darlington) in compliance with CSA standard N289.5 [NK30-DM-61150-10001, *Pickering Nuclear: Seismic Monitoring System*, September 2013; N-GUID-02004-10000-R00, Seismic Monitoring of OPG Nuclear Generating Stations, December 2010]. OPG has established procedures - Abnormal Incidents Manuals - that detail station response to earthquakes. Clear responsibilities are established to support monitoring and post-seismic response to an event [NK30-AIM-058-09013-6.0, *Abnormal Incident Manual: Seismic/Common Mode Event*; NA44-AIM-014-09013-06, *Abnormal Incident Manual: Seismic Event*]. OPG is also a contributor to the operation of the Southern Ontario Seismic Network (SOSN) which provides detailed free-field seismic records covering Southern Ontario. These systems support in-plant monitoring of the station's response to seismic events. The intent of the requirements has been met, and for these reasons, and as per the very low safety significance (Safety Significance Level 4), this is assessed as an Acceptable Deviation with no further action. GI-13-AD4 (for Units 1 and 4 only): This AD is related to a requirement in CSA N289.3 for the minimum number of cycles used for seismic fatigue analysis for the level of compliance of Pickering NGS plant structures supporting the operation of Pickering NGS reactors in the context of extended operation. Pickering Units 1,4 Class 1 systems and components are designed to ASME Code Section III which betters the CSA N289.3 requirement, including required cycles for fatigue analysis. Also, Seismic Margin Assessment is an accepted method for assessing seismic qualification and as such, this is an Acceptable Deviation for Pickering 1,4. For continuing analysis of potential new modifications, the requirements of CSA N289.3-10 with regards to number of cycles for seismic fatigue analysis would be met or bettered, consistent with the modelling methodology in place at OPG for such work as noted above. For these reasons, and as per the very low safety significance (Safety Significance Level 4), this is assessed as an

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		Acceptable Deviation for Pickering 1,4 with no further action. GI-43-AD1: This AD is related to a new requirement in CSA N291-15 for bolted connections in members that are part of the seismic load resisting system. The original code requirements for PNGS safety related structures [National Building Code of Canada] and the Seismic Margin Assessments (SMA) include requirements of CSA S16 “Design of Steel Structures”, which is the basis for the requirements for bolted connections of steel structures in CSA N291-15. Given that the SMA methodology and the plant design both include requirements of S16/S16.1 (predecessor of S16), the plant meets or has been assessed to these requirements. As it would not be practicable to make changes to the bolted steel connections in the existing structural design, and given the low safety significance (Significance Level 3), this is assessed as an Acceptable Deviation with no further action. PSR1-AD1 involves requirements for seismic-qualification. The following issues in PSR1-AD1 were raised in the Darlington and Pickering B ISRs. In terms of Safety Analysis, CSA N289.2 does not specify any specific requirements relating to the seismic success path. It is focused on a systematic approach to identify the seismic hazard. As such, there are no specific clauses relating to Safety Analysis. However, it is recognized that the magnitude and frequency of the spectra impacts on the qualification of SSCs and that if their required qualification were to change, the success path may need to change. The Darlington ISR raised a gap against Clause 5.5.2.1 of CSA N289.3 because the time histories are not statistically independent. The issues in PSR1-AD1 generally relate to the seismic design report of DNGS structures not explicitly accounting for effects due to • soil-structure interaction (SSI), • the difference between modeling techniques and analysis approaches for the evaluation of the site response analysis for the free-field case, and • complete interaction versus the sub-structure technique for the soil/foundation/structure system. These issues are also applicable to Pickering NGS. The Periodic Inspection of the concrete structures and routine leakage tests of the Containment envelope verify that there is no significant deterioration in the integrity of the Concrete Containment Structures. The rationale for considering gaps related to seismic qualification as an AD is that the seismic PRA confirms that Pickering NGS conforms to CNSC S-294 and to various international guides on seismic

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
		PRA, and that the results are acceptable. Consequently these ADs are considered to have no impact on the seismic resistance of qualified structures. This AD group is therefore deemed to have no impact on this aggregation assessment. GI-13-AD1, GI-13-AD4 and GI-43-AD1 have the same attribute “Seismic Design” insofar as the gaps are all related to the design programs by which the qualified SSCs were designed. While this is a common factor, the seismic design of Pickering NGS has been shown to be conservative such that all qualified SSCs will withstand the effects of a design basis seismic event. The seismic design aspect covered by PSR1-AD1 regarding soil structure interactions and GI-13-AD3 for monitoring have nothing in common with the other ADs with the attribute “Seismic Design”. Hence there is no aggregation effect. The aggregate impacts of these 6 seismic ADs on Pickering NGS seismic design integrity capability is negligible and hence has no impact on public safety.
O-269 – Safety Review Procedures	GI-30-AD1	GI-30-AD1 concerns methodology and processes for evaluation of instantaneous risk. An industry initiative is underway to develop a methodology and processes for evaluation of instantaneous risk. The current instantaneous risk monitoring uses the At-Power and Outage Internal Events PSA models. At-Power and Outage instantaneous risk Safety Goal monitoring is proceduralized and implemented regularly for online maintenance and during each unit outage. OPG is continuing to participate in the industry initiative on further evaluation of instantaneous risk. Operationalization of Probabilistic Safety Assessment is a Strength for Pickering NGS. PSA is used to support the conduct of engineering, maintenance and operation at Pickering NGS as follows: • Proposed modifications to plant operation, configuration or procedures that may significantly increase the risk of a severe accident are reviewed to quantify impact on risk and to assess its acceptability. Similarly, any modifications that may reduce risks are to quantify the benefits in terms of impact on risk as in input to decision-making. • PSA assumptions important to safety regarding inspection, testing and maintenance are identified and incorporated into operating and maintenance procedures. • PSA is used to identify accident scenarios. • PSA is used to support in-plant and ex-plant consequence analyses for event sequences beyond the design basis for use in understanding severe accident progression and management, as allowed by

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
		the scope and limitations of the PSA. There are other processes at Pickering NGS that provide assurance of defence-in-depth during temporary outage or maintenance alignments, and these deterministic considerations apply regardless of the potential hazard. Evaluation of instantaneous risk using additional instantaneous risk tools beyond the current at-power and outage risk assessments is considered an enhancement to the existing means that OPG has for determining risks during normal operation and planned events. Therefore, this issue has no impact on the aggregate assessment of ADs in this Level. There is only one AD associated with this principle, and therefore there is no aggregate impact for Safety Principle O-269, Safety Review Procedures.
M&C-246 – Safety Evaluation of Design	PSR1-AD15 PSR1-AD19 PSR1-AD22	The following AD groups are assigned to Safety Principle M&C-246. • PSR1-AD15 involves safety-related requirements for plant design. This AD group relates to Single Failure Criterion issues that were raised against IAEA NS-R-1 and CNSC RD-337 and assessed as ADs. The PSR2 review of REGDOC-2.5.2, which adopts principles from NS-R-1 and supersedes CNSC RD-337, included a detailed assessment of PSR1 ADs. Also, the PSR2 assessment of CSA N290.2-11, “Requirements for Emergency Core Cooling Systems of Nuclear Power Plants", identified an AD for the Pickering 1,4 ECI system, which contains singleton components required to operate for the system to operate successfully, on the basis that all practical modifications were made to the Pickering 1,4 ECI system during Pickering A Return to Service and the system meets its unavailability target. The Pickering NGS design permits testing, maintenance and other necessary activities to be completed while keeping the system available. As stated in the PSR2 code review for REGDOC-2.5.2 [P-REP-03680-00029-R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 1, 5, 6, and 7*, March 2, 2017], this issue was transferred to the Pickering B Continued Operations Plan, where it was closed on the basis that the Level 1 and Level 2 PSA demonstrated that the existing plant design satisfied all safety requirements. The applicability of the gap resolution and subsequent disposition is equally applicable to Pickering 1,4, where the PSAs have similarly been updated. Therefore, this issue is of low safety significance. • PSR1-AD19 involves deterministic safety analysis requirements. The ADs in this group are related to dual trip parameter coverage, the criterion for no dry-out for the first trip, the allowable duration of post dry-out operation for the second trip and interpretation of the acceptance criterion relating to the acceptable duration of post dry-out operation. However, the

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
		derived acceptance criteria outlined in the COG report [COG-13-9035*, Derived Acceptance Criteria For Deterministic Safety Analysis*, November 2014], developed by the industry taking into account feedback from the CNSC, are less restrictive than those outlined in the current Pickering 1,4 Safety Report [NA44-SR-01320-00002-R004, *Pickering Nuclear 1-4 Safety Report: Part 3 – Accident Analysis*, October 31, 2013] and Pickering 5-8 Safety Report [NK30-SR-01320-00003-R004, *Pickering Nuclear 5-8 Safety Report: Part 3 – Accident Analysis*, October 30, 2014], providing confidence that the derived acceptance criteria documented in COG-13-9035 will not adversely impact on existing margins as demonstrated in the Safety Reports. The PSR2 review confirmed that the PSR1 AD rationale for PSR1-AD19 continues to remain applicable and is not impacted by extended operation. • PSR1-AD22 - These ADs are related to requirements for development, verification and review of computer programs used for Safety Analysis, as well as requirements for software validation. OPG has ensured quality in the development of its safety analysis tools. OPG has been participating in an industry-wide COG program addressing software validation. Industry efforts towards industry standard toolsets and Generic Action Item 98G02 *Validation of Computer Programs Used in Safety Analysis of Power Reactors* detail some of the work ongoing to comply with CNSC guide G-149 (2000), Computer Programs Used In Design and Safety Analyses of Nuclear Power Plants and Research Reactors. CNSC Generic Action Item GAI 98G02 was closed with the CNSC concluding that OPG (and industry) established an acceptable process to improve computer code validation by achieving an overall level of baseline validation for a specific set of major computer codes used in safety analyses. Also, a review of CNSC guide G-149 (2000) was performed at a high level as part of the Pickering B ISR which confirmed that OPG's governance and processes adequately addressed the intent of the guide. Based on the justification above, the quality assurance of computer codes is considered to be sufficient at the overall level, so the effect of this issue is limited without impairing the capability of safety provisions. The results of the Darlington ISR from the perspective of G-149 are considered applicable to Pickering 1,4 and 5-8, given that the assessment was performed against OPG governance applicable to all of the OPG Nuclear fleet. The ADs in this group that relate to barriers in Safety Principle M&C-246, Safety Evaluation of Design, have no significant interaction because the issues are not cumulative in nature and individual groups have

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
		low safety significance or safety impact. Also, these ADs are related to issues which, for the most part, have been addressed to minimize any potential risk to plant operation. In the case of issues related to the Single Failure Criterion, Level 1 and Level 2 PSA demonstrate that the existing plant design satisfies all PSA-related safety requirements. The ADs related to Fuel and Fuel Channel derived acceptance criteria are addressed in the current Pickering 1,4 and Pickering 5-8 Safety Reports, providing confidence that the derived acceptance criteria documented in industry accepted report [COG-13-9035*, Derived Acceptance Criteria For Deterministic Safety Analysis*, November 2014] will not adversely impact on existing margins. For ADs related to requirements for development, verification and review of computer programs used for Safety Analysis, the quality assurance of computer codes is considered to be sufficient at the overall level, so that the effect of this issue is limited without impairing the capability of safety provisions. Therefore, the overall conclusion for M&C-246 is that the aggregate impact on defence-in-depth of these ADs is negligible.
M&C-249 – Achievement of Quality	PSR1-AD22	PSR1-AD22 is related to requirements for the development, verification and review of computer programs used for Safety Analysis, as well as requirements for software validation. The full scope of OPG governance establishes a QA program that is robust and compliant, and thus ensures the development of Scientific, Engineering and Safety Analysis software is adequately aligned with the requirements of CNSC G-149 and CSA N286.7. The OPG Program, N-PROG-MP-0009, *Design Management*, provides requirements relevant to the design and development of Scientific, Engineering and Safety Analysis software. The requirements for design inputs and design outputs are given in the Design Management program. This provision in the governance requires definition of the inputs and outputs as part of the software design process. This governing document also addresses requirements for a formal Design Plan, with instructions on the need for stakeholder input during the design process. This AD group is also assessed under M&C-246, Safety Evaluation of Design. The assessment concludes that the quality assurance of computer codes, including legacy codes, is sufficient at the overall level, so the effect of this issue is limited without impairing the capability of safety provisions. The results of the Darlington ISR from the perspective of G-149 are considered applicable to Pickering 1,4 and 5-8, given that the assessment was performed against OPG governance applicable to all of the OPG Nuclear fleet. There is only one AD group associated with this principle, and therefore there is no aggregate impact for

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
		Safety Principle M&C-249, Achievement of Quality.

Overall AD Aggregation for Defence-in-Depth Level 1
The ADs associated with the safety principles in Level 1 defence-in-depth cover the following diverse areas. • D-150 – Design Management –The ADs associated with Safety Principle D-150, Design Management, have no significant interactions because the issues are against unique SSCs, and individual groups have low or very low safety significance or safety impact. OPG's Design Management Program, OPG-PROG-MP-0009, specifies requirements for procurement engineering processes ensuring that implementation and maintenance of the nuclear facilities meet the design basis requirements. The Design Management Program provides assurance that design and procedure changes are prepared, reviewed, approved, documented and implemented in accordance with approved procedures, applicable regulatory requirements, standards and industry practices. • C-255 – Verification of Design and Construction - The findings during the PSR1 were assessed to be ADs given the robust design practices at the time of construction, together with ongoing periodic inspections and in-service testing. The AD rationale continues to remain applicable and is not impacted by the planned extended operation. There is only one AD group associated with this safety principle, and therefore there is no aggregate impact. • D-205 – Startup, Shutdown and Low Power Operations - As detailed in Appendix D for Safety Principle D-205, Pickering NGS has SSCs in place to control the reactor power and provide core cooling in response to events occurring during startup, shutdown and low power operation, confirming that this AD group is an AD for PSR2. There is only one AD group associated with this safety principle, and therefore there is no aggregate impact for Safety Principle D-205, Startup, Shutdown and Low Power Operation. • D-195 – Reactor Core Integrity – This safety principle has only one associated AD group, which is mitigated by multiple OPG programs related to management of design, surveillance of components and equipment, and equipment reliability. The Equipment Reliability Program, N-PROG-MA-0026, and the Major Components Program, N-PROG-MA-0025 are identified as Strengths for Pickering NGS. • D-186 – Inspectability of Safety Equipment – The associated AD group is related to concessions in the Periodic Inspection Plans for components that are deemed inaccessible. Although the issue relates to inspection requirements for inaccessible areas, the CNSC has been informed and accepted the revised PIPs in terms of meeting the requirements and the remedial steps identified in the CSA N285.5-08, *Periodic inspection of CANDU Nuclear Power Plant Containment Components*, Update No. 1 standard. This is not impacted by the planned extended operation beyond 2020. • D-188 – Radiation Protection in Design – The Safety Factor 15 Report has confirmed that Radiation Protection is adequately accounted for in the design and operation of Pickering NGS, and that Radiation Protection provisions (including design and equipment) provide adequate protection of persons from the harmful effects of radiation and ensure that contamination and radiation exposures and doses to persons are monitored and controlled and maintained ALARA. • S-136 – External Factors Affecting the Plant – Seismic PRAs have been issued for Pickering 1,4 and Pickering 5-8 that confirm that Pickering NGS conforms to CNSC S-294 and to various international guides on seismic PRA. Consequently these ADs have no impact on the seismic resistance of qualified structures. Therefore, these ADs do not contribute to aggregation or pose any significant risk. • O-269 - Safety Review Procedures – There are deterministic rules at Pickering NGS that provide assurance of defence-in-depth during

Overall AD Aggregation for Defence-in-Depth Level 1
temporary outage or maintenance alignments, and these deterministic considerations apply regardless of the potential hazard. Evaluation of instantaneous risk using PSA is considered an enhancement to the existing means that OPG has for determining risks during normal operation and planned events. • O-305 - Maintenance, Testing and Inspection – This is area has numerous ADs and interactions were identified. However, OPG has a robust and multi-faceted SSC inspection program, such that the current condition of the plant is well understood. This provides a sound baseline for extended operation as well as confidence that the requisite data will continue to be acquired. In addition, maintenance programs are aligned with industry best practice and are carried out in accordance with written procedures that are part of the overall management system. • O-284 - Operational Limits and Conditions – There are two groups of ADs for this safety principle. However, one related to Minimum Staff Complement has been closed. The other related to Human Factors Engineering has been addressed by means of a very strong and robust Human Factors Engineering program [N-MAN-06700-10002-R004, *Guide for OPG Human Factors Engineering Process,* December 18, 2015]. PSR2 review confirmed that Human Factors Engineering Program Plans prepared by OPG meet the requirements of CNSC G-276 and CNSC G-278, as well as applicable elements from NUREG-0711. • O-278 – Training – One AD was assessed for this safety principle, related to smaller modifications where explicitly documenting each validation element was not judged necessary. The PSR2 assessment in Safety Factor 1 Report confirmed that the Pickering NGS plant layout and facilities provide a safe working environment. The training and qualification processes (and certification processes for Control Room staff) in place for Operations positions ensure that the staff are competent to carry out functions assigned to them. • M&C-246 – Safety Evaluation of Design – The groups of ADs have no significant interaction because the issues are not cumulative in nature and individual groups have low safety significance or safety impact. Also, these ADs are related to issues which, for the most part, have been addressed to minimize any potential risk to plant operation. • M&C-249 – Achievement of Quality – One AD was assessed for this safety principle. The AD is related to requirements for the development, verification and review of computer programs used for Safety Analysis, as well as requirements for software validation. The full scope of OPG governance establishes a QA program that is robust and compliant, and thus ensures the development of Scientific, Engineering and Safety Analysis software is adequately aligned with the requirements of relevant codes and standards. Some of the ADs have the same attribute 'Design' insofar as the gaps are all related to the design program by which the qualified SSCs were designed. While this is a common factor, the design of Pickering NGS has been shown to be conservative, and conformance to the design basis is continually confirmed by analysis and by the ongoing maintenance, testing and inspection programs at OPG. Equipment Reliability and maintenance, testing and inspection have several ADs and interactions were identified. However, OPG has robust and multi-faceted SSC inspection, monitoring and assessment programs, such that the current condition of the plant is well understood. This provides a sound baseline for extended operation as well as confidence that the requisite data will continue to be acquired. Maintenance programs are aligned with industry best practice and are carried out in accordance with written procedures that are part of the overall

Overall AD Aggregation for Defence-in-Depth Level 1
management system. Regular preventive and predictive maintenance, inspection, testing and servicing of SSCs important to safety and reliability are conducted to maintain SSCs within their design basis. Implementation and execution of Pickering NGS's Equipment Reliability Program, N-PROG-MA-0026, is identified as a Strength for Pickering NGS, as is the Major Components Program, N-PROG-MA-0025. Common attributes related to Human Factors Engineering are addressed by the demonstrated safe and effective working environment at Pickering NGS. The programs for initial, refresher, and upgrade training and qualification processes (and certification processes for Main Control Room staff) are in place for Operations positions and ensure that the staff are competent to carry out functions assigned to them. The aggregate impact of the ADs for each safety principle is determined to have no significant impact on safety. Furthermore, the overall aggregate impact of all of the ADs associated to Level 1 defence-in-depth is assessed to be low.

AD Aggregation for Defence-in-Depth Level 2

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
D-158 – General Basis for Design	GI-31-AD1 GI-44-AD1	The following ADs are assigned to Safety Principle D-158. • GI-31-AD1 involves consideration of AOOs in the Pickering NGS Deterministic Safety Analysis. • GI-44-AD1 is related to REGDOC-2.5.2 requirements and limits for AOOs, DBAs and BDBAs. As part of the PSR2 review, modern versions of the codes and standards were assessed to determine the extent to which Pickering NGS conforms with safety-significant requirements of the new standards, namely CNSC REGDOC-2.5.2 and CNSC REGDOC-2.4.1. Required system performance under accident conditions is addressed in the Pickering 1,4 Safety Report [NA44-SR-01320-00002-R004, *Pickering Nuclear 1-4 Safety Report: Part 3 – Accident Analysis*, October 31, 2013] and Pickering 5-8 Safety Report [NK30-SR-01320-00003-R004, *Pickering Nuclear 5-8 Safety Report: Part 3 – Accident Analysis*, October 30, 2014]. Also, a full range of initiating events, including AOO-type sequences, is considered in the PSAs [P-REP-03611-00006-R00, *Pickering NGS PSA Update to Include Enhancements from the Fukushima Integrated Action Plan*, April 30, 2014]. However, the term AOO is not used in the current Pickering Safety Reports even though AOO-type sequences are assessed. Pickering NGS has a comprehensive list of events (Initiating Events) defined for the safety analyses that are modelled in the PSA and the risk is acceptably low. Consideration of the identification and analysis of AOOs is addressed under REGDOC-2.4.1 implementation. The implementation plan will be updated in accordance with the Licence Conditions Handbook and will identify any changes required to support the continued safe operation of Pickering NGS. These changes will be informed by the timeline of the Darlington REGDOC-2.4.1 Implementation Plan and the limited additional years of Pickering NGS operation. Level 2 defence-in-depth is achieved by detecting deviations from normal operating conditions by the Reactor Regulating System and plant process and control systems. The control systems in Pickering NGS maintain reactor operating conditions within the normal operating range, and effectively respond to anticipated transients to avoid the need for safety system action. Reactor control in Pickering NGS has a high degree of immunity to process upsets, measurement failures, etc., due to extensive redundancy in control devices and process measurements. The ability to maintain control in the presence of partial system failures, combined with high reliability of the Dual Computer Control System, leads to a very high availability of the Reactor Regulating System, which controls reactor power.

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
		As regards Level 2 defence-in-depth, both of these ADs cover the same requirements for consideration of AOOs. Both ADs are being addressed by OPG through updating the REGDOC-2.4.1 Implementation Plan for extended operation. Therefore, the aggregate impact of these ADs on defence-in-depth is assessed to be negligible.
D-205 – Startup, Shutdown and Low Power Operation	PSR1-AD10	PSR1-AD10 involves CSA N290.4, Requirements for Reactor Control Systems of Nuclear Power Plants, requirements for the Reactor Regulating System (RRS) related to Start-up Instrumentation, documentation, reliability, and human-system interface. These issues were identified during the Pickering A Return to Service (PARTS) review, and are related to the requirement for the design of the Reactor Regulating System and for the operator to initiate monitoring of the flux tilt and initiate power reduction as needed. The PSR2 confirms the continued applicability of ADs on the following basis: 1. The standard requires the Reactor Regulating System to be physically and functionally separated from the Special Safety Systems. (This is related to clause 5.2 in CSA N290.4-11). Special Safety Systems and the Reactor Regulating System are mostly independent of the major control loops, but there are three exceptions identified in the code review: • Sharing of taps and impulse lines from flow orifices in reactor flow measurement loops between RRS and Shutdown System A (SDSA). The code review states that irrational inputs to RRS control programs are rejected and SDSA channel flows are continually monitored in the Control Room to ensure they remain in the normal operating range. Therefore this is assessed as an AD with no safety impact. • There are a number of RRS hardware and program interlocks related to the shutdown function, e.g., gradually filling all zones when the reactor is tripped. These interlocks do not affect SDS functionality and are standard practice in CANDU design. Also, these interlocks drive RRS reactivity devices in the safe direction. Therefore this is assessed as an AD with no safety impact. • In the control of Moderator level, RRS and ECI share some components, i.e., the common Helium bubbler supply in the Calandria and Dump Tank level measurements. The loss of the bubblers does not result in a loss of regulation, therefore there will be no postulated demand on either the in-core logic or ECI injection from this event and, therefore, sharing the bubblers is acceptable with no safety impact. 2. An AD is identified with clause 4.3.1.2 of CSA N290.4, which requires that all components of the system (including power sources) shall be included in the reliability calculations. This clause is

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
		superseded by CSA N290.4-11 clauses 5.4.2 and 5.4.3, which require the design target reliability of RRS be established in the Probabilistic Safety Assessment. The Pickering 1,4 Probabilistic Safety Assessment [P-REP-03611-00006-R000, *Pickering NGS PSA Update to Include Enhancements from the Fukushima Integrated Action Plan,* April 30, 2014] demonstrates that the RRS reliability including any supporting systems is acceptable. Therefore this is assessed as an AD with no safety impact. 3. Clause 4.3.6.1 requires RRS to control neutron flux to obtain an acceptable spatial distribution and be capable of counteracting flux distortions that may otherwise cause Fuel bundle or channel power limits to be exceeded. (This is related to CSA N290.4-11 clause 4.1.2 (c)). Pickering 1,4 are not equipped with automatic reactor power setback or an alarm based on high flux tilt. Flux tilts are managed by operator monitoring and by the initiation of a power reduction as specified in operating manuals. This deviation has been accepted by the CNSC. Therefore this is assessed as an AD with no safety impact. 4. Clause 4.3.25.4 requires that where a number of channels of one system are in proximity, colour coding is the preferred identification method. This is not implemented in the Pickering 1,4 RRS. This requirement is referred to in clause 5.11.4 in CSA N290.4-11 as one of several methods of facilitating the human-system interface in the design and, therefore, is still relevant. However, the design Control Room panels for the RRS are clearly organized and labeled, such that colour coding is not required. Therefore this is assessed as an AD with no safety impact. Assessment of this safety principle in Appendix D of this report confirms that Pickering NGS has the SSCs in place to control the reactor power and provide fuel cooling in response to events occurring during startup, shutdown and low power operation. Level 2 defence-in-depth is achieved by detecting deviations from normal operating conditions by the RRS and plant process and control systems. The deviation from control is achieved by the control systems in Pickering NGS that maintain the reactor operating conditions within the normal operating range, and effectively respond to anticipated transients to avoid the need for safety system action. Reactor control in Pickering NGS has a high degree of immunity to process upsets, measurement failures, etc., due to extensive redundancy in control devices and process measurements. The ability to maintain control in the presence of partial system failures, combined with high reliability of the Dual Computer Control System, leads to a very high availability of the Reactor Regulating System, which controls reactor power. There is only one AD associated with this safety principle, and therefore there is no aggregate impact for Safety Principle D-205, Startup, Shutdown and Low Power Operation.

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
D-230 – Preservation of Control Capability	PSR1-AD10	PSR1-AD10 involves requirements for RRS related to Start-up Instrumentation, documentation, reliability, and human-system interface. The Control Rooms in Pickering NGS are designed to remain habitable during normal operation and postulated transients and DBAs. Should a Main Control Room become uninhabitable, the SDSE instrument rooms in Pickering 1,4 and the Unit Emergency Control Centres in Pickering 5-8 are designed to inform operations staff of key plant parameters and allow for critical safety system action to be initiated. There is only one AD associated with this safety principle, and therefore there is no aggregate impact for Safety Principle D-230, Preservation of Control Capability.
O-284 – Operational Limits and Conditions O-278 – Training	PSR1-AD2	PSR1-AD2 involves issues relating to the Human Factors Engineering Program and Verification and Validation during design process. The Pickering B ISR raised a gap on the basis that for smaller modifications, explicitly documenting each validation element was not judged necessary. Operating experience and improvements have been incorporated into the processes and design to improve the human-machine interfaces in many areas (e.g., Control Room annunciation upgrades as a result of changes to computer hardware and operator interface). Additionally, training and qualification processes (and certification processes for Control Room staff) for Operations positions ensure that the staff are competent to carry out functions assigned to them. The extensive use of Main Control Room simulators for training and validation of system modifications to assess their impact on other systems and human-machine interfaces provides OPG with a safe means of testing and training operators, thus strengthening the Level 2 defence-in-depth barrier. There is only one AD associated with these safety principles, and therefore no aggregate impact for Safety Principle O-284, Operational Limits and Conditions, or Safety Principle O-278, Training.

Overall AD Aggregation for Defence-in-Depth Level 2
The two PSR2 ADs assigned to Level 2 defence-in-depth, GI-31-AD1 and GI-44-AD1, are being addressed via the REGDOC-2.4.1 Implementation Plan for extended operation. The ADs are assessed to be low safety significance. The effects of the these ADs are mitigated by the robust engineered features of the control system that prevent deviation from control, a strong management system and well trained operating personnel. PSR1-AD2 involves issues relating to Human Factors Engineering Program and Verification and Validation during the design process. Operating experience and improvements that have been incorporated into the processes and design to improve the human-machine interfaces in many areas and the extensive use of Main Control Room simulators for training and validation of system modifications to assess their impact on other systems and human-machine interfaces provides OPG with an effective means of training and testing operators, thus strengthening the Level 2 defence-in-depth barrier. PSR1-AD10 involves requirements for RRS related to Start-up Instrumentation, documentation, reliability, and human-system interface. Level 2 defence-in-depth is achieved by detecting deviations from normal operating conditions by the Reactor Regulating System and plant process and control systems. Reactor control in Pickering NGS has a high degree of immunity to process upsets, measurement failures, etc., due to extensive redundancy in control devices and process measurements. The two PSR2 ADs and the two PSR1 AD groups have some elements in common, e.g., Human Factors elements appear in both of the PSR2 ADs. Overall, however, the issues associated with the ADs are sufficiently distinct and lack overlap, such that the aggregate effect has no impact on safety. Therefore, the overall aggregate impact of all of the ADs associated with Level 2 defence-in-depth is assessed to be negligible.

AD Aggregation for Defence-in-Depth Level 3

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
D-158 – General Basis for Design	GI-31-AD2 GI-39-AD1 GI-44-AD1 GI-44-AD8	The following ADs are assigned to Safety Principle D-158. • GI-31-AD2 concerns consideration of CSA N288.2-14 requirements for safety analysis using the ADDAM code. The ADDAM 1.4.2 code is being used in the update of the Pickering Safety Report Common Mode Events appendices, under the REGDOC-2.4.1 Implementation Plan for extended operation. As the REGDOC-2.4.1 Implementation Plan does not identify other updates of the Pickering Safety Reports' analyses to address changes in N288.2-14, the existing Safety Report analyses related to atmospheric dispersion and dose calculations continue to support the safety case for Pickering NGS. Also, ADDAM has been evaluated against the 2014 version of CSA N288.2 under a COG Project outside of PSR2. The section by section review showed that no major modifications to the ADDAM 1.4.2 code, methodology or manuals would be required [COG report ISTO-15-5057, *Assessment of Impact of CSA N288.2-14 on ADDAM*, March 2017]. On this basis, the issue is assessed as low safety significance (Safety Significance Level 3). • GI-39-AD1 involves legacy Real-Time Process Computing (RTPC) applications. OPG has a RTPC program in place that adequately deals with both legacy and new RTPC software installations in a manner that meets the intent of the CSA standard CSA-N290.14 and other standards (e.g., ISO/IEC). The OPG program document N-PROG-MP-0006, *Software*, identifies the processes and overall requirements for an effective Software Program that supports safe and efficient plant operation and that meets the intent of CSA-N290.14 and other standards. All OPG Nuclear software installations are subject to the OPG Software Program documents. The legacy software applications have decades of successful service, which indicates the adequacy of the current OPG approach and the programmatic implementation. Evaluation of the legacy software installations with respect to the N290.14-15 requirements is not practicable and would provide very little safety benefit. On this basis, this AD is assessed as very low safety significance (Safety Significance Level 4), and does not contribute to aggregation for this level of defence-in-depth. • GI-44-AD1 involves meeting REGDOC-2.5.2 requirements and limits for AOOs, DBAs and BDBAs. With the current deterministic analyses, dose limits for event sequences are as defined in the Licence Conditions Handbook [LCH-PNGS-R005, *Licence Conditions Handbook*, Pickering Nuclear

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
		Generating Station, Effective Date: November 7, 2016] (for single failure and dual failure sequences) and OPG is compliant with those requirements as demonstrated through the accident analyses in the Pickering Safety Reports. The DBA events bound the AOO sequences in terms of system response and consequences. In addition, the Pickering NGS PSAs consider a full range of event sequences – covering AOO-type events, DBAs and BDBAs. Pickering NGS meets its required risk-based Safety Goals as demonstrated through the PSAs. Furthermore, the REGDOC-2.4.1 Implementation Plan for the extended operation will be updated in accordance with the Licence Conditions Handbook and will identify any changes required to support the continued safe operation of Pickering NGS. These changes will be informed by the timeline of the Darlington REGDOC-2.4.1 Implementation Plan and the limited additional years of Pickering NGS operation. Additional practicable enhancements to further address this gap are not readily evident, and would provide limited safety benefit. On this basis, and given the low safety significance (Safety Significance Level 3), this is assessed as an AD and no further action will be taken. • GI-44-AD8 concerns operator action time credits. REGDOC-2.5.2 requirements for allowable times for operator action from the MCR or the field are more limiting than the corresponding requirements of REGDOC-2.4.1. The Pickering A and B Safety Report credits for operator actions from the MCR and in the field are consistent with REGDOC-2.4.1 requirements of 15 and 30 minutes, respectively. The ability to execute required actions within these time limits has been demonstrated through decades of operation, through effective training and testing programs. The gap against the corresponding REGDOC 2.5.2 requirements is a low safety significance issue and has been addressed to the extent practicable. Therefore, it is assessed as a low safety significance issue and no further action is planned. GI-31 addresses conformance with the safety significant requirements of REGDOC-2.4.1. For the purposes of this aggregation assessment, no additional enhancements to the OPG REGDOC-2.4.1 implementation plan are credited beyond those in the current plan. The Level 3 defence-in-depth aspect of GI-44-AD1 involves the requirements for DBAs, as well as for Level 3 mitigation of AOOs. GI-39-AD1 does not interact with any of the other ADs. There is interaction between GI-44-AD1, GI-44-AD8 and GI-31-AD2; that is, a change in operator action time credits (GI-44-AD8) could potentially affect predicted barrier conditions following DBAs (GI-44-AD1) and dose results calculated by ADDAM (GI-31-AD2). However, this interaction does not pose an aggregate impact for extended operation as it is already considered in the dispositioning of these ADs via the REGDOC-2.4.1

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
		Implementation Plan for the extended operation. Therefore, the aggregate impact on defence-in-depth of these ADs is assessed to be very low.
D-182 – Equipment Qualification	PSR1-AD1 PSR1-AD16	The following ADs are assigned to Safety Principle D-182, • PSR1-AD1 involves requirements for seismic-qualification. The following issues in PSR1-AD1 were raised in the Darlington and Pickering B ISRs. In terms of Safety Analysis, CSA N289.2 does not identify any specific requirements relating to the seismic success path. It is focused on a systematic approach to identifying the seismic hazard. As such, there are no specific clauses relating to Safety Analysis. However, it is recognized that the magnitude and frequency of the spectra impacts on the qualification of SSCs and that if their required qualification were to change, the success path may need to change. The Darlington ISR raised a gap against Clause 5.5.2.1 of CSA N289.3 because the time histories are not statistically independent. The issues in PSR1-AD1 generally relate to the seismic design report of DNGS structures not explicitly accounting for effects due to • soil-structure interaction (SSI), • the difference between modeling techniques and analysis approaches for the evaluation of the site response analysis for the free-field case, and • complete interaction versus the sub-structure technique for the soil/foundation/structure system. These issues are also applicable to Pickering NGS. The Periodic Inspection of the concrete structures and routine leakage tests of the Containment envelope verify that there is no significant deterioration in the integrity of the Concrete Containment Structures. The rationale for considering gaps related to seismic qualification as an AD is that the seismic PRA confirms that Pickering NGS conforms to CNSC S-294 and to various international guides on seismic PRA, and that the results are acceptable. Consequently these ADs are considered to have no impact on the seismic resistance of qualified structures. This AD group is therefore deemed to have no impact on this aggregation assessment. • PSR1-AD16 Requirements for Containment Coatings and Coverings. The gaps included in this issue concern non-metallic coatings and liners that perform a safety function, such as preserving the concrete Containment envelope. The basic requirement is that not only should

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
		these materials be carefully selected, and applied, but their deterioration (especially under post-accident conditions) should not impair the performance of other safety functions, such as blocking of ECI strainers. The seismic issues covered by PSR1-AD1 regarding soil-structure interactions and the issues covered by PSR1-AD16 regarding environmental qualification of non-metallic coatings and liners have nothing in common. Hence there is no aggregation effect.
D-150 – Design Management	PSR1-AD15 PSR1-AD19	These AD groups assigned to Safety Principle D-150, Design Management, have also been assessed under D-150 for their impact on Level 1. • PSR1-AD15 involves safety-related requirements for plant design. This AD group relates to Single Failure Criterion issues that were raised against IAEA NS-R-1 and CNSC RD-337 and assessed as ADs. The PSR2 review of REGDOC-2.5.2, which adopts principles from NS-R-1 and supersedes CNSC RD-337, included a detailed assessment of PSR1 ADs. Also, the PSR2 assessment of CSA N290.2-11, *Requirements for Emergency Core Cooling Systems of Nuclear Power Plants*, identified an AD for the Pickering 1,4 ECI system, which contains singleton components required to operate for the system to operate successfully, on the basis that all practical modifications were made to the Pickering 1,4 ECI system during Pickering A Return to Service and the system meets its unavailability target. Pickering NGS design permits testing, maintenance and other necessary activities to be completed while keeping the system available. This issue is addressed in the Pickering B Continued Operations Plan on the basis that the Level 1 and Level 2 PSA demonstrate that the existing plant design satisfies all safety requirements. The applicability of the gap resolution and subsequent disposition is considered to be equally applicable to Pickering 1,4, where the PSAs have similarly been updated [P-REP-03611-00006-R00, *Pickering NGS PSA Update to Include Enhancements from the Fukushima Integrated Action Plan*, April 30, 2014]. Therefore, the issue related to this group of ADs is assessed to be low safety significance. • PSR1-AD19 involves deterministic safety analysis requirements. The ADs in this group are related to dual trip parameter coverage, the criterion for no dry-out for the first trip, the allowable duration of post dry-out operation for the second trip and interpretation of the acceptance criteria relating to the acceptable duration of post dry-out operation. The derived acceptance criteria outlined in COG report [COG-13-9035*, Derived Acceptance Criteria For Deterministic Safety Analysis*, November 2014] are less restrictive than those outlined in the current Pickering 1,4 and Pickering 5-8 Safety Reports, providing confidence that the derived

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		acceptance criteria documented in COG-13-9035 will not adversely impact on existing margins as demonstrated in the Safety Reports. PSR2 confirmed that the PSR1 AD rationale for PSR1-AD19 continues to remain applicable and is not impacted by extended operation. The ADs in group PSR1-AD15 relate to the design features of the Special Safety Systems to perform their safety function when required. The Level 1 and Level 2 PSA demonstrate that the existing plant design satisfies the relevant safety requirements [P-REP-03611-00006-R00, *Pickering NGS PSA Update to Include Enhancements from the Fukushima Integrated Action Plan*, April 30, 2014]. The ADs in group PSR1-AD19 relate to the acceptance criteria for the acceptable duration of post dry-out operation. It has been confirmed that the derived acceptance criteria will not adversely impact on existing margins as demonstrated in the Safety Reports. There is no interaction between the two group of ADs associated with D-150 in Level 3 defence-in-depth.
D-186 – Inspectability of Safety Equipment	GI-16-AD1	GI-16-AD1 involves inspection requirements for inaccessible areas, and is also assessed under O-305. This AD group is related to concessions in the Periodic Inspection Plans for where components are deemed inaccessible. The CNSC has accepted the reasonableness of the solution to this issue and has determined that the revised PIPs satisfactorily meet the requirements of CSA N285.5-08 Update No. 1 standard. Furthermore, as indicated in the resolution for GI-16-AD1 in Appendix B of this report, where the inspection of a system or component would necessitate the dismantling of equipment, the required inspection should be performed when the equipment is dismantled for other reasons (i.e., maintenance). There is only one AD associated with this safety principle, and therefore there is no aggregate impact for Safety Principle D-186, Inspectability of Safety Equipment.
D-227 – Monitoring of Plant Safety Status	GI-34-AD1 GI-35-AD1 GI-36-AD2 GI-44-AD2 GI-48-AD1 GI-48-AD2	The following ADs are assigned to Safety Principle D-227. • GI-34-AD1 concerns remote tripping and monitoring capability. Clause 4.1.8.2 of CSA N290.1-13 is for a new plant and requires remote tripping and monitoring capability for both Shutdown Systems. Remote tripping capability is available for Pickering 5-8 SDS2 and Pickering 1,4 SDSE. However, Pickering 5-8 and Pickering 1,4 do not have remote tripping and monitoring capability for SDS1 or SDSA respectively.

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		However, there is dedicated remote tripping and monitoring capability at Pickering 1,4 for SDSE and at Pickering 5-8 for SDS2, and this meets the CSA N290.1 intent and requirements to the extent practicable. In addition, the failure to shutdown probability, as demonstrated in the PSAs [P-REP-03611-00006-R00, *Pickering NGS PSA Update to Include Enhancements from the Fukushima Integrated Action Plan*, April 30, 2014], is very low. • GI-35-AD1 involves Human Factors requirements for the Main Control Rooms. The Darlington ISR identified a gap against Clause 4.14.10 of N290.0-11 as a result of minimal design standards related to HFE or HFE activities being formally documented when the Control Rooms were originally designed and constructed. The original design phase of Pickering NGS recognized the need for focus on the operator interfaces in the control centres, and on recognition of the integrated whole of the control centre and the related human-systems interfaces. Pickering NGS, as a result, has decades of safe operation and operating experience with monitoring, operation, testing, maintenance and training (including simulators). With regards to plant modifications, all plant modifications are required to be completed in compliance with OPG Program, [N-PROG-MP-0001-R015, *Engineering Change Control*, May 12, 2017] which includes the requirements for Human Factors and Ergonomics to be considered in design. The technical, design and operator reviews, during and following the design process and via the Availability for Service process, ensure the usability requirements will be achieved. Based on the extensive operating experience and modifications processes, there is no justification for revisiting the overall Control Room design from a Human Factors perspective. On this basis, and as per the very low safety significance (Safety Significance Level 4) this is assessed as an AD and no further action will be taken. • GI-36-AD2 is an issue of monitoring ECI strainer condition. This issue is related to monitoring of ECI System debris strainer effectiveness. Pickering A and B comply with the requirements for new plant debris interceptors, with the exception of clause 5.14.11, for which the benefit of developing and implementing new instrumentation for post-accident effectiveness is assessed to be small. The intent of the requirement is met by monitoring ECI recovery pump performance and reactor building water level. Given the effectiveness of the ECI recovery phase with the installed strainer modules, this issue is assessed as low safety significance (Safety Significance Level 3).

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		• GI-44-AD2 deals with Human Factors requirements for design. Refer to the assessment for GI-35-AD1 above. • GI-48-AD1 involves Clause 7.2.1.13 of CSA N293-13 that states: "Electrical conductors... shall be capable of performing their intended functions for not less than 1 hour after the start of a fire." Modifications to the Fire Protection System meet the requirements by the use of Edwards System Technology (EST) that connects the fire alarm control panels via a data communication link with dual redundant circuit wiring paths. An assessment of the Pyrotronics detection/alarm system was completed [P-CORR-03680-0620819, Re: Resolution Plan for Pickering Periodic Safety Review 2 (PSR2) Global Issue #GI-48 Gap SF1-4, June 20, 2017], and concluded that the characteristics of this detection/alarm system (detection prior to potential alarm system fire damage, and signal latching function) as well as the physical separation in the equipment/cabling configuration, ensure that lack of a one hour fire rating for cabling in these systems would not cause failure to detect the fire or loss of the fire signal once detected. As such, manual suppression of the fire by Emergency Response Team personnel would not be impeded and there is no adverse nuclear safety impact. Therefore, this issue is assessed as an AD with low safety significance (Safety Significance Level 3). • GI-48-AD2 involves Clause 7.2.1.10.1 of CSA N293-12 that states: "A display and control centre shall be located in the MCR [Main Control Room]… capable of providing detailed information on the location and nature of the signal. In addition, the panel operator shall be able to control the fire alarm system without having to leave his or her station." An assessment of this issue was completed [P-CORR-03680-0620820, *Re: Resolution Plan for Pickering Periodic Safety Review 2 (PSR2) Global Issue #GI-48 Gap SF1-3*, June 2, 2017]. The assessment confirmed that the existing arrangement of annunciation/display (including a Fire Control Panel located immediately outside the MCR and annunciation within the MCR), together with emergency response protocols and staff training, meets the intent of the requirement in providing timely response to fire signals. This is a low safety significance issue (Safety Significance Level 3) and has been addressed to the extent practicable. Therefore, it is assessed as an AD. There is interaction among these ADs. However, the issue in GI-36-AD2 is a specific ECI monitoring requirement, whereas the other three ADs are more general in scope. Therefore, GI-36-AD2 does not significantly impact the more general ADs. GI-35-AD1 and GI-44-AD2 deal with the same Human Factors issue and therefore do not involve an aggregate impact. Remote tripping and monitoring provisions (GI-

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
		34-AD1) meet the CSA N290.1 intent, and therefore the interaction with the Human Factors ADs is very low. GI-48-AD1 and GI-48-AD2 are related to annunciation and monitoring for fire related issues, however the assessments have confirmed that sufficient mitigation and alternate means are in place to ensure timely response to internal fires. Therefore, the aggregate impact on defence-in-depth of these ADs is assessed to be very low.
O-305 – Maintenance, Testing and Inspection	GI-16-AD1 GI-18-AD1	The following ADs are assigned to Safety Principle O-305. • GI-16-AD1 involves inspection requirements for inaccessible areas. Refer to the assessment for GI-16-AD1 under D-186, Inspectability of safety equipment • GI-18-AD1 is related to CSA N287.7-08 clause 7.11.2 on the accuracy requirement for dew point temperature measurement and repeatability requirement for pressure transmitter measurements related to Containment leak rate testing. Current instrumentation does not meet the accuracy requirement for dew point temperature of ± 1 °C and for pressure transmitter measurement repeatability of ± 0.001% of full scale. Currently industry standard instrumentation can meet ± 2 °C for dew point temperature accuracy, and ± 0.05% full scale for pressure transmitter measurement repeatability. A CSA committee is working to update CSA N287.7 to account for accuracy and repeatability capability of standard available commercial sensors. Until such time as CSA N287.7 is revised, OPG will continue to request appropriate concessions from the CNSC when leak rate testing is required, as per the very low safety significance (Safety Significance Level 4). There is no interaction between inspection of inaccessible areas and dew point temperature measurement accuracy and repeatability. Therefore, the aggregate impact on defence-in-depth of these ADs is assessed to be negligible.
O-284 – Operating Limits and Conditions O-278 – Training	PSR1-AD2	PSR1-AD2 concerns CSA N290.1 Human Machine Interface Consideration in Design, specifically issues relating to Staffing, the Human Factors Engineering Program and Verification and Validation during design process. The Pickering B ISR raised a gap on the basis that for smaller modifications, explicitly documenting each validation element was not judged necessary. Operating experience and improvements have been incorporated into the processes and design to improve the human-machine interfaces in many areas (e.g., Control Room annunciation upgrades as a result of changes to computer hardware and operator interface). Additionally, training and qualification

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
		processes (and certification processes for Control Room staff) for Operations positions ensure that the staff are competent to carry out functions assigned to them. The extensive use of Main Control Room simulators for training and validation of system modifications to assess their impact on other systems and human-machine interfaces provides OPG with a safe means of testing and training operators on abnormal and accident operating conditions, thus strengthening the Level 3 defence-in-depth barrier. There is only one AD associated with these safety principles, and therefore there is no aggregate impact for Safety Principle O-284, Operating Limits and Conditions, or Safety Principle O-278, Training.
D-200 – Automatic Shutdown Systems	GI-14-AD1 GI-33-AD1 GI-34-AD1 GI-44-AD6 PSR1-AD5	The following ADs and AD groups are assigned to Safety Principle D-200. • GI-14-AD1 involves revision of the Environmental Qualification Assessment of Tefzel cables to reflect the change of qualified life of the Vertical Flux Detectors. OPG is actively progressing revision of the EQA for Tefzel cables outside of PSR2 to align with the requirements of [N-PROC-RA-0044, *Environmental Qualification Assessment* October 17, 2014] with respect to the Pickering 5-8 Vertical Flux Detector Tefzel cables. Since this issue is in progress and the gap will be addressed, this AD is not assessed any further. • GI-33-AD1 concerns legacy code Class 3 Liquid Injection Shutdown System (LISS) components. This is a legacy design issue. A limited number of LISS components that should have been code Class 1 were purchased and installed as code Class 3. However, a rationale was accepted and a code classification concession was granted by the CNSC to allow the system to remain as-is for the legacy modifications. OPG provided rationale to show that consequences following any failure of Class 3 or Class 6 portions of the systems continue to satisfy the requirements of CSA N285.0. Components have been operating for a very long time and continue to perform their required function. On this basis, this issue is low safety significance (Safety Significance Level 3). • GI-34-AD1 concerns remote tripping and monitoring capability. Clause 4.1.8.2 of CSA N290.1-13 is for a new plant and requires remote tripping and monitoring capability for both Shutdown Systems. Remote tripping capability is available for Pickering 5-8 SDS2 and Pickering 1,4 SDSE. However, Pickering 5-8 and Pickering 1,4 do not have remote tripping and monitoring capability for SDS1 or SDSA, respectively. However, there is dedicated remote tripping and monitoring capability at Pickering 1,4 for SDSE and at Pickering 5-8 for SDS2, and this meets the CSA N290.1 intent and requirements to the extent

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
		practicable. In addition, the failure to shutdown probability, as demonstrated in the PSAs, is very low [P-REP-03611-00006-R00, *Pickering NGS PSA Update to Include Enhancements from the Fukushima Integrated Action Plan*, April 30, 2014]. • GI-44-AD6 involves reliability requirements for safety systems. Clause 7.6 of REGDOC-2.5.2 requires all safety systems and their support systems to be designed to meet an on-demand failure rate less than 10^{-3} yrs/yr. This requirement is not met for several systems including Pickering 1,4 ECI. However, safety system performance is closely scrutinized and monitored, and established unavailability targets and performance of all safety systems are monitored and reported annually in the Annual Risk and Reliability Report [P-REP-09051.1-00016-R000, *Pickering NGS – 2016 Annual Risk and Reliability Report*, March 31, 2017]. The current engineered provisions provide sufficient functionality to ensure compliance with PSA Safety Goals. This is a low safety significance issue (Safety Significance Level 3). • PSR1-AD5 concerns CSA N290.1 requirements for separation and independence of the Shutdown Systems. The design principles in the N290.1-80 requirements that could not be fully complied with were requirements for independent Shutdown Systems with conceptually different reactivity components, and diverse trip parameters. SDSA and SDSE in Pickering 1,4 are independent trip logic components of the reactor protective system. These two Shutdown Systems are designed to be physically and operationally independent of each other and process systems. They are diverse in design to the extent that their logic components are supplied by different manufacturers. SDSE in Pickering 1,4 provides further defence-in-depth through increased Shutdown System reliability, trip coverage, reactivity depth, and protection against common-mode cross-link effects, such that the overall Shutdown System effectiveness approaches that normally attributed to reactors having two completely separate and independent Shutdown Systems. For the Pickering 1,4 Shutdown System, there are approved exceptions for SDSA and SDSE not being independent of each other. The AD rationale was accepted by the CNSC and the basis for the acceptability of the exception continues to be applicable and is not impacted by operation beyond 2020. GI-34-AD1 is independent of the other ADs and does not interact with them. GI-14-AD1 and GI-33-AD1

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
		are also independent and do not interact; however, both of these ADs have a limited interaction with Shutdown System reliability (GI-44-AD6). The aggregate impact is mitigated by the fact that OPG is actively progressing GI-14-AD1, and that a code classification concession was granted by the CNSC based on the rationale provided by OPG (GI-33-AD1). PSR1-AD5 involves reliability and redundancy design-related requirements. In general, these requirements do not interact directly with the other ADs. PSR1-AD5 represents diverse aspects of Shutdown System design that do not directly interact with the other ADs. Therefore, the aggregate impact on Defence-in-Depth of these ADs is assessed to be very low.
D-207 – Emergency Heat Removal	GI-36-AD1 GI-36-AD2 GI-38-AD1 GI-44-AD6 GI-44-AD7 GI-44-AD9 PSR1-AD6 PSR1-AD16	The following ADs and AD groups are assigned to Safety Principle D-207. • GI-36-AD1 involves ECI design requirements based on the least effective Shutdown System. Clause 5.2.1.2 of CSA N290.2-11 requires that ECI System design requirements be based on the assumption that the least effective of the Shutdown Systems has operated successfully. This requirement cannot be met for Pickering 1,4 since there is only one Shutdown System (albeit with tripping capability from separate SDSA and SDSE logic). This issue is related to ECI System capability with respect to Shutdown System operation. The ECI System capability is assessed using the available SDS and hence this meets the CSA N290.2 requirement to the extent practicable. The Pickering 1,4 ECI system reliability meets the licensing target, as demonstrated in [P-REP-09051.1-00016-R000, *Pickering NGS - 2016 Annual Risk and Reliability Report*, March 31, 2017]. On this basis, this is a low safety significance issue (Safety Significance Level 3). • GI-36-AD2 is an issue of monitoring ECI strainer condition. Clause 5.14.11 of CSA N290.2-11 requires instrumentation to be available to monitor post-accident effectiveness and to determine the extent of plugging of ECI System debris interceptors (strainers). While the relative health of a strainer can be inferred by a combination of ECI System recovery pump performance and reactor building water level, there is no direct correlation between these conditions and debris loading available. A detailed assessment [NK30-CORR-00531-05194 R001, *Pickering B – Generic Action Item 06G01 Emergency Core Cooling System Strainer Deposits – Status Update and Request for Closure*, June 30, 2009] of the potential sources of strainer debris and contaminants for Pickering 5-8 demonstrated sufficient margin to ensure post-accident ECI recovery strainer effectiveness. The issues identified in the Pickering 1,4 assessment [NA44-CORR-00531-06062 R000, *GAI 06G01: Emergency Core*

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
		Cooling System Strainer Deposits – Status Update, June 30, 2009] resulted in the installation of new strainer modules in Pickering 1,4. Since Pickering 1,4 and Pickering 5-8 comply with the requirements for new plant debris interceptors, with the exception of clause 5.14.11, the benefit of developing and implementing new instrumentation for post-accident effectiveness is assessed to be small. This is because although no direct measurement instrumentation is available to monitor post-accident effectiveness and to determine the extent of plugging of the debris interceptor, the intent of the requirement is met by monitoring ECI recovery pump performance and reactor building water level. Additional mitigating factors include (a) the demonstrated margin that ensures post-accident effectiveness of the ECI recovery phase with the installed strainer modules, (b) the low contribution of ECI System recovery strainer plugging to the PSA results [NA44-CORR-33350-0265268-R000, *Pickering A Risk Assessment ECI Strainer Plugging Following a Large LOCA*, September 23, 2008], and (c) mandatory inspections to ensure the ECI recovery flowpath is free from debris prior to restart of a Pickering 5-8 or Pickering 1,4 unit following a maintenance outage [NK30-SRS-E-082-R004, *ECI Recovery Flowpath Inspection*, March 27, 2014], [NA44-SRS-E-026-U14-R019, *ECI Recovery Flowpath Inspection*, February 21, 2017]. Given the low safety significance (Safety Significance Level 3), this is assessed as an AD and no further action will be taken. • GI-38-AD1 involves unavailability targets for the Shutdown Cooling System and other heat removal systems. Clause 5.6.1 of CSA N290.11-13 requires design reliability to be established for outage heat sinks. Design reliability requirements have not been established individually for all normal and back-up heat sinks used at Pickering NGS. The reliability of all outage heat sinks (including those without explicit targets) is managed under the Risk & Reliability Program [N-PROG-RA-0016-R009, *Risk and Reliability Program*, May 27, 2016] (both through unavailability models as well as through Probabilistic Safety Assessment [P-REP-03611-00006-R00, *Pickering NGS PSA Update to Include Enhancements from the Fukushima Integrated Action Plan*, April 30, 2014]), hence reactor safety impact is assessed and monitored. The reactor safety impact of not having explicit individual heat sink design reliability is assessed, monitored and is not a significant issue, and is assessed as Safety Significance Level 4. • GI-44-AD6 involves reliability requirements for safety systems. Clause 7.6 of REGDOC-2.5.2 requires all safety related systems to be designed to meet an on-demand failure rate less than 10^{-3} yrs/yr. However, safety system performance is closely scrutinized and monitored, and established

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
		unavailability targets of all safety systems are monitored and reported annually in the Annual Risk and Reliability Report. The current engineered provisions provide sufficient functionality to ensure compliance with PSA Safety Goals [P-REP-03611-00006-R00, *Pickering NGS PSA Update to Include Enhancements from the Fukushima Integrated Action Plan,* April 30, 2014]. This is a low safety significance issue (Safety Significance Level 3). • GI-44-AD7 involves sharing of ECI and Containment. Sharing of Safety Systems and Turbine Hall: Clause 7.6.5 of REGDOC-2.5.2 has a new requirement that precludes sharing of safety systems and the turbine generator building. Pickering units share ECI and Negative Pressure Containment, as well as the turbine hall. The main impacts of sharing of ECI and Containment are addressed since if either ECI or Containment is unavailable, all affected units are considered impaired and must be shut down within specified time limits, hence reducing the risk of a coincidental DBA. Moreover environmental conditions in the common turbine building have been assessed and credited provisions have been protected to ensure the ability to shutdown/control, cool and monitor remains available on non-accident units. Required Safety Goals are met. This is a low safety significance issue (Safety Significance Level 3). • GI-44-AD9 involves REGDOC-2.5.2 requirements for ECI Heat Exchanger Leak Detection. Detection/Isolation of ECI Heat Exchanger (HX) Tube Leak: Clause 8.5 of REGDOC-2.5.2 requires ECI recovery heat exchanger tube leak detection capability. Pickering 5-8 ECI recovery heat exchangers do not have leak detection capability on the cooling water side. Pickering 5-8 has ECI recovery piping, pumps and heat exchangers outside of Containment. Components penetrating and outside Containment are all DBE qualified and Nuclear Class 2. Since the intent of leakage detection is served by the system leakage collection, recovery and radiation monitoring in the vicinity, the added benefit of implementing a design modification for direct leakage detection is a low safety significance issue (Safety Significance Level 3). • PSR1-AD6 involves requirements for independence and redundancy in the design of the Emergency Core Cooling System. The issues associated with these ADs are specific to Pickering 1,4. The Pickering 1,4 ECI system utilizes the Moderator system in its low pressure mode, and a number of instances of single failure existed in the design of the ECI system. The issues were assessed as ADs based on the improvements made to the design to reduce the commonality between the Moderator system and ECI

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
		recovery, and some of the significant singletons were addressed by design changes. These design changes resulted in the greatest reduction in core damage frequency. Therefore, this is not a safety significant issue and is assessed as a low significance AD. • PSR1-AD16 Requirements for Containment Coatings and Coverings. The gaps included in this Issue concern non-metallic coatings and liners that perform a safety function, like preserving the concrete Containment envelope. The basic requirement is that not only should these materials be carefully selected, and applied, but their deterioration (especially under post-accident conditions) should not impair the performance of other safety functions, such as blocking of ECI strainers. A detailed assessment [NK30-CORR-00531-05194 R001, *Pickering B – Generic Action Item 06G01 Emergency Core Cooling System Strainer Deposits – Status Update and Request for Closure*, June 30, 2009] of the potential sources of strainer debris and contaminants for Pickering 5-8 demonstrated sufficient margin to ensure post-accident ECI recovery strainer effectiveness. The issues identified in the Pickering 1,4 assessment [NA44-CORR-00531-06062 R000, *GAI 06G01: Emergency Core Cooling System Strainer Deposits – Status Update*, June 30, 2009] resulted in the installation of new strainer modules in Pickering 1,4. Pickering 1,4 and Pickering 5-8 comply with the requirements for new plant debris interceptors, and given the low safety significance (Safety Significance Level 3), this is assessed as an AD. All of these ADs involve the ECI system except for GI-38-AD1 and therefore the latter has negligible interaction with the other ADs. GI-44-AD9 has negligible interaction with the other ADs as it is specific to HX tube leakage. GI-36-AD1 and GI-36-AD2 both deal with ECI performance; however, the strainer performance is not a function of the Shutdown System assumed in the design, so there is no interaction between these ADs. Safety system separation (GI-44-AD7) is normally addressed separately from design for reliability (GI-44-AD6), and therefore their interaction is very low. PSR1-AD6 involves requirements for independence and redundancy in the design of the Emergency Core Cooling System, which does not interact with the other ADs. PSR1-AD16 involves Requirements for Containment Coatings and Coverings which relates to performance of ECI recovery system following a postulated event. The synergy between the ADs is minimal and therefore, the aggregate impact on defence-in-depth of these ADs is assessed to be very low.
D-217 –	GI-42-AD1	The following ADs and AD groups are assigned to Safety Principle D-217.

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Confinement of Radioactive Material	GI-44-AD6 PSR1-AD3 PSR1-AD4 PSR1-AD7 PSR1-AD8	• GI-42-AD1 involves requirements for examination and testing of Concrete Containment Structures. The Concrete Containment Structures (CCSs) at Pickering A and B were built and tested to meet the 1965 and 1970 National Building Code of Canada requirements, respectively, prior to the initial issuance of CSA N287.5. No assessments exist which demonstrate that the requirements in effect during construction of Pickering NGS CCSs comply with the requirements of CSA N287.5. A PSR2 gap was raised for Pickering NGS given that conformance with the specific requirements of CSA N287.5 has not been demonstrated. The original Pickering construction included requirements for tests and quality control procedures which generally meet the intent of CSA N287.5. The controls in place at the time of construction and the ongoing controls in place for inspections, aging management and modifications adequately meet the intent of CSA N287.5-11. Ongoing confirmation that the Pickering NGS CCSs remain fit for service is demonstrated via periodic and in-service inspections conducted in accordance with the requirements of CSA N285.5 and CSA N287.7, and the resultant inspection reports attest to the quality of the design. The Engineering Change Control process ensures that that any design changes made to the Pickering CCSs will comply with CSA N287.5 going forward, as applicable. The residual elements of this issue are of low safety significance. • GI-44-AD6 involves reliability requirements for safety systems. Refer to assessment for GI-44-AD6 against D-207, Emergency heat removal. • PSR1-AD3 involves general requirements for the design of Concrete Containment Structures. These ADs involve general requirements for the design of Concrete Containment Structures. The original and subsequent revisions of the CSA N287 series of standards were issued after completion of design and construction of Pickering NGS. Therefore the details related to general requirements of Containment structure, parts, materials, design, fabrications, construction, inspection examination, testing and commissioning in accordance with the series of standards of CSA N287 were not available. At that time, the CCSs were designed to meet the requirements of the National Building Code (NBC). All CSA N287.1-93 (R2004) Pickering B ISR Review findings were assessed to be ADs in PSR1 given robust design practices at the time, together with ongoing periodic inspections and in-service testing. The standards that applied during the original construction of Pickering NGS included requirements for tests and quality control procedures to ensure that the concrete used in the as-built structures met

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
		the original design requirements. The PSR2 review of CSA N287.1-14 also outlines Programs, Procedures and Standards which are credited with the ability to detect and monitor any safety significant degradation mechanisms and thus to provide assurance of continued fitness for service of the Pickering NGS CCSs. • PSR1-AD4 involves specific requirements for the design of Concrete Containment Structures. These ADs involve specific requirements for the design of CCSs. This group of ADs applies to initial construction, installation, and fabrication of mechanical splices for Concrete Containment Structures. The original and subsequent revisions of the CSA N287 series of standards were issued after completion of design and construction of Pickering NGS. All CSA N287.3-93 (R2004) Pickering B ISR Review findings were assessed to be ADs in PSR1 given robust design practices at the time, together with ongoing periodic inspections and in-service testing. The standards that applied during original construction of Pickering NGS included requirements for tests and quality control procedures to ensure that the concrete used in the as-built structures met the original design requirements. The PSR2 review of CSA N287.3-14 also outlines Programs, Procedures and Standards which are credited with the ability to detect and monitor any safety significant degradation mechanisms and thus to provide assurance of continued fitness for service of the Pickering NGS CCSs. • PSR1-AD7 involves requirements for Containment pressure capability in operation and to respond to accidents. Clause 3.4.2 of CNSC R-7 identifies that the negative design pressure of Containment must not be greater than that predicted in the Safety Report for a set of postulated events with failure of dousing. It was not considered necessary to evaluate event combinations involving failure of dousing for negative Containment design pressure impacts, because these event combinations result in higher rather than lower Containment pressure. Clause 3.4.2 identifies that the negative design pressure of Containment must not be greater than that predicted in the Safety Report for a set of postulated events with failure of dousing. It is not necessary to evaluate event combinations involving failure of dousing for negative Containment design pressure impacts, because these event combinations result in higher rather than lower Containment pressure. Therefore, the intent is met. • PSR1-AD8 involves requirements for Containment related to Containment leakage. These PSR1 ADs involve requirements for Containment related to Containment leakage and the fact

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
		that only a positive pressure proof test was performed for Pickering NGS A. and, as required, a negative pressure test was not performed during commissioning. This requirement is not met because the Containment envelope was not tested to the lowest predicted negative pressure following an accident. However, the negative design pressure is limited due to the multi-unit design of the Pickering Containment and the PSR1 review demonstrates that leakage is not expected to be significant at these pressures. Examination and testing of CCSs (GI-42-AD1) contributes to confidence in the reliability of Containment (GI-44-AD6). Therefore, there is interaction between these two ADs, although it is not direct as they involve different activities. The general requirements for the CCSs (PSR1-AD3) primarily impact the Containment performance (e.g., leak-tightness) and are of lower significance for Containment reliability. This is also true for PSR1-AD4. PSR1-AD7 and PSR1-AD8 are diverse requirements which do not significantly interact among each other or the other ADs assessed for this safety principle. Therefore, the aggregate impact on Safety Principle D-217 of these ADs is assessed to be very low.
D-221 – Protection of Confinement Structure	PSR1-AD9	The following AD group is assigned to Safety Principle D-221. • PSR1-AD9 involves hydrogen monitoring requirements related to BDBAs. Clause 10.2.3 of CSA N290.3-11 states: "New builds shall monitor the conditions identified according to the requirements of Clause 10.2.2 [... (e.g., pressure, temperature, and hydrogen concentration inside Containment) that need to be monitored for BDBAs ...]" The absence of hydrogen monitoring instrumentation is acceptable, given installed hydrogen mitigation equipment together with SAMGs provide means to safely manage any hydrogen in Containment. This issue is considered to be an AD since the Post-LOCA Hydrogen Ignition System has been environmentally qualified for a mission time of 30 days, and would therefore likely be available for hydrogen mitigation in the short term following a LOCA plus loss of ECI. The PARS (Passive Auto-Catalytic Recombiners) are designed to consume long term hydrogen. There is only one AD associated with this safety principle, and therefore there is no aggregate impact for Safety Principle D-221, Protection of Confinement Structure.
D-174 –	GI-44-AD6	The following AD and AD group are assigned to Safety Principle D-174.

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
Reliability Targets	PSR1-AD11	• GI-44-AD6 involves reliability requirements for safety systems. Refer to the assessment for GI-44-AD6 against D-207, Emergency heat removal. • PSR1-AD11 Requirements for Electrical Power and Instrument Air Systems The gaps in AD#191 in PSR1-AD11 were raised because the practice used at Pickering 'A' was to provide a single rectifier for each Class I battery. This is considered acceptable since rectifier failures are annunciated and the associated battery has a 45 minute capacity, which allows time for operator action. Provision is made to connect a temporary charger if the main charger is out of service for long periods. Per Regulatory Document RD-98, reliability targets are set for Systems Important to Safety. The Class I power system is in the list of Systems Important to Safety and therefore it is a monitored system required to meet its reliability target. Since the reliability of the system has been demonstrated to be acceptable with its current design, the Acceptable Deviation remains applicable for extended operation. There is overlap between GI-44-AD6 and PSR1-AD11 in that both ADs are related to reliability requirements, but for different systems, and therefore there is no aggregate impact of these ADs on D-174 – Reliability Targets. Therefore, the aggregate impact on defence-in-depth of these ADs is assessed to be very low.
D-177 – Dependent Failures	PSR1-AD1 PSR1-AD18 PSR1-AD21	The following AD groups are assigned to Safety Principle D-177. • PSR1-AD1 involves issues relating to assessing seismicity at the Pickering NGS site. Seismic PSAs [P-REP-03611-00006-R00, *Pickering NGS PSA Update to Include Enhancements from the Fukushima Integrated Action Plan*, April 30, 2014] have been issued for both Pickering 1,4 and Pickering 5-8 and confirm that Pickering NGS conforms to CNSC S-294 and to various international guides on seismic PRA. Consequently these gaps are considered to have no impact on the seismic resistance of qualified structures. The periodic inspection of the concrete structures and routine leakage tests of the Containment envelope verify that there is no significant deterioration in the integrity of the Concrete Containment Structures. • PSR1-AD18 involves safety-related requirements for hazards (turbine orientation and firewater supply). AD #243: The orientation of the turbine generators relative to the safety related SSCs at Darlington

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
		NGS does not meet the requirement of CNSC RD-337. Turbine Failures at Pickering NGS are addressed in the PSA Hazard Screening addressed in the PSR2 Hazard Analysis Safety Factor Report. The Darlington ISR D278 issue resolution, which concluded that reactor can be safely shut down, cooled down and maintained subcritical following a postulated turbine disintegration, is also applicable to Pickering NGS. The effect of turbine missiles has been considered in the supporting design analysis. • PSR1-AD21 involves measures to prevent the spread of fire as well as to ensure staff safety and capability to respond. The scope of this AD group originated with deviations between the design of Darlington NGS and requirements of the NBCC-2005. For example, a new Article in NBCC-2005 calls for integrated testing of fire protection and life safety systems to ensure that they will operate together, as intended. However, documentation could not be found to confirm that an integrated systems test is performed for the life safety and fire protection systems installed at Darlington NGS. The gaps associated with this AD group were assessed as an AD for PSR2 based on the following key arguments: • Fire Hazard Analysis (FHA) and the Fire Safety Shutdown Analysis (FSSA) demonstrate that the safety objectives of Pickering 1,4 and 5-8 are met under postulated fire scenarios. • In the event of a fire at Pickering NGS, personnel, as per training and as instructed via the public address system, are to avoid the incident unit and area as well as to refrain from using the elevators. • Access to Pickering NGS is limited to trained personnel who are familiar with the hazards and safety features of site buildings. Multiple means of egress are available and existing signage is clearly recognizable and understood by the personnel accessing the Station buildings. • Although there is a lack of smoke detection near the entrance to certain areas of the Station, it unlikely that these fires would go undetected and unreported given the plant is served by multiple alternate means of egress. • With respect to electrical conductors, there are no high-rise buildings, areas of refuge, or contained use areas located at Pickering NGS. • Although the absence of electronic supervision for fire protection water supply valves could result in a valve being closed without sending an alert to central alarm monitoring, water supply valves without electronic supervision are locked in an 'open' position at Darlington which would prevent

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
		unintended closure. As per Section 11.3 of Appendix B of [NA44-DM-71400-00002 R000 *Design Manual for Pickering Generating Station, Fire Protection Systems*, February 2014] for Pickering 1,4 and Section 4.2 of [NK30-DM-71400-00001-R006, *Design Manual for Pickering Nuclear, Fire Protection System*, January 2016] for Pickering 5-8, the same holds true for Pickering NGS as well. As discussed under the PSR2 review for NFPA 24 [P-REP-03680-00029-R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 1, 5, 6, and 7*, March 2, 2017], a number of yard post indicator valves at Pickering NGS without electronic supervision do not have locking mechanisms. Resolution of this finding is currently in progress with locks now installed on the majority of the affected valves. This was identified as a PSR2 gap under the NFPA 24 review (PSR2 NFPA 24 Gap #1). Therefore, the ADs applicable to Darlington are also applicable to Pickering NGS with one exception, which is being tracked as a PSR2 gap. PSR1-AD1 is related to seismic resistance of qualified structures. PSR1-AD18 is related to the orientation of the turbine generators relative to the safety related SSCs. PSR1-AD21 is involves measures to prevent the spread of fire as well as to ensure staff safety and capability to respond. As can be observed, there is no synergy among the three AD groups and therefore, there is no aggregate impact of these ADs on D-177, Dependent Failures.
M&C-246 – Safety Evaluation of Design	PSR1-AD15 PSR1-AD19	The following AD groups are assigned to Safety Principle M&C-246. • PSR1-AD15 involves safety-related requirements for plant design. This AD group relates to Single Failure Criterion issues that were raised against IAEA NS-R-1 and CNSC RD-337 and assessed as ADs. The PSR2 review of REGDOC-2.5.2, which adopts principles from NS-R-1 and supersedes CNSC RD-337, included a detailed assessment of PSR1 ADs. As stated in the PSR2 code review for REGDOC-2.5.2 [P-REP-03680-00029-R000, *Pickering PSR2 Law, Regulation, Code and Standard Reviews Associated with Safety Factors 1, 5, 6, and 7,* March 2, 2017], this issue was transferred to the Pickering B Continued Operations Plan, where it was closed on the basis that the Level 1 and Level 2 PSA demonstrate that the existing plant design satisfies relevant safety requirements. The applicability of the gap resolution and subsequent disposition is considered to be equally applicable to Pickering 1,4, where the PSAs have similarly been updated.

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
		Therefore, this issue is of low safety significance. • PSR1-AD19 involves deterministic safety analysis requirements. The ADs in this group are related to dual trip parameter coverage, the criterion for no dry-out for the first trip, the allowable duration of post dry-out operation for the second trip and interpretation of the acceptance criterion relating to the acceptable duration of post dry-out operation. However, the derived acceptance criteria outlined in the COG report [COG-13-9035, *Derived Acceptance Criteria For Deterministic Safety Analysis*, November 2014], developed by the industry taking into account feedback from the CNSC, are less restrictive than those outlined in the current Pickering 1,4 and Pickering 5-8 Safety Reports, providing confidence that the derived acceptance criteria documented in COG-13-9035 will not adversely impact on existing margins as demonstrated in the Safety Reports. The PSR2 confirmed that the PSR1 AD rationale for PSR1-AD19 continues to remain applicable and is not impacted by extended operation. The ADs in this group that relate to Safety Principle M&C-246, Safety Evaluation of Design, have no significant interaction because the issues are not cumulative in nature and individual groups have very low safety significance or safety impact. Also, these ADs are related to issues which, for the most part, have been addressed to minimize any potential risk to plant operation. Therefore, the overall conclusion for Safety Principle M&C-246 is that the aggregate impact on defence-in-depth of these ADs is negligible.
S-136 – External Factors Affecting the Plant	GI-13-AD1 GI-13-AD2 GI-13-AD3 GI-13-AD4 GI-43-AD1 PSR1-AD1	The following ADs and AD groups are assigned to Safety Principle S-136. GI-13-AD1: This AD is related to the CSA N289.3 requirements to confirm that a) the generated time history used within seismic analyses of safety-related systems correctly represents the design ground response spectrum for the Pickering site in compliance with CSA N289.3-10, and b) the power spectral density (PSD) function of each time-history has been calculated and shown to not have any significant gaps in energy over the frequency intervals. The rationale for considering this gap as an AD is the time-history method is a typical method for seismic qualification that has been employed at Pickering NGS. Time-history ground motion inputs at Pickering NGS were established based on the requirements of CSA N289.3-M81 and that the seismic analysis method is conservative and adequate, which is confirmed by the Pickering Seismic PRA that concludes the seismic design of systems and structures are sound. Consequently these gaps are considered to have no impact on the seismic resistance of qualified structures. GI-13-AD2: This AD is related to the CSA N289.4-12 requirements for confirmation that aging effects that

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
		may impair seismic functionality shall be accounted for in seismic qualification by testing. Routine monitoring measures ensure that degradation effects will not be permitted to advance to the extent that seismic functionality is impaired. OPG's commitment [N-CORR-00531-05661, Design Codes and Standards Effective Dates for OPG Nuclear Fleet, April 2012] to using CSA N289.4-12 for testing procedures for seismic qualification of nuclear power plant SSCs will ensure that the requirements will be followed. Also, administrative controls [N-STD-MP-0025, *General Requirements For Seismic Qualification of OPG Nuclear Facilities*, October 27, 2016] ensure that seismic qualification of SSCs performing safety-related functions during and following an earthquake is maintained for the life of the facility. Consequently this gap is considered to have no impact on the seismic resistance of equipment qualified by test. GI-13-AD3: This AD is related to the CSA N289.5 requirement for seismic monitoring/recording. OPG has installed in-plant seismic instrumentation to monitor seismic activity at Pickering NGS (and at Darlington) in compliance with CSA standard N289.5 [NK30-DM-61150-10001, *Pickering Nuclear: Seismic Monitoring System*, September 2013; N-GUID-02004-10000-R00, Seismic Monitoring of OPG Nuclear Generating Stations, December 2010]. OPG has established procedures - Abnormal Incidents Manuals - that detail station response to earthquakes. Clear responsibilities are established to support monitoring and post-seismic response to an event [NK30-AIM-058-09013-6.0, *Abnormal Incident Manual: Seismic/Common Mode Event*; NA44-AIM-014-09013-06, *Abnormal Incident Manual: Seismic Event*]. OPG is also a contributor to the operation of the Southern Ontario Seismic Network (SOSN) which provides detailed free-field seismic records covering Southern Ontario. These systems support in-plant monitoring of the station's response to seismic events. The intent of the requirements has been met, and for these reasons, and as per the very low safety significance (Safety Significance Level 4), this is assessed as an Acceptable Deviation with no further action. GI-13-AD4 (for Units 1 and 4 only): This AD is related to a requirement in CSA N289.3 for the minimum number of cycles used for seismic fatigue analysis for the level of compliance of Pickering NGS plant structures supporting the operation of Pickering NGS reactors in the context of extended operation. Pickering Units 1,4 Class 1 systems and components are designed to ASME Code Section III which betters the CSA N289.3 requirement, including required cycles for fatigue analysis. Also, Seismic Margin Assessment is an accepted method for assessing seismic qualification and as such, this is an Acceptable Deviation for Pickering 1,4. For continuing analysis of potential new modifications, the requirements of CSA N289.3-10 with regards to number of cycles for seismic fatigue analysis would be met or bettered, consistent with the modelling methodology in place at OPG for such work as noted above. For these reasons, and as per the very low safety significance (Safety Significance Level 4), this is assessed as an

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
		Acceptable Deviation for Pickering 1,4 with no further action. GI-43-AD1: This AD is related to a new requirement in CSA N291-15 for bolted connections in members that are part of the seismic load resisting system. The original code requirements for PNGS safety related structures [National Building Code of Canada] and the Seismic Margin Assessments (SMA) include requirements of CSA S16 “Design of Steel Structures”, which is the basis for the requirements for bolted connections of steel structures in CSA N291-15. Given that the SMA methodology and the plant design both include requirements of S16/S16.1 (predecessor of S16), the plant meets or has been assessed to these requirements. As it would not be practicable to make changes to the bolted steel connections in the existing structural design, and given the low safety significance (Significance Level 3), this is assessed as an Acceptable Deviation with no further action. PSR1-AD1 involves requirements for seismic-qualification and testing. The following issues in PSR1-AD1 were raised in the Darlington and Pickering B ISRs. In terms of Safety Analysis, CSA N289.2 does not specify any specific requirements relating to the seismic success path. It is focused on a systematic approach to identify the seismic hazard. As such, there are no specific clauses relating to Safety Analysis. However, it is recognized that the magnitude and frequency of the spectra impacts on the qualification of SSCs and that if their required qualification were to change, the success path may need to change. The Darlington ISR raised a gap against Clause 5.5.2.1 of CSA N289.3 because the time histories are not statistically independent. The issues in PSR1-AD1 generally relate to the seismic design report of DNGS structures not explicitly accounting for effects due to • soil-structure interaction (SSI), • the difference between modeling techniques and analysis approaches for the evaluation of the site response analysis for the free-field case, and • complete interaction versus the sub-structure technique for the soil/foundation/structure system. These issues are also applicable to Pickering NGS. The Periodic Inspection of the concrete structures and routine leakage tests of the Containment envelope verify that there is no significant deterioration in the integrity of the Concrete Containment Structures. The rationale for considering gaps related to seismic qualification as an AD is that the seismic PRA confirms that Pickering NGS conforms to CNSC S-294 and to various international guides on seismic

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
		PRA, and that the results are acceptable. Consequently these ADs are considered to have no impact on the seismic resistance of qualified structures. This AD group is therefore deemed to have no impact on this aggregation assessment. GI-13-AD1, GI-13-AD4 and GI-43-AD1 have the same attribute “Seismic Design” insofar as the gaps are all related to the design programs by which the qualified SSCs were designed. While this is a common factor, the seismic design of Pickering NGS has been shown to be conservative such that all qualified SSCs will withstand the effects of a design basis seismic event. The seismic design aspect covered by PSR1-AD1 regarding soil structure interactions and GI-13-AD3 for monitoring have nothing in common with the other ADs with the attribute “Seismic Design”. Hence there is no aggregation effect. The aggregate impacts of these 6 seismic ADs on Pickering NGS seismic design integrity capability is negligible and hence has no impact on public safety.

Overall AD Aggregation for Defence-in-Depth Level 3
The ADs associated with the safety principles in Level 3 defence-in-depth include the following diverse areas: • D-158 – General Basis for Design – Four PSR2 AD groups were assessed under this safety principle. The ADs are related to deterministic safety analysis. Although there may be some interaction among the ADs in this area, this interaction does not pose an aggregate impact for extended operation as some of the issues have largely been addressed, are in progress or are already considered in the dispositioning of these ADs via the REGDOC-2.4.1 Implementation Plan for extended operation. • D-182 – Equipment Qualification – There are two AD groups associated with this Safety Principle. The seismic issues covered by PSR1-AD1 regarding soil-structure interactions and the issues covered by PSR1-AD16 regarding environmental qualification of non-metallic coatings and liners have nothing in common. Hence there is no interaction between the two ADs. • D-150 – Design Management – The two ADs associated with this safety principle have no interaction because the issues are against unique SSCs, and individual groups have low or very low safety significance or safety impact. OPG's Design Management Program, N-PROG-MP-0009, specifies requirements for procurement engineering processes ensuring implementation and maintenance of the nuclear facilities meet the design basis requirements. The Design Management Program provides assurance that design and procedure changes are prepared, reviewed, approved, documented and implemented in accordance with approved procedures, applicable regulatory requirements, standards and industry practices. • D-186 – Inspectability of Safety Equipment – There is only one AD associated with this safety principle. GI-16-AD1 involves inspection requirements for inaccessible areas. The CNSC has been informed and has found the current method in the program that provides for alternate means of assessing condition of the SSCs to be acceptable. • D-227 – Monitoring of Plant Safety Status – There is some interaction among the ADs assessed in this safety principle. One AD relates to monitoring of the ECI strainer condition, whereas the other ADs are more general in scope and are related to Human Factors issues and the ability for remote tripping and monitoring provisions. The ADs related to fire detection and monitoring are assessed to be of low safety significance due to the mitigations and the alternate means available for detection to ensure timely response to internal fires. The assessment confirms that the mitigations in place are adequate to provide the necessary defence-in-depth for control of events within design basis. • O-305 – Maintenance, Testing and Inspection – ADs involve inspection requirements for inaccessible areas and dew point temperature measurement accuracy and repeatability. The CNSC has been informed and has found the current method in the program that provides for alternate means of assessing condition of the SSCs to be acceptable. The issue related to the dew point temperature measurement is being addressed at an industry level in consultation with the relevant CSA committee. Both issues are considered to be of very low safety significance. • O-284 – Operating Limits and Conditions – The ADs assessed in this safety principle are related to human-machine interface considerations in design. Operating experience and improvements have been incorporated into the processes and design to improve the human-machine interfaces in many areas (e.g., Control Room annunciation upgrades as a result of changes to computer hardware and

Overall AD Aggregation for Defence-in-Depth Level 3
operator interface). • O-278 – Training – The same ADs that are assessed for O-284 are also assessed for O-278 with respect to Training. The review confirms that training and qualification processes (and certification processes for Control Room staff) for Operations positions ensure that the staff are competent to carry out functions assigned to them. In addition extensive use of Main Control Room simulators for training and validation of system modifications to assess their impact on other systems and human-machine interfaces provides OPG with a safe means of testing and training operators on abnormal and accident operating conditions, thus strengthening the Level 3 defence-in-depth barrier. • D-200 – Automatic Shutdown Systems – PSR1-AD5 involves reliability and redundancy design-related requirements. GI-34-AD1, related to remote tripping and monitoring capability, is independent of the other ADs and does not interact with them. GI-14-AD1, Environmental Qualification Assessment of Tefzel cables, and GI-33-AD1, legacy code classification of LISS components, are also independent and do not interact; however, both of these ADs have a limited interaction with Shutdown System reliability (GI-44-AD6). The aggregate impact is mitigated by the fact that OPG is actively progressing the GI-14-AD1 activity outside of PSR2, and that a code classification concession was granted by the CNSC based on the rationale provided by OPG (GI-33-AD1). In general, these requirements do not interact directly with the other ADs. • D-207-Emergency Heat Removal – All of these ADs involve the ECI system except for GI-38-AD1, which involves unavailability targets for the Shutdown Cooling System and other heat removal systems, and therefore the latter has negligible interaction with the other ADs. The Pickering 1,4 ECI system reliability meets the licensing target, as demonstrated in [NA44-REP-09051.1-00014, *2014 Annual Reliability Report – Pickering Units 1 & 4*, March 2015] and Level 1 and Level 2 PSA demonstrated that the existing plant design satisfies all safety requirements. • D-217 – Confinement of Radioactive Material – The ADs associated with this safety principle are related to Concrete Containment Structures. Pickering A and B were built and tested to meet the 1965 and 1970 National Building Code of Canada requirements, respectively, prior to the initial issuance of CSA N287.5. The original Pickering construction included requirements for tests and quality control procedures which generally meet the intent of N287.5. The controls in place at the time of construction and the ongoing controls in place for inspections, aging management and modifications adequately meet the intent of CSA N287.5-11. Ongoing confirmation that the Pickering NGS Concrete Containment Structures remain fit for service is demonstrated via periodic and in-service inspections conducted in accordance with the requirements of CSA N285.5 and N287.7, and the resultant inspection reports attest to the quality of the design. • D-221 – Protection of Confinement Structure – The ADs associated with this safety principle relate to hydrogen monitoring requirements for BDBAs. This issue is considered to be an AD since PLHIS has been environmentally qualified for a mission time of 30 days, and the PARS (Passive Auto-Catalytic Recombiners) are designed to consume long term hydrogen. • D-174 Reliability Targets – The ADs associated with this safety principle relate to requirements for Reliability Targets due to lack of required redundancy in some systems. As required by CNSC Regulatory Document RD/GD-98, Reliability Program, OPG has developed the lists of Systems Important to Safety for Pickering 1,4 and 5-8. The Systems Important to Safety, along with their unavailability targets

Overall AD Aggregation for Defence-in-Depth Level 3
are documented in P-REP-09051.1-00016 R000, *Pickering NGS - 2016 Annual Risk and Reliability Report*. Therefore, there is no significant safety issue associated with this safety principle. • D-177 – Dependent Failures – The ADs associated with this safety principle relate to common mode failures, such as seismic events. Seismic PRAs have been issued for both Pickering 1,4 and Pickering 5-8 and confirm that Pickering NGS conforms to CNSC S-294 and to various international guides on seismic PRA. Consequently these gaps have no impact on the seismic resistance of qualified structures. Also, the Periodic Inspection of the concrete structures and routine leakage tests of the Containment envelope verify that there is no significant deterioration in the integrity of the Concrete Containment Structures • M&C-246 – Safety Evaluation of Design – The ADs associated with this safety principle relate to requirements for redundancy, independence and separation, and trip effectiveness. The Level 1 and Level 2 PSA for Pickering NGS demonstrate that the existing plant design satisfies all safety requirements and the industry-supported, derived acceptance criteria documented in [COG-13-9035*, Derived Acceptance Criteria For Deterministic Safety Analysis*, November 2014] will not adversely impact on existing margins as demonstrated in the Safety Reports. • S-136 – External Factors Affecting the Plan – The ADs associated with this safety principle involve requirements for seismic-qualification and testing. Seismic PRAs have been issued for both Pickering 1,4 and Pickering 5-8 and confirm that Pickering NGS conforms to CNSC S-294 and to various international guides on seismic PRA. Consequently these issues have no impact on the seismic resistance of qualified structures. The ADs associated with the Safety Principles D-158, D-150, D-186, D-227, O-305, O-284, D-200, D-207, D-217, D-221, D-174, D-177, D-182, S-136, M&C-246, O-278 in Level 3 defence-in-depth cover diverse areas with no significant interactions. The aggregate impact of the ADs for each safety principle is determined to have no significant impact on safety. Furthermore, the overall aggregate impact of all of the ADs associated to Level 3 defence-in-depth is assessed to be low The dominant issues observed for Level 3 defence-in-depth are related to the design requirements for redundancy, independence and separation of Special Safety Systems. However, these issues do not interact sufficiently to aggregate or have significant safety impact. The issues are mitigated by ongoing confirmation that the Pickering NGS SSCs remain fit for service as demonstrated via periodic and in-service inspections. Implementation and execution of Pickering NGS's Equipment Reliability Program, N-PROG-MA-0026, is identified as a Strength. Confirmation of the effectiveness of the program is Pickering NGS Systems Important to Safety meeting the reliability targets [P-REP-09051.1-00016 R000, *Pickering NGS - 2016 Annual Risk and Reliability Report,* March 31, 2017]). In addition, Level 1 and Level 2 PSAs demonstrate that the existing plant design satisfies all safety requirements.

AD Aggregation for Defence-in-Depth Level 4

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
D-158 – General Basis for Design	GI-44-AD1 GI-44-AD3 GI-44-AD4	The following ADs are assigned to Safety Principle D-158. • GI-44-AD1 involves meeting REGDOC-2.5.2 requirements and limits for AOOs, DBAs and BDBAs. • GI-44-AD3 involves seismic margin requirements for Beyond Design Basis Earthquake. • GI-44-AD4 involves meeting REGDOC-2.5.2 Safety Goals. The Level 4 defence-in-depth aspect of GI-44-AD1 involves the requirements for BDBAs. GI-44-AD1 and GI-44-AD3 primarily affect calculated BDBA consequences and frequencies, whereas GI-44-AD4 deals with the safety criterion. Therefore, there is very low interaction between GI-44-AD4 and the other two ADs. Seismic analysis is included in the PSA and the REGDOC-2.4.1 Implementation Plan for deterministic safety analysis, so GI-44-AD3 has an interaction with GI-44-AD1; however, GI-44-AD3 can be considered as an aspect of the more general GI-44-AD1 that is also more fully addressed in GI-13 in the context of CSA N289 requirements. Therefore, the aggregate impact on defence-in-depth of these ADs is assessed to be very low.
D-207 – Emergency Heat Removal	GI-36-AD1 GI-36-AD2 GI-38-AD1 GI-44-AD6 GI-44-AD7 GI-44-AD9 PSR1-AD6 PSR1-AD16	The AD groups assigned to Safety Principle D-207, Emergency Heat Removal, have also been assessed for their impact on Level 3. The following AD groups are assigned to Safety Principle D-207. • GI-36-AD1 involves ECI design requirements based on the least effective Shutdown System. Clause 5.2.1.2 of CSA N290.2-11 requires that ECI System design requirements be based on the assumption that the least effective of the Shutdown Systems has operated successfully. This requirement cannot be met for Pickering 1,4 since there is only one Shutdown System (albeit with tripping capability from separate SDSA and SDSE logic). This issue is related to ECI System capability with respect to Shutdown System operation. The ECI System capability is assessed using the available SDS and hence this meets the CSA N290.2 requirement to the extent practicable. The Pickering 1,4 ECI system reliability meets the licensing target, as demonstrated in [P-REP-09051.1-00016-R000, *Pickering NGS - 2016 Annual Risk and Reliability Report*, March 31, 2017]. On this basis, this is a low safety significance issue (Safety Significance Level 3). • GI-36-AD2 is an issue of monitoring ECI strainer condition.

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
		Clause 5.14.11 of CSA N290.2-11 requires instrumentation to be available to monitor post-accident effectiveness and to determine the extent of plugging of ECI System debris interceptors (strainers). While the relative health of a strainer can be inferred by a combination of ECI System recovery pump performance and reactor building water level, there is no direct correlation between these conditions and debris loading available. A detailed assessment [NK30-CORR-00531-05194 R001, *Pickering B – Generic Action Item 06G01 Emergency Core Cooling System Strainer Deposits – Status Update and Request for Closure*, June 30, 2009] of the potential sources of strainer debris and contaminants for Pickering 5-8 demonstrated sufficient margin to ensure post-accident ECI recovery strainer effectiveness. The issues identified in the Pickering 1,4 assessment [NA44-CORR-00531-06062 R000, *GAI 06G01: Emergency Core Cooling System Strainer Deposits – Status Update*, June 30, 2009] resulted in the installation of new strainer modules in Pickering 1,4. Since Pickering 1,4 and Pickering 5-8 comply with the requirements for new plant debris interceptors, with the exception of clause 5.14.11, the benefit of developing and implementing new instrumentation for post-accident effectiveness is assessed to be small. This is because although no direct measurement instrumentation is available to monitor post-accident effectiveness and to determine the extent of plugging of the debris interceptor, the intent of the requirement is met by monitoring ECI recovery pump performance and reactor building water level. Additional mitigating factors include (a) the demonstrated margin that ensures post-accident effectiveness of the ECI recovery phase with the installed strainer modules, (b) the low contribution of ECI System recovery strainer plugging to the PSA results [NA44-CORR-33350-0265268-R000, *Pickering A Risk Assessment ECI Strainer Plugging Following a Large LOCA*, September 23, 2008], and (c) mandatory inspections to ensure the ECI recovery flowpath is free from debris prior to restart of a Pickering 5-8 or Pickering 1,4 unit following a maintenance outage [NK30-SRS-E-082-R004, *ECI Recovery Flowpath Inspection*, March 27, 2014], [NA44-SRS-E-026-U14-R019, *ECI Recovery Flowpath Inspection*, February 21, 2017]. Given the low safety significance (Safety Significance Level 3), this is assessed as an AD and no further action will be taken. • GI-38-AD1 involves unavailability targets for the Shutdown Cooling System and other heat removal systems. Clause 5.6.1 of CSA N290.11-13 requires design reliability to be established for outage heat sinks. Design reliability requirements have not been established individually for all normal and back-up

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
		heat sinks used at Pickering NGS. The reliability of all outage heat sinks (including those without explicit targets) is managed under the Risk & Reliability Program [N-PROG-RA-0016-R009, *Risk and Reliability Program*, May 27, 2016] (both through unavailability models as well as through Probabilistic Safety Assessment [P-REP-03611-00006-R00, *Pickering NGS PSA Update to Include Enhancements from the Fukushima Integrated Action Plan*, April 30, 2014]), hence reactor safety impact is assessed and monitored. The reactor safety impact of not having explicit individual heat sink design reliability is assessed, monitored and is not a significant issue, and is assessed as Safety Significance Level 4. • GI-44-AD6 involves reliability requirements for safety systems. Clause 7.6 of REGDOC-2.5.2 requires all safety related systems to be designed to meet an on-demand failure rate less than 10^{-3} yrs/yr. However, safety system performance is closely scrutinized and monitored, and established unavailability targets of all safety systems are monitored and reported annually in the Annual Risk and Reliability Report. The current engineered provisions provide sufficient functionality to ensure compliance with PSA Safety Goals [P-REP-03611-00006-R00, *Pickering NGS PSA Update to Include Enhancements from the Fukushima Integrated Action Plan,* April 30, 2014]. This is a low safety significance issue (Safety Significance Level 3). • GI-44-AD7 involves sharing of ECI and Containment. Sharing of Safety Systems and Turbine Hall: Clause 7.6.5 of REGDOC-2.5.2 has a new requirement that precludes sharing of safety systems and the turbine generator building. Pickering units share ECI and Negative Pressure Containment (NPC), as well as the turbine hall. The main impacts of sharing of ECI and Containment are addressed since if either ECI or Containment is unavailable, all affected units are considered impaired and must be shut down within specified time limits, hence reducing the risk of a coincidental DBA. Moreover environmental conditions in the common turbine building have been assessed and credited provisions have been protected to ensure the ability to shutdown/control, cool and monitor remains available on non-accident units. Required Safety Goals are met. This is a low safety significance issue (Safety Significance Level 3). • GI-44-AD9 involves REGDOC-2.5.2 requirements for ECI Heat Exchanger Leak Detection. Detection/Isolation of ECI Heat Exchanger (HX) Tube Leak: Clause 8.5 of REGDOC-2.5.2 requires

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
		ECI recovery heat exchanger tube leak detection capability. Pickering 5-8 ECI recovery heat exchangers do not have leak detection capability on the cooling water side. Pickering 5-8 has ECI recovery piping, pumps and heat exchangers outside of Containment. Components penetrating and outside Containment are all DBE qualified and Nuclear Class 2. Since the intent of leakage detection is served by the system leakage collection, recovery and radiation monitoring in the vicinity, the added benefit of implementing a design modification for direct leakage detection is a low safety significance issue (Safety Significance Level 3). • PSR1-AD6 involves requirements for independence and redundancy in the design of the Emergency Core Cooling System. The issues associated with these ADs are specific to Pickering 1,4. The Pickering 1,4 ECI system utilizes the Moderator system in its low pressure mode, and a number of instances of single failure susceptibility existed in the design of the ECI system. The issues were assessed as ADs based on the improvements made to the design to reduce the commonality between the Moderator system and ECI recovery, and some of the significant singletons were addressed by design changes. These design changes resulted in the greatest reduction in calculated core damage frequency. Therefore, this is not a safety significant issue and is assessed as a low significant AD. • PSR1-AD16 Requirements for Containment Coatings and Coverings. The gaps included in this Issue concern non-metallic coatings and liners that perform a safety function, such as preserving the concrete Containment envelope. The basic requirement is that not only should these materials be carefully selected, and applied, but their deterioration (especially under post-accident conditions) should not impair the performance of other safety functions, such as blocking of ECI strainers. All of these ADs involve the ECI system except for GI-38-AD1 and therefore the latter has negligible interaction with the other ADs. GI-44-AD9 has negligible interaction with the other ADs as it is specific to HX tube leakage. GI-36-AD1 and GI-36-AD2 both deal with ECI performance; however, strainer performance is not a function of the Shutdown System credited in the safety analysis, so there is no interaction between these ADs. Safety system separation (GI-44-AD7) is normally addressed separately from design for reliability (GI-44-AD6), and therefore the interaction of these two ADs is very low. PSR1-AD6 involves requirements for independence and redundancy in the design of the Emergency Core Cooling System, which does not interact with PSR1-AD16, which involves Requirements for Containment Coatings and Coverings which relates to performance of ECI recovery system following a

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
		postulated event. The synergy between the ADs is minimal and therefore, the aggregate impact on defence-in-depth of these ADs is assessed to be very low.
AM-326 – Engineered Features of Accident Management	PSR1-AD12	PSR1-AD12 involves requirements for post-accident monitoring design. The Pickering NGS design satisfies requirements for providing essential monitoring capability via built-in redundancy and qualification. Routine maintenance and calibration procedures in conjunction with redundant information chains and environmentally qualified sensors and cables ensure availability. In this way a need for immediate post-accident maintenance or calibration is minimized. Nevertheless, should a component fail, it may not be immediately repairable, depending on the component's location and the accident scenario. An assessment of the impact of this gap was provided in OPG Report [NK30-REP-03680-00016 R000, *OPG Response To CNSC Comments On Pickering NGS-B Integrated Safety Review-Plant Design, Safety Analysis, Safety Performance, Ageing And Equipment Qualification*, September 22, 2009]. It was concluded that there were sufficient instrument loops that would be accessible post-accident, there was sufficient redundancy and that calibrations for accuracy during post-accident mission were not necessary. Given the similarities between plant design and instrument loop locations, this item is also an AD for Pickering 1,4 There is only one AD, and therefore there is no aggregate impact for AM-326.
D-217 – Confinement of Radioactive Material	GI-44-AD5 PSR1-AD4	The following AD and AD group are assigned to Safety Principle D-217: • GI-44-AD5 involves a requirement for Containment leak-tightness for a period sufficient to implement off-site emergency measures. The requirement for Containment leak tightness for a period sufficient to implement off-site emergency measures cannot be explicitly demonstrated since an explicit set of BDBAs is not identified within the licensing basis. The current engineered provisions provide a sufficient Containment barrier to ensure compliance with existing Safety Goals. Beyond Design Basis provisions using Emergency Mitigating Equipment and SAMG provide additional means of protecting Containment integrity against potential accident sequences. Risk improvement initiatives continue to be developed to further mitigate this issue. Furthermore, due to the nature of the Pickering NGS design and construction, it is not practicable to retrofit a leak tightness barrier for a period sufficient to implement off-site emergency measures following certain BDBAs. This is a low

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
		safety significance issue (Safety Significance Level 3) and is being managed to the extent practicable. Therefore, it is assessed as an AD. • PSR1-AD4 involves specific requirements for the design of Concrete Containment Structures. The ADs associated with this group involve specific requirements for the design of Concrete Containment Structures. This group of ADs applies to initial construction, installation, and fabrication of mechanical splices for Concrete Containment Structures. The original and subsequent revisions of CSA N287 series of standards were issued after completion of design and construction of Pickering NGS. All N287.3-93 (R2004) Pickering B ISR Review findings were assessed to be ADs in PSR1 given robust design practices at the time, together with ongoing periodic inspections and in-service testing. The standards that applied during the original construction of Pickering NGS included requirements for tests and quality control procedures to ensure that the concrete used in the as-built structures met the original design requirements. The PSR2 review of N287.3-14 also outlines Programs, Procedures and Standards which are credited with the ability to detect and monitor any safety significant degradation mechanisms and thus to provide assurance of continued fitness for service of the Pickering NGS CCSs. While the two issues in this assessment are related to Containment structure design, their impacts are on different aspects of the design. GI-44-AD5 is related to the design requirement for Containment leak-tightness, while the second AD group, PSR1-AD4, is related to verification of the original design and construction. Therefore, there is no aggregate impact of the two ADs assessed under this safety principle.
D-221 – Protection of Confinement Structure	PSR1-AD9	PSR1-AD9 involves hydrogen monitoring requirements related to BDBAs. Clause 10.2.3 of N290.3-11 states: "New builds shall monitor the conditions identified according to the requirements of Clause 10.2.2 [... (e.g., pressure, temperature, and hydrogen concentration inside Containment) that need to be monitored for BDBAs ...] " The absence of hydrogen monitoring instrumentation is acceptable, given installed hydrogen mitigation equipment together with SAMGs provide means to safely manage any hydrogen in Containment. There is only one AD, and therefore no aggregate impact for D-221.
D-227 –	GI-44-AD10	GI-44-AD10 involves REGDOC-2.5.2 requirements for Beyond Design Basis qualification of the Safety

Safety Principle	AD #	Aggregate AD Assessment by Safety Principle
Monitoring of Plant Safety Status		Parameter Display System. The Safety Parameter Display System cannot be demonstrated to be qualified for BDBAs since an explicit set of BDBAs is not identified within the licensing basis. As part of the follow-up to the 2011 Fukushima accident, instrumentation to support critical parameters required to function for BDBAs has been evaluated for survivability in [N-REP-09013-10007 R000, *Ontario Power Generation Severe Accident Management Guidance Instrumentation and Equipment Survivability - Summary Report*, December 13, 2013]. The instrument loops associated with these critical parameters have been identified for use in Critical Safety Parameter Monitoring and BDBA accident management guidelines. The indications from these loops are not in one central location and, in some cases, require field action (e.g., power) to obtain data. The capability to monitor post-accident conditions remotely from the Main Control Room is provided in the SDSE Instrument Rooms for Units 1,4 and in the Unit Emergency Control Centres for Units 5-8. Since the intent of this requirement is met, the added benefit of implementing a design modification addressing this gap is not significant. This is a low safety significance issue and has been addressed to the extent practicable with enhancements already identified in [N-REP-09013-10009, *Information to Support Closure of FAI 1.8.1 - Survivability Assessments for Equipment and Instrumentation for Severe Accident Management*, December 17, 2013]. Therefore, it is assessed as an AD. There is only one AD, and therefore no aggregate impact for D-227.

Overall AD Aggregation for Defence-in-Depth Level 4
The ADs associated with the safety principles in Level 4 defence-in-depth include the following diverse areas: • D-158 – General Design Basis – The AD groups are related to deterministic safety analysis. Although there may be some interaction among the ADs in this area, this interaction does not pose an aggregate impact for extended operation as some of the issues have largely been addressed, are in progress or are already considered in the dispositioning of these ADs via the REGDOC-2.4.1 Implementation Plan for the extended operation. • D-207 – Emergency Heat Removal – All of these ADs involve the ECI system except for GI-38-AD1, which involves unavailability targets for the Shutdown Cooling System and other heat removal systems, and therefore the latter has negligible interaction with the other ADs. The Pickering 1,4 ECI system reliability meets the licensing target, as demonstrated in [P-REP-09051.1-00016-R000, Pickering NGS - 2016 Annual Risk and Reliability Report, March 31, 2017] and Level 1 and Level 2 PSA demonstrate that the existing plant design satisfies all safety requirements. On this basis, and as per the low safety significance of the issues related to the group of ADs assessed in this safety principle, there is no aggregated impact. • AM-326 – Engineered Features of Accident Management – The one AD group assessed in this safety principle involves requirements for post-accident monitoring design. The Pickering NGS design satisfies requirements for providing essential monitoring capability via built-in redundancy and qualification. Routine maintenance and calibration procedures in conjunction with redundant information chains and environmentally qualified sensors and cables ensure availability. • D-217 – Confinement of Radioactive Material – The ADs assessed for this safety principle are related to Concrete Containment Structures. GI-44-AD5 is related to the design requirement for Containment leak-tightness for a period sufficient to implement off-site emergency measures for BDBAs, while the second AD group, PSR1-AD47, is related to verification of the original design and construction which is periodically assessed through inspections and monitoring. The PSR2 review of N287.3-14 also outlines Programs, Procedures and Standards which provide the ability to detect and monitor any safety significant degradation mechanisms and thus to provide assurance of continued fitness for service of the Pickering NGS Concrete Containment Structures. • D-221 – Protection of Confinement - PSR1-AD9 involves hydrogen monitoring requirements related to BDBAs. The absence of hydrogen monitoring instrumentation is acceptable, given installed hydrogen mitigation equipment which together with SAMGs provides means to safely manage hydrogen in Containment. • D-227 – Monitoring of Plant Safety Status – The AD related to this safety principle (GI-44-AD10) involve REGDOC-2.5.2 requirements for Beyond Design Basis qualification of the Safety Parameter Display System. As part of the follow-up to the 2011 Fukushima accident, instrumentation to support critical parameters required to function for BDBAs has been evaluated for survivability in OPG Report [N-REP-09013-10007 R000, *Ontario Power Generation Severe Accident Management Guidance Instrumentation and Equipment Survivability - Summary Report*, December 13, 2013]. This issue is a low safety significance issue and has been addressed to the extent practicable. GI-32 addresses conformance with the safety significant requirements of CNSC REGDOC-2.4.2. For the purposes of this aggregation

Overall AD Aggregation for Defence-in-Depth Level 4
assessment, no additional enhancements to the OPG REGDOC-2.4.2 Implementation Strategy are credited beyond those currently planned. The ADs associated with the safety principles in Level 4 defence-in-depth cover diverse areas with no significant interactions among them. The aggregate impact of the ADs for each safety principle is determined to have no significant impact on safety. Furthermore, the overall aggregate impact of all of the ADs associated to Level 4 defence-in-depth is assessed to be negligible.

AD Aggregation for Defence-in-Depth Level 5

There are no ADs associated with the Safety Principles in Level 5 defence-in-depth.

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CNSC Correspondence Routing Sheet

Section 1 – Correspondence Preparation and Review (Co-ordinated by Author)

Title: Pickering NGS Periodic Safety Review 2 – Submission of Global Assessment Report Revision 1

CD# P-CORR-00531-05292	**CNSC Due Date:** ~~16~~ 12 Feb 2018
This letter completes a: ☐ REGO ☐ REGC ☐ REGM ☐ MGMT ☐ N/A	Associated AR No. ~~N/A~~ 28207858-14.

- Correspondence package satisfies applicable requirements of N-PROC-RA-0047, Communications with the Canadian Nuclear Safety Commission. ☒
- Correspondence package has been prepared using the guidelines in N-GUID-00531-10001, Preparation of Correspondence to the Canadian Nuclear Safety Commission. ☒
- Actions documented in letter are planned in accordance with N-PROC-RA-0006, Regulatory Action Management, and **accepted by assignees**. ☒ ☐ N/A
- Complete Section 3 of this form – Work Impact – New Commitments ☒ ☐ N/A

REVIEWERS (identified by Author and Line Management) (use extra page if required)

(1) **Name/Title (Sign):** Austin Dilts *Austin Dilts* **Date:** Jan. 31, 2018

(2) **Name/Title (Sign):** ~~John O'Connor~~ **Date:**

(3) **Name/Title (Sign):** **Date:**

This submission has received adequate review and is accurate and complete in its technical and general statements of fact, and that **comments** from all **reviewers** have been appropriately **dispositioned**.

AUTHOR (Print/Sign): Ricahrd MacEacheron [signature] **Date:** Jan 30, 2018

AUTHOR'S SECTION MANAGER or MANAGER (Print/Sign): Kristina Bramma *KBramma* **Date:** 31 Jan 18

AUTHOR'S MANAGER or DIRECTOR (Print/Sign): Mike Ruffolo [signature] **Date:** 2 Feb 18

Section 2 - Final Review and Routing for Signature (Completed by Regulatory Affairs)

RA SPOC: [signature] **Date:** Feb 5/18

RA Manager or RA Site Section Manager: P. Herrera [signature] **Date:** Feb 7/18

Others to Concur (as determined by Regulatory Affairs):

(1) **Name:** Jason Wight [signature] **Date:** 8-FEB-2018

(2) **Name:** **Date:**

(3) **Name** **Date:**

The attached submission has been reviewed/concurred as indicated above and is ready for signature and issuance.

Designated Licensing Authority: P. Herrera [signature] **Date:** Feb 8, 2018

CNSC Correspondence Routing Sheet

Section 3 – Work Impact – New Commitments (Completed by Author)

Instructions:

1. Complete the table below for each new commitment made in the letter (REGO, REGC, REGM, or MGMT). Copy this page if more than 2 commitments are made. Guidance can be found in N-PROC-RA-0006 and N-PROC-AS-0019.
2. Provide the Commitment Owner Alert Group, the overall due date, and the Subject and Description of the commitment Action Request as it should appear in Action Tracking.
3. For each commitment Action Request, provide the associated Action Tracking assignment subject, the responsible person's name, the Alert Group and assignment due date.
4. To add table rows, place cursor in the last cell of the last row and press Tab.

New Commitment Type: REGO ☐ REGC ☐ REGM ☐ MGMT ☐		Commitment Owner Alert Group:			Overall Due Date
Subject of Action Request:					
Description of Action Request:					
No.	Assignment Subject	Assignment Details	Responsibility	Alert Group	Due Date
1.					
2.					
3.					
4.					

New Commitment Type: REGO ☐ REGC ☐ REGM ☐ MGMT ☐		Commitment Owner Alert Group:			Overall Due Date
Subject of Action Request:					
Description of Action Request:					
No.	Assignment Subject	Assignment Details	Responsibility	Alert Group	Due Date
1.					
2.					

3.					
4.					

Prepared by (Name): R. MacEacheron Date:

Directorate of Power Reactor Regulation

e-Doc 5470609
File 4.01.02
RIB # 12135

March 2, 2018

Mr. Randy Lockwood
Senior Vice President
Pickering Nuclear
Ontario Power Generation Inc.
1675 Montgomery Park Road, P41, 3G-4
Pickering, ON L1V 2R5

Subject: Pickering NGS: CNSC Staff Acceptance of Pickering NGS Periodic Safety Review 2 (PSR2) Integrated Implementation Plan (IIP), Revision 1

Dear Mr. Lockwood:

Canadian Nuclear Safety Commission (CNSC) staff have completed the review of OPG Integrated Implementation Plan (IIP) R001 [1], the last deliverable under Pickering NGS Periodic Safety Review 2 (PSR2), conducted pursuant to REGDOC-2.3.3 [2]. The initial revision of the IIP [3] was submitted by OPG on November 30, 2017 and was reviewed and commented by CNSC staff [4].

Under the provisions of the CNSC/OPG Protocol [5] for the conduct of PSR in support of Pickering NGS licence renewal, numerous meetings and communication exchanges took place between CNSC staff and OPG, to clarify the regulatory expectations, and receive additional detailed information regarding planned IIP activities and their corresponding target completion dates. In addition, CNSC staff visited the Pickering NGS site on February 14, 2018 to collect information specifically related to measures for containment controlled filtered venting. Furthermore, CNSC staff and OPG met on February 26, 2018 to ensure that the revised IIP is clear, consistent with the PSR2 Global Assessment Report and supported by OPG administrative programs and databases for tracking and reporting.

CNSC staff review of the IIP R001 [1] determined that OPG has satisfactorily dispositioned CNSC staff comments [4] on IIP revision 0 [3] and incorporated sufficient information regarding the actions to be undertaken under IIP R001.

CNSC staff also determined that the IIP, as submitted in [1, 3], fulfils the regulatory requirements of CNSC REGDOC-2.3.3, meets CNSC staff expectations and thus is a satisfactory foundation for plant safety enhancements for the extended operations of Pickering NGS. As such, CNSC staff conclude that Pickering Periodic Safety Review Integrated Implementation Plan [1] is acceptable.

Should you have any questions, please contact Dr. Al Omar at al.omar@canada.ca or at 613-995-0565.

Sincerely,

Alexandre Viktorov, Ph.D.
Regulatory Program Director
Pickering Regulatory Program Division

c.c.: PickeringRPD, A. Omar, B. Rzentkowski
P. Herrera, R. MacEacheron

References:

1. OPG letter, R. Lockwood to A. Viktorov, "Pickering NGS Periodic Safety Review 2 – Submission of Integrated Implementation Plan Revision 1", March 1, 2018 CD# P-CORR-00531-05311, e-Doc 5470841; with Enclosure — OPG Document, "Pickering NGS PSR2 Integrated Implementation Plan", P-REP-03680-00031-R001, February 28, 2018.

2. CNSC Regulatory Document, "Periodic Safety Reviews", REGDOC-2.3.3, April 2015.

3. OPG letter, R. Lockwood to A. Viktorov, "Pickering NGS Periodic Safety Review 2 – Submission of Integrated Implementation Plan", November 30, 2017. CD# P-CORR-00531-05085, e-Doc 5406515; Enclosure — OPG Document, "Pickering NGS Periodic Safety Review 2 (PSR2) Integrated Implementation Plan", P-REP-03680-000310 R000, November 30, 2017.

4. CNSC Letter, A. Viktorov to R. Lockwood, "Pickering NGS: Periodic Safety Review 2 - CNSC Staff Review of OPG Integrated Implementation Plan (IIP), Rev. 000", February 13, 2018, CD# P-CORR-00531-05315, e-Doc 5455745.

5. Protocol, "OPG-CNSC Protocol for the Conduct of a Periodic Safety Review in Support of Pickering NGS Licence Renewal", January 17, 2017, CD# P-CORR-00531-04725 R001, e-Doc 5143721.

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1675 Montgomery Park Road, P.O. Box 160, Pickering, Ontario L1V 2R5

March 1, 2018 **OPG PROPRIETARY**

CD# P-CORR-00531-05311

DR. A. VIKTOROV
Director
Pickering Regulatory Program Division

Canadian Nuclear Safety Commission
280 Slater Street
Ottawa, Ontario
K1P 5S9

Dear Dr. Viktorov:

Pickering NGS Periodic Safety Review 2 – Submission of Integrated Implementation Plan Revision 1

The purpose of this letter is to submit for Canadian Nuclear Safety Commission (CNSC) staff acceptance the Pickering Periodic Safety Review 2 (PSR2) Integrated Implementation Plan (IIP), Report P-REP-033680-00031-R001 (Enclosure 1).

In April 2015, the CNSC informed Ontario Power Generation (OPG) that a Periodic Safety Review (PSR) was required to support extension to commercial operation of Pickering NGS beyond 2020 (Reference 1).

In support of OPG's decision to continue Pickering NGS commercial operation to the end of 2024 (Reference 2), OPG has conducted a PSR which is an internationally recognized process defined in IAEA Specific Safety Guide 25, and regulated in Canada under CNSC REGDOC-2.3.3 – Periodic Safety Reviews.

The objective of Pickering's PSR, referred to as PSR2, as it builds on previous assessments, was to confirm that the design, operation, structures, systems, and components support continued safe operation, and to recommend reasonable and practical safety enhancements to further improve the already low risk of plant operation.

The PSR2 was conducted thoroughly by senior industry experts with different teams independently executing each of the four major phases:

- **PSR2 Basis Document:** A PSR2 Basis Document defining the scope of the PSR2 process, and documenting how the PSR2 was to be conducted was prepared (Reference 3) and accepted by the CNSC (Reference 4) in July 2016.

- **Safety Factor Reviews:** Fifteen Safety Factor and two Complementary Reviews identifying compliances and gaps were completed and submitted for CNSC review in fall 2016 / winter 2017.

- **Global Assessment:** A Global Assessment that consolidated gaps from the Safety Factor and Complementary Reviews into Global Issues with proposed resolutions which were assigned safety significance, prioritized and ranked was completed. The results from the Global Assessment are documented in a PSR2 Global Assessment Report (GAR) submitted (Reference 5) for CNSC review in October 2017. A revision to the GAR was subsequently submitted (Reference 6) in February 2018 and acknowledged by the CNSC that it satisfied requirements of CNSC REGDOC 2.3.3. (Reference 7).

  The Global Assessment concludes that the current Pickering NGS design, operation, processes and management system will ensure continued safe operation of Units 1,4 and 5-8, both in the short term, and for extended operation. The GAR recommends reasonable and practical resolutions that further enhance safety.

- **Integrated Implementation Plan (IIP):** The Pickering PSR2 IIP represents the final step in the comprehensive PSR process which documents the actions with target completion dates that support the reasonable and practical resolutions from the Global Assessment. Revision 0 of the PSR2 IIP was submitted (Reference 8) in November 2017. This revision to the PSR2 IIP incorporates minor enhancements and comments received from the CNSC (References 9) that are dispositioned in Attachment 1.

In accordance with REGDOC 2.3.3, OPG has in place the required organizational structure and administrative instructions for ensuring successful execution of the IIP.

The Pickering PSR2 confirms that there are no safety issues for continued safe operation of the Pickering NGS through 2024, and the actions within this IIP will further enhance safety.

OPG's comprehensive programs are aligned with industry best practices for ensuring the condition of important Structures, Systems and Components are well understood and well maintained. The Pickering PSR2 and the enclosed IIP represents OPG's commitment to continual improvement in plant condition, operation and performance.

If you have any questions, please contact Paulina Herrera, Manager, Pickering Regulatory Affairs at 905-839-1151 extension 3235.

Randy Lockwood
Senior Vice President
Pickering Nuclear

cc: CNSC Site Office – Pickering
CNSC Pickering Regulatory Program Division (copy to each staff)

References:

1. CNSC Letter, M. Santini to B. McGee, "Pickering NGS: CNSC Staff Assessment of 2014 COP, SOP, and CALs", June 18, 2015, e-Doc 4782433 CD# P-CORR-00531-04493.

2. OPG Letter, R. Lockwood to G. Frappier, "End of Commercial Operation of Pickering NGS", June 28, 2017, CD# P-CORR-00531-04930.

3. OPG Letter, B. McGee to H. Khouaja, "Submission of Pickering NGS Periodic Safety Review 2 Basis Document Revision 002", July 6, 2016, CD# P-CORR-00531-04780.

4. CNSC Letter, H. Khouaja to B. McGee, "Pickering NGS: CNSC Staff Acceptance of Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document", July 8, 2016, e-Doc 5037314, CD# P-CORR-00531-04789

5. OPG Letter, R. Lockwood to A. Viktorov, "Pickering NGS Periodic Safety Review 2 - Submission of Global Assessment Report", October 30, 2017, CD# P-CORR-00531-05084.

6. OPG Letter, R. Lockwood to A. Viktorov, "Pickering NGS Periodic Safety Review 2 - Submission of Global Assessment Report Revision 1", February 12, 2018, CD# P-CORR-00531-05292.

7. CNSC Letter, A. Viktorov to R. Lockwood, "Pickering NGS: Periodic Safety Review 2 - CNSC Staff Review of OPG Global Assessment Report (GAR), Revision 1", February 19, 2018, e-Doc 5461487, CD# P-CORR-00531-05322.

8. OPG Letter, R. Lockwood to A. Viktorov, "Pickering NGS Periodic Safety Review 2 - Submission of Integrated Implementation Plan", November 30, 2017, CD# P-CORR-00531-05085.

9. CNSC Letter, A. Viktorov to R. Lockwood, "Pickering NGS: Periodic Safety Review 2 - CNSC Staff Review of OPG Integrated Implementation Plan (IIP), Rev. 000", February 13, 2018, e-Doc 5455745, CD# P-CORR-00531-05315.

Attachments:

1. "Disposition of CNSC Comments on P-REP-03680-00031"

Enclosures:

1. OPG Document, "Pickering NGS PSR2 Integrated Implementation Plan", February 28, 2018, P-REP-03680-00031-R001.

Attachment 1

Disposition of CNSC Comments on P-REP-03680-00031-R000

Attachment: Results of CNSC Staff Review of OPG Integrated Implementation Plan, Rev 000 [A1]

#	Page #	Section # / Paragraph # / Figure #	The Issue	CNSC Staff Comments and Recommendations	OPG Dispositions
1.		General		OPG is to disposition remaining CNSC staff comments [A2] on OPG Draft IIP of 3 Nov 2017.	Remaining comments dispositioned.
2.		General		OPG is to consider if some of the TCDs need to be adjusted based on latest available OPG information and discussions with CNSC staff.	TCDs adjusted by OPG on a number of actions with the latest information.
3.		General		OPG is to add some wording in the IIP describing the "Enhancement Value", which is mentioned in detail in the PSR Database	Paragraph summarizing the "Enhancement Value" added to section 3.1.3 of the IIP.
4.		General		OPG is to ensure that the latest revisions to the GAR [A3] are reflected and are consistent with the IIP.	GAR R01 information reflected and is consistent with the IIP.
5.	12	Figure 3	Using 0.00E+00 on the vertical scale is not correct.	Modify the vertical scale and re-configure the figure accordingly.	Figure 3 vertical scale modified to a "log scale" and figure 3 re-configured and updated with data values.
6.	14	1.3.3 Heading	Consistency of expressing "Extended Operation Period"	**Deterministic Safety Analysis to Support ~~Extended Operating Period~~ Operation to the End of 2024 (GI-24)**	Updated to read "**Deterministic Safety Analysis to Support ~~Extended Operating Period~~ Operation to the End of 2024 (GI-24)**"

#	Page #	Section # / Paragraph # / Figure #	The Issue	CNSC Staff Comments and Recommendations		OPG Dispositions
7.	17	2.1 Paragraphs 3 and 4	The statement in paragraph 3, *The review of Safety Factors 1 through 15 and other complementary assessments confirmed that the design, condition and operation of Pickering Units 1, 4 and 5-8 (as well as common systems) will support continued safe operation for the extended operating period,* contradicts the information provided in Section 1.3.2.1 which states that condition assessments must	Until all condition assessments are completed, it is not appropriate to state that the SFR reviews confirmed that the condition of SSCs will support safe operation for the extended operating period. Plus there are activities remaining to confirm the fitness-for-service of fuel channels for the extended operating period. This statement together with the statement in Paragraph 4 in Section 2.1 should be further qualified.	OPG should consider changing the sentence to read " … (as well as common systems) will ~~support~~ ensure continued safe …"	Sentence updated to read " … (as well as common systems) will ~~support~~ ensure continued safe …"

#	Page #	Section # / Paragraph # / Figure #	The Issue	CNSC Staff Comments and Recommendations	OPG Dispositions
			be completed for various components.		
8.	18	1st paragraph	OPG stated that "Furthermore, the Global Assessment confirmed that OPG and Pickering NGS Senior Leadership are committed to investing in the plant, and focusing the organization to strive for continued improvement in the plant condition, operation and performance."	OPG should indicate where in the Global Assessment process as — reported in the GAR — that some analysis was performed the results of which confirmed that "OPG and Senior Leadership" are committed to investing in the plant, and focusing ..."OPG needs to refer to some pages or quotes from the GAR to support this type of statements. Otherwise this sentence should be removed from this section 2.2.	Wording updated to read: "Furthermore, OPG Pickering NGS Senior Leadership is committed to investing in the plant, and focusing the organization to strive for continued improvement in the plant condition, operation and performance." This paragraph moved from Section 2.2 to the end of section 2.2.1.
9.	20 and 21	2.2.3, 1st paragraph, 1st sentence,	It is stated that "Through PSR2, 143 gaps were identified from	Part of the 143 gaps is a sum of 23 Type III Additional Gaps that were identified by CNSC staff. OPG should modify this sentence and make it consistent with the information	Section 2.2.3 modified to make it consistent with the information included in Figure 4 page 16 and more clearly describes where the numbers come from.

#	Page #	Section # / Paragraph # / Figure #	The Issue	CNSC Staff Comments and Recommendations	OPG Dispositions
		and 1st paragraph on page 21	various sources (e.g. Safety Factor Reports, Complementary Assessments and Expert Panel review) and were integrated into the Global Assessment."	included in Figure 4 on page 16. In addition, first paragraph on page 21 has to be reconciled and edited to make the first and second paragraphs of 2.2.3 consistent with Figure 4 on page 16.	
10.	21	Second paragraph	Consistency of expressing "Extended Operation Period"	Replace the 2 occurrences of "extended operation period" with "operation to the end of 2024"	IIP document updated to replace the 2 occurrences of "extended operation period" with "operation to the end of 2024"
11.	22, and elsewhere	3rd paragraph	Consistency of the use of "IIP Administration Instruction"	Here and on pages 23, 25 OPG uses "IIP Administration process", while on page 27 and Reference 14 OPG uses "IIP Administration Instruction "IIP Administration", respectively. OPG should use consistent terminology that is. As well, consistent with the terminology used in OPG's IIP Administration Instruction document. Also, OPG used "IIP administrative and change control process on pages 16 and 114,	IIP document revised to ensure consistent use of the term "IIP Administration Instruction" [R-14] throughout the document as appropriate.
12.	24	Figure 5		The text supporting and describing Figure 5 does not cover all the boxes illustrated in the	Figure 5 showing IIP structure removed from the document. Refer to the IIP Administration

#	Page #	Section # / Paragraph # / Figure #	The Issue	CNSC Staff Comments and Recommendations	OPG Dispositions
				Figure. OPG should complete the description of all the boxes and roles. In addition, OPG should ensure that Figure 5 is consistent with the IIP Administrative Instruction being developed by OPG.	Instruction for figure and description of roles and accountabilities.
13.	25	3.1.3	Consistency of the use of the term "IIP Administration Instruction"	OPG is to ensure consistency between this section 3.1.3 and the "IIP Administration Instruction" being developed by OPG; especially the management of the "IIP change control process mentioned in 2nd paragraph under section 3.1.3. In addition, ensure consistency of the entire section 3.1.3 with the IIP Administration Instruction being developed.	IIP Document revised to remove section 3.1.3. Refer to the IIP Administration Instruction for detailed information regarding execution management and oversight.
14.	27	3.2.2	Consistency with the IIP Administrative Instruction being developed by OPG	OPG to ensure that the contents of Section 3.2.2 are reflected in and consistent with the IIP Administration Instruction being developed by OPG.	IIP Document revised to remove section 3.2.2. Refer to the IIP Administration Instruction for detailed information regarding change control.
15.	39, 69	Appendix A, B	GI-12 was assigned to "06-Fitness for service" SCA, which is not in line with CNSC SCA structure.	According to CNSC SCA structure, Environmental Qualification of Equipment belongs to "Physical Design" SCA. GI-12 should be listed under "SCA 05 Physical Design"	GI-12 SCA will be updated throughout the document to be consistent with SCA 05 Physical Design instead of SCA 06.

#	Page #	Section # / Paragraph # / Figure #	The Issue	CNSC Staff Comments and Recommendations	OPG Dispositions
16.	52	Appendix B	**G01-RS4-06-04** - The scope of this action doesn't explicitly indicate that the Technical Basis Document for the FC LCMP will be updated from that of 2011, as identified under SF2-AG1.	It is proposed to consider making the following changes (indicated in red): Under **G01-RS4-06-04.2**: "Update and Submit 2018 Fuel Channel Life Cycle Management Plan (LCMP), the Technical Basis Document (TBD) used in the FC LCMP, and the Pickering NGS Fuel Channel Readiness Plan in Support of Operation to the end of 2024 (FCRP2024)" Under **Action**: Incorporate the results of the Pickering NGS FCRP2024 activities into Fuel Channel assessments/evaluations and identify actions to mitigate aging effects, as required. Update the FCRP2024, the TBD used in FC LCMP, and the 2018 LCMP accordingly. Under **Completion Criteria**: This action is considered complete when the FCRP2024 (P-PLAN-31100-00002) update, the TBD used in FC LCMP update, and the 2018 Fuel Channel LCMP (N-PLAN-01060-10002) update have been submitted to CNSC.	The submission of the Technical Basis Document will be managed outside of the IIP. No change to the IIP document wording.
17.	69	Appendix B, IIP Action G12-RS1-06-17.1	Action G12-RS1-06-17.1 text states "Assess existing EQAs for Environmentally	To facilitate gauging the successful completion of this action, CNSC staff expect OPG's outputs to include: • The total number of EQAs that need to be assessed	The plan for OPG to address this action are as follows: 1. Assessment of 205 EQAs to identify potential gaps and recommendations

#	Page #	Section # / Paragraph # / Figure #	The Issue	CNSC Staff Comments and Recommendations	OPG Dispositions
			Qualified (EQ) life-limited components to support commercial operation to 2024." The target completion date is 2019-12-31.	• The plan to assess these EQAs, say starting in early 2018, including the number or percentage of EQAs that is planned to be completed in 2018 and 2019.	for updated PNGS end of commercial operations. 2. Upon completion of step 1, updating of EQAs as required to address the gaps and recommendations. Target to have 50 percent completed by end of 2018 and remaining completed in 2019.
18.	79	Appendix B	Wrong reference in the success criteria	Change "N-PROG-RA-0013" to "N-PROG-RA-0016"	Reference changed to "N-PROG-RA-0016"
19.	79	G27-RS2-04-24	**IIP action G27-RS2-04-24 states that:** "Investigate and implement additional practicable design, operational and/or analytical enhancements to further improve Pickering 1, 4 Severe Core Damage	The current description of G27-RS2-04-24 is too general to address specific CNSC staff concerns regarding the demonstration of effectiveness of strategies (e.g. EME, fire water injection, etc.) to show that they can arrest core degradation and ensure in-vessel retention (SF6-AG3). OPG is to include a new action under G27-RS2-04-24 to record the results of the investigation and to address the demonstration of effectiveness of strategies (e.g. EME, fire water injection, etc.) analytically and through exercises/drills to show that they can arrest core degradation	New action added under G27-RS2-04-24 titled "**Investigate additional practicable design, operational and/or analytical enhancements**" which will record the results of the investigation and demonstrate effectiveness. A new action added under G40-RS1-10-28 titled "**Complete reassessment of Pickering NGS Beyond Design Basis Containment Integrity**" To document OPGs plan to submit a letter to address the assessment of options that can be used to mitigate the consequences of a severe accident beyond IVR to ensure the controlled release from the vacuum building.

#	Page #	Section # / Paragraph # / Figure #	The Issue	CNSC Staff Comments and Recommendations	OPG Dispositions
			Frequency and Large Release Frequency (e.g., alternative emergency cooling water makeup)."	and **ensure in-vessel retention (IVR)** in a timely manner (SF6-AG3). In addition, OPG is to include the assessment of options that can be used to mitigate the consequences of a severe accident beyond IVR to ensure the controlled release from the vacuum building.	
20.	116	Item D 3.19	OPG indicates that "The majority (52 of 75) of the AGs were Type I & II (requests for additional information) that will be addressed by OPG before March 15, 2018 [D-6]. None of these AGs invalidate the conclusions of the associated report.	Although this statement is true at the time of developing Rev 0 of the IIP, the results of the preliminary CNSC review [A4] of the AG's may lead to implementation a number of additional IIP actions or the modification of the scope of existing IIP actions to disposition such AGs by OPG.	Comments related to AG's will be discussed and disposition under the matters related to the AGs. No change to the IIP document.

References:

[A1] OPG letter, R. Lockwood to A. Viktorov, "Pickering NGS Periodic Safety Review 2 – Submission of Integrated

Implementation Plan", November 30, 2017, e-Doc 5406515, Enclosure 1, OPG Report, "OPG Report, "Pickering NGS Periodic Safety Review 2 (PSR2) Integrated Implementation Plan", CD# P-REP-03680-00031-R000, e-Doc 5406515.

[A2] OPG e-mail, R. MacEacheron to A. Omar, "Comments &Disposition Table for CNSC's Comments on Draft IIP", December 19, 2017, e-Doc 5455777.

[A3] CNSC letter, A. Viktorov to R. Lockwood, "Pickering NGS: Periodic Safety Review 2 – CNSC Staff Review of OPG Global Assessment Report (GAR)", January 29, 2018, e-Doc 5441553.

[A4] CNSC Staff e-mail, A. Omar to R. MacEacheron, "List of AGs – Table of OPG Follow-up Actions", attachment with e-Doc 5453451, February 02, 2018, e-Doc 5455611.

Enclosure 1 to OPG Letter, R. Lockwood to A. Viktorov, "Pickering NGS Periodic Safety Review 2 – Submission of Integrated Implementation Plan Revision 1", CD# P-CORR-00531-05311.

Enclosure 1

Pickering NGS PSR2 Integrated Implementation Plan

P-REP-03680-00031-R001

(120 pages including this coversheet)

Pickering NGS Periodic Safety Review 2 (PSR2) Integrated Implementation Plan

P-REP-03680-00031-R001
2018-02-28

Order Number: N/A
Other Reference Number:

OPG Proprietary

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Authorized by: Mar 1, 2018
R. Lockwood Date
Senior Vice President
Pickering

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Revision Summary

Revision Number	Date	Comments
R000	2017-11-30	Initial issue. Prepared by RCM Technologies P-REP-03680-00031-R000.
R001	2018-02-28	Final issue. Prepared by OPG incorporating e-Doc 5455745 “CNSC Staff Review of OPG Integrated Implementation Plan (IIP) Rev.000 dated February 13, 2018.”

Executive Summary

The Pickering Nuclear Generating Station has been operated safely by Ontario Power Generation (OPG) for over 40 years. Continuous investments in plant condition, driven by strong reliability programs have enhanced safety and reliability throughout the plant's life. OPG and the leadership team at Pickering remain committed to continued safety and reliability.

In June 2015, the Canadian Nuclear Safety Commission (CNSC) informed OPG that a Periodic Safety Review (PSR) was required to support extension to commercial operation of Pickering NGS beyond 2020.

In support of OPG's plan to extend commercial operation of Pickering NGS to the end of 2024, OPG has conducted a Periodic Safety Review (PSR) which is an internationally recognized process defined in IAEA Specific Safety Guide 25, and regulated in Canada under CNSC REGDOC-2.3.3 – *Periodic Safety Reviews*. The objective of Pickering's PSR, referred to as PSR2 as it builds on previous assessments, was to confirm that the design, operation, structures, systems, and components (SSCs) support continued safe operation, and to recommend reasonable and practicable safety enhancements to further improve the already low risk of plant operation.

This document, Pickering NGS PSR2 Integrated Implementation Plan (IIP), which is submitted to the CNSC for review and acceptance, represents the final step in a comprehensive two-year process that further demonstrates OPG's commitment to safe, reliable operation, and maintaining a healthy nuclear safety culture.

Pickering PSR2 has acknowledged and credited many actions that were already in progress to enhance safety and reliability. The PSR2 review found strength in managed systems and programs. Additional initiatives under existing programs have been identified that will ensure safety and reliability are maintained and enhanced throughout the extended operations period to the end of 2024.

OPG is committed to continuous improvement in safety at all of its nuclear facilities and has robust comprehensive programs in place that are aligned with industry best practices for ensuring the condition of SSCs important to safety is well understood and well maintained. Pickering NGS reactor units will be operated only if fitness for service of SSCs important to safety is assured.

Pickering PSR2 confirms that there are no safety issues that preclude continued safe operation of Pickering NGS through 2024. The actions within this IIP will further enhance safety and reliability.

1.0 OVERVIEW

OPG is responsible for approximately half of the electricity generated in the Province of Ontario, where Pickering Nuclear Generating Station (NGS) supplies 14% of Ontario's electricity needs.

In June 2015, the CNSC informed OPG [R-16] that a PSR would be required to support Pickering NGS commercial operation beyond 2020. As a result, and in support of OPG's decision [R-17] to extend commercial operation of Pickering NGS to the end of 2024, OPG has completed a PSR in accordance with Regulatory Document REGDOC-2.3.3 [R-9]. This IIP document represents the final deliverable in the PSR process.

Continued commercial operation of Pickering through 2024 will ensure that the province has a reliable source of Green-House-Gas (GHG)-free, baseload electricity. This represents an estimated reduction of GHG emissions of 17 million tonnes per year of extended operation. Continuing to operate Pickering NGS through 2024 not only contributes to the province meeting its environmental initiative, it is also associated with 4,500 direct and indirect jobs across the province.

1.1 Pickering NGS Performance

1.1.1 Safety Performance

Safety at all of OPG's facilities is an overriding priority and essential in the pursuit of achieving high performance goals.

Combining a safe robust design with mature programs that meet or exceed industry standards and regulatory requirements has allowed OPG to operate Pickering NGS safely for over 40 years. Continuous investments in plant condition, driven by strong reliability programs and an OPG leadership team that is committed to safety have enhanced performance throughout the plant's life.

Pickering NGS continues to have strong performance in all areas of safety with a conventional safety performance rating that is in the industry's top quartile.

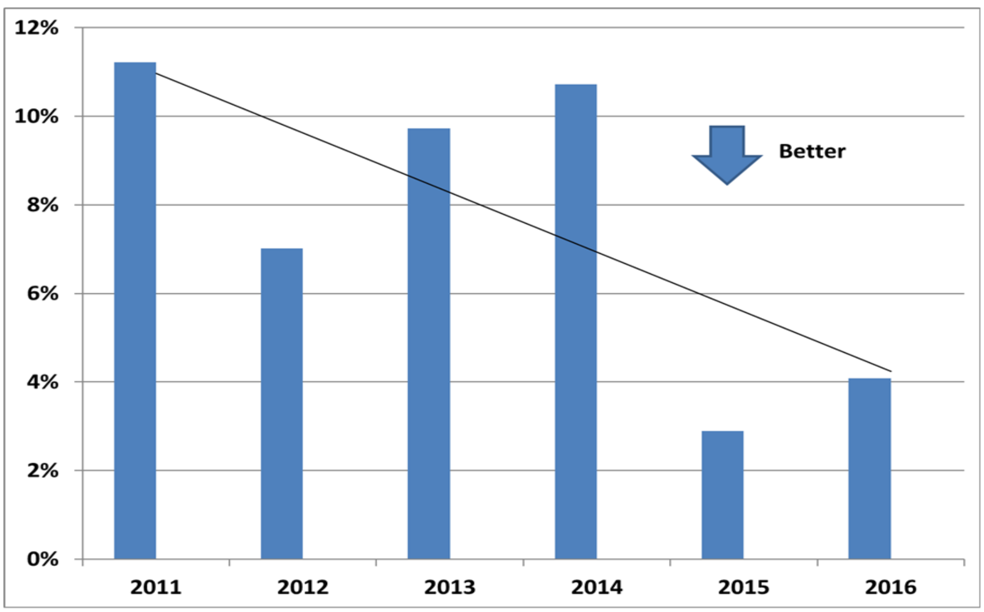
OPG has comprehensive Probabalistic Safety Assessments (PSA) in place for Pickering NGS 1,4 and Pickering NGS 5-8 that demonstrate the likelihood of a serious accident remains very low. OPG continues to invest to further enhance safety at its nuclear facilities as demonstrated by OPG's post Fukushima actions and the implementation of Emergency Mitigating Equipment (EME).

1.1.2 Reliability Performance

OPG has comprehensive programs in place that ensure the condition of Structures, Systems and Components important to safety at Pickering NGS is well understood, and that actions are effectively taken to continually improve plant condition. This is evident in the five year trend for Forced Loss Rate (Figure 1), an internationally recognized indicator of plant reliability and recognized by the CNSC[1] as an important indicator for program effectiveness.

For Pickering NGS, operational reliability has improved significantly over the years, with two of Pickering's units having record operational runs of 632 days for Unit 5 and 622 days for Unit 1. This, combined with the best Forced Loss Rate performance in site history at approximately 3% and 4% in 2015 and 2016, demonstrates the effectiveness of OPG programs and commitment to continuous improvement.

Figure 1: Pickering NGS Generation Forced Loss Rate (%)



1.1.3 Condition of Major Components

OPG has in place well established programs and processes that meet or exceed applicable regulatory requirements for ensuring that the physical condition of SSCs remain fit-for-service.

The current regulatory requirement is that major components, including fuel channels, be inspected during unit planned outages. The inspection results are compared against acceptance criteria defined in the appropriate CSA Standard and when required by the CSA standard, OPG submits fitness-for service assessments for CNSC acceptance to support return of the unit to service. In addition, inspection

[1] Section 12 of CNSC Regulatory Document 3.1.1, *"Reporting Requirements for Nuclear Power Plants*", a low station Forced Loss Rate reflects the effectiveness of stations programs and practices in maintaining systems available for safe electrical generation.

results are reported to CNSC following unit restart in accordance with the reporting criteria defined in the appropriate CSA Standard.

In keeping with OPG's "safety first principle", OPG will not operate a unit unless there is high confidence that all SSCs will remain fit-for-service for the operating cycle (next planned outage).

1.2 Pickering NGS PSR2

In support of licence renewal and continued commercial operation of Pickering NGS to the end of 2024, a Periodic Safety Review (PSR) has been completed in accordance with CNSC REGDOC-2.3.3 that was a comprehensive assessment of the Pickering NGS design and operation.

The objective of the PSR, referred to as PSR2 as it builds on the previous assessments was to confirm that the design, operation and safety-significant SSCs support continued safe operation of Pickering NGS to the end of 2024.

The PSR2 was conducted thoroughly by senior industry experts with different teams independently executing each of the four major phases:

1. **PSR2 Basis Document:** A PSR2 Basis Document defining the scope of the PSR2 process, and documenting the conduct of the PSR2 was prepared by OPG and accepted by the CNSC in July 2016.

2. **Safety Factor Reports:** Prepared by AMEC Foster-Wheeler and Tetra Tech, fifteen Safety Factor and two Complementary Reviews identifying compliances and gaps were submitted for CNSC review between July 2016 and March 2017.

3. **Global Assessment Report (GAR):** Prepared by Candesco, a division of Kinectrics Inc., that consolidated gaps based on the findings from the Safety Factor Reports and Complementary Reviews into Global Issues, prioritized Global Issues, developed Proposed Resolution Plans (which form the scope of the IIP), and ranked Proposed Resolution Plans. The GAR was completed and submitted for CNSC review in October 2017.

4. **Integrated Implementation Plan (IIP):** Prepared by RCM Technologies and OPG, documents specific actions that support the Global Issue Resolution Statements, referred to as Resolution Actions and IIP Actions (this document).

All of the above PSR2 phases were subjected to a third party Expert Panel review. The Expert Panel comprised senior industry leaders familiar with station design and operation of Pickering NGS and other nuclear stations, whose purpose was to provide guidance and counsel throughout the PSR2 process.

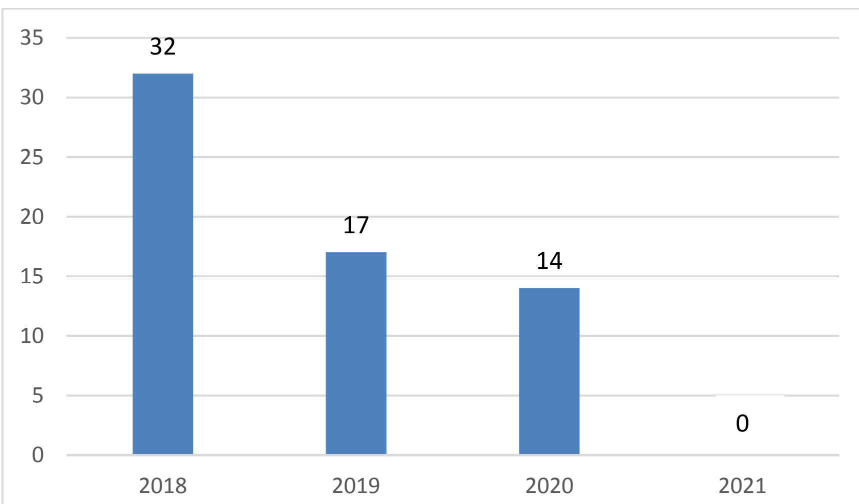
An overview of the four phases of the PSR2 process culminating in the development of the IIP Actions is summarized in section 2.0. Details of the activities carried out over the two year period on the PSR2 process are provided in Appendix D.

1.3 IIP Actions to Address Global Issues

Appendix B documents 63 IIP Actions that have been developed to address the proposed Resolution Statements for 23 Global Issues (GI) identified in phase 3 of PSR2. The 63 IIP Actions, based on current planning assumptions, have completion dates distributed over the next three years as shown in Figure 2, and baseline milestones as shown in Figure 8 in Appendix D.

Each IIP Action is scheduled for completion either through OPG's work management system for plant activities or through departmental work programs (e.g. OPG's *Integrated On-line Work Schedule*, N-PROC-MA-0022).

Figure 2: Distribution of IIP Actions and Timeline for Completing Actions



In addition to implementing programmatic improvements, this IIP contains actions for Plant Modifications, Fitness for Service, and Safety Analyses. The following key GIs are highlighted due to their significance to safety for continued operation to the end of 2024.

1.3.1 IIP Actions Involving Plant Modifications

1.3.1.1 Firewater System Enhancement (GI-48)

Canadian Standard Association (CSA) standard CSA N293-12, *Fire Protection for Nuclear Power Plants*, requires independent electrically and diesel driven firewater supply pumps. On Pickering NGS 1,4 this requirement is satisfied. However, a PSR2 gap was identified for Pickering NGS 5-8, as firewater is supplied from electrically driven pumps with redundant power supplies. To address this gap, changes to the existing Firewater system are included in the IIP scope to allow the Firewater from Pickering NGS 1,4 diesel driven Firewater pumps to supply Pickering NGS 5-8 through station interconnection. This interconnection will allow the Pickering site fire protection system to meet the safety intent of modern standards (CSA N293-12) for the redundancy and diversity of firewater supply.

1.3.1.2 Beyond Design Basis Accident (BDBA) Accident Management (GI-40)

OPG's response to the Fukushima accident included completion of:

1. Phase 1 Emergency Mitigation Equipment (EME) to prevent severe accident progression following a sustained station loss of power.

2. Phase 2 EME to restore critical containment functions.

These EME modifications greatly enhance safety at Pickering NGS; they were explicitly designed to mitigate the consequences of a sustained station loss of power event. The Phase 2 EME modifications included in Appendix B are intended to further enhance EME and BDBA coverage.

In addition to the EME modifications, OPG is updating the assessment (P-REP-09013-00002-R001, 2014-01-27, *Pickering NGS Beyond Design Basis Containment Integrity*) to cover operation to the end of 2024. This reassessment will include post-BDBA containment controlled filtered venting.

Lastly, OPG will upgrade the EME Phase 2 modifications to include the necessary power and support service connections to restore functionality of the Main Volume Vacuum Pump (MVVP).

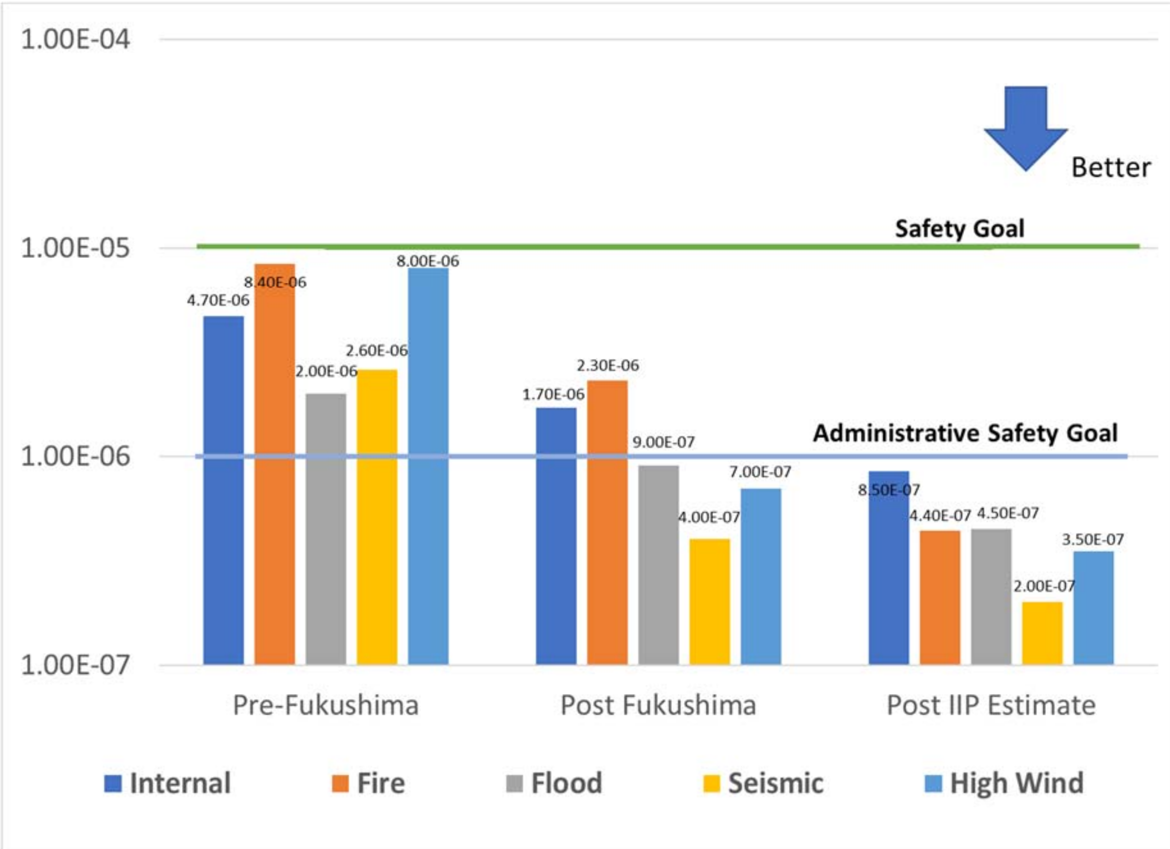
1.3.1.3 Pickering NGS 1,4 Probabilistic Safety Assessment (GI-27)

Even though Pickering PSA Safety Goals are met, OPG has set more challenging expectations through an Administrative Safety Goal. To meet the more challenging goals, OPG has implemented Fukushima lessons learned that have enhanced plant safety. Following Fukushima lessons learned action implementation, Pickering NGS 5-8 meets the Administrative Safety Goals in all areas.

Pickering NGS Units 1,4 PSA Large Release Frequency is already better than the Safety Goal. To ensure Pickering NGS 1,4 also meet the more challenging Administrative Safety Goal, IIP Actions have been established to install piping modifications to provide make-up water to Unit 1 and Unit 4 Calandria, Heat Transport System and Steam Generators to ensure continuous post-BDBA fuel cooling and protection of containment.

Following the completion of these enhancements as per the IIP Actions, Pickering NGS 1,4 PSA estimated LRF will be better than (i.e. lower than) the Administrative Safety Goal — further improving on already implemented Fukushima lessons learned actions, as shown in Figure 3.

Figure 3: Pickering NGS 1,4 PSA Large Release Frequency Improvements Following GI-27 IIP Action Completion



1.3.2 IIP Actions for Fitness for Service

Of the 23 Global Issues, there are three Global Issues of interest with regards to Fitness for Service, summarized below.

1.3.2.1 Completion/Updating of the Condition Assessments (GI-08 and GI-43)

The condition of the plant has been reviewed through Condition Assessments (CA), which ensures that appropriate maintenance, testing and monitoring are ongoing at

Pickering NGS. OPG continues to invest in the plant and to perform periodic component inspections to ensure that Pickering meets or exceeds industry standards.

Over a half million components covering all plant SSCs were reviewed through a defined process. Findings and recommendations were documented in over one thousand Condition Assessment (CA) reports. The CA recommendations cover operation beyond the expected commercial operation period to the end of 2024.

The goal of Global Issues GI-08 and GI-43 is to confirm the completeness of Pickering Condition Assessments to support commercial operation to the end of 2024.

The relevant IIP Actions that relate to Condition Assessments include:

1. Develop and implement into OPG governance a risk-based approach for aging management of critical passive components including piping systems, cables and civil structures.

2. Update Condition Assessment reports for various components including non-containment civil structures per the OPG's Aging Management Process (N-PROC-MP-0060).

3. Develop and implement a database for tracking and reporting on the implementation of CA actions in support of extended operation.

1.3.2.2 Fitness for Service for Fuel Channels (GI-01)

The goal of GI-01, "Fitness for Service for Fuel Channels", is to ensure that Fuel Channels remain fit-for-service for the extended operating period to the end of 2024.

As part of OPG's licence renewal application, OPG has requested approval to operate beyond the current Commission approved limit of 247,000 Effective Full Power Hours (EFPH) for the Pickering NGS 5-8 fuel channels, to 295,000 EFPH for the lead Pickering reactor unit.

This IIP contains the following actions that address Fuel Channel Fitness-for-Service:

1. Provide Revised OPG CSA N285.8 Compliance Plan.

2. Update Pickering NGS Fuel Channel Periodic Inspection Plan (PIP) for Operation to the end of 2024.

3. Submit 2018 Fuel Channel Life Cycle Management Plan (LCMP) Update that includes Pickering NGS 1,4 Operation to the end of 2024.

4. Prepare and update, as necessary, the "Pickering NGS Fuel Channel Readiness Plan in Support of Operation to 2024" that documents and provides the status of the work required in support of Pickering NGS operation to the end of 2024.

Annual submission of the Fuel Channel LCMP to the CNSC will continue per existing regulatory requirements.

1.3.3 Deterministic Safety Analysis to Support Extended Operation to the End of 2024 (GI-24)

OPG is committed to continuous improvement in safety at all of its nuclear facilities and has robust, comprehensive programs in place that are aligned with industry best practices for ensuring the condition of SSCs important to safety are well understood and well maintained. OPG maintains and routinely updates the Pickering NGS safety analysis to include aging effects, as required by CNSC REGDOC-2.6.3 [R-13]. The safety analysis demonstrates that the public risk from Pickering NGS remains very low.

The goal of GI-24, “Safety Analysis to Support the Extended Operating Period”, is to demonstrate the adequacy of the safety margins of the plant covering the extended operating period.

The most recent Heat Transport System (HTS) aging safety analysis (Small Break Loss of Coolant Accident (SBLOCA) and Loss of Flow (LOF)) are valid to January 31, 2019 for Pickering NGS 1,4 and June 20, 2019 for Pickering NGS 5-8. Neutron Overpower (NOP) safety analysis is also included in the scope of GI-24.

This IIP contains actions to perform and submit updated safety analysis per CNSC REGDOC-2.4.1, to support continued safe operation for extended operation.

1.4 Pickering PSR2 Conclusions

The PSR process has been thoroughly conducted over a two-year period by industry experts. The process has identified plant modifications that will enhance safety and reliability, and has highlighted where additional work is required to support commercial operation to the end of 2024. The PSR reviews confirmed that there are no management program gaps.

This IIP document has produced a baseline schedule of activities which are presented in Appendix B, and which will be controlled through regular OPG monitoring and reporting. Any changes to the baseline schedule (shown in Figure 8 in Appendix D) will be managed through the IIP Administration Instruction [R-14]. With a robust design, established mature programs in place that meet or exceed industry standards, and a leadership team that is committed to safety and continuous improvement, Pickering NGS will continue to operate safely and reliably through 2024.

2.0 PERIODIC SAFETY REVIEW (PSR2) SCOPE

The Pickering NGS PSR2 was conducted by senior industry experts, building on the review basis of earlier OPG PSR work (PSR1) and other associated assessments, consisting of:

- The Pickering B Integrated Safety Review (ISR), completed in 2009 and performed in support of the proposed refurbishment and continued operation (at that time planned for an additional 30 years) of Pickering NGS Units 5-8.
- Pickering NGS 1,4 integrated safety assessments performed during the Pickering A Return to Service (PARTS) work (circa 2000), in support of approval to restart Units 1 and 4.
- The relevant programmatic aspects of the Darlington ISR completed in October 2011 in support of refurbishment and continued operation of the Darlington units (programmatic parts are applicable to Pickering where programs and practices are common for the OPG fleet).

In accordance with CNSC REGDOC-2.3.3, Periodic Safety Reviews [R-9], the elements of PSR2 consisted of the following four phases submitted to the CNSC:

1. Preparation of a Basis Document [R-8].
2. Conduct of the Safety Factor (SF) reviews and identification of Compliances and Gaps.
3. Consideration in the Global Assessment process of the five levels of defence-in-depth, identified strengths of Pickering NGS design, operations and performance, analysis of Gaps, and identification of potential safety enhancements for Pickering NGS.
4. Preparation of an Integrated Implementation Plan (IIP) for the implementation of safety enhancements (this document).

An overview of the PSR2 process is shown in Figure 4. The PSR2 timeline is shown in Figure 9 in Appendix D. The activities carried out during the four phases over the two year period on the PSR2 process are detailed in Appendix D.

Figure 4: Overview of Pickering NGS PSR2

Pickering PSR2 Basis Document

- Prepared in accordance with CNSC REGDOC-2.3.3
- Documents how PSR2 was to be conducted
- CNSC Accepted

↓

Safety Factor & Complementary Reviews

- 15 Safety Factor and 74 Code & Standard reviews completed in accordance with CNSC REGDOC-2.3.3 resulted in 93 Gaps identified.
- 2 Complementary Reviews (COP & FAI Reassessments) completed which resulted in 26 Gaps identified.
- 23 Type III CNSC Additional Gaps and 1 Expert Panel Gap identified.

143 gaps identified for Global Assessment

↓

Global Assessment

- 143 gaps consolidated into 51 Global Issues (GIs)
- 117 proposed resolution plans developed for all 51 GIs
- 23 of the 51 GIs resulted in 35 proposed resolution plans with actions and followed the prioritization process
- 82 proposed resolution plans did not require progression to the IIP (22 NFAs, 35 ADs, 25 XRFs)
- 24 Strengths identified
- Defence-in-Depth reviews completed
- Ranking of Resolution Statements performed

23 GIs having 35 Resolution Plans required follow-up action in the IIP

↓

Integrated Implementation Plan

- 23 Global Issues carried into the IIP
- 35 Resolution Actions developed
- Completion and Success Criteria developed for Resolution Actions
- 63 IIP actions with completion criteria developed with Target Completion Dates
- IIP administrative and change control process developed

↓

IIP Actions Scheduled

- IIP activities scheduled for completion align with safety significance and unit planned outage schedule
- Tracking of IIP actions to completion and submission of closure requests for CNSC acceptance
- Quarterly and annual reporting on IIP progress

A summary of the Safety Factor reviews and Global Assessment are detailed below in Sections 2.1 and 2.2.

2.1 Safety Factor Review

Following the PSR2 Basis Document phase which defined the approach for how PSR2 was to be conducted, the second phase of the PSR2 process involved completion of 15 Safety Factor reviews, which covered all aspects important to the safety of Pickering NGS.

The results of the Safety Factor reviews were documented in 15 Safety Factor Reports. The Safety Factor Reports addressed the Review Tasks derived from IAEA SSG-25 [R-10] and from CNSC REGDOC-2.3.3 [R-9], and documented the results of the assessments of Pickering NGS with respect to applicable modern Laws, Regulations, Codes and Standards. Other complementary assessments, such as reassessment of the Pickering B ISR Continued Operations Plan (COP) actions and the Fukushima Action Items (FAIs) were also completed in the Safety Factor review phase.

The Safety Factor Reports and complementary assessments concluded that there are no fundamental safety issues at Pickering NGS, and that OPG has effective programs and processes in place to support continued safe operation for the extended operating period to the end of 2024.

The Safety Factor Reports identified compliances and gaps with respect to the review elements in the PSR2 assessment basis. Gaps identified in the Safety Factor Reports and complementary assessments, were then consolidated into Global Issues in the Pickering NGS Global Assessment during phase three of PSR2.

2.2 Global Assessment

The third phase of the PSR2 process, the Global Assessment, was the review of the findings of the Safety Factor reports, the Pickering B ISR COP and FAI complementary assessments, and Expert Panel reviews, to provide an overall assessment of the safety of the station, and confirm its safe operation for the extended operating period to the end of 2024. The Global Assessment included consideration of the identified design, operational and performance strengths, the five levels of defence-in-depth, the enhancements identified through proposed resolution plans, in order to make a conclusion on the overall acceptability of operation of the plant over the extended operating period to the end of 2024.

The Global Assessment concluded that the current plant design, operation, processes and management system will ensure continued safe operation of Pickering NGS Units 1, 4 and Pickering NGS Units 5-8 both in the short term, and for extended operation to the end of 2024.

The results of the Global Assessment are documented in the Pickering NGS Global Assessment Report [R-1].

The Global Assessment considered Safety Factor and complementary assessment gaps and compliances, identified Pickering NGS Strengths, and Defence-in-Depth Assessment conclusions. These factors were considered in aggregate to identify Global Issues and Resolution Plans, from which the IIP scope and schedule was derived.

2.2.1 Global Assessment Pickering NGS Strengths

As part of the Global Assessment, Strengths in Pickering NGS design, operations and performance were identified.

REGDOC-2.3.3 defines Strengths as current practices that are “equivalent to or better than those established in modern codes and standards, practices”. The Safety Factor Compliances (and groups of Compliances) were taken into the Global Assessment for consideration as Strengths.

The review to identify Strengths considered the following sources:

- Safety Factor Reports
- Codes and Standards Assessments
- Complementary Assessments (COP and FAIs)
- Independent Third Party Assessments (CNSC’s Regulatory Oversight Report for 2015 [R-12] and assessments/reviews by international organizations)

The methodology and the list of Strengths were reviewed by the Pickering NGS Global Assessment Expert Panel with extensive knowledge of the Pickering NGS PSR2 project and design/operation of Pickering NGS.

A total of 24 Strengths were identified for Pickering NGS, which are detailed in the Pickering NGS Global Assessment Report [R-1]. The Pickering NGS Strengths were used in the Defence-in-Depth Assessment to demonstrate the extent to which the safety requirements of defence in depth are fulfilled, as required by CNSC REGDOC-2.3.3.

Furthermore, OPG Pickering NGS Senior Leadership is committed to investing in the plant, and focusing the organization to strive for continued improvement in the plant condition, operation and performance.

2.2.2 Defence-in-Depth Assessment

As part of the Global Assessment, a Defence-in-Depth assessment was performed which supported extended operation at Pickering NGS by demonstrating the extent to which the safety requirements of defence-in-depth are fulfilled at Pickering NGS. The overall assessment was an important element in supporting the proposed enhancement plans and the planned operational strategy over the period of PSR2.

The following five levels of defence, listed below are defined in IAEA INSAG-10 [R-2], Defence in Depth in Nuclear Safety:

Level 1: Prevention of abnormal operation and failures

Level 2: Control of abnormal operation and detection of failures

Level 3: Control of accidents within the design basis

Level 4: Control of severe plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents

Level 5: Mitigation of radiological consequences of significant releases of radioactive materials

The Defence-in-Depth assessment considered the overall plant, as well as the identified strengths, acceptable deviations, and the proposed resolutions to the Global Issues listed in the Global Assessment.

The defence-in-depth concept applied to the Global Assessment was consistent with IAEA INSAG-10, Defence in Depth in Nuclear Safety [R-2]. The assessment used elements of the process described in IAEA SRS-46, Assessment of Defence in Depth for Nuclear Power Plants [R-3].

It was confirmed that the applicable safety principles from IAEA SRS-46 [R-3] for the concept of defence-in-depth was applied at the Pickering NGS design stage and throughout its operation over a period of several decades. At the design stage, the focus was on the first three levels of defence-in-depth: prevention of operation outside normal operating conditions, control of abnormal conditions, and provision of safety systems to effectively mitigate Design Basis Accidents (DBA). The capability of station systems and processes for responding to emergencies to mitigate the consequences of BDBAs, including severe accidents, was considered for defence-in-depth Levels 4 and 5.

The Defence-in-Depth assessment confirmed that at Pickering NGS, effective Level 1 barriers are ensured through the original conservative design, supplemented by design enhancements implemented since initial operation, comprehensive operating and maintenance programs in place, and ongoing continuous improvements based on national and international OPEX. Given the focus and priority placed on equipment reliability to address the findings in the areas of the equipment condition, this level of defence will continue to be strong and effective for Pickering NGS.

The assessment of defence-in-depth Level 2 confirmed that the provisions in place at Pickering NGS are mature and robust for detecting changes from normal operating conditions by the Reactor Regulating System, plant process control systems and the Special Safety Systems. The control systems in Pickering NGS maintain the reactor operating conditions within the normal operating range, and effectively respond to anticipated transients to avoid the need for safety system action.

The assessment of defence-in-depth Level 3 confirmed that effective provisions for the control of accidents within the design basis are provided at Pickering NGS. Operators have indications and alarms as well as the capability to perform actions from the Main Control Room for this purpose. The review confirmed that Pickering NGS has strong Level 3 barriers due to the high quality of the design, which is supported by a robust set of safety analyses, and the improvement initiatives to enhance equipment reliability.

The assessment of defence-in-depth Level 4 confirmed that Pickering NGS has additional design features and effective procedural provisions in place. Pickering NGS Units 1,4 and 5-8 have complementary design features for BDBAs. Operating Manuals and Abnormal Incident Manuals include Emergency Mitigating Equipment Guidelines to prevent accident progression. Severe Accident Management Guidelines for mitigating accident progression in the very unlikely event of a BDBA have been implemented. Furthermore, a mature emergency response infrastructure is in place, and the requisite qualified staffing and expertise are maintained.

The assessment of defence-in-depth Level 5 confirmed that the coordinated emergency response capability of the various response organizations and the implementation of OPEX from the Fukushima event supports the Level 5 defence-in-depth provisions. Implementation of the planned Fukushima improvement initiatives will further enhance the barriers for Level 5 at Pickering NGS.

The Defence-in-Depth assessment concluded that Pickering NGS Units 1,4 and 5-8 design and operation have effective barriers in all levels of defence-in-depth and that significant enhancements have been implemented since the plant was put into service.

2.2.3 Global Issues

OPG identified 120 gaps from various sources (e.g. Safety Factor Reports, Complementary Assessments and Expert Panel review) that were evaluated in the Global Assessment. The gaps were consolidated and grouped based on topical similarities, as Global Issues. This consolidation facilitated the analysis of any interfaces between Safety Factors and the aggregate impact of Global Issues. Each Global Issue was prioritized with respect to nuclear safety, and assigned a corresponding Safety Significance Level.

Subsequent to the conduct of the initial Global Assessment process, 23 Additional Gaps (AG) were identified from the CNSC staff review of the Safety Factor Reports and complementary reviews. In total through the PSR, 143 Gaps were incorporated into the Global Assessment resulting in 51 Global Issues.

None of the Global Issues identify a safety concern that requires additional planned or urgent action to be taken.

2.2.4 Development of Proposed Resolution Plans

Proposed resolution plans were developed to address the 51 Global Issues (GI) with consideration of safety benefit and practicability. The proposed Resolution Plans for each of the 51 Global Issues consisted of the following Resolution types:

- **Resolution Statements (RS):** An activity intended to address the Global Issue. There were 35 Resolution Statements covering 23 Global Issues (some GIs had more than one RS)

- **No Further Action (NFA):** An activity which had already been completed or had actions already underway outside of PSR2 to address the related Global Issue or where information had been found that addressed the Global Issue have been categorized as requiring No Further Action (NFA) within PSR2. 35 proposed resolution plans were categorized as NFA during the Global Assessment.

- **Acceptable Deviation (AD):** An activity for which it was determined that the proposed resolution had Low/Very Low Safety Significance or that practicable resolution(s) were not readily evident. There were 22 proposed resolution plans categorized as AD during the Global assessment.

- **Cross-Reference (XRF):** An activity that was covered by another resolution as Cross-Reference (XRF). 25 proposed resolution plans were categorized as XRF during the Global Assessment.

The Global Assessment process resulted in 23 Global Issues that have 35 Resolution Statements with defined actions, some of which reflect existing work programs and plans at the station that are already in progress.

All Global Issue Resolution Statements were ranked per the Global Assessment normalized ranking, from 1 to 35 in order of decreasing importance, as shown in Appendix A.

The 35 Resolution Actions, and the supporting 63 IIP Actions which define the scope of the IIP, are scheduled with ranking considered, and managed through a structured IIP Administration Instruction [R-14].

3.0 INTEGRATED IMPLEMENTATION PLAN

The fourth and final phase of the PSR2 process is the Integrated Implementation Plan (IIP) that defines Resolution Actions to address Global Issues. Each Resolution Action is completed through the execution of one or more IIP Actions.

Appendices A, B and C in this document define the Resolution Actions and their supporting IIP Actions. Such actions will be closed when CNSC is satisfied that their corresponding success criteria are met.

OPG has established a schedule to manage the completion of the Resolution Actions, and their supporting IIP Actions, with baseline target completion dates, as illustrated in Appendix D Figure 8.

Within OPG, accountability for the successful execution of the IIP has been assigned at the appropriate level, ensuring the commitment and engagement of the organization.

Furthermore, OPG has established IIP change control process and reporting arrangements – as part of OPG's Regulatory Management Governance framework – under a new instruction document, P-INS-03680-00001 [R-14].

3.1 Resolution Action and IIP Action Identification

The Resolution Actions and their associated IIP Actions are assigned unique identifiers which trace their origin and classification within the PSR2 process.

	Global Issue	Resolution Statement	Safety and Control Area	IIP Resolution Action	IIP Assignment
Resolution Action:	**G04**	**-RS2**	**-06**	**-08**	
IIP Action:	G04	-RS2	-06	-08	.1
	G04	-RS2	-06	-08	.2

For example: "G04-RS2-06-08" identifies the eighth Resolution Action ("-08") of the IIP associated with the second Global Assessment Resolution Statement ("-RS2-") related to the fourth Global Issue ("G04-") associated with CNSC Safety and Control Area six ("-06"). ".1" identifies the first IIP Action related to Resolution Action "G04-RS2-06-08". The IIP Actions are listed by SCA in Appendix C of this document.

The IIP Resolution Actions were transitioned into OPG's Asset Suite Action Tracking as Regulatory Commitment (REGC) Action Requests (ARs), and the IIP Actions were transitioned into Action Tracking as Regulatory Management (REGM) "assignments" associated with the corresponding REGC ARs. Within OPG's Action Tracking, the Resolution Actions are identified by specific AR numbers.

3.2 Completion and Success Criteria

Resolution Actions and their supporting IIP Actions have been developed by Senior Industry Experts and OPG Subject Matter Experts and were provided specific and measurable Completion Criteria.

The Completion Criteria establish the verifiable evidence that needs to be provided to consider a specific action as having fulfilled its intent as developed through the PSR2 process. The Completion Criteria may include measures such as completed and documented analysis, system inspections, or installed modifications.

The Resolution Actions were also provided specific and measurable Success Criteria. The Success Criteria establish the verifiable evidence that needs to be provided to consider the Resolution Action as having satisfied its purpose as developed through the PSR2 process. The Success Criteria may include measures such as submitted aggregate analysis results, system inspection results, or confirmation that installed modifications are Available for Service (AFS).

4.0 CONCLUSION

The Pickering Nuclear Generating Station (PNGS) has been operated safely by Ontario Power Generation (OPG) for over 40 years. Investments in plant condition, driven by strong reliability programs and Leadership have enhanced plant safety and reliability.

The opportunity to extend the plant life was recognised early in 2015. Over a two year period, the first three steps of a PSR were conducted as per CNSC REGDOC-2.3.3. The Global Assessment Report identified the work scope that supports continued safe and reliable commercial operation to the end of 2024.

The scope of the IIP comprises 63 IIP Actions that have been converted to a baseline schedule (Appendix B), with milestones shown in Appendix D Figure 8. In developing the IIP, the PSR2 process did not eliminate any potential nuclear safety enhancements based on cost alone.

The management infrastructure and processes are in place [R-14] to ensure that IIP schedule progress is monitored and reported, risks are identified early and mitigated as appropriate, and changes are controlled by approved procedure.

OPG is committed to continuous safety enhancement at its nuclear facilities and has robust comprehensive programs in place aligned with industry best practices. The PSR identified no safety issues for continued safe operation of Pickering NGS through 2024, and the actions within this IIP will further enhance safety.

5.0 DEFINITIONS AND ACRONYMS

ACU	Air Conditioning Unit
AD	Acceptable Deviation
AG	Additional Gap
AIFB	Auxiliary Irradiated Fuel Bay
ALARA	As Low As Reasonably Achievable
AR	Action Request
BDBA	Beyond Design Basis Accident
CANDU	Canada Deuterium Uranium
CA	Condition Assessment
CCA	Component Condition Assessment
CHR	Component Health Report
CNSC	Canadian Nuclear Safety Commission
CME	Common Mode Event
COP	Continued Operations Plan
CSA	Canadian Standards Association
CSI	CANDU Safety Issue
CT	Calandria Tube
DBA	Design Basis Accident
EBWS	Emergency Boiler Water System
ECI	Emergency Coolant Injection
EQ	Environmental Qualification
EQA	Environmental Qualification Assessment
EME	Emergency Mitigating Equipment
FAI	Fukushima Action Items
FFS	Fit For Service
GAR	Global Assessment Review
GI	Global Issue
HTS	Heat Transport System
IAEA	International Atomic Energy Agency
IFB	Irradiated Fuel Bay
IIP	Integrated Implementation Plan

INSAG	International Nuclear Safety Advisory Group
ISR	Integrated Safety Review
LBLOCA	Large Break Loss of Coolant Accident
LISS	Liquid Injection Shutdown System
LCH	License Condition Handbook
LCMP	Life Cycle Management Plan
LOCA	Loss of Coolant Accident
LOF	Loss of Flow
NFA	No Further Action
NGS	Nuclear Generating Station
NOP	Neutron Overpower Protection
OPEX	Operating Experience
OPG	Ontario Power Generation
PARTS	Pickering A Return to Service
PCA	Probabilistic Core Assessment
PHT	Primary Heat Transport
PIP	Periodic Inspection Program
PLBB	Probabilistic Leak-Before Break
PNGS	Pickering Nuclear Generating Station
PM	Preventative Maintenance
PROL	Power Reactor Operating License
PSA	Probabilistic Safety Assessment
PRD	Pressure Relief Duct
PSR	Periodic Safety Review
PSR2	Periodic Safety Review 2
RA	Resolution Action
RB	Reactor Building
R&D	Research and Development
RP	Radiation Protection
RS	Resolution Statement
SHR	System Health Report
SBLOCA	Small Break Loss of Coolant Accident

SCA	CNSC Safety Control Area
SCR	Station Condition Record
SES	Site Electrical System
SF	Safety Factor
SFR	Safety Factor Report
SOE	Safe Operating Envelope
SRS	Safety Reports Series
SSC	Structures, Systems and Components
SSG	Specific Safety Guide
VB	Vacuum Building
XRF	Cross-reference

6.0 REFERENCES

[R-1] OPG Report, *Pickering NGS Global Assessment Report*, P-REP-03680-00032-R001, February 08, 2018.

[R-2] IAEA Report, *Defence in Depth in Nuclear Safety*, INSAG-10, June 1996.

[R-3] IAEA Report, *Assessment of Defence in Depth for Nuclear Power Plants*, Safety Report Series No. 46, February 2005.

[R-4] OPG Program, *Equipment Reliability*, N-PROG-MA-0026-R002, June 4, 2015.

[R-5] OPG Program, *Integrated Aging Management*, N-PROG-MP-0008-R006B, May 2, 2016.

[R-6] OPG Program, *Consolidated Nuclear Emergency Plan*, N-PROG-RA-0001-R015, December 22, 2016.

[R-7] OPG Standard, *Beyond Design Basis Accident Management*, N-STD-MP-0019-R002, July 13, 2016.

[R-8] OPG Report, *Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document*, P-REP-03680-00001-R002, June 2016.

[R-9] CNSC Regulatory Document, *Periodic Safety Reviews*, REGDOC-2.3.3, April 2015.

[R-10] IAEA Specific Safety Guide, *Periodic Safety Review of Nuclear Power Plants*, SSG-25, March 2013.

[R-11] OPG Procedure, *Communications with the Canadian Nuclear Safety Commission*, N-PROC-RA-0047

[R-12] CNSC Report, *Regulatory Oversight Report for Canadian Nuclear Power Plants: 2015*, October 2016.

[R-13] CNSC Regulatory Document, *Fitness for Service: Aging Management*, REGDOC-2.6.3, March 2014.

[R-14] OPG Instruction, *Pickering Integrated Implementation Plan Administration*, P-INS-03680-00001.

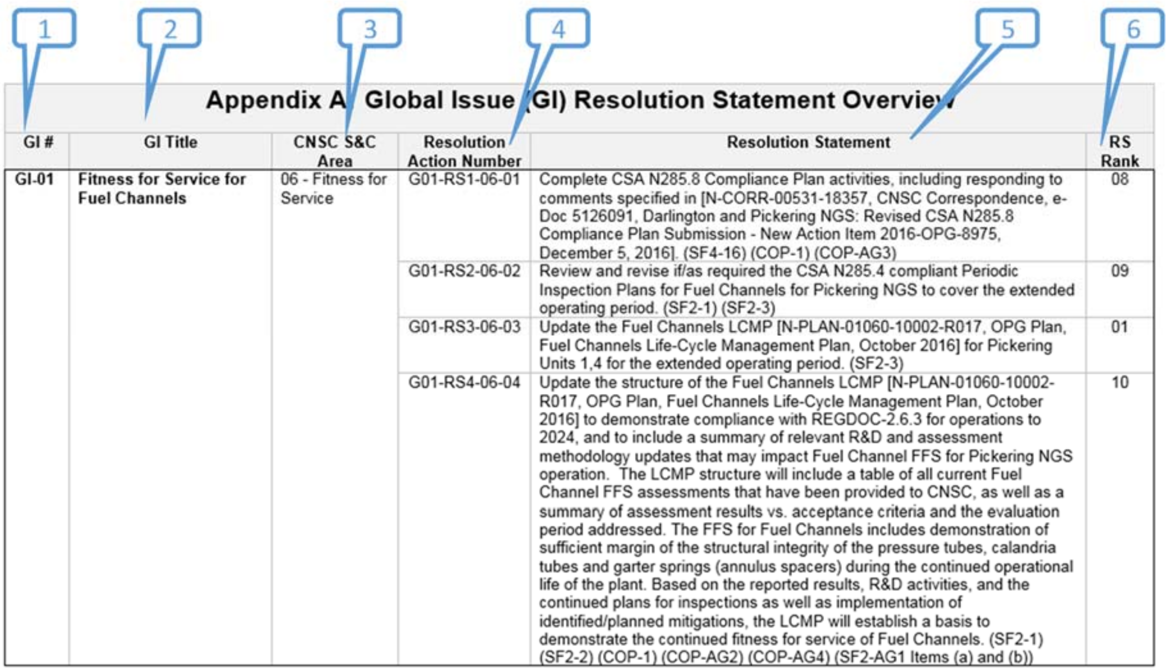
[R-15] OPG Procedure, *Action Item Management*, N-PROC-AS-0019.

[R-16] CNSC Letter, M. Santini to B. McGee, *Pickering NGS: CNSC Staff Assessment of 2014 COP, SOP, and CALs*, June 18, 2015, CD P-CORR-00531-04493, e-Doc 4782433.

[R-17] OPG Letter, R. Lockwood to G. Frappier, *End of Commercial Operation of Pickering NGS*, June 28, 2017, CD P-CORR-00531-04930.

Appendix A: Global Issue (GI) Resolution Statement Overview

Resolution Statements for Global Issues identified in the Global Assessment Report are listed in Appendix A. Resolution Statements are proposed action statements to address a Global Issue, and are each given a unique Resolution Action numerical identifier for management through [R-14]. IIP Actions (Appendix B) have been produced as sub-actions of Resolution Actions, the completion of which, will complete the associated Resolution Action.



Appendix A Global Issue (GI) Resolution Statement Overview

GI #	GI Title	CNSC S&C Area	Resolution Action Number	Resolution Statement	RS Rank
GI-01	Fitness for Service for Fuel Channels	06 - Fitness for Service	G01-RS1-06-01	Complete CSA N285.8 Compliance Plan activities, including responding to comments specified in [N-CORR-00531-18357, CNSC Correspondence, e-Doc 5126091, Darlington and Pickering NGS: Revised CSA N285.8 Compliance Plan Submission - New Action Item 2016-OPG-8975, December 5, 2016]. (SF4-16) (COP-1) (COP-AG3)	08
			G01-RS2-06-02	Review and revise if/as required the CSA N285.4 compliant Periodic Inspection Plans for Fuel Channels for Pickering NGS to cover the extended operating period. (SF2-1) (SF2-3)	09
			G01-RS3-06-03	Update the Fuel Channels LCMP [N-PLAN-01060-10002-R017, OPG Plan, Fuel Channels Life-Cycle Management Plan, October 2016] for Pickering Units 1,4 for the extended operating period. (SF2-3)	01
			G01-RS4-06-04	Update the structure of the Fuel Channels LCMP [N-PLAN-01060-10002-R017, OPG Plan, Fuel Channels Life-Cycle Management Plan, October 2016] to demonstrate compliance with REGDOC-2.6.3 for operations to 2024, and to include a summary of relevant R&D and assessment methodology updates that may impact Fuel Channel FFS for Pickering NGS operation. The LCMP structure will include a table of all current Fuel Channel FFS assessments that have been provided to CNSC, as well as a summary of assessment results vs. acceptance criteria and the evaluation period addressed. The FFS for Fuel Channels includes demonstration of sufficient margin of the structural integrity of the pressure tubes, calandria tubes and garter springs (annulus spacers) during the continued operational life of the plant. Based on the reported results, R&D activities, and the continued plans for inspections as well as implementation of identified/planned mitigations, the LCMP will establish a basis to demonstrate the continued fitness for service of Fuel Channels. (SF2-1) (SF2-2) (COP-1) (COP-AG2) (COP-AG4) (SF2-AG1 Items (a) and (b))	10

Content

(1) **GI #:** Global Issue (GI) number, as identified in the GAR.

(2) **GI Title:** Description of the GI.

(3) **CNSC S&C Area:** Resolution Action associated REGDOC-2.3.3 Safety and Control Area.

(4) **Resolution Action Number:** Unique numerical Resolution Action reference number used to manage Resolution Actions.

(5) **Resolution Statement:** The Resolution Statement for the associated GI as defined in the Global Assessment Report. The associated Safety Factor Report and Complementary Assessment gaps are listed in parentheses following each statement.

(6) **RS Rank:** Resolution Statement (RS) ranking as identified in the Global Assessment Report, ranked in order of the priority to address the GI, based on the magnitude and timeliness of the benefit to be achieved by its resolution.

Appendix A: Global Issue (GI) Resolution Statement Overview

GI #	GI Title	CNSC S&C Area	Resolution Action Number	Resolution Statement	RS Rank
GI-01	**Fitness for Service for Fuel Channels**	06 - Fitness for Service	G01-RS1-06-01	Complete CSA N285.8 Compliance Plan activities, including responding to comments specified in [N-CORR-00531-18357, CNSC Correspondence, e-Doc 5126091, Darlington and Pickering NGS: Revised CSA N285.8 Compliance Plan Submission - New Action Item 2016-OPG-8975, December 5, 2016]. (SF4-16) (COP-1) (COP-AG3)	08
			G01-RS2-06-02	Review and revise if/as required the CSA N285.4 compliant Periodic Inspection Plans for Fuel Channels for Pickering NGS to cover the extended operating period. (SF2-1) (SF2-3)	09
			G01-RS3-06-03	Update the Fuel Channels LCMP [N-PLAN-01060-10002-R017, OPG Plan, Fuel Channels Life-Cycle Management Plan, October 2016] for Pickering Units 1,4 for the extended operating period. (SF2-3)	01
			G01-RS4-06-04	Update the structure of the Fuel Channels LCMP [N-PLAN-01060-10002-R017, OPG Plan, Fuel Channels Life-Cycle Management Plan, October 2016] to demonstrate compliance with REGDOC-2.6.3 for operations to 2024, and to include a summary of relevant R&D and assessment methodology updates that may impact Fuel Channel FFS for Pickering NGS operation. The LCMP structure will include a table of all current Fuel Channel FFS assessments that have been provided to CNSC, as well as a summary of assessment results vs. acceptance criteria and the evaluation period addressed. The FFS for Fuel Channels includes demonstration of sufficient margin of the structural integrity of the pressure tubes, calandria tubes and garter springs (annulus spacers) during the continued operational life of the plant. Based on the reported results, R&D activities, and the continued plans for inspections as well as implementation of identified/planned mitigations, the LCMP will establish a basis to demonstrate the continued fitness for service of Fuel Channels. (SF2-1) (SF2-2) (COP-1) (COP-AG2) (COP-AG4) (SF2-AG1 Items (a) and (b))	10

Appendix A: Global Issue (GI) Resolution Statement Overview

GI #	GI Title	CNSC S&C Area	Resolution Action Number	Resolution Statement	RS Rank
GI-02	**Fitness for Service for Feeders**	06 - Fitness for Service	G02-RS1-06-05	Update the Feeders LCMP [N-PLAN-01060-10001-R018, OPG Plan, Feeders Life Cycle Management Plan, October 2016], based on updated Fitness for Service assessment for Pickering Units 1,4 for the extended operating period. The LCMP update is to support continued demonstration that predicted Feeder condition, with identified and planned mitigations, is acceptable for the intended operation, including any potential impact of Fuel Channel elongation on Feeder integrity. The Feeders LCMP update is to include a planned Feeder replacement plan/schedule to address the extended operating period. (SF2-4) (SF2-5) (COP-8)	02
GI-03	**Fitness for Service for Steam Generators**	06 - Fitness for Service	G03-RS1-06-06	Update the Steam Generators LCMP [N-PLAN-33110-10009-R007, OPG Plan, Steam Generators Life Cycle Management Plan, October 2016], based on updated Fitness for Service assessment for Pickering Units 1,4 for the extended operating period. The LCMP update is to support continued demonstration that predicted Steam Generator condition, with identified and planned mitigations, is acceptable for the intended operation. (SF2-6) (SF2-7) (COP-9)	03
GI-04	**Fitness for Service for Reactor Components and Structures**	06 - Fitness for Service	G04-RS1-06-07	Update the Reactor Components and Structures LCMP [N-PLAN-01060-10003-R014, OPG Plan, Reactor Components and Structures Life Cycle Management Plan, October 2016], based on updated Fitness for Service assessment and an updated Technical Basis Document [N-PLAN-01060-10008 R00, Reactor Components & Structures Life Cycle Management Plan: Technical Basis Document, 2010] for Pickering Units 1,4 for the extended operating period. The LCMP update is to support continued demonstration that predicted Reactor Components and Structures condition, with identified and planned mitigations, is acceptable for the intended operation. (SF2-8) (SF2-9) (COP-10) (COP-11) (COP-12) (COP-13) (COP-26)	04
			G04-RS2-06-08	Perform measurements, as required, of CT-LISS nozzle gaps on Units 5-8 to refine the gap closure rates. Using this new measurement data, update analyses as required, to demonstrate Fitness for Service. Implement mitigation strategies if CT-LISS nozzle contact is predicted within the extended operating period. (SF2-8) (COP-2) (COP-11) (SF2-AG1 Item (c)).	05

Appendix A: Global Issue (GI) Resolution Statement Overview

GI #	GI Title	CNSC S&C Area	Resolution Action Number	Resolution Statement	RS Rank
GI-05	**Completeness of Class 1 Piping / Components Service Limits Assessment (Excluding Major Components)**	06 - Fitness for Service	G05-RS1-06-09	Confirm the adequacy of the service limits assessments for Nuclear Class 1 Piping after accounting for impact of environmental factors (for example: irradiation, temperature, humidity). Note – This Resolution Statement does not address Major Components. (SF2-10)	11
GI-06	**Impact of the Revised Criticality Coding on the Cable Surveillance Program**	06 - Fitness for Service	G06-RS1-06-10	Reassess the impact of the changes in the cable Criticality Coding and update the scope of the cable surveillance plan accordingly. (SF2-13)	24
GI-07	**Pickering Buried Piping Fitness for the Extended Operating Period**	06 - Fitness for Service	G07-RS1-06-11	Update the Buried Piping Program asset management plan [N-PLAN-04916-10002 R003, Buried Piping Program Asset Management Plan, January 2017] and risk ranking [P-MAN-04916-00001-R002, Pickering Strategy Manual for Selection of Systems and Components for Inspection – Buried Piping, March 2014] for the extended operating period. (SF2-14)	25
			G07-RS2-06-12	Update governance to reflect a graded approach in the event that leakage in fuel oil piping occurs. This graded approach recognizes the nuclear safety importance of the systems being supplied with fuel oil, and would allow these systems to be temporarily repaired while awaiting further corrective action, allowing the systems to remain in service. This will involve document revision of Buried Piping Program Requirements [N-PROC-MA-0088-R003, Buried Piping Program Requirements, April 7, 2015]. (SF1-35)	35
GI-08	**Completion / Updating of the Condition Assessments**	06 - Fitness for Service	G08-RS1-06-13	Complete and update Condition Assessments (CA) for the piping systems and commodity groups in PSR2 scope for station operation for the extended operating period. Resulting recommendations will be assessed and included, as appropriate, in the CA action plans in the System and Component Health Reports. OPG is actively progressing with this work. (SF2-12) (SF2-15) (SF2-AG8 (b))	14
			G08-RS2-06-14	Develop and implement a process to track and report aging-management-related actions from the Condition Assessment recommendations. (SF2-12) (SF2-15) (SF2-AG4)	15

Appendix A: Global Issue (GI) Resolution Statement Overview

GI #	GI Title	CNSC S&C Area	Resolution Action Number	Resolution Statement	RS Rank
GI-09	**Seismic Capacity of the Conveyor Tube and Fuel Basket Stacking Arrangement**	05 – Plant Design	G09-RS1-05-15	Complete the required assessment to support the current fuel basket stacking arrangements in the Pickering IFBs. This seismic related issue was noted in the response to Fukushima Action Item FAI 2.1.2. Additional investigation is required to support the current spent fuel basket stacking arrangements in the Pickering IFBs. (SF2-17)	21
GI-10	**IFB Condition**	06 - Fitness for Service	G10-RS1-06-16	Complete the Pickering 5-8 IFB Leakage Mitigation Project [P-CORR-00531-04865, OPG Correspondence, Status Update: Pickering B Irradiated Fuel Bay Leak Mitigation Project #13-40703, Action Item 2014-48-5386, November 17, 2016] to mitigate leaks from IFB-B to the interspace. (SF2-16)	16
GI-12	**Extending the Environmental Qualification of Equipment**	05 – Physical Design	G12-RS1-05-17	Complete EQA re-assessments to support the extended operating period. (SF3-1)	17
GI-19	**FFS of Containment for the Extended Operating Period**	06 - Fitness for Service	G19-RS1-06-18	Demonstrate the FFS of the foundation steel H-piles for the Pickering A Reactor Building, Vacuum Building, and Pressure Relief Duct at the Pickering site for the extended operating period, as specified in [P-CORR-00531-04896, Pickering NGS: Continued Operations Plan (COP) Actions F06 and I15-6B - Periodic Safety Review Reassessment for Operation Beyond 2020, January 23, 2017] and in [P-CORR-00531-04973, Pickering NGS: CNSC Staff Review of OPG's Reassessment of COP Actions for Consideration in the PSR2, February 24, 2017]. (COP-25)	18
GI-24	**Safety Analysis to Support the Extended Operating Period**	04 - Safety Analysis	G24-RS1-04-19	Update Heat Transport System aging safety analysis models and perform the required safety analysis of the events most impacted by aging (SBLOCA, LOF and Neutron Overpower (NOP)) to support extended operation as per the existing practices [N-CORR-00531-18427, OPG Correspondence, Progress Report on OPG Heat Transport System Aging Safety Analysis, February 24, 2017]. (SF5-1) (COP-AG1)	06

Appendix A: Global Issue (GI) Resolution Statement Overview

GI #	GI Title	CNSC S&C Area	Resolution Action Number	Resolution Statement	RS Rank
GI-25	**Category 3 CANDU Safety Issues**	04 - Safety Analysis	G25-RS1-04-20	Complete the re-categorization of the Large Break LOCA (LBLOCA) CANDU Safety Issues to Category 2. OPG submitted an update to CNSC staff on the resolution of the LBLOCA issue [N-CORR-00531-18022, OPG Correspondence, Resolution of Large Break LOCA (LBLOCA) Safety Analysis Margin Issue, April 25, 2016]. An OPG update on the status of CSIs and their resolution is submitted to the CNSC annually, the latest being [N-CORR-00531-18052, Progress Update On Category 3 CANDU Safety Issues Implementation of Risk Control Measures, June 15, 2016]. Given the recent progress by industry in addressing the findings of CNSC staff reviews, it is expected that the remaining Category 3 CSIs will be re-categorized to Category 2 in 2017. OPG is actively progressing this work. (SF5-2) (COP-20)	34
			G25-RS2-04-21	Complete the re-categorization of CANDU Safety Issue CSI-IH6 for Pickering to Category 2. Complete the assessment of the layout of high-energy piping and Safety-Related Systems inside of the Reactor Buildings of Pickering Units 1 and 4 as per [P-REP-04960-00001 R002, OPG Report, Methodology of High-Energy Line Break Assessment for Piping Inside the Pickering Reactor Buildings, June 14, 2016]. For Pickering Units 5-8, the assessment is complete [N-CORR-00531-18052, OPG Correspondence, Progress Update on Category 3 CANDU Safety Issues – Implementation of Risk Control Measures, June 15, 2016] and a request for re-categorisation has been made [N-CORR-00531-18288, OPG Correspondence, Re-Categorization Request for CANDU Safety Issue IH6 for Pickering NGS 5-8 and Status for Pickering NGS 1-4, December 5, 2016]. For Pickering 1,4, a re-categorization request is planned for June 2018 [N-CORR-00531-18618, OPG Correspondence, Progress Update on Category 3 CANDU Safety Issues – Implementation of Risk Control Measures, June 23, 2017]. OPG is actively progressing this work. (SF5-2) (SF7-1) (SF1-9)	22
GI-26	**Emergency Response Projection Software**	10 - Emergency Management and Fire Protection	G26-RS1-10-22	Complete the emergency response projection enhancements identified in OPG Correspondence [N-CORR-00531-18136, Status Update for Action Item 2016-OPG-7469: Implementation of Emergency Response Projection Computer Code Upgrades, July 22, 2016], which are currently underway. (SF13-2)	20

Appendix A: Global Issue (GI) Resolution Statement Overview

GI #	GI Title	CNSC S&C Area	Resolution Action Number	Resolution Statement	RS Rank
GI-27	**Pickering 1,4 Probabilistic Safety Assessment**	04 - Safety Analysis	G27-RS1-04-23	Complete actions from PSA improvement Plan [P-CORR-00531-04946, OPG Correspondence, Pickering NGS: Risk Improvement Plan Update, February 28, 2017]. (SF6-1) (SF6-2)	23
			G27-RS2-04-24	Investigate and implement additional practicable design, operational and/or analytical enhancements to further improve Pickering 1,4 Severe Core Damage Frequency and Large Release Frequency (e.g., alternative emergency cooling water makeup). (SF6-1) (SF6-2) (FAI-AG1) (SF6-AG2) (SF6-AG3) (SF1-AG16)	07
GI-31	**Deterministic Safety Analysis**	04 - Safety Analysis	G31-RS1-04-25	Complete the Pickering NGS Implementation Plan for REGDOC-2.4.1 [N-PLAN-03500-0500515 R003, REGDOC-2.4.1 Implementation Plan, May 25, 2015]. The Implementation Plan at Pickering NGS was summarized in the PROL Amendment request as follows: “In alignment with current Pickering licensing requirements, and with the graded approach permitted by REGDOC-2.4.1 requirements, OPG will be upgrading the Pickering safety reports only to the extent that a new appendix will be included to address the development and analysis of common mode events in 2017. The analysis of common mode events represents the single largest gap in the Pickering Safety Reports with respect to REGDOC-2.4.1.” OPG is progressing this activity. (SF5-3)	30
			G31-RS2-04-26	Prepare Implementation Plan update for REGDOC-2.4.1 including consideration of the impact of the extended operating period. (SF5-4) (COP-21)	31
GI-32	**Implementation of REGDOC-2.4.2 PSA Requirements**	04 - Safety Analysis	G32-RS1-04-27	Complete the activities in the REGDOC-2.4.2 Implementation Strategy, as identified in Section 5.1, Safety Analysis Program, of [P-CORR-00531-04886, CNSC Correspondence, e-Doc 5121102, Pickering NGS: Licence Conditions Handbook, LCH-PNGS-R005, November 10, 2016] and update the Strategy in the context of the additional operating period. OPG is progressing this activity in support of extended operation at Pickering NGS. (SF6-4)	32

Appendix A: Global Issue (GI) Resolution Statement Overview

GI #	GI Title	CNSC S&C Area	Resolution Action Number	Resolution Statement	RS Rank
GI-40	**Accident Management**	10 - Emergency Management and Fire Protection	G40-RS1-10-28	Complete the planned Phase 2 EME implementation. This includes supplying cooling water, and power to essential loads via EME generators, to allow for operation of Air Cooling Units (ACUs) and Hydrogen Igniters [P-CORR-00531-04945, OPG Correspondence, Pickering NGS – CNSC Action Item 2016-48-7470 Status Update on Emergency Mitigating Equipment and Telecommunications Projects, February 16, 2017]. OPG is actively progressing this work in support of extended operation at Pickering NGS. (SF1-33) (SF1-AG4)	12
GI-43	**Safety-Related Structures (Non-Containment) for Nuclear Power Plants**	06 - Fitness for Service	G43-RS1-06-29	Perform the scope of inspections for non-Containment safety-significant civil structures as per the established Preventive Maintenance program (PM 00121151). (SF1-21) (SF1-22) (SF2-11) (SF4-13 Action #31)	19
			G43-RS2-06-30	Develop program governance using a risk based approach for aging management of safety-significant civil structures for the extended operating period. This applies to non-Containment Safety-Related Civil Structures (SF1-21) (SF1-22) (SF2-11) (SF4-13 Action #31)	28
			G43-RS3-06-31	Prepare Condition Assessments as appropriate for safety-significant civil structures for the extended operating period. Recommendations from these Condition Assessments will be tracked and reported along with those related to GI-8. This applies to non-Containment Safety-Related Civil Structures. (SF1-21) (SF1-22) (SF2-11) (SF4-13 Action #31)	26
GI-47	**Compliance With Fire Protection Code NFPA 24**	05 – Plant Design	G47-RS1-05-32	To resolve deviation #13301 from NK30-REP-71400-10001 R001 [NK30-REP-71400-10001 R001, OPG Report, Fire Protection Code Compliance Review Pickering Nuclear Generating Station B, November 23, 2010] the following work orders need to be completed to install wrenches and locks on the 058 Yard Fire Protection System: WO 3259862, 3259894, 3259893 (SF1-23)	27
GI-48	**CSA N293-12 Fire Protection of Nuclear Power Plants**	05 – Plant Design	G48-RS1-05-33	Provide, as necessary, design and/or operational changes and commissioning/testing to facilitate required interconnection of Pickering 1,4 and Pickering 5-8 Fire Protection System water supplies to meet the safety intent of CSA N293-12 Clause 7.3.2.2 (d). (SF1-5)	13

Appendix A: Global Issue (GI) Resolution Statement Overview

GI #	GI Title	CNSC S&C Area	Resolution Action Number	Resolution Statement	RS Rank
GI-50	**N285.4 PIP / Documentation Revision**	06 – Fitness for Service	G50-RS1-06-34	Revise the CSA N285.4 PIPs and governance to align with elements of N285.4-14, including making reference to CSA N285.4-14, addressing erosion and corrosion inspection requirements, reflecting extended life inspection schedules, and addressing assessment of the prior non-conforming state when dispositioning inspection results. (SF4-3) (SF4-5) (SF4-6) (SF4-7) (SF4-8)	33
			G50-RS2-06-35	Assess the impact of extended operation on concessions against CSA N285.4. (SF2-AG10)	29

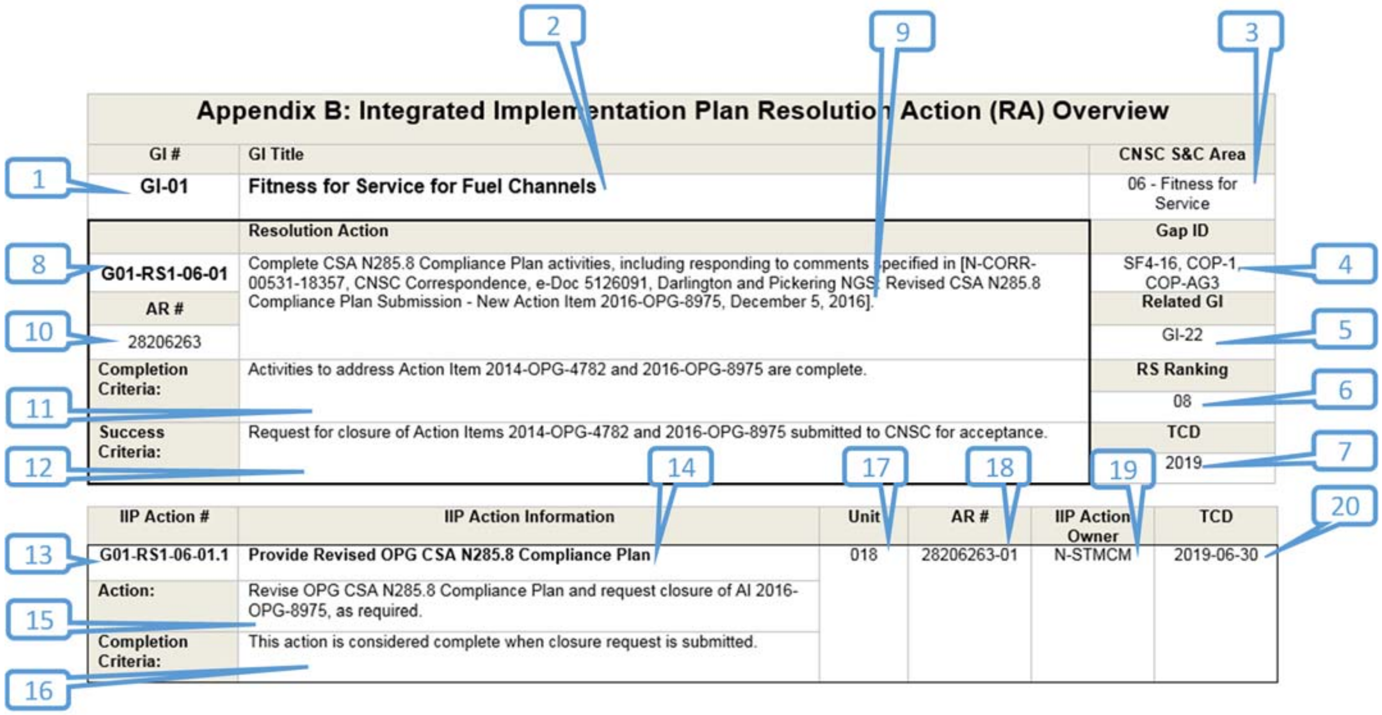
Appendix B: Integrated Implementation Plan Resolution Action Overview

Resolution Actions and their supporting IIP Actions are listed in Appendix B. The table is divided into three boxed areas: Global Issue information, Resolution Action, and IIP Action(s), each of which is described in detail below.

Global Issue information is listed at the top of the first table, to correlate the GI Resolution Statement to the associated gap(s), and to link any cross-referenced GIs. The information is used to facilitate understanding of the connection between the Safety Factor Reports, the Global Assessment Report, and the IIP.

The Resolution Action box is bolded to highlight the Resolution Action that will address the GI Resolution Statement. The information within the box identifies a unique numerical identifier and Action Request (AR) number used to manage the Resolution Action, and defines the Resolution Action, and the Resolution Action completion and success criteria. The Resolution Action is supported by IIP Actions.

The IIP Actions which support the Resolution Action are listed beneath the Resolution Action box. Each IIP Action is given a unique number, as well as an AR number to be tracked through [R-14]. The ranked IIP Actions and completion criteria are defined, as well as the associated Pickering NGS unit, IIP Action Owner, and IIP Action target completion date.



Content

Global Issue (GI) Information

(1) **GI #:** Global Issue (GI) number, as identified in the Global Assessment Report.

(2) **GI Title:** Description of the GI.

(3) **CNSC S&C Area:** Resolution Action associated REGDOC-2.3.3 Safety and Control Area.

(4) **Gap ID:** Safety Factor Report and Complementary Review gaps associated with the GI.

(5) **Related GI:** Cross-referenced (XRF) GIs which will be addressed by the Resolution Action.

(6) **RS Ranking:** Resolution Statement (RS) ranking as identified in the Global Assessment Report, ranked in order of the priority to address the RS, based on the magnitude and timeliness of the benefit to be achieved by its resolution.

(7) **TCD:** Resolution Action target completion date, by year.

Resolution Action Information

(8) **Resolution Action Number:** Unique numerical Resolution Action reference number used to manage Resolution Actions.

(9) **Resolution Action:** Global Issue Resolution Statement as defined in the Global Assessment Report.

(10) **AR #:** Action Request number generated in Asset Suite used to track the Resolution Action to completion through OPG's Action Item Management process.

(11) **Completion Criteria:** Specific criteria for the Resolution Action to be approved as complete by the IIP Manager.

(12) **Success Criteria:** Specific criteria for the Resolution Action to be accepted as closed by the CNSC.

IIP Action Information

(13) **IIP Action #:** Unique numerical IIP Action tracking reference, as a sub-action of the Resolution Action number, used to manage the IIP Action.

(14) **IIP Action Title:** Brief description of the IIP Action that will support completion of the Resolution Action.

(15) **IIP Action Description:** Detailed description of the IIP Action that will support completion of the Resolution Action.

(16) **IIP Action Completion Criteria:** Specific criteria for the IIP Action to be approved as complete by the IIP Action Owner.

(17) **Unit:** Pickering NGS unit(s) that the IIP Action is applicable to. 014 represents Pickering NGS Units 1,4, 058 represents Pickering NGS Units 5-8, and 018 represents common systems of Pickering NGS.

(18) **AR #:** Action Request number generated in Asset Suite used to track the IIP Action to completion through OPG's Action Item Management process.

(19) **IIP Action Owner:** IIP Action Owner responsible OPG department.

(20) **TCD:** IIP Action target completion date.

Appendix B: Integrated Implementation Plan Resolution Action (RA) Overview

GI #	GI Title	CNSC S&C Area
GI-01	**Fitness for Service for Fuel Channels**	06 - Fitness for Service
	Resolution Action	**Gap ID**
G01-RS1-06-01	Complete CSA N285.8 Compliance Plan activities, including responding to comments specified in [N-CORR-00531-18357, CNSC Correspondence, e-Doc 5126091, Darlington and Pickering NGS: Revised CSA N285.8 Compliance Plan Submission - New Action Item 2016-OPG-8975, December 5, 2016].	SF4-16, COP-1, COP-AG3
AR #		**Related GI**
28206263		GI-22
Completion Criteria:	Activities to address Action Item 2014-OPG-4782 and 2016-OPG-8975 are complete.	**RS Ranking** 08
Success Criteria:	Request for closure of Action Items 2014-OPG-4782 and 2016-OPG-8975 submitted to CNSC for acceptance.	**TCD** 2019

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G01-RS1-06-01.1	**Provide Revised OPG CSA N285.8 Compliance Plan**	018	28206263-01	N-STMCM	2019-06-30
Action:	Revise OPG CSA N285.8 Compliance Plan and request closure of AI 2016-OPG-8975, as required.				
Completion Criteria:	This action is considered complete when closure request is submitted.				

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G01-RS1-06-01.2	**Define Nominal Cooldown Transient for use in Probabilistic Leak-Before-Break (PLBB) Analyses**	018	28206263-02	N-STMCM	2018-06-30
Action:	Propose Nominal Cooldown Transient for use in Probabilistic Leak-Before-Break (PLBB)				
Completion Criteria:	OPG to submit closure request for the relevant condition associated with Action Item 2014-OPG-4782 to CNSC.				
G01-RS1-06-01.3	**Provide to CNSC documented evidence that validation of PLBB code is complete**	018	28206263-03	N-STMCM	2018-03-31
Action:	OPG to document evidence that validation of PLBB code is complete.				
Completion Criteria:	OPG to submit closure request for the relevant condition associated with Action Item 2014-OPG-4782 to CNSC.				
G01-RS1-06-01.4	**Address the Probabilistic Core Assessment (PCA) Flaw Removal Issue**	018	28206263-04	N-STMCM	2018-06-30
Action:	OPG to document resolution to address the Probabilistic Core Assessment Flaw Removal issue.				
Completion Criteria:	OPG to submit closure request of the Probabilistic Core Assessment Flaw Removal issue.				

Appendix B: Integrated Implementation Plan Resolution Action (RA) Overview

GI #	GI Title	CNSC S&C Area
GI-01	**Fitness for Service for Fuel Channels**	06 - Fitness for Service
	Resolution Action	**Gap ID**
G01-RS2-06-02	Review and revise if/as required the CSA N285.4 compliant Periodic Inspection Plans for Fuel Channels for Pickering NGS to cover the extended operating period.	SF2-1, SF2-3
AR #		**Related GI**
28206265		N/A
Completion Criteria:	Pickering Fuel Channel Periodic Inspection Plan updated to support operation to the end of 2024.	**RS Ranking** 09
Success Criteria:	The updated Fuel Channel PIPs for Pickering NGS to support operations to the end of 2024 submitted to CNSC for acceptance.	**TCD** 2018

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G01-RS2-06-02.1	**Update Pickering NGS Fuel Channel Periodic Inspection Plan (PIP) for Operation to the end of 2024.**	018	28206265-01	N-STMCM	2018-12-31
Action:	Update the Fuel Channels Periodic Inspection Plans in support of Pickering NGS operation to the end of 2024.				
Completion Criteria:	This action is considered complete when the updated Periodic Inspection Plans has been submitted to CNSC.				

Appendix B: Integrated Implementation Plan Resolution Action (RA) Overview

GI #	GI Title	CNSC S&C Area
GI-01	**Fitness for Service for Fuel Channels**	06 - Fitness for Service
	Resolution Action	**Gap ID**
G01-RS3-06-03	Update the Fuel Channels LCMP [N-PLAN-01060-10002-R017, OPG Plan, Fuel Channels Life-Cycle Management Plan, October 2016] for Pickering Units 1,4 for the extended operating period.	SF2-3
AR #		**Related GI**
28206266		GI-22, GI-33
Completion Criteria:	Fuel Channel Life Cycle Management Plan (LCMP) updated for Pickering NGS Units 1,4 for operation to the end of 2024.	**RS Ranking** 01
Success Criteria:	2018 Fuel Channel LCMP submitted to CNSC.	**TCD** 2018

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G01-RS3-06-03.1	**Submit 2018 Fuel Channel Life Cycle Management Plan (LCMP) Update that includes Pickering NGS U1 and U4 Operation to the end of 2024**	014	28206266-01	N-STMCM	2018-11-30
Action:	Update 2018 Fuel Channel LCMP to address operation of Pickering NGS Unit 1 and Unit 4 until the end of 2024.				
Completion Criteria:	This action is considered complete when the 2018 Fuel Channel LCMP N-PLAN-01060-10002 update has been submitted to CNSC.				

Appendix B: Integrated Implementation Plan Resolution Action (RA) Overview

GI #	GI Title	CNSC S&C Area
GI-01	**Fitness for Service for Fuel Channels**	06 - Fitness for Service
	Resolution Action	**Gap ID**
G01-RS4-06-04	Update the structure of the Fuel Channels LCMP [N-PLAN-01060-10002-R017, OPG Plan, Fuel Channels Life-Cycle Management Plan, October 2016] to demonstrate compliance with REGDOC-2.6.3 for operations to 2024, and to include a summary of relevant R&D and assessment methodology updates that may impact Fuel Channel FFS for Pickering NGS operation. The LCMP structure will include a table of all current Fuel Channel FFS assessments that have been provided to CNSC, as well as a summary of assessment results vs. acceptance criteria and the evaluation period addressed. The FFS for Fuel Channels includes demonstration of sufficient margin of the structural integrity of the pressure tubes, calandria tubes and garter springs (annulus spacers) during the continued operational life of the plant. Based on the reported results, R&D activities, and the continued plans for inspections as well as implementation of identified/planned mitigations, the LCMP will establish a basis to demonstrate the continued fitness for service of Fuel Channels.	SF2-1, SF2-2, SF2-AG1, COP-1, COP-AG2, COP-AG4
AR #		**Related GI**
28206267		N/A
Completion Criteria:	Fuel Channel Life Cycle Management Plan (LCMP) updated in support of Pickering NGS operations to the end of 2024.	**RS Ranking**
		10
Success Criteria:	Correspondence confirming that the fuel channel elements of the Major Components aging management program, which includes the fuel channel LCMP, complies with REGDOC-2.6.3 and includes all activities required to demonstrate Fuel Channel Fitness for Service in support of Pickering NGS operations to the end of 2024 is submitted to CNSC for acceptance.	**TCD**
		2020

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G01-RS4-06-04.1	**Develop Pickering NGS Fuel Channel Readiness Plan in Support of Operation to the end of 2024 (FCRP2024)**	018	28206267-01	N-STMCM	2018-03-31
Action:	Prepare and update, as necessary, the “Pickering NGS Fuel Channel Readiness Plan in Support of Operation to 2024” (P-PLAN-31100-00002) that documents and provides the status of the work required in support of Pickering NGS operation to the end of 2024. The plan shall include, but not necessarily be limited to: • A roadmap that demonstrates compliance of Pickering’s Fuel Channel program with REGDOC-2.6.3. • Relevant R&D and assessment methodology updates. • A summary of required inspections. A summary of required assessments/evaluations and mitigation strategies.				
Completion Criteria:	This action is considered complete when the “Pickering NGS Fuel Channel Readiness Plan in Support of Operation to 2024” (P-PLAN-31100-00002) has been submitted to CNSC.				
G01-RS4-06-04.2	**Update and Submit 2018 Fuel Channel Life Cycle Management Plan (LCMP) and the Pickering NGS Fuel Channel Readiness Plan in Support of Operation to the end of 2024 (FCRP2024)**	018	28206267-02	N-STMCM	2018-11-30
Action:	Incorporate the results of the Pickering NGS FCRP2024 activities into Fuel Channel assessments/evaluations and identify actions to mitigate aging effects, as required. Update the FCRP2024 and the 2018 LCMP accordingly.				
Completion Criteria:	This action is considered complete when the FCRP2024 (P-PLAN-31100-00002) update and the 2018 Fuel Channel LCMP (N-PLAN-01060-10002) update have been submitted to CNSC.				

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G01-RS4-06-04.3	**Update and Submit 2019 Fuel Channel Life Cycle Management Plan (LCMP) and the Pickering NGS Fuel Channel Readiness Plan in Support of Operation to the end of 2024 (FCRP2024)**	018	28206267-03	N-STMCM	2019-11-30
Action:	Incorporate the results of the Pickering NGS FCRP2024 activities into Fuel Channel assessments/evaluations and identify actions to mitigate aging effects, as required. Update the FCRP2024 and the 2019 LCMP accordingly.				
Completion Criteria:	This action is considered complete when the FCRP2024 (P-PLAN-31100-00002) update and the 2019 Fuel Channel LCMP (N-PLAN-01060-10002) update have been submitted to CNSC.				
G01-RS4-06-04.4	**Update and Submit 2020 Fuel Channel Life Cycle Management Plan (LCMP) and the Pickering NGS Fuel Channel Readiness Plan in Support of Operation to the end of 2024 (FCRP2024)**	018	28206267-04	N-STMCM	2020-11-30
Action:	Incorporate the results of the Pickering NGS FCRP2024 activities into Fuel Channel assessments/evaluations and identify actions to mitigate aging effects, as required. Update the FCRP2024 and the 2020 LCMP accordingly.				
Completion Criteria:	This action is considered complete when the FCRP2024 (P-PLAN-31100-00002) update and the 2020 Fuel Channel LCMP (N-PLAN-01060-10002) update have been submitted to CNSC.				
G01-RS4-06-04.5	**Submit Confirmatory Fuel Channels Fitness for Service Correspondence**	018	28206267-05	N-STMCM	2020-11-30
Action:	Prepare correspondence confirming that the 2020 Fuel Channel LCMP complies with REGDOC-2.6.3 and includes all activities required to demonstrate Fuel Channel Fitness for Service in of support Pickering NGS operations to the end of 2024.				
Completion Criteria:	This action is considered complete when the correspondence has been submitted to CNSC.				

Appendix B: Integrated Implementation Plan Resolution Action (RA) Overview

GI #	GI Title	CNSC S&C Area
GI-02	**Fitness for Service for Feeders**	06 – Fitness for Service
	Resolution Action	**Gap ID**
G02-RS1-06-05	Update the Feeders LCMP [N-PLAN-01060-10001-R018, OPG Plan, Feeders Life Cycle Management Plan, October 2016], based on updated Fitness for Service assessment for Pickering Units 1,4 for the extended operating period. The LCMP update is to support continued demonstration that predicted Feeder condition, with identified and planned mitigations, is acceptable for the intended operation, including any potential impact of Fuel Channel elongation on Feeder integrity. The Feeders LCMP update is to include a planned Feeder replacement plan/schedule to address the extended operating period.	SF2-4, SF2-5, COP-8
AR #		**Related GI**
28206269		GI-22, GI-33
Completion Criteria:	Feeders Life Cycle Management Plan (LCMP) N-PLAN-01060-10001 updated for Pickering NGS Units 1, 4 operations to the end of 2024.	**RS Ranking**
		02
Success Criteria:	Updated Feeders Life Cycle Management Plan (LCMP) N-PLAN-01060-10001 submitted to CNSC for review.	**TCD**
		2018

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G02-RS1-06-05.1	**Submit 2018 Feeders Life Cycle Management Plan (LCMP) Update that includes Pickering NGS U1 and U4 operations to the end of 2024**	014	28206269-01	N-STMCM	2018-11-30
Action:	Update Feeders LCMP that addresses operation of Pickering NGS Unit 1 and Unit 4 to the end of 2024 and submit to CNSC.				
Completion Criteria:	This action is considered complete when: • Feeders LCMP N-PLAN-01060-10001 has been updated to include Pickering NGS Unit 1 and Unit 4 operation to the end of 2024. • Updated LCMP has been submitted to CNSC.				

Appendix B: Integrated Implementation Plan Resolution Action (RA) Overview

GI #	GI Title	CNSC S&C Area
GI-03	**Fitness for Service for Steam Generators**	06 – Fitness for Service
	Resolution Action	**Gap ID**
G03-RS1-06-06	Update the Steam Generators LCMP [N-PLAN-33110-10009-R007, OPG Plan, Steam Generators Life Cycle Management Plan, October 2016], based on updated Fitness for Service assessment for Pickering Units 1,4 for the extended operating period. The LCMP update is to support continued demonstration that predicted Steam Generator condition, with identified and planned mitigations, is acceptable for the intended operation.	SF2-6, SF2-7, COP-9
AR #		**Related GI**
28206270		GI-22, GI-33
Completion Criteria:	Steam Generators Life Cycle Management Plan (LCMP) N-PLAN-33110-10009 updated for Pickering NGS Units 1, 4 operations to the end of 2024.	**RS Ranking**
		03
Success Criteria:	Updated Steam Generators Life Cycle Management Plan (LCMP) N-PLAN-33110-10009 submitted to CNSC for review.	**TCD**
		2018

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G03-RS1-06-06.1	**Submit 2018 Steam Generators Life Cycle Management Plan (LCMP) update that includes Pickering NGS U1 and U4 operations to the end of 2024**	014	28206270-01	N-STMCM	2018-11-30
Action:	Update Steam Generators Life Cycle Management Plan (LCMP) that addresses operation of Pickering Unit 1 and Unit 4 to the end of 2024 and submit to CNSC.				
Completion Criteria:	This action is considered complete when: • Steam Generators LCMP N-PLAN-33100-10009 has been updated to include Pickering NGS Unit 1 and Unit 4 operation to the end of 2024. • Updated LCMP has been submitted to CNSC.				

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GI #	GI Title	CNSC S&C Area
GI-04	**Fitness for Service for Reactor Components and Structures**	06 – Fitness for Service
	Resolution Action	**Gap ID**
G04-RS1-06-07	Update the Reactor Components and Structures LCMP [N-PLAN-01060-10003-R014, OPG Plan, Reactor Components and Structures Life Cycle Management Plan, October 2016], based on updated Fitness for Service assessment and an updated Technical Basis Document [N-PLAN-01060-10008 R00, Reactor Components & Structures Life Cycle Management Plan: Technical Basis Document, 2010] for Pickering Units 1,4 for the extended operating period. The LCMP update is to support continued demonstration that predicted Reactor Components and Structures condition, with identified and planned mitigations, is acceptable for the intended operation.	SF2-8, SF2-9, COP-10, COP-11, COP-12, COP-13, COP-26
AR #		**Related GI**
28206271		GI-22, GI-33
Completion Criteria:	Reactor Components and Structures Life Cycle Management Plan (LCMP) N-PLAN-01060-10003 updated for Pickering NGS Units 1,4 operations to the end of 2024.	**RS Ranking**
		04
Success Criteria:	Updated Reactor Components and Structures Life Cycle Management Plan (LCMP) N-PLAN-01060-10003 submitted to CNSC for review.	**TCD**
		2018

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G04-RS1-06-07.1	**Submit 2018 Reactor Components Life Cycle Management Plan (LCMP) update that includes Pickering NGS U1 and U4 Operation to the end of 2024**	014	28206271-01	N-STMCM	2018-11-30
Action:	Update Reactor Components and Structures Life Cycle Management Plan (LCMP) that addresses operation of Pickering NGS Unit 1 and Unit 4 to the end of 2024 and submit to CNSC.				
Completion Criteria:	This action is considered complete when: • Reactor Components and Structures LCMP N-PLAN-01060-10003 has been updated to include Pickering NGS Unit 1 and Unit 4 operation to the end of 2024. • Updated LCMP has been submitted to CNSC.				

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GI #	GI Title	CNSC S&C Area
GI-04	**Fitness for Service for Reactor Components and Structures**	06 – Fitness for Service
	Resolution Action	**Gap ID**
G04-RS2-06-08	Perform measurements, as required, of CT-LISS nozzle gaps on Units 5-8 to refine the gap closure rates. Using this new measurement data, update analyses as required, to demonstrate Fitness for Service. Implement mitigation strategies if CT-LISS nozzle contact is predicted within the extended operating period.	SF2-8, SF2-AG1, COP-2, COP-11
AR #		**Related GI**
28206273		GI-22
Completion Criteria:	Measurement data as required and supporting analysis demonstrate adequate Calandria Tube – Liquid Injection Shutdown System (CT-LISS) nozzle gaps for operation to the end of 2024 of Pickering NGS Unit 5 and Unit 6.	**RS Ranking** 05
Success Criteria:	Post planned outage FFS report confirming that projected CT-LISS nozzle clearance precludes contact or analysis demonstrates continued operation with predicted contact is submitted to CNSC for review.	**TCD** 2020

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G04-RS2-06-08.1	**Perform CT-LISS nozzle gap measurements as required on Pickering NGS Unit 6**	6	28206273-01	N-STMCM	2018-12-31
Action:	Per LCMP (N-PLAN-01060-10003-2.3.4.1), perform Pickering NGS Unit 6 CT-LISS nozzle gap inspections and conduct FFS assessment.				
Completion Criteria:	Post planned outage FFS report has been submitted to CNSC.				

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G04-RS2-06-08.2	**Perform CT-LISS nozzle gap measurements as required on Pickering NGS Unit 5**	5	28206273-02	N-STMCM	2020-09-30
Action:	Per LCMP (N-PLAN-01060-10003-2.3.4.1), perform Pickering NGS Unit 5 CT-LISS nozzle gap inspections and conduct FFS assessment.				
Completion Criteria:	Post planned outage FFS report has been submitted to CNSC.				

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GI #	GI Title	CNSC S&C Area
GI-05	**Completeness of Class 1 Piping / Components Service Limits Assessment (Excluding Major Components)**	06 – Fitness for Service
	Resolution Action	**Gap ID**
G05-RS1-06-09	Confirm the adequacy of the service limits assessments for Nuclear Class 1 Piping after accounting for impact of environmental factors (for example: irradiation, temperature, humidity). Note – This Resolution Statement does not address Major Components.	SF2-10
AR #		**Related GI**
28206274		GI-33
Completion Criteria:	Pickering NGS formal service limits assessment including environmental factors complete.	**RS Ranking**
		11
Success Criteria:	Service limits assessments including impact of environmental factors are confirmed to be adequate and submitted to CNSC for review.	**TCD**
		2020

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G05-RS1-06-09.1	**Confirm service limits assessments for Nuclear Class 1 Piping include environmental factors**	018	28206274-01	P-PDCMM	2020-12-31
Action:	Prepare a formal report on service limits assessments based on P-CORR-33000-00001. The formal report will include an assessment of the impact of environmental factors.				
Completion Criteria:	Pickering NGS formal service limits assessment report including impact of environmental factors is complete.				

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GI #	GI Title	CNSC S&C Area
GI-06	**Impact of the Revised Criticality Coding on the Cable Surveillance Program**	06 – Fitness for Service
	Resolution Action	**Gap ID**
G06-RS1-06-10	Reassess the impact of the changes in the cable Criticality Coding and update the scope of the cable surveillance plan accordingly.	SF2-13
AR #		**Related GI**
28206275		N/A
Completion Criteria:	Assessment of changes in cable criticality coding are complete, and Cable Surveillance Plan updated to reflect changes.	**RS Ranking**
		24
Success Criteria:	The updated Cable Surveillance Plan to reflect the impacts of the changes in cable criticality coding is submitted to CNSC for review.	**TCD**
		2018

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G06-RS1-06-10.1	**Assess the impact of the changes in criticality coding on cables**	018	28206275-01	P-PECM	2018-12-31
Action:	Review changes in criticality coding on cables as a result of the Criticality Code review project performed by OPG and update the cable surveillance plan Electrical Cable Equipment Strategy Instruction [P-ESI-57100-00001] as required.				
Completion Criteria:	Electrical Cable Equipment Strategy Instruction [P-ESI-57100-00001] updated to reflect changes in criticality coding.				

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GI #	GI Title	CNSC S&C Area
GI-07	**Pickering Buried Piping Fitness for the Extended Operating Period**	06 – Fitness for Service
	Resolution Action	**Gap ID**
G07-RS1-06-11	Update the Buried Piping Program asset management plan [N-PLAN-04916-10002 R003, Buried Piping Program Asset Management Plan, January 2017] and risk ranking [P-MAN-04916-00001-R002, Pickering Strategy Manual for Selection of Systems and Components for Inspection – Buried Piping, March 2014] for the extended operating period.	SF2-14
AR #		**Related GI**
28206278		N/A
Completion Criteria:	N-PLAN-04916-10002 updated to include buried piping risks and Asset Management requirements to support commercial operation to the end of 2024.	**RS Ranking**
		25
Success Criteria:	The updated Buried Piping Asset Management Plan to support commercial operation to the end of 2024 submitted to CNSC for review.	**TCD**
		2019

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G07-RS1-06-11.1	**Update the Buried Piping Program Asset Management Plan (N-PLAN-04916-10002) and Risk Ranking document**	018	28206278-01	N-CED	2019-03-31
Action:	Update the Buried Piping Program Asset Management Plan (N-PLAN-04916-10002) and Risk Ranking document to support commercial operation to the end of 2024.				
Completion Criteria:	N-PLAN-04916-10002 updated to include buried piping Asset Management requirements to support commercial operation to the end of 2024.				

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GI #	GI Title	CNSC S&C Area
GI-07	**Pickering Buried Piping Fitness for the Extended Operating Period**	06 – Fitness for Service
	Resolution Action	**Gap ID**
G07-RS2-06-12	Update governance to reflect a graded approach in the event that leakage in fuel oil piping occurs. This graded approach recognizes the nuclear safety importance of the systems being supplied with fuel oil, and would allow these systems to be temporarily repaired while awaiting further corrective action, allowing the systems to remain in service. This will involve document revision of Buried Piping Program Requirements [N-PROC-MA-0088-R003, Buried Piping Program Requirements, April 7, 2015].	SF1-35
AR #		**Related GI**
28206283		N/A
Completion Criteria:	OPG Buried Piping Program Requirements [N-PROC-MA-0088] updated to include a graded approach to repairs of fuel oil piping.	**RS Ranking**
		35
Success Criteria:	The updated N-PROC-MA-0088 reflecting graded approach for fuel oil piping leakage mitigation submitted to CNSC for review.	**TCD**
		2020

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G07-RS2-06-12.1	**Update Buried Piping Program Requirements [N-PROC-MA-0088-R003]**	018	28206283-01	N-CED	2020-03-31
Action:	Update the Buried Piping Program Requirements [N-PROC-MA-0088-R003] to include a graded approach for repairs of fuel oil piping.				
Completion Criteria:	OPG Buried Piping Program Requirements [N-PROC-MA-0088] updated to include a graded approach for repairs of fuel oil piping.				

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GI #	GI Title	CNSC S&C Area
GI-08	**Completion / Updating of the Condition Assessments**	06 – Fitness for Service
	Resolution Action	**Gap ID**
G08-RS1-06-13	Complete and update Condition Assessments (CA) for the piping systems and commodity groups in PSR2 scope for station operation for the extended operating period. Resulting recommendations will be assessed and included, as appropriate, in the CA action plans in the System and Component Health Reports.	SF2-12, SF2-15, SF2-AG8
AR #		**Related GI**
28206279		GI-10, GI-20, GI-21, GI-22, GI-29, GI-49
Completion Criteria:	Condition Assessments complete per the Integrated Aging Management Program (N-PROG-MP-0008) and the Aging Management Process (N-PROC-MP-0060), and resulting recommendations assessed and included in the corresponding CHRs/SHRs	**RS Ranking** 14
Success Criteria:	Condition Assessments complete, consequential actions plans integrated in CHRs/SHRs, as per N-PROG-MP-0008 and N-PROC-MP-0060, and supporting correspondence submitted to CNSC for review.	**TCD** 2019

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G08-RS1-06-13.1	**Develop a risk based approach for aging management of critical piping systems**	018	28206279-01	P-AMSIM	2018-09-30
Action:	Develop a risk based approach for aging management of critical piping systems, and incorporate that approach in OPG governance.				
Completion Criteria:	Piping system risk based approach methodology established and incorporated into OPG governance.				

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G08-RS1-06-13.2	**Complete the Condition Assessments consistent with the revised Reactor Safety Criticality Codes (XRF from GI-20)**	018	28206279-02	P-AMSIM	2019-03-31
Action:	Complete Scoping and Screening per the Aging Management Process N-PROC-MP-0060.				
Completion Criteria:	Scoping and Screening complete for commodity groups in PSR2 scope.				
G08-RS1-06-13.3	**Complete Condition Assessments for the piping systems in PSR2 scope to support Pickering NGS commercial operation to the end of 2024 (GI-08 and XRF from GI-22)**	018	28206279-03	P-AMSIM	2019-06-30
Action:	Complete Condition Assessments for piping systems per the Aging Management Process N-PROC-MP-0060.				
Completion Criteria:	Condition Assessments for piping systems are complete and action plans incorporated into associated health reports.				
G08-RS1-06-13.4	**Complete Condition Assessments for commodity groups in PSR2 scope to support Pickering NGS commercial operation to the end of 2024**	018	28206279-04	P-AMSIM	2019-03-31
Action:	Complete Condition Assessments per the Aging Management Process N-PROC-MP-0060.				
Completion Criteria:	Condition Assessments are complete and action plans incorporated into associated health reports.				
G08-RS1-06-13.5	**Complete Condition Assessments for the Irradiated Fuel Bays (IFB) to support Pickering NGS commercial operation to the end of 2024. (XRF from GI-10)**	018	28206279-05	P-AMSIM	2018-06-30
Action:	Complete Condition Assessments for IFBs per the Aging Management Process N-PROC-MP-0060.				
Completion Criteria:	Condition Assessments for IFBs are complete and action plans incorporated into associated health reports.				

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G08-RS1-06-13.6	**Complete Condition Assessments for the Deaerators and the Deaerator Storage Tanks to support Pickering NGS commercial operation to the end of 2024. (XRF from GI-21)**	018	28206279-06	P-AMSIM	2018-06-30
Action:	Complete Condition Assessments for DA and DA Storage tanks per the Aging Management Process N-PROC-MP-0060.				
Completion Criteria:	Condition Assessments for DA and DA Storage tanks are complete and action plans incorporated into associated health reports				
G08-RS1-06-13.7	**Complete Condition Assessments for the Fueling Machines and FM Ball Screws to support Pickering NGS commercial operation to the end of 2024 (XRF from GI-29)**	018	28206279-07	P-AMSIM	2018-06-30
Action:	Complete Condition Assessments for FM and FM ball screws per the Aging Management Process N-PROC-MP-0060.				
Completion Criteria:	Condition Assessments for FM and FM ball screws are complete and action plans incorporated into associated health reports.				
G08-RS1-06-13.8	**Complete Condition Assessments for the Primary Heat Transport auxiliary piping system, Primary Heat Transport pump discharge valves, and boiler inlet and outlet valves to support Pickering NGS commercial operation to the end of 2024 (XRF from GI-49)**	018	28206279-08	P-AMSIM	2018-06-30
Action:	Complete Condition Assessments for PHT piping, pump discharge valves and boiler inlet & outlet valves per the Aging Management Process N-PROC-MP-0060.				
Completion Criteria:	Condition Assessments are complete and action plans incorporated into associated health reports.				

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GI #	GI Title	CNSC S&C Area
GI-08	**Completion / Updating of the Condition Assessments**	06 – Fitness for Service
	Resolution Action	**Gap ID**
G08-RS2-06-14	Develop and implement a process to track and report aging-management-related actions from the Condition Assessment recommendations.	SF2-12, SF2-15, SF2-AG4
AR #		**Related GI**
28206286		GI-10, GI-21, GI-22, GI-29, GI-43, GI-49
Completion Criteria:	Develop and implement a Condition Assessment (CA) action tracking database for tracking and reporting of CA action status.	**RS Ranking** 15
Success Criteria:	The process, including a database, to track aging-management-related actions is implemented and a correspondence is submitted to CNSC indicating that the database is available for review.	**TCD** 2018

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G08-RS2-06-14.1	**Develop and implement Condition Assessment action tracking and reporting process including a database**	018	28206286-01	P-AMSIM	2018-09-30
Action:	Develop and implement a Condition Assessment (CA) action tracking process and database for reporting of CA action status.				
Completion Criteria:	CA action tracking process and database are developed and populated with CA actions.				

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GI #	GI Title	CNSC S&C Area
GI-09	**Seismic Capacity of the Conveyor Tube and Fuel Basket Stacking Arrangement**	05 – Physical Design
	Resolution Action	**Gap ID**
G09-RS1-05-15	Complete the required assessment to support the current fuel basket stacking arrangements in the Pickering IFBs. This seismic related issue was noted in the response to Fukushima Action Item FAI 2.1.2. Additional investigation is required to support the current spent fuel basket stacking arrangements in the Pickering IFBs.	SF2-17
AR #		**Related GI**
28206287		N/A
Completion Criteria:	Pickering NGS IFB / AIFB fuel basket stacking arrangement assessment is complete.	**RS Ranking**
		21
Success Criteria:	Pickering NGS IFB / AIFB fuel basket stacking assessment confirms operational stacking limits and supporting correspondence submitted to CNSC for review.	**TCD**
		2019

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G09-RS1-05-15.1	**Complete Pickering NGS IFB / AIFB fuel basket stacking arrangement assessment**	018	28206287-01	P-PEFHM	2019-03-31
Action:	Complete and document Pickering NGS IFB / AIFB fuel basket stacking arrangement assessment.				
Completion Criteria:	Pickering NGS IFB / AIFB fuel basket stacking arrangement assessment for frame stacking complete.				

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GI #	GI Title	CNSC S&C Area
GI-10	**IFB Condition**	06 – Fitness for Service
	Resolution Action	**Gap ID**
G10-RS1-06-16	Complete the Pickering 5-8 IFB Leakage Mitigation Project [P-CORR-00531-04865, OPG Correspondence, Status Update: Pickering B Irradiated Fuel Bay Leak Mitigation Project #13-40703, Action Item 2014-48-5386, November 17, 2016] to mitigate leaks from IFB-B to the interspace.	SF2-16
AR #		**Related GI**
28206289		N/A
Completion Criteria:	Pickering NGS Units 5-8 Irradiated Fuel Bay (IFB) Leak Mitigation Project #13-40703 complete.	**RS Ranking**
		16
Success Criteria:	Pickering NGS Units 5-8 IFB leakage to Units 5-8 IFB sump has been reduced to acceptable levels, as defined by Leak Mitigation Project #13-40703, and supporting correspondence submitted to CNSC for review.	**TCD**
		2019

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G10-RS1-06-16.1	**Complete Pickering NGS Units 5-8 Irradiated Fuel Bay (IFB) Leakage Mitigation Project #13-40703**	058	28206289-01	P-PEFHM	2019-09-30
Action:	Complete Pickering NGS Units 5-8 IFB Leakage Mitigation Project #13-40703.				
Completion Criteria:	Pickering NGS Units 5-8 IFB Leak Mitigation Project has been completed.				

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GI #	GI Title	CNSC S&C Area
GI-12	**Extending the Environmental Qualification of Equipment**	05 – Physical Design
	Resolution Action	**Gap ID**
G12-RS1-05-17	Complete EQA re-assessments to support the extended operating period.	SF3-1
AR #		**Related GI**
28206291		N/A
Completion Criteria:	Environmental Qualification Assessments (EQAs) are complete to support commercial operation to the end of 2024.	**RS Ranking** 17
Success Criteria:	EQAs completed and any identified Pickering NGS Environmentally Qualified (EQ) actions documented for resolution through OPG EQ program, and supporting documentation submitted to CNSC for review.	**TCD** 2019

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G12-RS1-05-17.1	**Complete Environmental Qualification Assessments (EQA) to support Pickering NGS extended operations**	018	28206291-01	P-PEFHM	2019-12-31
Action:	Assess existing EQAs for Environmentally Qualified (EQ) life-limited components to support commercial operation to the end of 2024.				
Completion Criteria:	Assessment of EQAs has been completed and any actions are documented for OPG EQ program resolution.				

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GI #	GI Title	CNSC S&C Area
GI-19	**FFS of Containment for the Extended Operating Period**	06 – Fitness for Service
	Resolution Action	**Gap ID**
G19-RS1-06-18	Demonstrate the FFS of the foundation steel H-piles for the Pickering A Reactor Building, Vacuum Building, and Pressure Relief Duct at the Pickering site for the extended operating period, as specified in [P-CORR-00531-04896, Pickering NGS: Continued Operations Plan (COP) Actions F06 and I15-6B - Periodic Safety Review Reassessment for Operation Beyond 2020, January 23, 2017] and in [P-CORR-00531-04973, Pickering NGS: CNSC Staff Review of OPG's Reassessment of COP Actions for Consideration in the PSR2, February 24, 2017].	COP-25
AR #		**Related GI**
28206292		GI-43
Completion Criteria:	FFS of foundation H-piles for Pickering 1,4 Reactor Buildings (RB), Vacuum Building (VB) and Pressure Relief Duct (PRD) has been demonstrated to support commercial operation to the end of 2024.	**RS Ranking** 18
Success Criteria:	FFS of foundation H-piles has been demonstrated for Pickering 1,4 RBs, VB and PRD to support commercial operations to the end of 2024, and supporting correspondence submitted to CNSC for review.	**TCD** 2019

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G19-RS1-06-18.1	**Demonstrate FFS of foundation H-piles for Pickering 1,4 Reactor Buildings (RB), Vacuum Building (VB) and Pressure Relief Duct (PRD)**	018	28206292-01	P-PDCMM	2019-06-30
Action:	Per P-CORR-00531-04896, demonstrate fitness for service of the foundation steel H-piles supporting the Pickering 1,4 Reactor Buildings (RB), Vacuum Building (VB), and Pressure Relief Duct (PRD) to support commercial operation to the end of 2024.				
Completion Criteria:	FFS of foundation H-piles for Pickering 1,4 RB, VB and PRD has been demonstrated.				

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GI #	GI Title	CNSC S&C Area
GI-24	**Safety Analysis to Support the Extended Operating Period**	04 - Safety Analysis
	Resolution Action	**Gap ID**
G24-RS1-04-19	Update Heat Transport System aging safety analysis models and perform the required safety analysis of the events most impacted by aging (SBLOCA, LOF and Neutron Overpower (NOP)) to support extended operation as per the existing practices [N-CORR-00531-18427, OPG Correspondence, Progress Report on OPG Heat Transport System Aging Safety Analysis, February 24, 2017].	SF5-1, COP-AG1
AR #		**Related GI**
28206294		GI-22
Completion Criteria:	The impact of Heat Transport System (HTS) component aging on the Small-Break Loss of Coolant (SBLOCA), Loss of Flow (LOF) and Neutron Overpower (NOP) accident scenarios are assessed to demonstrate that adequate safety margins exist for Pickering NGS operations to the end of 2024. Results are documented and submitted to CNSC.	**RS Ranking** 06
Success Criteria:	Correspondence confirming that Pickering NGS safety analysis results demonstrate adequate safety margins to support commercial operations to the end of 2024 are submitted to CNSC for acceptance.	**TCD** 2019

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G24-RS1-04-19.1	**Update Heat Transport Aging safety analysis models (Pickering 1,4)**	014	28206294-01	N-SAIP	2018-08-31
Action:	Update Safety Analysis models accounting for heat transport system aging for Pickering NGS Units 1,4 operations to the end of 2024.				
Completion Criteria:	This action will be considered complete when Heat Transport Aging safety analysis model for Pickering NGS Units 1,4 has been updated and submitted to CNSC.				

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G24-RS1-04-19.2	**Complete Loss of Flow (LOF) Safety Analysis accounting for Heat Transport Aging Methodology (Pickering 1,4)**	014	28206294-02	N-SAIP	2018-11-30
Action:	Complete required Loss of Flow safety analysis for operation to the end of 2024 for Pickering NGS Units 1,4.				
Completion Criteria:	This action will be considered complete when the required Loss of Flow Safety Analysis for P014 has been submitted to CNSC, addressing the impact of Heat Transport System (HTS) component aging, and demonstrating that adequate safety margins exist for Pickering NGS Units 1,4 operations to the end of 2024.				
G24-RS1-04-19.3	**Complete Small Break Loss of Coolant Accident (SBLOCA) safety analysis accounting for Heat Transport Aging Methodology (Pickering 1,4)**	014	28206294-03	N-SAIP	2018-11-30
Action:	Complete required Small Break LOCA safety analysis for operation to the end of 2024 for Pickering NGS Units 1,4.				
Completion Criteria:	This action will be considered complete when the required SBLOCA Safety Analysis for P014 has been submitted to CNSC, addressing the impact of Heat Transport System (HTS) component aging, and demonstrating that adequate safety margins exist for Pickering NGS Units 1,4 operations to the end of 2024.				
G24-RS1-04-19.4	**Complete Neutron Overpower safety analysis accounting for Heat Transport Aging (Pickering 1,4)**	014	28206294-04	N-SAIP	2018-11-30
Action:	Complete required Neutron Overpower safety analysis for operation to the end of 2024 for Pickering NGS Units 1,4.				
Completion Criteria:	This action will be considered complete when the required NOP Safety Analysis for P014 has been submitted to CNSC, addressing the impact of Heat Transport System (HTS) component aging, and demonstrating that adequate safety margins exist for Pickering NGS Units 1,4 operations to the end of 2024.				

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G24-RS1-04-19.5	**Update Heat Transport Aging safety analysis models (Pickering 5-8)**	058	28206294-05	N-SAIP	2019-01-31
Action:	Update Safety Analysis models accounting for heat transport system aging for Pickering NGS Units 5-8 for operation to the end of 2024.				
Completion Criteria:	This action will be considered complete when Heat Transport Aging safety analysis model for Pickering NGS Units 5-8 has been updated by OPG for operations to the end of 2024.				
G24-RS1-04-19.6	**Complete Loss of Flow (LOF) Safety Analysis accounting for Heat Transport Aging Methodology (Pickering 5-8)**	058	28206294-06	N-SAIP	2019-05-30
Action:	Complete required Loss of Flow safety analysis for operation to the end of 2024 for Pickering NGS Units 5-8.				
Completion Criteria:	This action will be considered complete when the required Loss of Flow Safety Analysis for P058 has been submitted to CNSC, addressing the impact of Heat Transport System (HTS) component aging, and demonstrating that adequate safety margins exist for Pickering NGS Units 5-8 operations to the end of 2024.				
G24-RS1-04-19.7	**Complete Small Break Loss of Coolant Accident (SBLOCA) safety analysis accounting for Heat Transport Aging Methodology (Pickering 5-8)**	058	28206294-07	N-SAIP	2019-05-30
Action:	Complete required Small Break LOCA safety analysis for operation to the end of 2024 for Pickering NGS Units 5-8.				
Completion Criteria:	This action will be considered complete when the required SBLOCA Safety Analysis for P058 has been submitted to CNSC, addressing the impact of Heat Transport System (HTS) component aging, and demonstrating that adequate safety margins exist for Pickering NGS Units 5-8 operations to the end of 2024.				

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G24-RS1-04-19.8	**Complete Neutron Overpower safety analysis accounting for Heat Transport Aging (Pickering 5-8)**	058	28206294-08	N-SAIP	2019-05-30
Action:	Complete required Neutron Overpower safety analysis for operation to the end of 2024 for Pickering NGS Units 5-8.				
Completion Criteria:	This action will be considered complete when the required NOP Safety Analysis for P058 has been submitted to CNSC, addressing the impact of Heat Transport System (HTS) component aging, and demonstrating that adequate safety margins exist for Pickering NGS Units 5-8 operations to the end of 2024.				

Appendix B: Integrated Implementation Plan Resolution Action (RA) Overview

GI #	GI Title	CNSC S&C Area
GI-25	**Category 3 CANDU Safety Issues**	04 – Safety Analysis
	Resolution Action	**Gap ID**
G25-RS1-04-20	Complete the re-categorization of the Large Break LOCA (LBLOCA) CANDU Safety Issues to Category 2. OPG submitted an update to CNSC staff on the resolution of the LBLOCA issue [N-CORR-00531-18022, OPG Correspondence, Resolution of Large Break LOCA (LBLOCA) Safety Analysis Margin Issue, April 25, 2016]. An OPG update on the status of CSIs and their resolution is submitted to the CNSC annually, the latest being [N-CORR-00531-18052, Progress Update On Category 3 CANDU Safety Issues - Implementation of Risk Control Measures, June 15, 2016]. Given the recent progress by industry in addressing the findings of CNSC staff reviews, it is expected that the remaining Category 3 CSIs will be re-categorized to Category 2 in 2017.	SF5-2, COP-20
AR #		**Related GI**
28206295		N/A
Completion Criteria:	Updated LBLOCA analysis is completed per N-CORR-00531-18618 (which contains the most current status update of Category 3 CANDU Safety Issues). Analysis results submitted to CNSC as part of request to re-categorize LBLOCA issues to Category 2.	**RS Ranking** 34
Success Criteria:	Updated LBLOCA analysis submitted as part of request to re-categorize to Category 2 to CNSC for review.	**TCD** 2020

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G25-RS1-04-20.1	**Re-categorization of the Large Break Loss of Coolant Accident (LBLOCA) CANDU Safety Issues (CSI) to Category 2**	018	28206295-01	N-SAIP	2020-06-30
Action:	Per N-CORR-00531-18618 (which contains the most current status update of Category 3 CANDU Safety Issues) use a modified limit of operating envelope (LOE) safety analysis methodology to update the LBLOCA analysis and re-categorize LBLOCA CSI to Category 2.				
Completion Criteria:	Updated LBLOCA analysis has been completed and submitted to CNSC as part of request to re-categorize LBLOCA CSI to Category 2.				

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GI #	GI Title	CNSC S&C Area
GI-25	**Category 3 CANDU Safety Issues**	04 – Safety Analysis
	Resolution Action	**Gap ID**
G25-RS2-04-21	Complete the re-categorization of CANDU Safety Issue CSI-IH6 for Pickering to Category 2. Complete the assessment of the layout of high-energy piping and Safety-Related Systems inside of the Reactor Buildings of Pickering Units 1 and 4 as per [P-REP-04960-00001 R002, OPG Report, Methodology of High-Energy Line Break Assessment for Piping Inside the Pickering Reactor Buildings, June 14, 2016]. For Pickering Units 5-8, the assessment is complete [N-CORR-00531-18052, OPG Correspondence, Progress Update on Category 3 CANDU Safety Issues – Implementation of Risk Control Measures, June 15, 2016] and a request for re-categorization has been made [N-CORR-00531-18288, OPG Correspondence, Re-Categorization Request for CANDU Safety Issue IH6 for Pickering NGS 5-8 and Status for Pickering NGS 1-4, December 5, 2016]. For Pickering 1,4, a re-categorization request is planned for June 2018 [N-CORR-00531-18618, OPG Correspondence, Progress Update on Category 3 CANDU Safety Issues – Implementation of Risk Control Measures, June 23, 2017].	SF5-2, SF7-1, SF1-9
AR #		**Related GI**
28206296		N/A
Completion Criteria:	Assessment of high energy pipe-line failures is completed per N-CORR-00531-18618 (which contains the most current status update of re-categorization request for CSI IH6 for Pickering Units 1,4)	**RS Ranking**
		22
Success Criteria:	Assessment results have been submitted as part of request to re-categorize CANDU Safety Issue CSI-IH6 to Category 2 to CNSC for review.	**TCD**
		2020

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G25-RS2-04-21.1	**Re-categorize the CANDU Safety Issue CSI-IH6 to Category 2**	014	28206296-01	N-CNENSATM	2020-06-30
Action:	Per N-CORR-00531-18618 (which contains the most current status update of re-categorization request for CSI IH6 for Pickering Units 1,4) , an assessment has been completed of Pickering NGS Units 1 and 4 high energy pipe-line failures to support re-categorization of CANDU Safety Issue CSI-IH6 to Category 2.				
Completion Criteria:	Assessment of high energy pipe-line failures has been completed per N-CORR-00531-18618, and assessment results have been submitted to CNSC as part of request to re-categorize CANDU Safety Issue CSI-IH6 to Category 2.				

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GI #	GI Title	CNSC S&C Area
GI-26	**Emergency Response Projection Software**	10 - Emergency Management and Fire Protection
	Resolution Action	**Gap ID**
G26-RS1-10-22	Complete the emergency response projection enhancements identified in OPG Correspondence [N-CORR-00531-18136, Status Update for Action Item 2016-OPG-7469: Implementation of Emergency Response Projection Computer Code Upgrades, July 22, 2016], which are currently underway.	SF13-2
AR #		**Related GI**
28206297		N/A
Completion Criteria:	Emergency response projection computer tools are developed.	**RS Ranking**
		20
Success Criteria:	Emergency response projection computer tools functional testing demonstrated and implemented and supporting correspondence submitted to CNSC for review.	**TCD**
		2018

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G26-RS1-10-22.1	**Develop and implement upgrades to the computer codes used for emergency response projections**	018	28206297-01	N-CNENSATM	2018-09-30
Action:	Develop upgrades to the computer codes used for emergency response projections.				
Completion Criteria:	Emergency response projection computer tools are developed.				

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GI #	GI Title	CNSC S&C Area
GI-27	**Pickering 1,4 Probabilistic Safety Assessment**	04 - Safety Analysis
	Resolution Action:	**Gap ID**
G27-RS1-04-23	Complete actions from PSA improvement Plan [P-CORR-00531-04946, OPG Correspondence, Pickering NGS: Risk Improvement Plan Update, February 28, 2017].	SF6-1, SF6-2
AR #		**Related GI**
28206298		N/A
Completion Criteria:	Development and implementation of Phase 2 fire-model refinements identified In P-CORR-00531-04946.	**RS Ranking** 23
Success Criteria:	Correspondence demonstrating that Pickering NGS Unit 1, 4 Severe Core Damage Frequency (SCDF) and Large Release Frequency (LRF) are further improved towards meeting OPG administrative Safety Goals (as per N-PROG-RA-0016) is submitted to CNSC for review.	**TCD** 2018

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G27-RS1-04-23.1	**Complete Fire Modeling Refinements**	014	28206298-01	N-CNENSATM	2018-09-30
Action:	Complete Phase 2 fire-model refinements per P-CORR-00531-04946.				
Completion Criteria:	Phase 2 fire-model refinements completed and reported to CNSC.				

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GI #	GI Title	CNSC S&C Area
GI-27	**Pickering 1,4 Probabilistic Safety Assessment**	04 - Safety Analysis
	Resolution Action:	**Gap ID**
G27-RS2-04-24	Investigate and implement additional practicable design, operational and/or analytical enhancements to further improve Pickering 1,4 Severe Core Damage Frequency and Large Release Frequency (e.g., alternative emergency cooling water makeup).	SF6-1, SF6-2, , SF1-AG16 SF6-AG2, SF6-AG3, FAI-AG1
AR #		**Related GI**
28206300		GI-40
Completion Criteria:	Design, analysis, installation and commissioning of Pickering NGS Unit 1,4 emergency cooling capability enhancements. [Project #13-83561 “Firewater Supply to Critical Nuclear Systems”] complete.	**RS Ranking** 07
Success Criteria:	Correspondence demonstrating that Pickering NGS Unit 1,4 Severe Core Damage Frequency (SCDF) and Large Release Frequency (LRF) are further improved towards meeting OPG administrative Safety Goals (as per N-PROG-RA-0016) is submitted to CNSC for review.	**TCD** 2020

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G27-RS2-04-24.1	**Investigate additional practicable design, operational and/or analytical enhancements**	014	28206300-04	P-AMSIM	2018-12-31
Action:	Document the results of investigations into practicable design, operational and/or analytical assessments to provide additional mitigation measures to reduce SCDF and LRF towards Admin Safety Goals.				
Completion Criteria:	Investigation into practicable design, operational and/or analytical enhancements completed, documented and submitted to the CNSC.				

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G27-RS2-04-24.2	**Upgrade Pickering NGS Unit 1,4 Emergency Boiler Water System (EBWS)**	014	28206300-01	N- PMDPPM	2020-12-31
Action:	Upgrade Pickering NGS Unit 1,4 emergency cooling connections from the Pickering NGS firewater system to Pickering NGS Unit 1,4 EBWS.				
Completion Criteria:	Emergency cooling pipe connections from the Pickering NGS Firewater system to Pickering NGS Unit 1,4 EBWS designed, installed, commissioned, and Available For Service (AFS).				
G27-RS2-04-24.3	**Upgrade Pickering NGS Unit 1, 4 Emergency Cooling to Heat Transport System (HTS) Makeup**	014	28206300-02	N- PMDPPM	2020-12-31
Action:	Upgrade Pickering NGS Unit 1, 4 emergency cooling to provide additional connections from the Pickering NGS Firewater system to provide to the heat transport system.				
Completion Criteria:	Emergency cooling pipe connections from the Pickering NGS Firewater system to Pickering NGS Unit 1,4 HTS makeup designed, installed, commissioned, and Available For Service (AFS).				
G27-RS2-04-24.4	**Upgrade Pickering NGS Unit 1,4 Calandria Makeup**	014	28206300-03	N- PMDPPM	2020-12-31
Action:	Upgrade Pickering NGS Unit 1, 4 emergency cooling by providing pipe connections from the Pickering NGS Firewater system to provide to the Pickering NGS Unit 1,4 Calandria.				
Completion Criteria:	Emergency cooling pipe connections from the Pickering NGS Firewater system to Pickering NGS Unit 1,4 Calandria designed, installed, commissioned, and Available For Service (AFS).				

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GI #	GI Title	CNSC S&C Area
GI-31	**Deterministic Safety Analysis**	04 - Safety Analysis
	Resolution Action	**Gap ID**
G31-RS1-04-25	Complete the Pickering NGS Implementation Plan for REGDOC-2.4.1 [N-PLAN-03500-0500515 R003, REGDOC-2.4.1 Implementation Plan, May 25, 2015]. The Implementation Plan at Pickering NGS was summarized in the PROL Amendment request as follows: “In alignment with current Pickering licensing requirements, and with the graded approach permitted by REGDOC-2.4.1 requirements, OPG will be upgrading the Pickering safety reports only to the extent that a new appendix will be included to address the development and analysis of common mode events in 2017. The analysis of common mode events represents the single largest gap in the Pickering Safety Reports with respect to REGDOC-2.4.1.	SF5-3
AR #		**Related GI**
28206303		N/A
Completion Criteria:	Per N-CORR-00531-18239, N-CORR-00531-18078, and REGO 28189400-02, this action will be complete when analysis of common mode events (CMEs) is complete.	**RS Ranking**
		30
Success Criteria:	Pickering NGS CME analysis is submitted to CNSC for review.	**TCD**
		2018

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G31-RS1-04-25.1	**Provide Pickering NGS Safety Report Analysis of Common Mode Events**	018	28206303-01	N-SAIP	2018-03-31
Action:	Complete Pickering Units 1,4 and Units 5-8 safety analysis of common mode events.				
Completion Criteria:	Pickering Units 1,4, and Units 5-8 safety analyses of common mode events has been completed.				

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GI #	GI Title	CNSC S&C Area
GI-31	**Deterministic Safety Analysis**	04 - Safety Analysis
	Resolution Action	**Gap ID**
G31-RS2-04-26	Prepare Implementation Plan update for REGDOC-2.4.1 including consideration of the impact of the extended operating period.	SF5-4, COP-21
AR #		**Related GI**
28206305		N/A
Completion Criteria:	OPG REGDOC-2.4.1 Implementation Plan has been updated to support commercial operation to the end of 2024.	**RS Ranking** 31
Success Criteria:	Updated OPG REGDOC-2.4.1 Implementation Plan has been submitted to CNSC for review.	**TCD** 2018

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G31-RS2-04-26.1	**Update OPG REGDOC-2.4.1 Implementation Plan**	018	28206305-01	N-SAIP	2018-03-31
Action:	Update the OPG REGDOC-2.4.1 Implementation Plan to include consideration of commercial operation to the end of 2024.				
Completion Criteria:	OPG REGDOC-2.4.1 Implementation Plan has been updated to support commercial operation to the end of 2024.				

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GI #	GI Title	CNSC S&C Area
GI-32	**Implementation of REGDOC-2.4.2 PSA Requirements**	04 – Safety Analysis
	Resolution Action	**Gap ID**
G32-RS1-04-27	Complete the activities in the REGDOC-2.4.2 Implementation Strategy, as identified in Section 5.1, Safety Analysis Program, of [P-CORR-00531-04886, CNSC Correspondence, e-Doc 5121102, Pickering NGS: Licence Conditions Handbook, LCH-PNGS-R005, November 10, 2016] and update the Strategy in the context of the additional operating period.	SF6-4
AR #		**Related GI**
28206328		N/A
Completion Criteria:	REGDOC-2.4.2 Implementation Strategy per the graded approach described in P-CORR-00531-04557 is complete.	**RS Ranking**
		32
Success Criteria:	REGDOC-2.4.2 Implementation Strategy activities have been completed to support commercial operations to the end of 2024, and submitted to CNSC for review.	**TCD**
		2020

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G32-RS1-04-27.1	**Complete the REGDOC-2.4.2 Implementation Strategy**	018	28206328-01	N-CNENSATM	2020-12-31
Action:	Per P-CORR-00531-04557 complete the REGDOC-2.4.2 Implementation Strategy				
Completion Criteria:	REGDOC-2.4.2 Implementation Strategy per the graded approach described in P-CORR-00531-04557 is complete.				

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GI #	GI Title	CNSC S&C Area
GI-40	**Accident Management**	10 – Emergency Management and Fire Protection
	Resolution Action	**Gap ID**
G40-RS1-10-28	Complete the planned Phase 2 EME implementation. This includes supplying cooling water, and power to essential loads via EME generators, to allow for operation of Air Cooling Units (ACUs) and Hydrogen Igniters [P-CORR-00531-04945, OPG Correspondence, Pickering NGS – CNSC Action Item 2016-48-7470 Status Update on Emergency Mitigating Equipment and Telecommunications Projects, February 16, 2017].	SF1-33, SF1-AG4
AR #		**Related GI**
28206332		GI-27, GI-37
Completion Criteria:	Pickering NGS Emergency Mitigating Equipment (EME) Phase 2 project #13-41027 and Phase 2 EME upgrade that includes restoration of functionality of a Main Volume Vacuum Pump (MVVP) have been completed.	**RS Ranking** 12
Success Criteria:	Correspondence indicating completion of the Pickering NGS EME Phase 2 project #13-41027 and upgrades to EME Phase 2 regarding functionality of an MVVP is submitted to CNSC for review.	**TCD** 2019

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G40-RS1-10-28.1	**Complete Pickering NGS Emergency Mitigating Equipment (EME) Phase 2 project 13-41027**	018	28206332-01	N-PMDPPM	2018-06-30
Action:	Complete Pickering NGS EME Phase 2 project, which includes supplying cooling water and restoring power to essential loads via EME generators to allow for operation of Air Cooling Units (ACUs), Hydrogen Igniters, and FADs.				
Completion Criteria:	Pickering NGS EME Phase 2 modifications have been completed and declared Available For Service (AFS).				
G40-RS1-10-28.2	**Complete reassessment of Pickering NGS Beyond Design Basis Containment Integrity**	018	28206332-02	P-AMSIM	2018-12-31
Action:	Complete and document reassessment of Pickering NGS Beyond Design Basis Containment Integrity (P-REP-09013-00002-R001, 2014-01-27) for post-BDBA containment controlled filtered venting.				

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
Completion Criteria:	Reassessment of Pickering NGS Beyond Design Basis Containment Integrity (P-REP-09013-00002-R001, 2014-01-27) for post-BDBA containment controlled filtered venting — including the requirement of a Main Vacuum Volume Pump (MVVP) — completed, documented and submitted to the CNSC.				
G40-RS1-10-28.3	**Upgrade Emergency Mitigating Equipment (EME) Phase 2 to restore the functionality of a Main Vacuum Volume Pump (MVVP)**	018	28206332-03	N-PMDPPM	2019-06-30
Action:	Complete the necessary power and support service connections required to restore the functionality of an MVVP via EME Phase 2.				
Completion Criteria:	Restoration of functionality of an MVVP through Pickering NGS EME Phase 2 is complete and declared Available For Service (AFS).				

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GI #	GI Title	CNSC S&C Area
GI-43	**Safety-Related Structures (Non-Containment) for Nuclear Power Plants**	06 – Fitness for Service
	Resolution Action	**Gap ID**
G43-RS1-06-29	Perform the scope of inspections for non-Containment safety-significant civil structures as per the established Preventive Maintenance program (PM 00121151).	SF1-21, SF1-22, SF2-11, SF4-13
AR #		**Related GI**
28206336		N/A
Completion Criteria:	This action will be completed when an inspection of Pickering NGS non-Containment safety-significant civil structures is performed.	**RS Ranking** 19
Success Criteria:	This action is successful when Pickering NGS non-Containment safety-significant civil structures inspections are completed. Supporting correspondence is submitted to CNSC for review.	**TCD** 2018

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G43-RS1-06-29.1	**Complete inspections of Pickering NGS non-Containment safety-significant civil structures**	018	28206336-01	P-PDCMM	2018-06-30
Action:	Complete inspections of non-Containment safety-significant civil structures per P-CORR-20000-0608706.				
Completion Criteria:	Inspections of non-Containment safety-significant civil structures have been completed.				

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GI #	GI Title	CNSC S&C Area
GI-43	**Safety-Related Structures (Non-Containment) for Nuclear Power Plants**	06 – Fitness for Service
	Resolution Action	**Gap ID**
G43-RS2-06-30	Develop program governance using a risk based approach for aging management of safety-significant civil structures for the extended operating period. This applies to non-Containment Safety-Related Civil Structures.	SF1-21, SF1-22, SF2-11, SF4-13
AR #		**Related GI**
28206339		N/A
Completion Criteria:	A risk-based approach has been developed for aging management of non-Containment safety significant civil structures and implemented into the OPG aging management program.	**RS Ranking** 28
Success Criteria:	Correspondence demonstrating that a risk based approach has been developed/applied for aging management of non-Containment safety-significant civil structures and has been incorporated into OPG governance, for CNSC review.	**TCD** 2018

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G43-RS2-06-30.1	**Develop a risk-based approach for aging management of non-Containment safety-significant civil structures**	018	28206339-01	P-AMSIM	2018-09-30
Action:	Develop a risk based approach for aging management of non-Containment safety-significant civil structures, and incorporate that approach into OPG governance.				
Completion Criteria:	Non-Containment safety-significant civil structures risk-based approach methodology has been established and incorporated into OPG governance.				

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GI #	GI Title	CNSC S&C Area
GI-43	**Safety-Related Structures (Non-Containment) for Nuclear Power Plants**	06 – Fitness for Service
	Resolution Action	**Gap ID**
G43-RS3-06-31	Prepare Condition Assessments as appropriate for safety-significant civil structures for the extended operating period. Recommendations from these Condition Assessments will be tracked and reported along with those related to GI-8. This applies to non-Containment Safety-Related Civil Structures.	SF1-21, SF1-22, SF2-11, SF4-13
AR #		**Related GI**
28206342		GI-22
Completion Criteria:	Non-Containment safety-significant civil structures Condition Assessments (CA) have been completed per the Integrated Aging Management Program (N-PROG-MP-0008) to support commercial operation to the end of 2024.	**RS Ranking**
		26
Success Criteria:	Per N-PROG-MP-0008, non-containment safety-significant civil structures Condition Assessments have been completed, actions plans have been integrated with CHRs/SHRs, and supporting correspondence submitted to CNSC for review.	**TCD**
		2019

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G43-RS3-06-31.1	**Prepare Condition Assessments as appropriate for non-Containment safety-significant civil structures for Pickering NGS extended operation**	018	28206342-01	P-AMSIM	2019-06-30
Action:	Complete Condition Assessments for non-containment safety-significant civil structures per the Aging Management Process N-PROC-MP-0060.				
Completion Criteria:	Pickering NGS Condition Assessments for non-Containment safety-significant civil structures have been completed and action plans integrated with CHRs/SHRs.				

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GI #	GI Title	CNSC S&C Area
GI-47	**Compliance with Fire Protection Code NFPA 24**	05 – Physical Design
	Resolution Action	**Gap ID**
G47-RS1-05-32	To resolve deviation #13301 from NK30-REP-71400-10001 R001 [NK30-REP-71400-10001 R001, OPG Report, Fire Protection Code Compliance Review Pickering Nuclear Generating Station B, November 23, 2010] the following work orders need to be completed to install wrenches and locks on the 058 Yard Fire Protection System: WO 3259862, 3259894, 3259893	SF1-23
AR #		**Related GI**
28206344		N/A
Completion Criteria:	Installation of wrenches and locks on the Pickering NGS 058 Yard Fire Protection System Yard Post indicator valves completed, as per WO 3259862, 3259894, 3259893.	**RS Ranking** 27
Success Criteria:	Deviation #13301 from NK30-REP-71400-10001 R001 [NK30-REP-71400-10001 R001, OPG Report, Fire Protection Code Compliance Review Pickering Nuclear Generating Station B, November 23, 2010] has been resolved and Fire Protection code is in compliance with NFPA 1970 Section 3601.	**TCD** 2019

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G47-RS1-05-32.1	**Install wrenches and locks on the Pickering NGS 058 Yard Fire Protection System Yard Post indicator valves 058-71450-V37, 058-71450-V3027, and 058-71450-V36**	058	28206344-01	P-PECM	2019-06-30
Action:	Complete installation of wrenches and locks on the Pickering NGS 058 Yard Fire Protection System Yard Post indicator valves, as per WO 3259862, 3259894, 3259893.				
Completion Criteria:	Installation of wrenches and locks on the Pickering NGS 058 Fire Protection System Yard Post indicator, as per WO 3259862, 3259894, 3259893 complete.				

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GI #	GI Title	CNSC S&C Area
GI-48	**Compliance with CSA N293-12 Fire Protection of Nuclear Power Plants**	05 – Physical Design

	Resolution Action	Gap ID
G48-RS1-05-33	Provide, as necessary, design and/or operational changes and commissioning/testing to facilitate required interconnection of Pickering 1,4 and Pickering 5-8 Fire Protection System water supplies to meet the safety intent of CSA N293-12 Clause 7.3.2.2 (d).	SF1-5
AR #		**Related GI**
28206346		N/A
Completion Criteria:	Pickering NGS Units 5-8 Fire Protection System recommended design and/or operational changes implemented to support the intent of CSA N293-12 Clause 7.3.2.2 (d).	**RS Ranking** 13
Success Criteria:	Correspondence indicating that design and/or operational changes have been implemented to meet the safety intent of CSA N293-12 Clause 7.3.2.2 (d), is submitted to CNSC for review.	**TCD** 2020

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G48-RS1-05-33.1	**Design and/or operational changes to Pickering Units 1,4 and Pickering Units 5-8 Fire Protection System interconnection.**	058	28206346-01	N- PMDPPM	2020-12-31
Action:	Implement design and/or operational changes to interconnect Pickering Units 1,4 and Pickering Units 5-8 Fire Protection System water supplies.				
Completion Criteria:	Pickering NGS Units 5-8 Fire Protection System recommended design and/or operational changes have been implemented and Available For Service (AFS).				

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GI #	GI Title	CNSC S&C Area
GI-50	**N285.4 PIP / Documentation Revision**	06 – Fitness for Service
	Resolution Action	**Gap ID**
G50-RS1-06-34	Revise the CSA N285.4 PIPs and governance to align with elements of N285.4-14, including making reference to CSA N285.4-14, addressing erosion and corrosion inspection requirements, reflecting extended life inspection schedules, and addressing assessment of the prior non-conforming state when dispositioning inspection results.	SF4-3, SF4-5, SF4-6, SF4-7, SF4-8
AR #		**Related GI**
28206347		N/A
Completion Criteria:	Per P-CORR-00531-05087, revise the required program documents.	**RS Ranking**
		33
Success Criteria:	Revised documents for the identified elements of N285.4-14 are submitted to the CNSC for review.	**TCD**
		2020

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G50-RS1-06-34.1	**Provide CNSC with an update on OPG implementation plans for identified elements of CSA N285.4-14**	018	28206347-01	N-CED	2018-09-30
Action:	Per P-CORR-00531-05087, provide CNSC with an update on OPG implementation plans for identified elements of CSA N285.4-14.				
Completion Criteria:	Update of implementation plans for identified elements of CSA N285.4-14 provided to CNSC.				

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G50-RS1-06-34.2	**Documents identified in implementation plans updated and issued as required**	018	28206347-02	N-CED	2020-12-31
Action:	Documents identified in implementation plans updated and issued, as per P-CORR-00531-05087.				
Completion Criteria:	Required program documents revised, as per P-CORR-00531-05087.				

Appendix B: Integrated Implementation Plan Resolution Action (RA) Overview

GI #	GI Title	CNSC S&C Area
GI-50	**N285.4 PIP / Documentation Revision**	06 – Fitness for Service
	Resolution Action	**Gap ID**
G50-RS2-06-35	Assess the impact of extended operation on concessions against CSA N285.4.	SF2-AG10
AR #		**Related GI**
28206348		N/A
Completion Criteria:	Impact of N285.4 CNSC accepted program implementation concessions assessed for the extended operating period, and OPG implementation plans for identified elements for N285.4-14 updated as necessary.	**RS Ranking** 29
Success Criteria:	Update of implementation plans for identified elements of CSA N285.4-14 provided to CNSC for review.	**TCD** 2020

IIP Action #	IIP Action Information	Unit	AR #	IIP Action Owner	TCD
G50-RS2-06-35.1	**Assess the impact of extended operation on concessions against CSA N285.4**	018	28206348-01	N-CED	2020-12-31
Action:	Per P-CORR-00531-05099, assess the impact of N285.4 CNSC accepted program implementation concessions for the extended operating period, and update OPG implementation plans for identified elements for N285.4-14 as necessary.				
Completion Criteria:	Implementation plans updated for identified elements of CSA N285.4-14 as necessary.				

Appendix C: PSR2 Safety and Control Area (SCA) IIP Action Status List

IIP Actions defined in the IIP Resolution Action Overview (Appendix B), are listed by REGDOC-2.3.3 Safety and Control Areas in this Appendix C. The Appendix provides an overview of the IIP Action, associated Pickering NGS unit, and IIP Action target completion dates. The table provides a snapshot of the completion status of the IIP Actions at the time of submission of the IIP Report. IIP Action completion status following the IIP submission is managed through [R-14], with IIP Action status reported annually to the CNSC.

Figure 5: PSR2 Safety and Control Area (SCA) IIP Action Status List Content

Appendix C: PSR2 Safety and Control Area (SCA) IIP Action Status List

SCA	SCA Description	IIP Action	IIP Action Title	Unit	IIP Action Target Completion Date	IIP Action Completion Date
4	Safety Analysis	G24-RS1-04-19.1	Update Heat Transport Aging safety analysis models (Pickering 1,4)	014	2018-06-30	
		G24-RS1-04-19.2	Complete Loss of Flow (LOF) Safety Analysis accounting for Heat Transport Aging Methodology (Pickering 1,4)	014	2018-09-30	
		G24-RS1-04-19.3	Complete Small Break Loss of Coolant Accident (SBLOCA) safety analysis accounting for Heat Transport Aging Methodology (Pickering 1,4)	014	2018-09-30	
		G24-RS1-04-19.4	Complete Neutron Overpower safety analysis accounting for Heat Transport Aging (Pickering 1,4)	014	2018-09-30	
		G24-RS1-04-19.5	Update Heat Transport Aging safety analysis models (Pickering 5-8)	058	2018-11-30	
		G24-RS1-04-19.6	Complete Loss of Flow (LOF) Safety Analysis accounting for Heat Transport Aging Methodology (Pickering 5-8)	058	2019-03-31	
		G24-RS1-04-19.7	Complete Small Break Loss of Coolant Accident (SBLOCA) safety analysis accounting for Heat Transport Aging Methodology (Pickering 5-8)	058	2019-03-31	
		G24-RS1-04-19.8	Complete Neutron Overpower safety analysis accounting for Heat Transport Aging (Pickering 5-8)	058	2019-03-31	
		G25-RS1-04-20.1	Re-categorization of the Large Break Loss of Coolant Accident (LBLOCA) CANDU Safety Issues (CSI) to Category 2	018	2020-06-30	

Content

(1) **SCA:** CNSC Safety and Control Area number, as defined in CNSC REGDOC-2.3.3.

(2) **SCA Description:** CNSC Safety and Control Area description.

(3) **IIP Action:** Unique numerical IIP Action tracking reference, as a sub-action of the Resolution Action number, used to manage the IIP Action.

(4) **IIP Action Title:** Brief description of the IIP action that will support completion of the Resolution Action.

(5) **Unit:** Pickering NGS unit(s) that the IIP Action is applicable to. 014 represents Pickering NGS Units 1,4, 058 represents Pickering NGS Units 5-8, and 018 represents common systems of Pickering NGS.

(6) **IIP Action Target Completion Date:** IIP Action target completion date.

(7) **IIP Action Completion Date:** IIP Action completion date, approved by the IIP Action Owner.

Appendix C: PSR2 Safety and Control Area (SCA) IIP Action Status List

SCA	SCA Description	IIP Action	IIP Action Title	Unit	IIP Action Target Completion Date	IIP Action Completion Date
4	**Safety Analysis**	G24-RS1-04-19.1	Update Heat Transport Aging safety analysis models (Pickering 1,4)	014	2018-08-31	
		G24-RS1-04-19.2	Complete Loss of Flow (LOF) Safety Analysis accounting for Heat Transport Aging Methodology (Pickering 1,4)	014	2018-11-30	
		G24-RS1-04-19.3	Complete Small Break Loss of Coolant Accident (SBLOCA) safety analysis accounting for Heat Transport Aging Methodology (Pickering 1,4)	014	2018-11-30	
		G24-RS1-04-19.4	Complete Neutron Overpower safety analysis accounting for Heat Transport Aging (Pickering 1,4)	014	2018-11-30	
		G24-RS1-04-19.5	Update Heat Transport Aging safety analysis models (Pickering 5-8)	058	2019-01-31	
		G24-RS1-04-19.6	Complete Loss of Flow (LOF) Safety Analysis accounting for Heat Transport Aging Methodology (Pickering 5-8)	058	2019-05-30	
		G24-RS1-04-19.7	Complete Small Break Loss of Coolant Accident (SBLOCA) safety analysis accounting for Heat Transport Aging Methodology (Pickering 5-8)	058	2019-05-30	
		G24-RS1-04-19.8	Complete Neutron Overpower safety analysis accounting for Heat Transport Aging (Pickering 5-8)	058	2019-05-30	
		G25-RS1-04-20.1	Re-categorization of the Large Break Loss of Coolant Accident (LBLOCA) CANDU Safety Issues (CSI) to Category 2	018	2020-06-30	

Appendix C: PSR2 Safety and Control Area (SCA) IIP Action Status List

SCA	SCA Description	IIP Action	IIP Action Title	Unit	IIP Action Target Completion Date	IIP Action Completion Date
		G25-RS2-04-21.1	Re-categorize the CANDU Safety Issue CSI-IH6 to Category 2	014	2020-06-30	
		G27-RS1-04-23.1	Complete Fire Modeling Refinements	014	2018-09-30	
		G27-RS2-04-24.1	Investigate additional practicable design, operational and/or analytical enhancements	014	2018-12-31	
		G27-RS2-04-24.2	Upgrade Pickering NGS Unit 1,4 Emergency Boiler Water System (EBWS)	014	2020-12-31	
		G27-RS2-04-24.3	Upgrade Pickering NGS Unit 1, 4 Emergency Cooling to Heat Transport System (HTS) Makeup	014	2020-12-31	
		G27-RS2-04-24.4	Upgrade Pickering NGS Unit 1,4 Calandria Makeup	014	2020-12-31	
		G31-RS1-04-25.1	Provide Pickering NGS Safety Report Analysis of Common Mode Events	018	2018-03-31	
		G31-RS2-04-26.1	Update OPG REGDOC-2.4.1 Implementation Plan	018	2018-03-31	
		G32-RS1-04-27.1	Complete the REGDOC-2.4.2 Implementation Plan	018	2020-12-31	
5	**Physical Design**	G09-RS1-05-15.1	Complete Pickering NGS IFB / AIFB fuel basket stacking arrangement assessment	018	2019-03-31	

Appendix C: PSR2 Safety and Control Area (SCA) IIP Action Status List

SCA	SCA Description	IIP Action	IIP Action Title	Unit	IIP Action Target Completion Date	IIP Action Completion Date
		G47-RS1-05-32.1	Install wrenches and locks on the Pickering NGS 058 Yard Fire Protection System Yard Post indicator valves 058-71450-V37, 058-71450-V3027, and 058-71450-V36	058	2019-06-30	
		G48-RS1-05-33.1	Design and/or operational changes to Pickering Units 1,4 and Pickering Units 5-8 Fire Protection System interconnection.	058	2020-12-31	
		G12-RS1-05-17.1	Complete Environmental Qualification Assessments (EQA) to support Pickering NGS extended operations	018	2019-12-31	
6	**Fitness for Service**	G01-RS1-06-01.1	Provide Revised OPG CSA N285.8 Compliance Plan	018	2019-06-30	
		G01-RS1-06-01.2	Define Nominal Cooldown Transient for use in Probabilistic Leak-Before-Break (PLBB) Analyses	018	2018-06-30	
		G01-RS1-06-01.3	Provide to CNSC documented evidence that validation of PLBB code is complete	018	2018-03-31	
		G01-RS1-06-01.4	Address the Probabilistic Core Assessment (PCA) Flaw Removal Issue	018	2018-06-30	
		G01-RS2-06-02.1	Update Pickering NGS Fuel Channel Periodic Inspection Plan (PIP) for Operation to the end of 2024.	018	2018-12-31	
		G01-RS3-06-03.1	Submit 2018 Fuel Channel Life Cycle Management Plan (LCMP) Update that includes Pickering NGS U1 and U4 Operation to the end of 2024	014	2018-11-30	

Appendix C: PSR2 Safety and Control Area (SCA) IIP Action Status List

SCA	SCA Description	IIP Action	IIP Action Title	Unit	IIP Action Target Completion Date	IIP Action Completion Date
		G01-RS4-06-04.1	Develop Pickering NGS Fuel Channel Readiness Plan in Support of Operation to the end of 2024 (FCRP2024)	018	2018-03-31	
		G01-RS4-06-04.2	Update and Submit 2018 Fuel Channel Life Cycle Management Plan (LCMP) and the Pickering NGS Fuel Channel Readiness Plan in Support of Operation to the end of 2024 (FCRP2024)	018	2018-11-30	
		G01-RS4-06-04.3	Update and Submit 2019 Fuel Channel Life Cycle Management Plan (LCMP) and the Pickering NGS Fuel Channel Readiness Plan in Support of Operation to the end of 2024 (FCRP2024)	018	2019-11-30	
		G01-RS4-06-04.4	Update and Submit 2020 Fuel Channel Life Cycle Management Plan (LCMP) and the Pickering NGS Fuel Channel Readiness Plan in Support of Operation to the end of 2024 (FCRP2024)	018	2020-11-30	
		G01-RS4-06-04.5	Submit Confirmatory Fuel Channels Fitness for Service Correspondence	018	2020-11-30	
		G02-RS1-06-05.1	Submit 2018 Feeders Life Cycle Management Plan (LCMP) Update that includes Pickering NGS U1 and U4 operations to the end of 2024	014	2018-11-30	
		G03-RS1-06-06.1	Submit 2018 Steam Generators Life Cycle Management Plan (LCMP) update that includes Pickering NGS U1 and U4 operations to the end of 2024	014	2018-11-30	
		G04-RS1-06-07.1	Submit 2018 Reactor Components Life Cycle Management Plan (LCMP) update that includes Pickering NGS U1 and U4 Operation to the end of 2024	014	2018-11-30	

Appendix C: PSR2 Safety and Control Area (SCA) IIP Action Status List

SCA	SCA Description	IIP Action	IIP Action Title	Unit	IIP Action Target Completion Date	IIP Action Completion Date
		G04-RS2-06-08.1	Perform CT-LISS nozzle gap measurements as required on Pickering NGS Unit 6	6	2018-12-31	
		G04-RS2-06-08.2	Perform CT-LISS nozzle gap measurements as required on Pickering NGS Unit 5	5	2020-09-30	
		G05-RS1-06-09.1	Confirm service limits assessments for Nuclear Class 1 Piping include environmental factors	018	2020-12-31	
		G06-RS1-06-10.1	Assess the impact of the changes in criticality coding on cables	018	2018-12-31	
		G07-RS1-06-11.1	Update the Buried Piping Program Asset Management plan (N-PLAN-04916-10002) and Risk Ranking document	018	2019-03-31	
		G07-RS2-06-12.1	Update Buried Piping Program Requirements [N-PROC-MA-0088-R003]	018	2020-03-31	
		G08-RS1-06-13.1	Develop a risk based approach for aging management of critical piping systems	018	2018-09-30	
		G08-RS1-06-13.2	Complete the Condition Assessments consistent with the revised Reactor Safety Criticality Codes (XRF from GI-20)	018	2019-03-31	
		G08-RS1-06-13.3	Complete Condition Assessments for the in-scope piping systems in PSR2 scope to support Pickering NGS commercial operation to the end of 2024 (GI-08 and XRF from GI-22)	018	2019-06-30	

Appendix C: PSR2 Safety and Control Area (SCA) IIP Action Status List

SCA	SCA Description	IIP Action	IIP Action Title	Unit	IIP Action Target Completion Date	IIP Action Completion Date
		G08-RS1-06-13.4	Complete Condition Assessments for commodity groups in PSR2 scope to support Pickering NGS commercial operation to the end of 2024	018	2019-03-31	
		G08-RS1-06-13.5	Complete Condition Assessments for the Irradiated Fuel Bays (IFB) to support Pickering NGS commercial operation to the end of 2024. (XRF from GI-10)	018	2018-06-30	
		G08-RS1-06-13.6	Complete Condition Assessments for the Deaerators and the Deaerator Storage Tanks to support Pickering NGS commercial operation to the end of 2024. (XRF from GI-21)	018	2018-06-30	
		G08-RS1-06-13.7	Complete Condition Assessments for the Fueling Machines and FM Ball Screws to support Pickering NGS commercial operation to the end of 2024 (XRF from GI-29)	018	2018-06-30	
		G08-RS1-06-13.8	Complete Condition Assessments for the Primary Heat Transport auxiliary piping system, Primary Heat Transport pump discharge valves, and boiler inlet and outlet valves to support Pickering NGS commercial operation to the end of 2024 (XRF from GI-49)	018	2018-06-30	
		G08-RS2-06-14.1	Develop and implement Condition Assessment action tracking and reporting process including a database	018	2018-09-30	
		G10-RS1-06-16.1	Complete Pickering NGS Units 5-8 Irradiated Fuel Bay (IFB) Leakage Mitigation Project #13-40703	058	2019-09-30	
		G19-RS1-06-18.1	Demonstrate FFS of foundation H-piles for Pickering 1,4 Reactor Buildings (RB), Vacuum Building (VB) and Pressure Relief Duct (PRD)	018	2019-06-30	

Appendix C: PSR2 Safety and Control Area (SCA) IIP Action Status List

SCA	SCA Description	IIP Action	IIP Action Title	Unit	IIP Action Target Completion Date	IIP Action Completion Date
		G43-RS1-06-29.1	Complete inspections of Pickering NGS non-Containment safety-significant civil structures	018	2018-06-30	
		G43-RS2-06-30.1	Develop a risk-based approach for aging management of non-Containment safety-significant civil structures	018	2018-09-30	
		G43-RS3-06-31.1	Prepare Condition Assessments as appropriate for non-Containment safety-significant civil structures for Pickering NGS extended operation	018	2019-06-30	
		G50-RS1-06-34.1	Provide CNSC with an update on OPG implementation plans for identified elements of CSA N285.4-14	018	2018-09-30	
		G50-RS1-06-34.2	Documents identified in implementation plans updated and issued	018	2020-12-31	
		G50-RS2-06-35.1	Assess the impact of extended operation on concessions against CSA N285.4	018	2020-12-31	
10	**Emergency Management and Fire Protection**	G26-RS1-10-22.1	Develop and implement upgrades to the computer codes used for emergency response projections	018	2018-09-30	
		G40-RS1-10-28.1	Complete Pickering NGS Emergency Mitigating Equipment (EME) Phase 2 project 13-41027	018	2018-06-30	
		G40-RS1-10-28.2	Complete reassessment of Pickering NGS Beyond Design Basis Containment Integrity	018	2018-12-31	
		G40-RS1-10-28.3	Upgrade Emergency Mitigating Equipment (EME) Phase 2 to restore the functionality of a Main Vacuum Volume Pump (MVVP)	018	2019-06-30	

Appendix D: PSR2 Process Overview

In support of extended operation of the six Pickering units to the end of 2024 and licence renewal, a Periodic Safety Review (PSR) was conducted in accordance with CNSC Regulatory Document 2.3.3, *Periodic Safety Reviews* and International Atomic Energy Agency's (IAEA) Safety Standards Series, Specific Safety Guide No. SSG-25, *Periodic Safety Review for Nuclear Power Plants.*

The following provides details on the PSR process used by OPG that lead to the actions specified in this Integrated Implementation Plan.

D 1.0 Introduction

CNSC REGDOC-2.3.3 and IAEA SSG-25 identify that subsequent PSRs should focus on changes in requirements, facility conditions, operating experience and new information, rather than repeating activities conducted in previous safety reviews. As such it is forward looking, focusing on: changes to requirements since the last applicable assessment, confirmation that the condition of Pickering NGS supports the additional years of commercial operation, and new operating experience since the last assessments.

The objective of Pickering's PSR was to confirm that the design, operation and safety-significant structures, systems, and components support continued safe operation and to determine reasonable and practical safety enhancements to further improve the already low risk of plant operation.

The subsequent PSR, referred to as PSR2, builds on earlier OPG PSR work (referred to as PSR1) and other associated assessments, specifically:

1. The Pickering B Integrated Safety Review (ISR), which included a comprehensive review of Codes and Standards that was completed in 2009 to support potential refurbishment and continued operation of Pickering NGS 5-8 for an additional 30 years.

   It was decided to not refurbish Pickering NGS 5-8, instead the option to extend operations to the end of 2020 without the replacement of the major reactor components. As a result, the Continued Operations Plan (COP) was developed and implemented for Pickering NGS 5-8 with an end of life of 2020.

2. Pickering NGS 1,4 integrated safety assessments were performed during the Pickering A Return to Service (PARTS) work in support of approval to restart Units 1 and 4 following the extended shutdown of these units. (The pressure tubes on these units had previously been replaced in the late 1980's and early 1990's). Based on the results of these safety assessments, termed Systematic Review of Safety, Pickering Units 1 and 4 were restarted. Pickering Units 2 and 3 were not restarted

and were placed in the safe storage state (fuel and water removed, systems isolated/de-energized, and separation from common containment).

3. The Darlington ISR was performed in support of refurbishment and continued operation of the Darlington units for an additional 30 years. Extensive reviews (primarily clause-by-clause reviews) of Codes and Standards were completed. Much of the compliance assessment and evaluation of Safety Factor health for the Darlington ISR was based on programs and practices that apply across OPG's nuclear operations. As a result, Darlington ISR programmatic conclusions are applicable to the Pickering PSR2 for nuclear programs and practices that are relevant to Pickering.

Pickering PSR1 results were determined to be applicable to PSR2 if there was an open PSR1 gap or if a closed PSR1 gap could be affected by extended operation. If so, these gaps were carried forward into the PSR2 for consideration in the Global Assessment.

D 2.0 PSR2 Scope

The safety of Pickering NGS is regularly and thoroughly assessed, verified and assured through several processes that are part of the current licensing framework. OPG also applies routine comprehensive safety assessment and improvement programs that deal with specific safety issues, significant events and changes in standards and operating practices as they arise. These programs allow assessment of safety and plant operation to be improved on a continuous basis that can be correlated to all of the Safety Factors reviewed in PSR2. They include programs that ensure safe operations, effective configuration management, equipment reliability, life cycle management, aging management, periodic inspection and maintenance. Programs are also in place in the area of organization management and safety culture that focus on safety-related behaviours and accountability.

D 2.1 Current Laws, Regulations, Codes and Standards Applicable to PSR2

The PSR evaluated the extent to which the plant meets current laws, regulations, codes and standards. The process to identify those documents applicable to the PSR2 assessment basis involved first creating a broad list from multiple sources (potential candidate laws, regulations, codes and standards) and then filtering them to identify those that were applicable to the PSR2 scope.

D 2.2 Structures, Systems and Components within the Scope of the PSR2 Review

The Structures, Systems and Components (SSC) within the scope of the PSR2 review encompassed the Systems Important to Safety (SIS) and the Safe Operating Envelope (SOE) systems and was restricted to the facilities that are regulated under the Pickering NGS Power Reactor Operating Licence. Therefore, the Pickering Waste Management Facility, which has a separate operating licence, was not considered within the Pickering PSR2 scope.

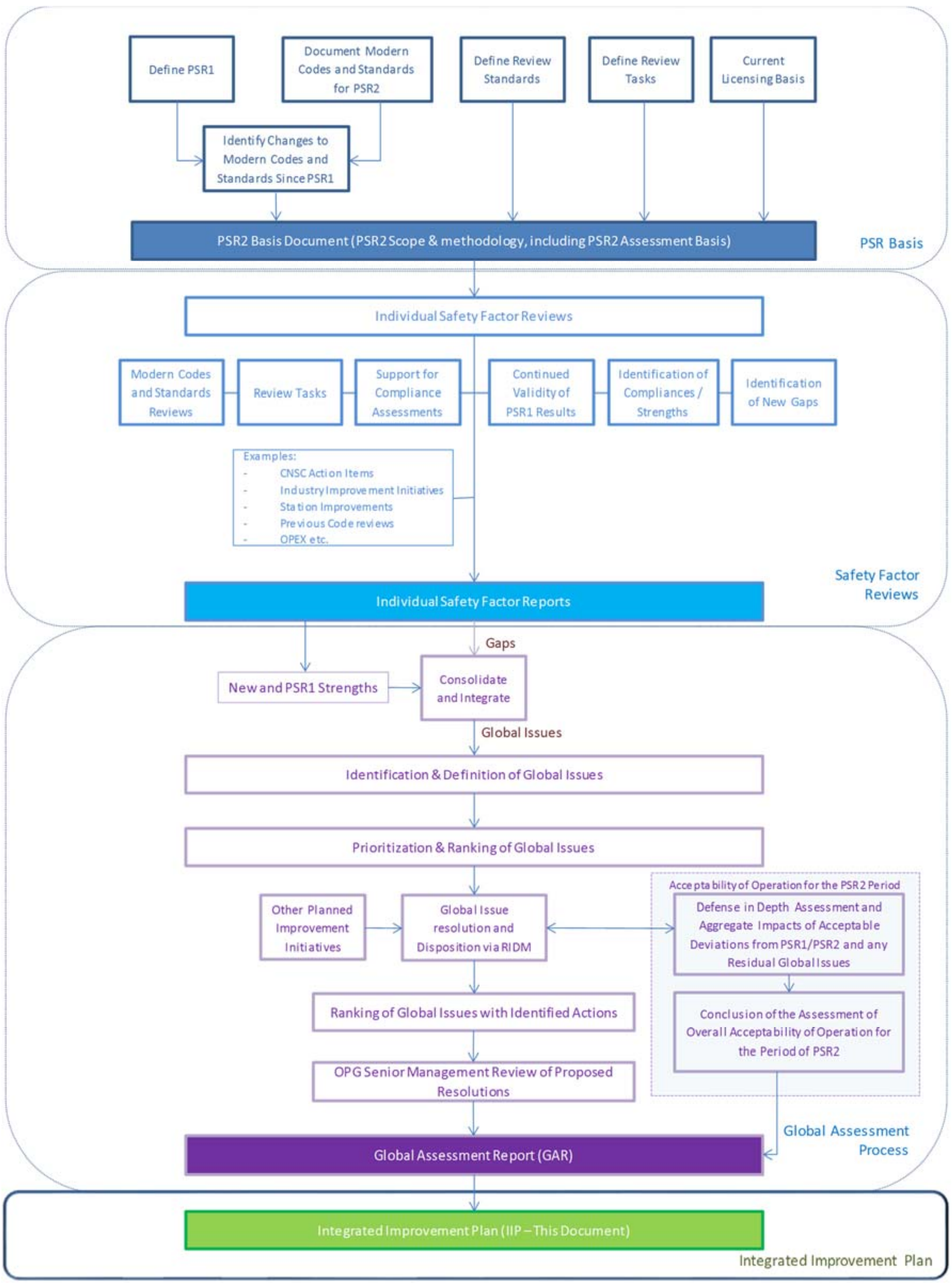
D 3.0 Details of the PSR2 Process

The general process overview for PSR2 is shown in Figure 6. Pickering's PSR2 was comprised of the following four key elements which are explained in the sections that follow:

1. PSR2 Basis Document
2. Safety Factor reviews
3. Global Assessment
4. Integrated Implementation Plan

Two additional assessments, referred to as "complementary assessments", (COP and Fukushima Action Items reassessments) were also performed to confirm the impact of extended operation beyond 2020. Where there were implications for extended operation, an associated gap was identified for consideration in the Global Assessment.

Figure 6: Pickering PSR2 Process Flowchart



D 3.1 PSR2 Basis Document

Pickering PSR2 Basis Document

- Prepared in accordance with CNSC REGDOC-2.3.3
- Documents how PSR2 was to be conducted
- CNSC Accepted

The Pickering PSR2 basis document, which was prepared by OPG and accepted by the CNSC in References D-2 and D-5 respectfully, defined the approach for completing the PSR2, specifically;

- The proposed operating strategy of the facility,
- Scope and methodology, including the conduct of Safety Factor reviews and identification of compliances and gaps,
- The process for categorizing, prioritizing, tracking and resolving Gaps arising from the Safety Factor reviews,
- Conduct of the Global Assessment,
- The methodology for preparing the Integrated Implementation Plan,
- Applicable current versions of Laws, Regulations, Codes and Standards,
- The major milestones, including the freeze date for document revisions, and,
- The project management and quality management processes.

D 3.2 Safety Factor Reviews

Safety Factor & Complementary Reviews

- 15 Safety Factor and 74 Code & standard reviews completed in accordance with CNSC REGDOC-2.3.3 resulted in 93 Gaps identified.
- 2 Complementary Reviews (COP & FAI Reassessments) completed which resulted in 26 Gaps identified.
- 23 Type III CNSC Additional Gaps and 1 Expert Panel Gap identified.

143 gaps identified for Global Assessment

Safety Factors cover all aspects important to the safety of an operating nuclear power plant, shown below in Table 1. There are 15 Safety Factors completed by Amec Foster-Wheeler used in the PSR2 review; 14 are identified in IAEA SSG-25, and one additional Safety Factor (Radiation Protection) as identified in CNSC REGDOC-2.3.3.

Table 1: PSR2 Safety Factors

Subject Area	Safety Factor	
The Plant	SF1	Plant Design
	SF2	Actual Condition of Structures, Systems and Components Important to Safety
	SF3	Equipment Qualification (environmental and seismic)
	SF4	Aging
Safety Analysis	SF5	Deterministic Safety Analysis
	SF6	Probabilistic Safety Assessment
	SF7	Hazard Analysis
Performance and Feedback from Operating Experience	SF8	Safety Performance
	SF9	Use of Experience from other NPPs and Research Findings
Management	SF10	Organization, the Management System and Safety Culture
	SF11	Procedures
	SF12	Human Factors
	SF13	Emergency Planning
Environment	SF14	Radiological Impact on the Environment
Radiation Protection	SF15	Radiation Protection

The results of the Safety Factor reviews were documented in Safety Factor Reports that have been submitted to CNSC. These reports include:

- The scope of the review,
- Applicable elements of the PSR2 Assessment Basis (Review Tasks and applicable Laws, Regulations, Codes and Standards),
- Review methodology,
- Assessment of compliance with Review Tasks,

- Effectiveness review of OPG programs supporting compliance assessments,
- Review findings (Compliances and Gaps),
- Impacts on other Safety Factor reviews,
- Overall assessment of the Safety Factor.

These reports concluded that there are no safety issues and that OPG has in place effective programs and processes for continued safe operation of Pickering NGS to the end of 2024. 143 gaps were identified from these Safety Factor reviews that were assessed in the Global Assessment.

As a subsequent PSR, the PSR2 Safety Factor reviews focused on changes in requirements (Laws, Regulations, Codes and Standards), updated plant conditions, operating experience and information from research, rather than repeating the activities of previous reviews. The methodology for performing the Safety Factor reviews takes full advantage of the safety assessments and Law, Regulation, Code and Standard compliance work previously completed by OPG.

This approach was in accordance with the guidance provided by the CNSC in REGDOC-2.3.3 that the effort required to undertake a subsequent PSR should require considerably less effort, subject to confirmation that previous conclusions remain valid.

D 3.3 Safety Factor Results and Reports

The Safety Factor reviews identified compliances and gaps with respect to the review elements in the PSR2 assessment basis. Specifically:

Compliance:

- For Clause-by-Clause reviews of current Laws, Regulations, Codes and Standards, Compliance indicates that the safety requirement is met.
- Where a High Level review has been performed, Compliance indicates that the intent of the safety requirement is met.
- Where an Incremental review has been performed, Compliance indicates that the change in the safety requirement, per the topical review, is met.
- For reviews of Safety Factor Review Tasks, Compliance indicates that either the safety requirement or the intent of the Review Task is met.

Gap:

- For Clause-by-Clause reviews of current Laws, Regulations, Codes and Standards, a gap indicates that the safety requirement is not met.
- Where a High Level review has been performed, a gap indicates that the intent of the standard is not met.
- Where an Incremental review has been performed, a gap indicates that the change in the standard, per the topical review, is not met.
- For reviews of Safety Factor Review Tasks, a gap indicates that the intent of the Review Task is not met.

Compliances that are equivalent to or surpassed PSR2 assessment basis requirements or practices were forwarded into the Global Assessment process for consideration as “strengths”. 24 strengths were identified and carried into the Global Assessment.

D 3.4 Global Assessment:

Global Assessment

- 143 gaps consolidated into 51 Global Issues (GIs)
- 117 proposed resolution plans developed for all 51 GIs
- 23 of the 51 GIs resulted in 35 proposed resolution plans following the prioritization process
- 82 proposed resolution plans did not require progression to the IIP (22 NFAs, 35 ADs, 25 XRFs)
- 24 Strengths identified
- Defence-in-Depth reviews completed
- Ranking of Resolution Statements performed

23 GIs having 35 Resolution Plans required follow-up action in the IIP

The objective of the Global Assessment was to provide an overall assessment of the safety of the plant, and to arrive at a judgement of the plant’s suitability for continued operation on the basis of a balanced view of the results from the reviews of the separate Safety Factors. This judgement takes into account the safety enhancements identified in the Global Assessment (plant and process modifications), strengths and residual Global Issues/acceptable deviations that impact on aggregate effects of the results, and consideration of existing planned safety enhancements and recent overall station safety performance.

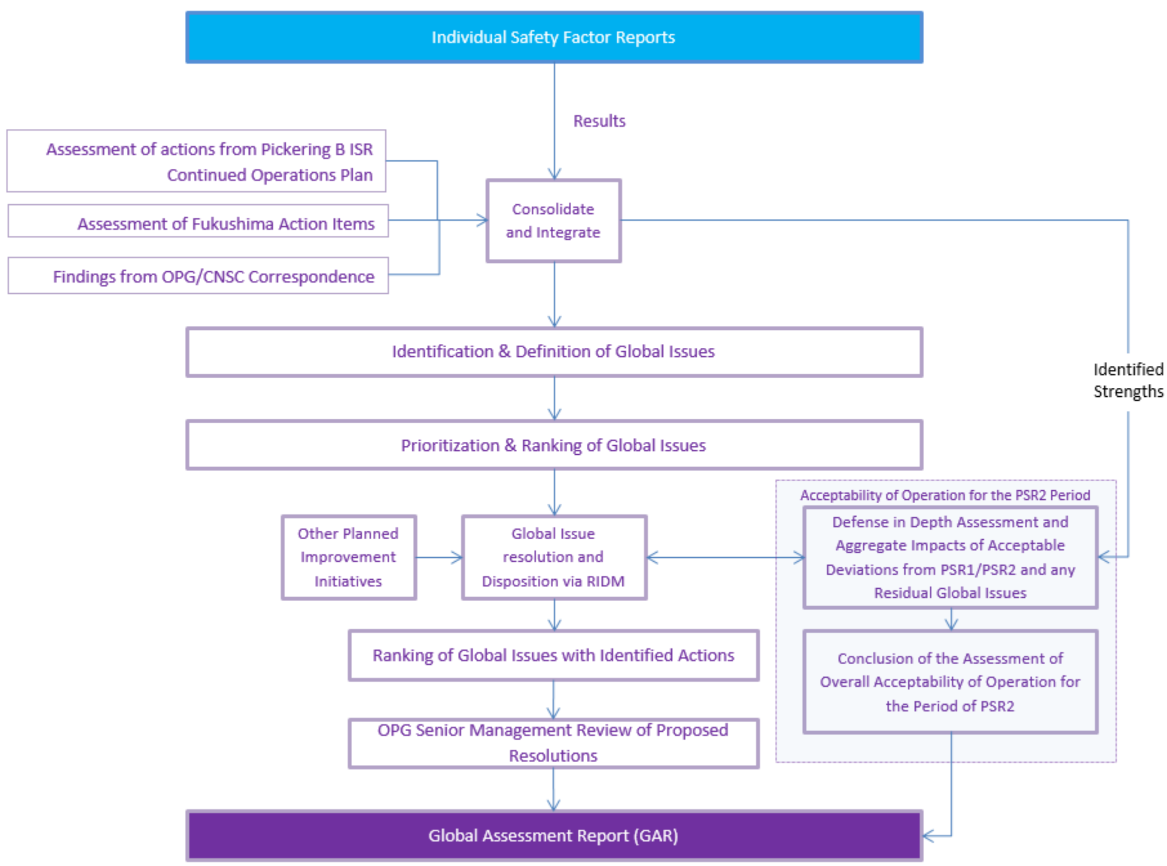
Consistent with the requirements of IAEA SSG-25, the Global Assessment was conducted by an interdisciplinary team (Candesco, a division of Kinectrics) with appropriate expertise in Operations, Design and plant safety, including appropriate participants from the safety factor reviews, and members who are independent from the Safety Factor review teams.

The Global Assessment Process consists of the following elements, shown below in Figure 8:

1. Identification and consolidation of Strengths and Gaps from the Safety Factor Reports.
2. Identification of Global Issues.
3. Assessment of interfaces between the various Safety Factors, Aggregate Impact of Global Issues.
4. Prioritization of Global Issues.

5. Development of Resolutions / Dispositions of Global Issues (and Gaps).
6. Consideration of defence-in-depth and aggregate impact of residual Global Issues / Acceptable Deviations.
7. Ranking of Global Issues with identified actions.
8. Senior Management Scope Review Board approval of proposed modifications for the purposes of PSR2.
9. Assessment of overall acceptability of operation of the plant over the period considered in PSR2.
10. Preparation of the Global Assessment Report to summarize the assessments, and document the Global Assessment.

Figure 7: Pickering NGS Global Assessment Process



D 3.5 Identification of Global Issues:

The gaps from the 15 individual Safety Factor Reports and two Complementary Reviews were consolidated and grouped by topic area into 51 Global Issues. The consolidation of gaps into Global Issues provided a means to assemble gaps of a common nature, facilitating the assessment of safety impact and identifying and assessing practical and effective resolutions.

D 3.6 Interfaces between the Various Safety Factors, Aggregate Impact of Global Issues:

With the assembly of Global Issues and strengths, and considering the recommendations from component condition assessments, the aggregate impact of the Global Issues was assessed to identify the interaction between issues.

D 3.7 Prioritization of Global Issues

Consistent with OPG prioritization processes used in previous Integrated Safety Reviews and industry practice, the Global Issues were prioritized with respect to their importance to nuclear safety.

The Safety Significance Level considered deterministic and probabilistic safety analysis impact, as appropriate. The assignment of safety significance values for prioritization was derived based on OPG experience and takes into account the priority values from the OPG guidelines for evaluating and prioritizing Safety Report Issues, the COG benefit-cost analysis processes, and the OPG station condition record categorization process. Probability levels selected for delineation between categories were based on significance and engineering judgement, and account for overall safety impact and align, where appropriate, with requirements and limits in relevant safety standards. The relationship between Safety Significance Level and impact on nuclear safety is shown in the Table 2.

Table 2: Relationship between Safety Significance Level

and Impact on Nuclear Safety

Safety Significance Level	Impact on Nuclear Safety
1	High
2	Medium
3	Low
4	Very Low

D 3.8 Development of Resolutions/Dispositions of Global Issues (and Gaps)

Resolution options were developed and assessed using risk-informed decision making techniques utilizing the following strategy:

- In assessing potential dispositions, defence-in-depth elements were considered.
- In developing the resolutions, consideration of overall safety significance will guide the resolution process.
- For Global Issue resolution – the process involved:

  - Evaluate the Global Issue to understand the safety basis, and intent of the requirement.
  - Consider possible options for resolution/mitigation. Consider safety significance and defence-in-depth elements.
  - Evaluate options with respect to effectiveness, cost, schedule, practicality. For potential plant modifications, this may require an evaluation of the safety impact, via both deterministic and probabilistic methods. If it is not practicable to fully resolve a Global Issue, other mitigation options were considered for enhancements.
  - Practicality of a proposed resolution was evaluated in terms of cost, resources, schedule, and considered in relation to the overall safety impact.
  - Propose recommended resolution/mitigation.
  - Document the decision making process.

- Items of High or Medium impact on nuclear safety (Safety Significance Levels 1 and 2) required more in-depth analysis to fully understand the issue and potential impact, and to develop the proposed resolution/mitigation. This required deterministic and/or probabilistic assessments to determine the nuclear safety impact of modifications and more detailed evaluation of the cost/practicality of proposed resolutions. Insights from available probabilistic safety analyses were used in evaluating the benefit/practicality of potential options.

- Items of Very Low Impact on Nuclear Safety (Safety Significance Level 4) were generally be deemed as Acceptable Deviations within the context of PSR2 (with the rationale provided). While these items were not tracked beyond the Global Assessment, they were shared with the accountable organizations for consideration as potential enhancement initiatives for their future work program planning purposes. A similar treatment was applied for items of Low Impact on Nuclear Safety (Safety Significance Level 3) for which a practicable solution was not readily evident.

- Proposed resolutions were categorized as follows:
  i) Programmatic (changes to procedures and programs),
  ii) Engineering (plant modifications), or
  iii) Analytical (e.g., safety analysis)

- In some cases, the development of resolutions/dispositions to the Global Issues were part of an OPG or industry initiative underway or planned. Or, the resolution and development of options required more detailed analysis and assessment, extending beyond the timelines for submission of PSR2. In these instances, the status of the initiative and plans were included in the disposition. The work was included in the Global Assessment to facilitate continued tracking.

- If the assessment determined that a Global Issue/gap had been closed, either by work done in the interim or by other processes, the rationale was documented and the Global Issue/gap was set to No Further Action (NFA) within the PSR process.

D 3.9 Consideration of Defence-in-Depth and Aggregate Impact of Residual Global Issues / Acceptable Deviations

An important element of the development of proposed recommendations were to assess the overall defence-in-depth and aggregate impact of the residual Global Issues/acceptable deviations. After evaluating a range of resolutions for Global Issues, and determining a recommended resolution to be selected, the impact on defence-in-depth, considering both deterministic and probabilistic elements, were evaluated to assess the aggregate impact on overall safety. This overall assessment was an important element in supporting the enhancement plans and the planned operational strategy over the period of PSR2.

D 3.10 Ranking of Global Issues with Identified Actions

All Global Issues whose resolution involves identified actions were ranked in accordance with overall safety significance, with consideration of factors such as impact to Nuclear Safety and timeliness to realize the benefit. This was based on engineering judgement applied by the Global Assessment team. The ranking process considered factors such as the priority previously determined (Safety Significance Level), the contribution to defence-in-depth, and the significance of the source (e.g., the type of document that generated the gap leading to the global issue). The ranking process also accounted for the extent of impact on multiple safety factors or areas.

D 3.11 Senior Management Scope Review Board Approval of Proposed Modifications for the Purposes of PSR2

The enhancements identified in the PSR2 Global Assessment Report, with their priority and safety basis, were presented to the OPG Senior Management Scope Review Board for approval. This review ensured alignment with the resolutions proposed, their basis and context, and was the means to obtain concurrence that the proposed enhancements are practicable and would be effective. Consistent with OPG project management processes, additional approval gates were required as the resolution development continues towards full implementation.

D 3.12 Assessment of Overall Acceptability of Operation of the Plant over the Period Considered in PSR2

As a final step in the assessment process, the team confirmed the overall acceptability of operation of the plant over the period considered in the PSR2. This entailed a review of the results of the safety factor reviews, a consideration of enhancements planned (both newly identified in PSR2 and from other station initiatives) and a consideration of plant performance.

D 3.13 Global Assessment Report

Preparation of the Global Assessment Report was conducted to summarize the Safety Factor Reports and Complementary Re-assessments and to document the Global Assessment by presenting the results, assessing the overall defence-in-depth of the plant, and documenting the conclusions, corrective actions, and enhancements to be considered. Appendix C of the Global Assessment Report includes a ranked list of those Global Issues with proposed resolution statements, with rationale for the ranking.

The Global Assessment Report also includes a statement of OPG's assessment of the overall acceptability of operation of the plant. Reviews and approval of the report were conducted as required under the OPG Management System.

As documented in Reference D-7, the GAR (P-REP-03680-00032-R001) was submitted to the CNSC in February 2018.

D 3.14 Integrated Implementation Plan:

Integrated Implementation Plan

- 23 Global Issues carried into the IIP
- 35 Resolution Actions developed
- Completion and Success Criteria developed for Resolution Actions
- 63 IIP actions with completion criteria developed with Target Completion Dates
- IIP administrative and change control process developed

The proposed enhancements resulting from the Global Assessment are documented in the Integrated Implementation Plan (IIP). Prepared by RCM Technologies, the IIP documents the enhancements (IIP actions) for addressing the Global Assessment Resolution Statements with target completion dates.

The IIP actions represent incremental enhancements that are in addition to OPG's continual improvement activities currently underway.

The enhancements summarized in the IIP were mapped to the CNSC Safety and Control Areas (per Appendix B of CNSC REGDOC-2.3.3), which are listed by SCA in Appendix C of the IIP.

D 3.15 Integrated Implementation Plan Logistics

The IIP listing of enhancements will include those resulting from the Global Assessment Report, including both new modifications proposed as part of the resolution of Global Issues, and also considering the existing planned station modifications that were integral to the overall assessment of safety.

A review was conducted with program owners and appropriate managers to derive plans for implementation based on priority and resources. These plans were developed with consideration of the other important initiatives underway or planned at Pickering NGS as part of continual improvement.

The initiatives were tabularized in Appendix B of this IIP, with owners assigned and planned implementation dates. Existing initiatives integral to the overall assessment of safety during the Global Assessment will also be included in this listing. The listing will include the priority and the basis for the priority. The implementation of the initiatives will be tracked and reported.

D 3.16 Integrated Implementation Plan Format

The IIP has been structured to allow a reader to understand the implementation plan and the basis for the plan. Appendices A, B and C within the IIP documents the IIP Actions along with their target completion dates. The IIP Actions include new initiatives that came from the Safety Factor Reports and the Global Assessment Report, and the existing initiatives that were integral to the overall assessment of safety.

The IIP was presented in a manner aligned with the CNSC Safety and Control Areas. The report summarizes the implementation tracking and reporting process and the IIP change management process. These processes will allow tracking of initiatives to completion or resolution in an auditable manner, consistent with OPG's management system.

D 3.17 Complementary Review: Continued Operations Plan (COP) Reassessment

In accordance with the PSR Basis Document [D-2], the Pickering NGS 5-8 Continued Operations Plan (COP) actions were reviewed to determine if there were implications for PSR2. Specifically, the COP actions pertaining to the Pickering NGS 5-8 Integrated Safety Review from 2009 and the Fitness for Service actions were reassessed for implications given the intent to operate Pickering NGS 5-8 beyond 2020.

In addition, implications for Pickering NGS 1,4 were also identified. Where there are implications for extended operation of Pickering NGS 5-8, or for Pickering NGS 1,4, a PSR2 gap was identified that was considered in the Global Assessment process described above.

D 3.18 Complementary Review: Fukushima Action Plan Reassessment

Following the events at Fukushima Daiichi in March 2011, the CNSC issued Fukushima Action Items to the Canadian Nuclear Utilities to ensure that the lessons learned from the event were appropriately incorporated into Canadian nuclear operations.

OPG has been recognized for its achievements in operational and management excellence in its response to the Fukushima Daiichi event and has confirmed that its stations remain safe with systems and procedures in place to deal with beyond design basis events.

OPG has taken the key lessons learned from the Fukushima event and incorporated changes to further enhance the safety of OPG's nuclear facilities. In 2015, all Phase 1 Fukushima Action Items (FAIs) for the Darlington and Pickering Units were closed.

In accordance with the PSR2 Basis Document, all of the FAIs pertaining to Pickering NGS were reassessed to determine if the basis for their closure remained valid in the context of intended extension of commercial operations of the station beyond 2020. This FAI reassessment, which was submitted to the CNSC in March 2017 did not identify any gaps for PSR2, however, two items were carried over to the Global Assessment as additional gaps as identified by the CNSC.

D 3.19 Findings from CNSC Staff Reviews of Safety Factor Reports and Complementary Reviews

From the CNSC staff review of the 15 Safety Factor Reports and Complementary Reviews, 77 Additional Gaps (AGs) were identified which were grouped into the following categories:

a) Type I & II: Provision of information. There were 54 AGs related to CNSC requests for additional supporting information to demonstrate program effectiveness or additional evidence for statements made by OPG in the Safety Factor and Complementary Review Reports.

b) Type III: Specific technical issues. There were 23 AGs related to CNSC identification of technical concerns, or demonstration of adequacy of implementation/response to issues of concern to CNSC.

The majority (54 of 77) of the AGs were Type I & II (requests for additional information) that will be addressed by OPG before March 15, 2018 [D-6]. None of these AGs invalidate the conclusions of the associated report.

The remaining 23 AG's Type III AGs were assessed in the Global Assessment and consolidated with other related PSR2 Gaps for the development of appropriate resolution statements.

D 4.0 PSR2 Results

The 15 Safety Factor review reports conclude that there are no fundamental safety issues and that OPG has in place effective programs and processes for continued safe operation of Pickering NGS until the end of 2024.

1. Organization, Management Systems and Safety Culture was reviewed as a safety factor for Pickering PSR2. Specifically, this review confirmed that the Pickering NGS organization, management system and safety culture are effective.

2. Human Factors was reviewed as a safety factor for Pickering PSR2. Specifically, this review confirmed that the various human factors that may affect the safe operation of Pickering NGS have been appropriately addressed, and are effective.

3. Safety Performance was reviewed as a safety factor for Pickering PSR2. Specifically, this review confirmed that the safety performance indicators and records of operating experience, including the evaluation of root causes of plant events, exist and are utilized.

4. OPEX and Research Findings was reviewed as a safety factor for Pickering PSR2. Specifically, this review confirmed that for Pickering NGS there is adequate feedback of relevant experience from other nuclear power plants and from findings of research, and that this is used to introduce reasonable and practicable safety improvements at the plant or in the operating organization.

5. The area of Procedures was reviewed as a safety factor for Pickering PSR2. Specifically, this review confirmed that the Pickering NGS processes for managing, implementing and adhering to operating and working procedures and for maintaining compliance with operational limits and conditions and regulatory requirements are adequate and effective and ensure plant safety.

6. Deterministic Safety Analysis was reviewed as a safety factor for Pickering PSR2. Specifically, this review confirmed that the deterministic safety analysis programs and procedures at OPG are comprehensive, resulting in a systematic and disciplined approach to identifying, prioritizing and addressing any safety analysis related issues.

7. Hazard Analysis was reviewed as a safety factor for Pickering PSR2. Specifically, this review confirmed that Pickering NGS has robust protection against internal and external hazards, taking into account the plant design, site characteristics, the actual condition of the Structures, Systems and Components (SSCs) important to safety.

8. Probabilistic Safety Assessment (PSA) was reviewed as a safety factor for Pickering PSR2. Specifically, this review has confirmed that the PSA programs and procedures at OPG are comprehensive, resulting in a systematic and disciplined approach to identifying, prioritizing and addressing safety analysis related issues.

9. Plant Design was reviewed as a safety factor for Pickering PSR2. This review confirmed, by assessment against the current licensing basis and applicable standards, requirements and practices that the physical design and documentation supports continued safe operation of Pickering NGS.

10. Equipment environmental and seismic qualifications were reviewed as a safety factor for Pickering PSR2. Specifically, this review confirmed that the Pickering NGS equipment important to safety has been properly environmentally and seismically qualified and that these qualifications are being maintained through maintenance, inspection and testing programs.

11. Actual condition of Structures, Systems and Components (SSCs) important to safety was reviewed as a safety factor for Pickering PSR2. Specifically, this review concluded that the majority of the plant's SSCs are in good condition and support safe extended station operation to the end of 2024. Recommendations for improvement have been made where required, of which many are in progress. For this life extension period, no major concerns have been identified and the SSCs Important to Safety continue to operate as per the design basis requirements.

12. Plant aging was reviewed as a safety factor for Pickering PSR2. Specifically, this review confirmed that aging aspects affecting SSCs important to safety are being effectively managed and that an effective aging management program is in place.

13. Radiation Protection was reviewed as a safety factor for Pickering PSR2. Specifically, this review has confirmed that radiation protection has been accounted for in the design and operation of Pickering NGS, and that radiation protection provisions (including design and equipment) protect workers from radiation and ensure that contamination and radiation exposures and doses to persons are monitored and controlled and maintained As Low As Reasonably Achievable (ALARA).

14. Radiological Impact on the Environment was reviewed as a safety factor for Pickering PSR2. Specifically, this review has confirmed that Pickering NGS has in place an effective program for monitoring the radiological impact of the plant on the environment, which ensures that emissions are properly controlled and are as low as reasonably achievable.

15. Emergency Planning was reviewed as a safety factor for Pickering PSR2. Specifically, this review has confirmed that OPG Nuclear has in place adequate plans, staff, facilities and equipment for dealing with emergencies. In addition, arrangements are in place for regular emergency training and exercises, and interaction and coordination with local and national authorities.

D 5.0 References

[D-1] Protocol, “OPG-CNSC Protocol for the Conduct of a Periodic Safety Review in Support of Pickering NGS Licence Renewal”, January 17, 2017, e-Doc 5143721, CD# P-CORR-00531-04725 R001.

[D-2] OPG Letter, B. McGee to H. Khouaja, “Submission of Pickering NGS Periodic Safety Review 2 Basis Document Revision 002”, July 6, 2016, CD# P-CORR-00531-04780.

[D-3] OPG Letter, B. McGee to A. Viktorov, "Pickering NGS Periodic Safety Review 2 - Submission of Continued Operations Plan (COP) Reassessment", February 13, 2017, CD# P-CORR-00531-04927.

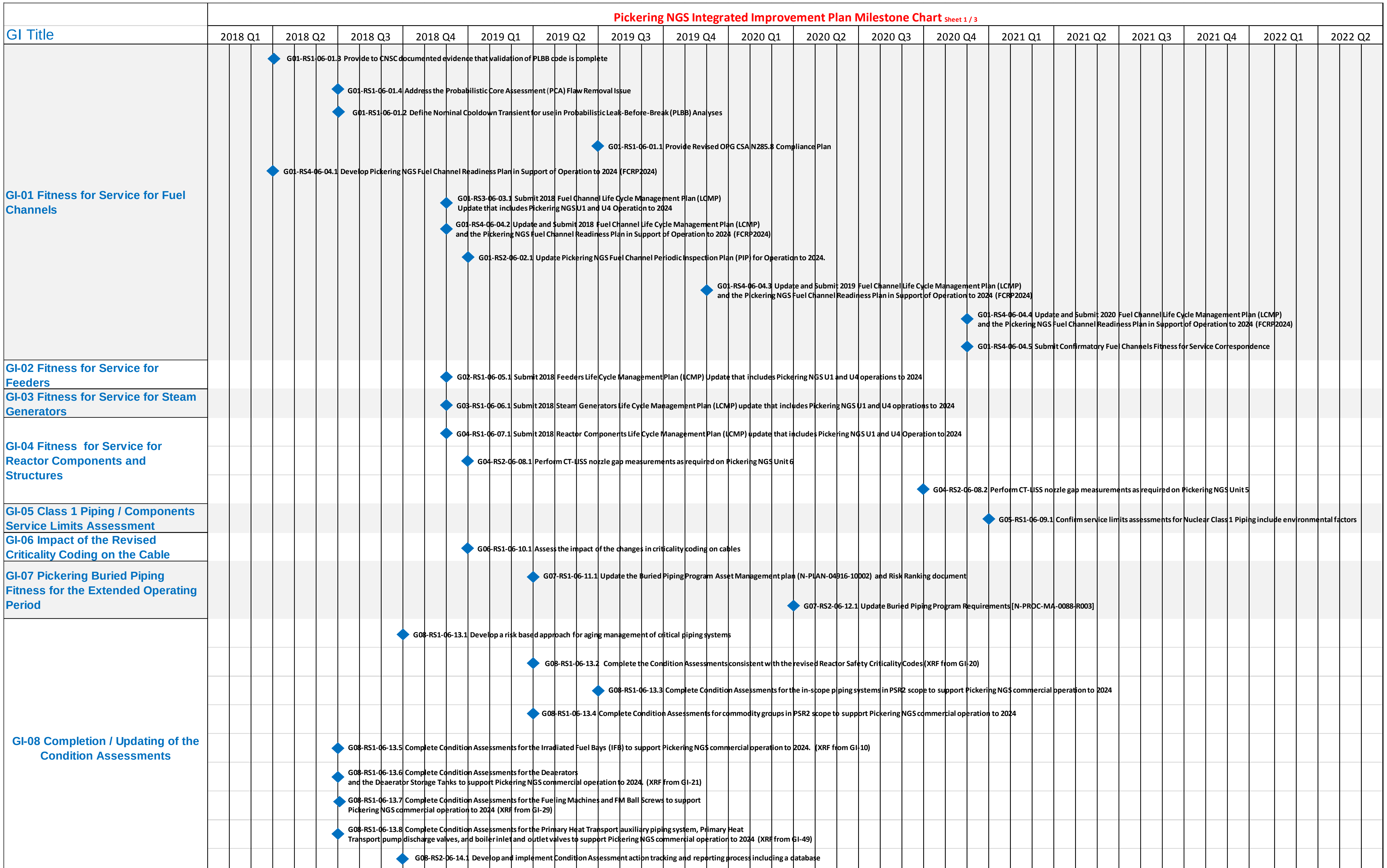
[D-4] OPG Letter, W.S. Woods to M. Santini and F. Rinfret, "OPG Progress Report No. 7 on CNSC Action Plan - Fukushima Action Items", November 30, 2015, CD# N-CORR-00531-06906.

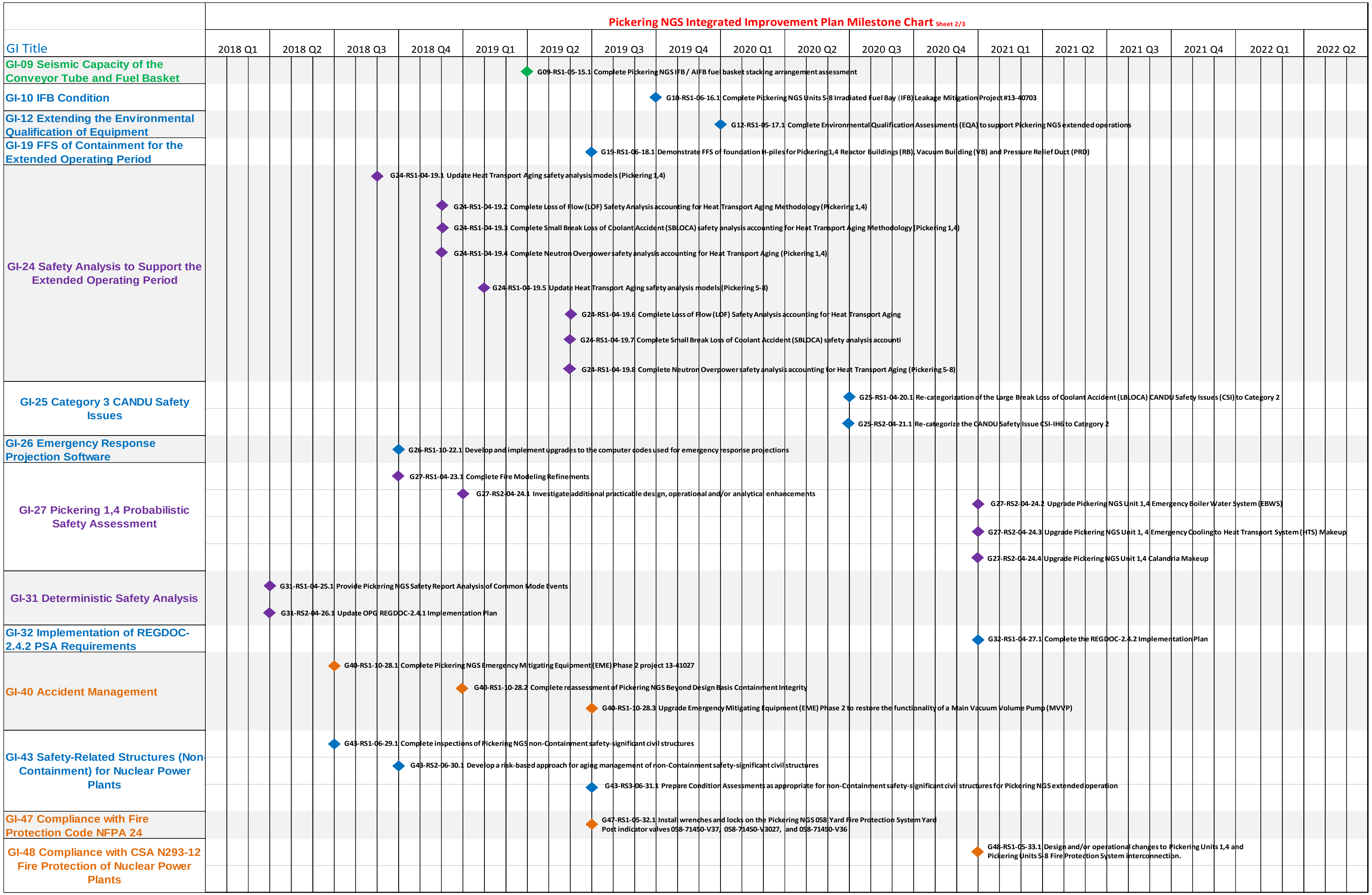
[D-5] CNSC Letter, H. Khouaja to B. McGee, “Pickering NGS: CNSC Staff Acceptance of Pickering NGS Periodic Safety Review 2 (PSR2) Basis Document”, July 8, 2016, e-Doc 5037314, CD# P-CORR-00531-04789.

[D-6] OPG Letter, R. Lockwood to A. Viktorov, Pickering Periodic Safety Review 2: Process for Addressing CNSC Identified Additional Gaps, P-CORR-00531-05132, September 18, 2017.

[D-7] OPG Report, *Pickering NGS Global Assessment Report*, P-REP-03680-00032-R001, February 08, 2018.

Figure 8: Pickering NGS Integrated Improvement Plan Milestones





Pickering NGS Integrated Improvement Plan Milestone Chart Sheet 2/3
GI Title
2018 Q1
2018 Q2
2018 Q3
2018 Q4
2019 Q1
2019 Q2
2019 Q3
2019 Q4
2020 Q1
2020 Q2
2020 Q3
2020 Q4
2021 Q1
2021 Q2
2021 Q3
2021 Q4
2022 Q1
2022 Q2
GI-09 Seismic Capacity of the Conveyor Tube and Fuel Basket
G09-RS1-05-15.1 Complete Pickering NGS IFB / AIFB fuel basket stacking arrangement assessment
GI-10 IFB Condition
G10-RS1-06-16.1 Complete Pickering NGS Units 5-8 Irradiated Fuel Bay (IFB) Leakage Mitigation Project #13-40703
GI-12 Extending the Environmental Qualification of Equipment
G12-RS1-05-17.1 Complete Environmental Qualification Assessments (EQA) to support Pickering NGS extended operations
GI-19 FFS of Containment for the Extended Operating Period
G19-RS1-06-18.1 Demonstrate FFS of foundation H-piles for Pickering 1,4 Reactor Buildings (RB), Vacuum Building (VB) and Pressure Relief Duct (PRD)
GI-24 Safety Analysis to Support the Extended Operating Period
G24-RS1-04-19.1 Update Heat Transport Aging safety analysis models (Pickering 1,4)
G24-RS1-04-19.2 Complete Loss of Flow (LOF) Safety Analysis accounting for Heat Transport Aging Methodology (Pickering 1,4)
G24-RS1-04-19.3 Complete Small Break Loss of Coolant Accident (SBLOCA) safety analysis accounting for Heat Transport Aging Methodology (Pickering 1,4)
G24-RS1-04-19.4 Complete Neutron Overpower safety analysis accounting for Heat Transport Aging (Pickering 1,4)
G24-RS1-04-19.5 Update Heat Transport Aging safety analysis models (Pickering 5-8)
G24-RS1-04-19.6 Complete Loss of Flow (LOF) Safety Analysis accounting for Heat Transport Aging
G24-RS1-04-19.7 Complete Small Break Loss of Coolant Accident (SBLOCA) safety analysis accounti
G24-RS1-04-19.8 Complete Neutron Overpower safety analysis accounting for Heat Transport Aging (Pickering 5-8)
GI-25 Category 3 CANDU Safety Issues
G25-RS1-04-20.1 Re-categorization of the Large Break Loss of Coolant Accident (LBLOCA) CANDU Safety Issues (CSI) to Category 2
G25-RS2-04-21.1 Re-categorize the CANDU Safety Issue CSI-IH6 to Category 2
GI-26 Emergency Response Projection Software
G26-RS1-10-22.1 Develop and implement upgrades to the computer codes used for emergency response projections
GI-27 Pickering 1,4 Probabilistic Safety Assessment
G27-RS1-04-23.1 Complete Fire Modeling Refinements
G27-RS2-04-24.1 Investigate additional practicable design, operational and/or analytical enhancements
G27-RS2-04-24.2 Upgrade Pickering NGS Unit 1,4 Emergency Boiler Water System (EBWS)
G27-RS2-04-24.3 Upgrade Pickering NGS Unit 1, 4 Emergency Cooling to Heat Transport System (HTS) Makeup
G27-RS2-04-24.4 Upgrade Pickering NGS Unit 1,4 Calandria Makeup
GI-31 Deterministic Safety Analysis
G31-RS1-04-25.1 Provide Pickering NGS Safety Report Analysis of Common Mode Events
G31-RS2-04-26.1 Update OPG REGDOC-2.4.1 Implementation Plan
GI-32 Implementation of REGDOC-2.4.2 PSA Requirements
G32-RS1-04-27.1 Complete the REGDOC-2.4.2 Implementation Plan
GI-40 Accident Management
G40-RS1-10-28.1 Complete Pickering NGS Emergency Mitigating Equipment (EME) Phase 2 project 13-41027
G40-RS1-10-28.2 Complete reassessment of Pickering NGS Beyond Design Basis Containment Integrity
G40-RS1-10-28.3 Upgrade Emergency Mitigating Equipment (EME) Phase 2 to restore the functionality of a Main Vacuum Volume Pump (MVVP)
GI-43 Safety-Related Structures (Non-Containment) for Nuclear Power Plants
G43-RS1-06-29.1 Complete inspections of Pickering NGS non-Containment safety-significant civil structures
G43-RS2-06-30.1 Develop a risk-based approach for aging management of non-Containment safety-significant civil structures
G43-RS3-06-31.1 Prepare Condition Assessments as appropriate for non-Containment safety-significant civil structures for Pickering NGS extended operation
GI-47 Compliance with Fire Protection Code NFPA 24
G47-RS1-05-32.1 Install wrenches and locks on the Pickering NGS 058 Yard Fire Protection System Yard Post indicator valves 058-71450-V37, 058-71450-V3027, and 058-71450-V36
GI-48 Compliance with CSA N293-12 Fire Protection of Nuclear Power Plants
G48-RS1-05-33.1 Design and/or operational changes to Pickering Units 1,4 and Pickering Units 5-8 Fire Protection System interconnection.

Pickering NGS Integrated Improvement Plan Milestone Chart Sheet 3/3

GI Title	2018 Q1	2018 Q2	2018 Q3	2018 Q4	2019 Q1	2019 Q2	2019 Q3	2019 Q4	2020 Q1	2020 Q2	2020 Q3	2020 Q4	2021 Q1	2021 Q2	2021 Q3	2021 Q4	2022 Q1	2022 Q2
GI-50 N285.4 PIP / Documentation Revision				G50-RS1-04-34.1 Revise the CSA N285.4 PIPs and governance as required to align with elements of N285.4-14									G50-RS1-04-34.2 Documents identified in implementation plans updated and issued as required G50-RS2-04-35.1 Assess the impact of extended operation on concessions against CSA N285.4					

Figure 9: Pickering NGS PSR2 Timeline