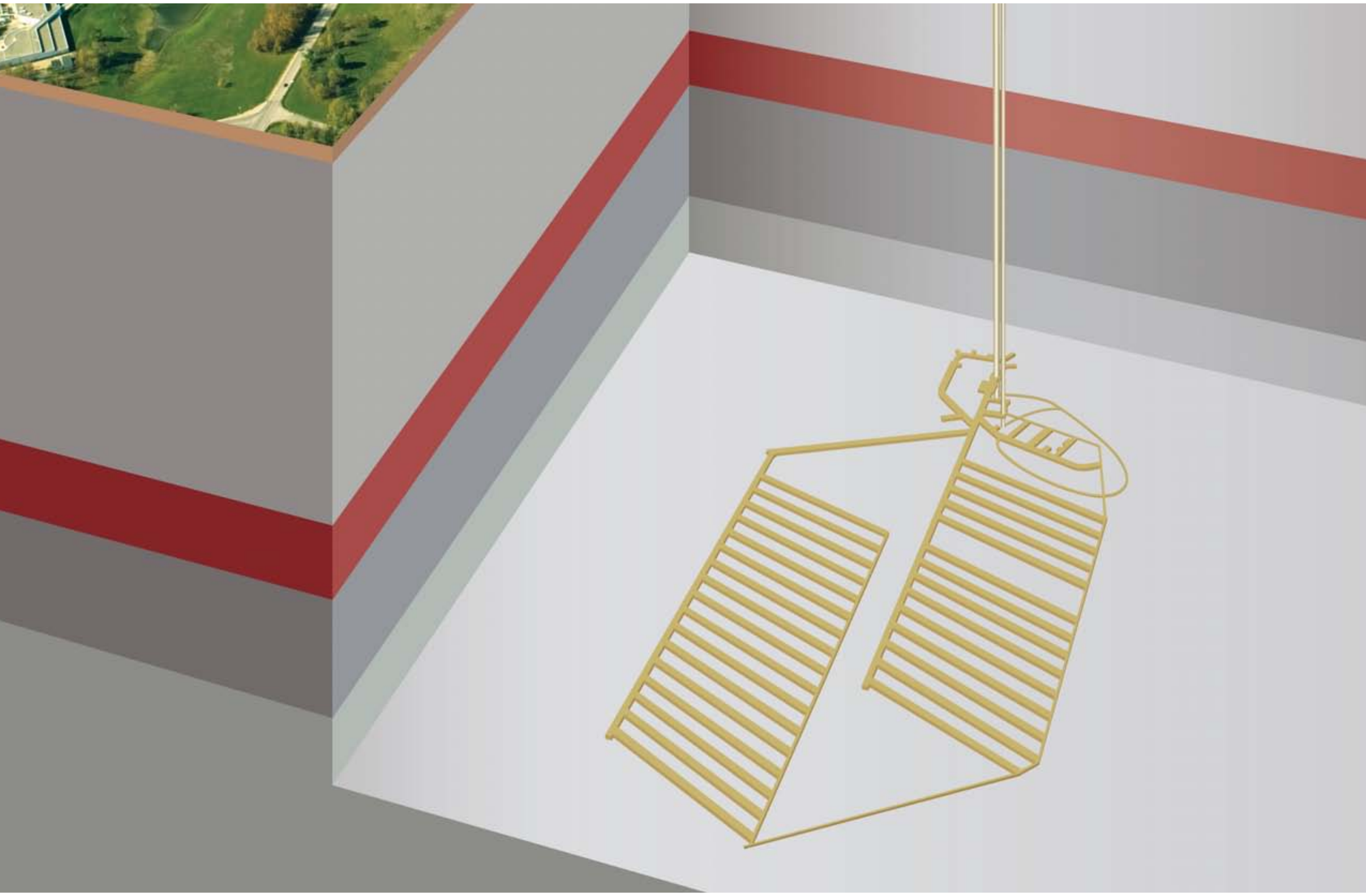
OPG's DEEP GEOLOGIC REPOSITORY PROJECT

For Low & Intermediate Level Waste

March 2011



Preliminary Safety Report

00216-SR-01320-00001 R000

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REPOSITORY

FOR LOW & INTERMEDIATE LEVEL WASTE

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Prepared by:
Nuclear Waste Management Organization

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1. INTRODUCTION

Low and Intermediate Level Waste (L&ILW) has been safely stored at Ontario Power Generation’s (OPG) Western Waste Management Facility (WWMF) at the Bruce nuclear site for approximately 40 years. However, the facilities to store L&ILW at the WWMF are designed for interim storage, and some of the wastes can remain radioactive for thousands of years. A Deep Geologic Repository (DGR), described and assessed in this Preliminary Safety Report (PSR), can meet the need for safe long-term management of L&ILW. The DGR location is adjacent to the WWMF on the Bruce nuclear site, in the Municipality of Kincardine.

The DGR is the long-term management solution for the operational and refurbishment L&ILW currently stored at the WWMF, as well as the future operational and refurbishment L&ILW produced as a result of operation of OPG-owned or operated nuclear reactors. The DGR is:

- Consistent with federal government policy (NRCAN96);
- Preferred by the host municipality over the other technical options that have been evaluated, including long-term storage in the existing facilities (KC04); and
- Consistent with best international practice.

1.1 Purpose and Scope of the Preliminary Safety Report

The purpose of the PSR is to provide information necessary to obtain a site preparation and construction licence for the DGR under the Nuclear Safety and Control Act (NSCA97) and associated regulations. While most of the informational requirements are addressed in this PSR, some remaining requirements are addressed through supplementary information submitted separately in support of OPG’s licence application (OPG07a).

The scope of the PSR includes describing information required to build a safety case, and presenting a safety case that demonstrates clearly that the DGR is safe to construct, operate and decommission, and that it will provide safe long-term management of OPG’s L&ILW.

1.2 DGR Project Overview

The DGR project consists of site preparation, construction, and operation of above ground and underground facilities for the long-term management of OPG’s L&ILW. The DGR project also includes decommissioning, abandonment and long-term performance. The underground facilities, located at a nominal depth of 680 m, are comprised of access-ways (shafts and tunnels), emplacement rooms and various underground service areas and installations. The surface facilities consist of buildings in the main shaft area including main shaft headframe, Waste Package Receiving Building (WPRB), office, main control room and amenities building, buildings in the ventilation shaft area including ventilation shaft headframe and hoist house, and the Waste Rock Management Area (WRMA).

The L&ILW is transferred to the DGR from the WWMF, where it is stored in various above ground and in-ground structures. The DGR is located on and beneath a parcel of land retained by OPG on the Bruce nuclear site, most of which was leased by OPG to Bruce Power in May 2001. The location of the DGR within the Bruce nuclear site is shown in Figure 1-1. The DGR site is shown in Figure 1-2. Schematic of the DGR is shown in Figure 1-3.

The project comprises the following phases:

1. Regulatory Approvals Phase

During the Regulatory Approvals (RA) phase, Environmental Assessment (EA) for the project is conducted, and project documentation in support of licensing, consisting mainly of an Environmental Impact Statement (EIS) and this PSR, are submitted to a Joint Review Panel. The Panel conducts a review of the project documentation in accordance with the Joint Review Panel Agreement (MOE09a), and submits a report to the Minister of Environment, followed by a decision on the site preparation and construction licence for the DGR project.

2. Design and Construction Phase

In the Design and Construction Phase, design of the DGR is finalized and, after obtaining the site preparation and construction licence, the site is prepared and the DGR is constructed in accordance with the licence to carry out these activities. Subsequently, the facility is commissioned and declared in-service.

3. Operations Phase

Operation of the DGR begins after an operating licence has been obtained from the Canadian Nuclear Safety Commission (CNSC). During operation of the DGR, the L&ILW is placed in the emplacement rooms. The facility is considered operational until a decommissioning licence is obtained and decommissioning activities start.

4. Decommissioning Phase

A separate EA is expected to be conducted for decommissioning of the DGR, as required under the Canadian Environmental Assessment Act (CEAA92), at the time that decommissioning is pursued. Decommissioning includes dismantling surface facilities and sealing the shafts.

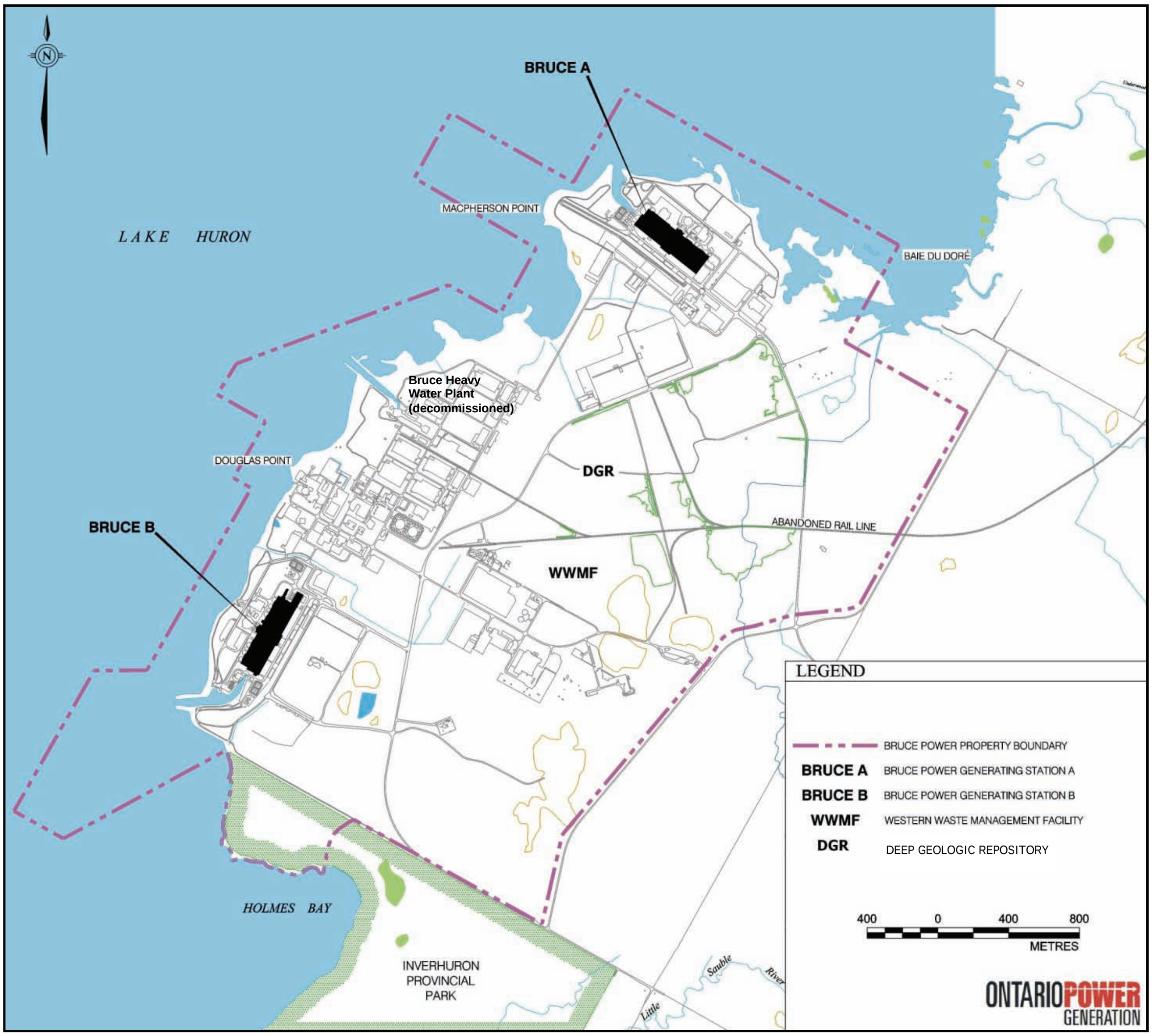

Figure 1-1: Location of the DGR within the Bruce Nuclear Site

Figure 1-2: DGR Site

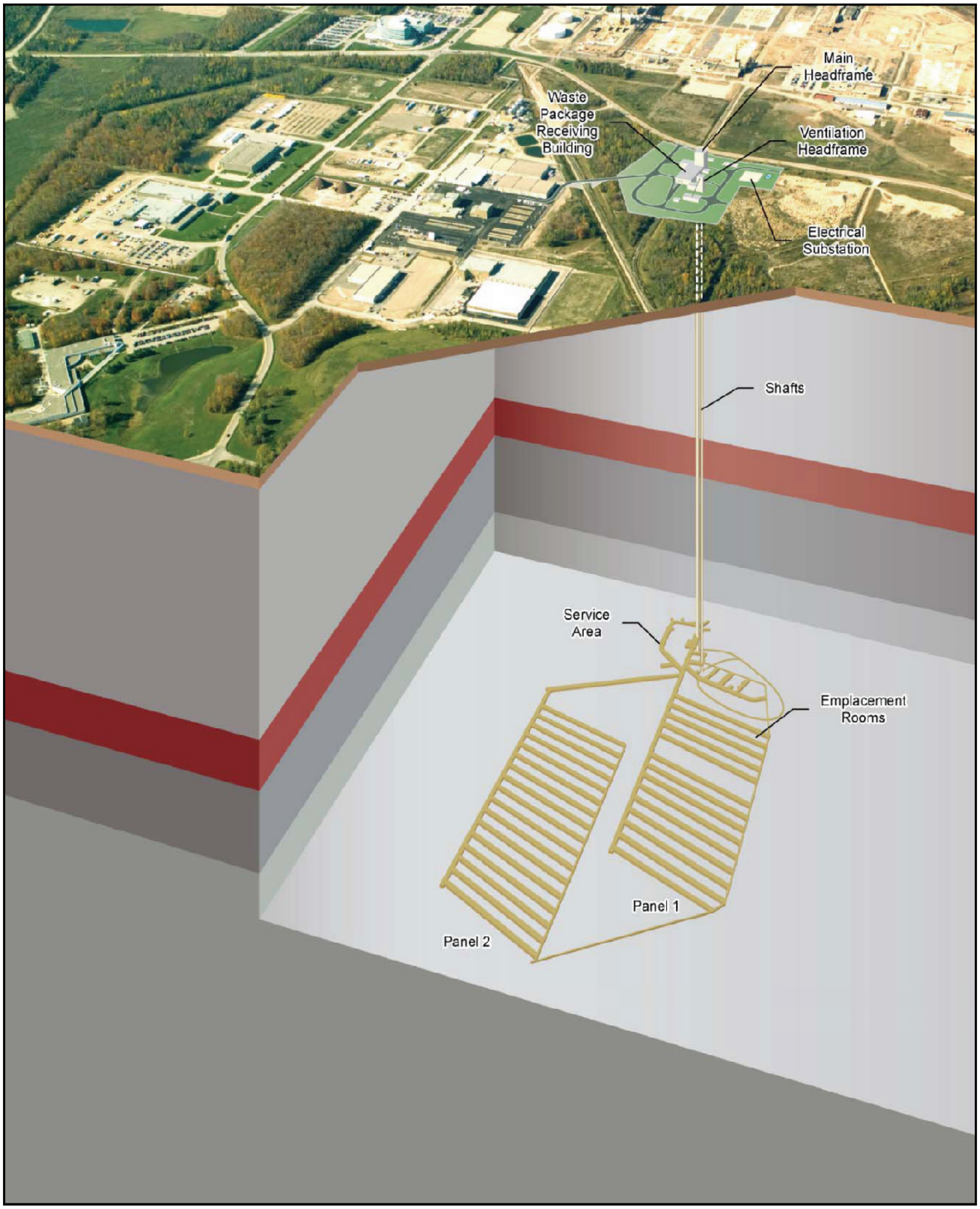

Figure 1-3: Schematic of the DGR

1.3 Time Frames

The temporal boundaries associated with the DGR, after it becomes operational, are divided into two main periods, i.e., preclosure and postclosure, and are described in the following sections.

1.3.1 Preclosure Period

The period of time post-construction and before decommissioning of the DGR is referred to as the preclosure period. It covers the period during which waste is being emplaced in the DGR, as well as the period of decommissioning of all components of the DGR. Activities include receipt and on-site transfer of waste packages, transfer underground and emplacement of L&ILW in rooms in the DGR, activities necessary to support and monitor operations, and decommissioning activities.

The operations phase is assumed to last approximately 40 years with waste being emplaced for the first 35 years. The preclosure period includes approximately 5 years for decommissioning.

1.3.2 Postclosure Period

The postclosure period starts at the end of decommissioning of the DGR, after the shafts have been sealed and the surface facilities have been dismantled. The site is expected to remain under institutional controls[1] for an extended period after decommissioning, which would prevent inappropriate land use including drilling, deep excavation or disruption of the shaft seals.

Following decommissioning of the repository, institutional controls will be put in place in order to prevent inappropriate land use, including drilling, deep excavation or disruption of the shaft seals. A period of 300 years is assumed over which such controls, including societal memory, are effective, consistent with international practice. Beyond this period, there are no expectations in this safety assessment with respect to any ongoing societal control, monitoring or memory of the site.

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[1] Based on CNSC Regulatory Guide G-320 (CNSC06a), and IAEA Safety Series No. 111-F (IAEA95), institutional controls can be defined as, *"the control of residual risks at a site (by a designated Institution or Authority) after it has been decommissioned."* These controls can include both active measures (requiring activities on the site such as monitoring and maintenance) and passive measures (not requiring activities on the site, such as land use restrictions and markers, as well as societal memory).

CNSC Regulatory Policy P-290 (CNSC04a) requires that, *"the assessment of future impacts of radioactive waste on the health and safety of persons and the environment encompasses the period of time when the maximum impact is predicted to occur."* Therefore, a time period of 1,000,000 years is selected as baseline for the postclosure calculations. This encompasses the period of highest radioactivity, including in particular the decay of C-14, as well as the time frame in which the residual radioactivity drops below that of the overlying rock at the Bruce nuclear site. An assessment of the continued behaviour of the DGR is provided for this time scale, including reasoned arguments for the stability and durability of the geosphere. However, calculated peak impacts (although small) associated with transport in groundwater might not occur for more than 1,000,000 years due to the isolation and containment provided by the repository system. Therefore, some illustrative calculations are extended for timescales in excess of 1,000,000 years.

1.4 Regulatory Context

This section presents the regulatory context for the information presented in this report. It outlines the regulatory requirements set under the Nuclear Safety and Control Act (NSCA) and its associated regulations, as well as the international guidance on safety of a deep geologic repository for radioactive waste.

It is OPG's intention that the DGR project will meet or exceed all regulatory requirements during site preparation, construction, operation and beyond.

1.4.1 Regulatory Requirements

In accordance with paragraph 2(g) of NSCA and paragraph 1(e) of the Class I Nuclear Facilities Regulations (SOR/2000-204) the DGR is a Class 1B nuclear facility.

Under the NSCA, paragraph 26(e) states that, *"subject to the Regulations, no person shall, except in accordance with a licence…prepare a site for, construct, operate, modify, decommission or abandon a nuclear facility".* The following licences are, therefore, required from the CNSC over the life of the DGR:

- Site preparation licence;
- Construction licence;
- Licence to operate;
- Decommissioning licence; and
- Licence to abandon.

The detailed requirements to obtain site preparation and construction licences are described in Section 3 of the General Nuclear Safety and Control Regulations (SOR/2000-202) and in Sections 3, 4 and 5 of the Class I Nuclear Facilities Regulations. Other applicable regulations include the Radiation Protection Regulations (SOR/2000-203), which apply to all nuclear facilities. Uranium Mines and Mills Regulations (SOR/2000-206), while not directly applicable to the DGR because it is not a uranium mine, have been taken into consideration due to similarities of some aspects of the DGR project to a mining project.

In addition to the regulations, a number of CNSC regulatory documents in the following categories are also applicable:

- Regulatory policies, which describe general principles that will be applied by the CNSC in their review;
- Regulatory guides, which set out regulatory expectations; and
- Regulatory standards, which establish regulatory standards.

CNSC regulatory documents applicable to the DGR are listed in Table 1-1.

The primary regulatory guidance setting out CNSC's expectations for the assessment of long-term safety of radioactive waste management is given in the CNSC Regulatory Guide G-320 (CNSC06a).

CNSC Regulatory Policy P-290 (CNSC04a) identifies the need for long-term management of radioactive waste and hazardous waste arising from licensed activities. The principles espoused by CNSC Regulatory Policy P-290 that relate to the need for long-term management include the following:

- The management of radioactive waste is commensurate with its radiological, chemical, and biological hazard to the health and safety of persons and the environment, and to national security; and
- The assessment of future impacts of radioactive waste on the health and safety of persons and the environment encompasses the period of time when the maximum impact is predicted to occur.

The predicted impact on the health and safety of persons and the environment from the management of radioactive waste is no greater than the impact that is permissible in Canada at the time of the regulatory decision.

Table 1-1: CNSC Regulatory Documents Applicable to the DGR Project during Site Preparation and Construction Phase

Document No.	Title
P-119	Policy on Human Factors (CNSC00a)
P-211	Compliance (CNSC01a)
P-223	Protection of the Environment (CNSC01b)
P-290	Managing Radioactive Waste (CNSC04a)
G-129 Rev.1	Keeping Radiation Exposures and Doses "As Low as Reasonably Achievable (ALARA)" (CNSC04b)
G-206	Financial Guarantees for the Decommissioning of Licensed Activities (CNSC00b)
G-217	Licensee Public Information Programs (CNSC04c)
G-219	Decommissioning Planning for Licensed Activities (CNSC00c)
G-221	A Guide to Ventilation Requirements for Uranium Mines and Mills (CNSC03a)
G-224	Environmental Monitoring Program at Class I Nuclear Facilities and Uranium Mines and Mills (Draft) (CNSC04d)
G-225	Emergency Planning at Class I Nuclear Facilities and Uranium Mines and Mills (CNSC01c)
G-276	Human Factors Engineering Program Plans (CNSC03b)
G-278	Human Factors Verification and Validation Plans (CNSC03c)
G-4	Measuring Airborne Radon Progeny at Uranium Mines and Mills (CNSC03d)
G-296	Developing Environmental Protection Policies, Programs and Procedures at Class I Nuclear Facilities and Uranium Mines and Mills (CNSC06b)
S-296	Environmental Protection Policies, Programs and Procedures at Class I Nuclear Facilities and Uranium Mines and Mills (CNSC06c)
G-320	Assessing the Long Term Safety of Radioactive Waste Management (CNSC06a)

Key concepts for long-term management are based on **containment** and **isolation** of the waste, in accordance with the CNSC Regulatory Guide G-320 (CNSC06a). The guide states that, *"containment can be achieved through a robust design based on multiple barriers providing defence-in-depth. Isolation is achieved through proper site selection and, when necessary, institutional controls to limit access and land use".*

Some regulatory requirements from the provincial jurisdiction, in particular Ontario's Occupational Health and Safety Act (OHSA90) and conventional occupational safety standards, including those pertaining to mining aspects of the DGR, are applicable to the DGR workers. Applicable provincial acts and regulations are referenced, as appropriate, later in the PSR.

Guidance on Maximum Acceptable (radionuclide) Concentrations (MACs) is given by other agencies, for example the Guidelines for Canadian Drinking Water Quality which include MACs of radionuclides in water supplies, based on a corresponding dose of 10% of the legal limit for members of the public (HC10). These guidelines also provide criteria for non-radioactive contaminants, as do the Ontario Drinking Water Standards (Reg. 169/03) and the Soil, Groundwater and Sediment Standards (MOE09b).

1.4.2 International Guidance

The DGR project takes into account applicable international guidance, as appropriate.

The development and safety of deep geologic repositories has been the subject of international attention by the International Atomic Energy Agency (IAEA) and the Nuclear Energy Agency (NEA) for many years.

A number of technical documents are available that provide guidance on best international practices with respect both to achieving safety, and on the demonstration of safety. Particular international documents relevant to the development and safety of the DGR are listed in Table 1-2.

Structured approach to safety assessment developed under the IAEA Improvement of Safety Assessment Methodologies (ISAM) program (IAEA04a) has been followed in the DGR program, as suggested in the CNSC Regulatory Guide G-320 (CNSC06a).

Specific guidance on radiation protection criteria and their application for disposal of long-lived radioactive waste has been provided by the International Commission on Radiological Protection (ICRP) in ICRP-81 (ICRP00). This guidance has been taken into account in the CNSC Regulatory Guide G-320, and has been taken into account in the DGR project.

Table 1-2: International Guidance Applicable to the DGR

Document No.	Title
IAEA SF-1	IAEA Safety Fundamentals: Fundamental Safety Principles (IAEA06a)
IAEA WS-R-4	Geological Disposal of Radioactive Waste – Safety Requirements (IAEA06b)
IAEA DS-334	Geological Disposal of Radioactive Waste (draft) (IAEA07)
IAEA DS-354	Disposal of Radioactive Waste (draft) (IAEA06c)
IAEA DS-355	The Safety Case and Safety Assessment for Radioactive Waste Disposal (draft) (IAEA08a)
IAEA SS 111-F	The Principles of Radioactive Waste Management (IAEA95)
IAEA SS 111-G-4 1	Siting of Geological Disposal Facilities (IAEA94)
IAEA-ISAM-1	Safety Assessment Methodologies for Near Surface Disposal Facilities (IAEA04a)
NEA 3679	Postclosure Safety Case for Geological Repositories (NEA04)
ICRP 81	Radiation Protection Recommendations as Applied to the Disposal of Long-Lived Solid Radioactive Waste (ICRP00)

1.5 Safety Objective

According to IAEA guidance (IAEA06b), geological disposal of radioactive waste is aimed at:

- Containing the waste until most of the radioactivity, and especially that associated with shorter-lived radionuclides, has decayed;
- Isolating the waste from the biosphere and to substantially reduce the likelihood of inadvertent human intrusion into the waste;

- Delaying any significant migration of radionuclides to the biosphere until a time in the far future when much of the radioactivity will have decayed; and

- Ensuring that any levels of radionuclides eventually reaching the biosphere are such that possible radiological impacts in the future are acceptably low.

Consistent with the above IAEA guidance and the NSCA (subparagraph 9(a) (i)), the overall **safety objective** of the DGR is:

To provide safe long-term management of low and intermediate level waste without posing unreasonable risk to the environment or health and safety of humans.

1.6 Demonstrating Compliance with the Safety Objective

Conclusions on whether the overall safety objective is met by the DGR can be made by comparing the predicted performance of the DGR with performance criteria based on regulatory requirements. To allow such comparisons, specific design and safety criteria have been established for the DGR, as discussed in Section 1.7.

In addition, long-term safety of the DGR during the postclosure period is judged through how well the following **safety functions** are fulfilled by the repository after decommissioning:

- **Isolation** of the waste away from the biosphere; and

- Long-term **containment** of the waste.

The overall safety objective can be concluded to be met if it can be demonstrated that:

- The DGR provides long-term isolation and containment;

- Preclosure and Postclosure safety criteria are met;

- The DGR system is robust; and

- The DGR can be constructed, operated and decommissioned safely.

To demonstrate that preclosure and postclosure safety criteria are met, the specific safety criteria established for these time periods in Chapters 7 and 8, respectively, are used to judge the results of the detailed safety assessments.

To demonstrate that the DGR system is robust, the intent is to show that:

- Results of the various analyses performed to assess DGR safety and the associated uncertainties show that the DGR system safety is robust; and

- Uncertainty and sensitivity analyses provide a reasonable level of confidence in postclosure safety assessments.

To demonstrate that the DGR can be constructed, operated and decommissioned safely, the intent is to show that:

- The DGR has been designed for safe construction, operation and decommissioning, incorporating good engineering practices and use of known technologies; and

- Experience with facilities similar to the DGR demonstrates a strong operational record.

To demonstrate that the existence of natural analogues provides confidence in DGR safety, the intent is to show that the geosphere can retain gases over very long time periods.

Detailed evidence to demonstrate that the DGR safety objective is met is presented throughout the PSR and summarized in Chapter 14.

1.7 Design and Safety Criteria

Specific criteria have been established for the DGR design and safety, based on either the regulations under the NSCA or guidance from federal and provincial authorities, and international guidance. These criteria are used in DGR design and in confirming conclusions on DGR safety, reached through various assessments during both the preclosure and postclosure periods.

Design criteria are provided in Section 6.1.1. Safety criteria for the preclosure period are provided in Section 7.1.2 and for the postclosure period, in Section 8.1.

1.8 Environmental Protection

Environmental implications of site preparation, construction and operation of the DGR are discussed in detail in the EIS (OPG11a). The prevention, mitigation and accommodation of abnormal operating conditions and credible accident conditions have been considered in the facility design and planned operations.

Continuous improvement of environmental performance will be through the implementation of an Environmental Management System, as described in Section 10.3.

The acceptance criteria for radiological and non-radiological protection of non-human biota are described in Chapter 8, Sections 8.1.3 and 8.1.4.

1.9 Strategies Used for the DGR Project

Consistent with international guidance (NEA04), this section presents the strategies used for the DGR project in the areas of project management, site characterization, design and assessments.

1.9.1 Management Strategy

The management strategy is to direct and control development of the DGR by a system of governance and work management both within OPG and within the Nuclear Waste Management Organization (NWMO), contracted by OPG to manage and conduct activities to obtain the site preparation and construction licence, and to design and construct the DGR. The project is committed to ensuring that developing, constructing, operating, decommissioning, and closing the DGR will be carried out in a manner that protects workers, the public and the environment, and meets or exceeds applicable regulatory requirements.

Quality Assurance aspects of the project are described in Chapter 11. The managed system, quality principles, and applicable governing documents during the RA phase are described in the Project Quality Plan (PQP) (NWMO09a). The program for design and construction of the DGR and its organizational structure to manage the DGR design, construction and commissioning are provided in the Design and Construction Phase Management System document (NWMO11a).

Figure 1-4 illustrates the specific studies that contribute towards building a safety case for the DGR project in the RA phase, presented in Chapter 14.

National and international peer reviews are regularly conducted. The groups that conducted peer reviews during the RA phase are shown in Figure 1-5. The project also benefits from the technical knowledge obtained through NWMO's international cooperation agreements (Sweden, France, Finland, Switzerland).

1.9.2 Site Characterization Strategy

The major focus of site characterization for the DGR was confirmation of the geologic setting. The natural characteristics of the site play a vital role in the performance of the DGR. Site feasibility studies and planning of site characterization activities were, therefore, given major consideration in the overall development of the DGR project.

The site characterization results are used as input to repository design and safety assessment, and in building the safety case.

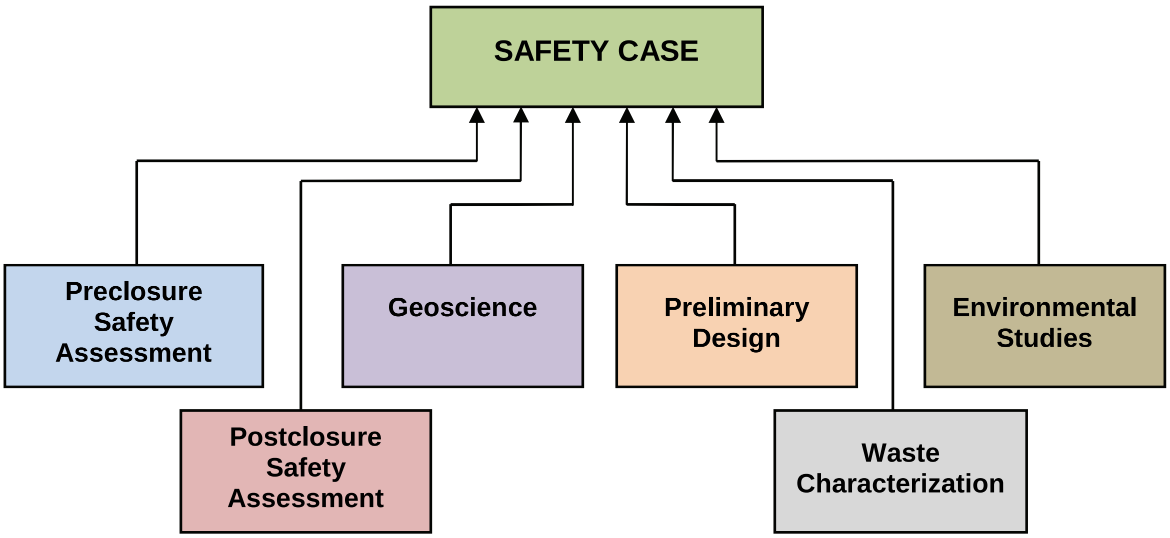


Figure 1-4: Studies Contributing to the Safety Case - RA Phase

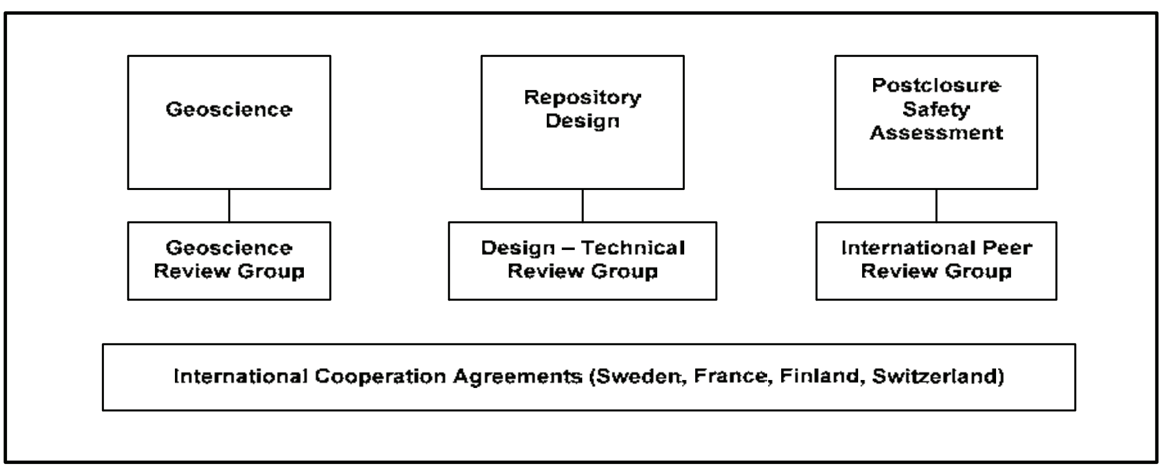


Figure 1-5: National and International Reviews and International Cooperation

The strategy for site characterization was founded upon the following key elements:

- A multi-year phased and iterative site characterization approach for the development, testing and refinement of a descriptive site-specific geosphere model;
- An assessment of internationally accepted site-specific geoscience attributes relevant to understanding technical site acceptability;
- Review of the Geoscientific Site Characterization Plan (GSCP) by federal authorities (CNSC and Geological Survey of Canada) and peer review by the independent Geoscience Review Group (GRG);
- Integration of the GSCP with ongoing regional geologic and hydrogeologic studies in south-western Ontario relevant to assessing concepts of long-term DGR safety;
- Use of scientific visualization techniques to improve transparency and traceability of multidisciplinary data interpretation and, hence, the ability to communicate GSCP results to stakeholders;
- Initiation of complementary geoscience analogue studies to assist with the explanation of geoscience phenomena related to, and to enhance confidence in, the understanding of long-term DGR safety;
- Direct inclusion of international geoscience site characterization experience in investigating deep sedimentary formations for long-term radioactive waste management purposes;
- Participation in various international fora focused on development of geoscience approaches and methods for demonstrating safety of geological disposal in sedimentary formations;
- Selection and scheduling of site characterization activities to optimize achievement of project objectives; and
- Acquisition and archiving of site characterization data following an appropriate quality assurance system, consistent with the PQP for the RA phase.

Site characterization was needed to collect enough site-specific data to design and assess the facility, and to prepare the required documentation for environmental assessment and licensing (CNSC05). Comments from federal authorities were solicited for the GSCP; their feedback during planning and carrying out of site characterization helped build confidence that the data to be collected and the manner in which it is collected would meet regulatory expectations.

The site characterization work was aimed at providing the information necessary to develop a comprehensive descriptive geosphere model that:

- Provides an understanding of the current condition of the site (baseline), its past evolution and likely future natural evolution over the postclosure period;
- Establishes a baseline for detecting potential short-term and long-term environmental impacts caused by the construction, operation and decommissioning of the facility; and
- Provides the necessary information and data to design the facility and perform safety assessments and optimizations for environmental assessment and licensing.

The descriptive geosphere site model will continue to be updated as further information becomes available, including during the construction and operations phases.

1.9.3 Repository Design Strategy

The main elements of the design strategy are:

- Advance design in multiple steps;
- Use of proven technology;
- Safe constructability and operability; and
- Design optimization.

The design has advanced from early conceptual, to conceptual, to its current preliminary stage in discrete steps. Each step was accompanied by safety assessment, and internal and external reviews such as those provided by international experts and the Technical Review Group.

Use of proven technology is made to ensure that DGR Facility design and construction are feasible and consistent with repository designs of a similar type. For example, proven methods for underground construction and well-accepted methods for shaft hoisting and waste package handing will be employed.

Safety is a key consideration in design and construction of DGR. Potential hazards were identified and assessed through all stages of design to date and will continue to be assessed as the design is advanced. Features have been incorporated into DGR design to mitigate hazards and construction methods will be selected to mitigate any hazards associated with construction of the facility.

The CNSC Regulatory Guide G-320 (CNSC06a) requires optimization of the design below the dose constraint. Design optimization has been carried out in several areas, including radiological safety, shaft design and sealing, facility location and layout, configuration of selected waste packages and underground waste package handling.

Repository design is an iterative process and the design continues to evolve based on:

- New data about the site generated during subsurface investigations, for example information related to mechanical stability such as stress magnitudes, orientation and bedrock bedding;
- The results of safety assessment, in particular the preclosure safety assessment and occupational radiation dose ALARA assessment and conventional safety considerations; and
- Further definition of the inventory and categories of waste to be emplaced.

1.9.4 Assessment Strategy

The assessment strategy was to perform detailed analyses that would allow formulation of robust arguments supported by multiple lines of reasoning, facilitated by further detailed assessments and analyses, as appropriate.

The following constitute elements of the assessment strategy:

- The analyses followed regulatory guidance described in Section 1.4;
- Discussions were initiated early with regulatory authorities to ensure that the regulatory expectations were clear, and that there was agreement on the acceptance criteria for radiological and non-radiological protection of humans and the environment;
- Information transfer, for example from the site characterization program for use in safety assessment, was controlled through a data clearance process to ensure the information is current and appropriate for the intended use;
- The analyses are transparent and traceable;
- The interim postclosure safety assessment results were peer reviewed;

- Safety assessment, and subsequently the safety case formulation, followed an iterative process, with the results from each iteration used to guide further development work, as shown in Table 1-3;

- Safety analyses made use of a range of safety and performance indicators; and

- Assessment included analysis of the associated uncertainties in scientific understanding, data or models.

Table 1-3: Iterative Process for the Safety Case Formulation in the Regulatory Approvals Phase

Site Characterization	Inventory	Design	Safety Assessment	Safety Case
Generic data (non-site)	Inventory Report (Draft)	Conceptual Design (2006)	V0 (Dry Run)	Early Draft PSR
Phase I Geosynthesis	Inventory Report (2008)	Conceptual Design (2008)	V1 (Peer Review)	Draft PSR
Phase II Geosynthesis	Inventory Report (2010)	Preliminary Design	V2 for SP&C Licence	PSR

1.10 Structure of the Preliminary Safety Report

The structure of the PSR is as follows:

Chapter 1 ***Introduction -*** An overview of the DGR project and context for the PSR.

Chapter 2 ***Site Description -*** Information on the site and regional environment.

Chapter 3 ***Site Evaluation and Characterization -*** Information on site evaluation and characterization.

Chapter 4 ***Geoscience -*** Geoscience information relevant to establishing the suitability of the site geology for the DGR project.

Chapter 5 ***Waste Inventory -*** Waste and inventory description.

Chapter 6	***Facility Description -*** Description of preliminary design for the DGR Facility.
Chapter 7	***Preclosure Safety Assessment -*** Provides an evaluation of potential radiological and non-radiological impacts on public and the workers due to DGR operation under normal and abnormal operating conditions and credible accident conditions.
Chapter 8	***Postclosure Safety Assessment -*** Provides an evaluation of potential radiological and non-radiological impacts on humans and non-human biota during normal evolution and disruptive scenarios.
Chapter 9	***Site Preparation and Construction -*** Information on how the site will be prepared and DGR constructed.
Chapter 10	***Operational Programs -*** Information on the operational programs that will be in place during DGR operation.
Chapter 11	***Quality Assurance -*** Quality assurance program for the DGR project.
Chapter 12	***Public Information Program -*** The program to keep the public informed and involved in the DGR project.
Chapter 13	***Preliminary Decommissioning Plan -*** The preliminary decommissioning plan for the DGR.
Chapter 14	***Conclusions -*** Overall conclusions on how the DGR meets its safety objective; the safety case for the DGR for demonstrating why emplacement of L&ILW in the DGR is considered to be a safe long-term management approach.
Chapter 15	***References -*** Provides all references used in the PSR.
Chapter 16	***Special Terms -*** Includes units, abbreviations and acronyms, and glossary of terms.
Chapter 17	***Engineering Drawings***

2. SITE DESCRIPTION

This chapter provides a summary of the physical, biological, and social environments within the areas near the DGR. The physical environment is characterized in terms of the atmospheric, aquatic, terrestrial, and social and economic environments (the geophysical environment is more fully discussed in Chapter 4). The biological environment is characterized in terms of terrestrial and aquatic habitats and the biota those habitats support. The social environment is characterized in terms of land use, socio-economic conditions, physical and cultural heritage and Aboriginal interests.

2.1 Site Location and General Description

2.1.1 DGR Site Location

The DGR site is located on the Bruce nuclear site. The Bruce nuclear site is located in the Municipality of Kincardine about mid-way between Kincardine and Port Elgin, at a longitude of 81°30' west and latitude 44°20' north, on the eastern shore of Lake Huron. The Bruce nuclear site is located on the Douglas Point promontory, a feature of relatively low relief that juts 2.5 to 3.0 km into the lake over a lateral distance of approximately 5 km between Inverhuron Bay to the southwest and Baie du Doré in the northeast.

The DGR Facility will be located within OPG-retained lands on the Bruce nuclear site, identified in Figure 2-1. The DGR surface buildings and infrastructure and underground facilities are situated approximately 1 km inland from Lake Huron, about 2 km from Bruce Nuclear Generating Station (NGS) A and 1.6 km from Bruce NGS B. There are no other major rivers or lakes in the vicinity of the site.

The DGR site is adjacent to and north of the existing WWMF. The location of the DGR site, within the secured Bruce nuclear site, ensures that access is security-controlled at all times.

2.1.2 DGR Site Geology

The geology at the site is favourable to locating a L&ILW DGR Facility, and is described in Chapter 4. A number of characteristics of the site contribute to its suitability for the long-term management of L&ILW. These characteristics have been identified and verified through site studies and technical research.

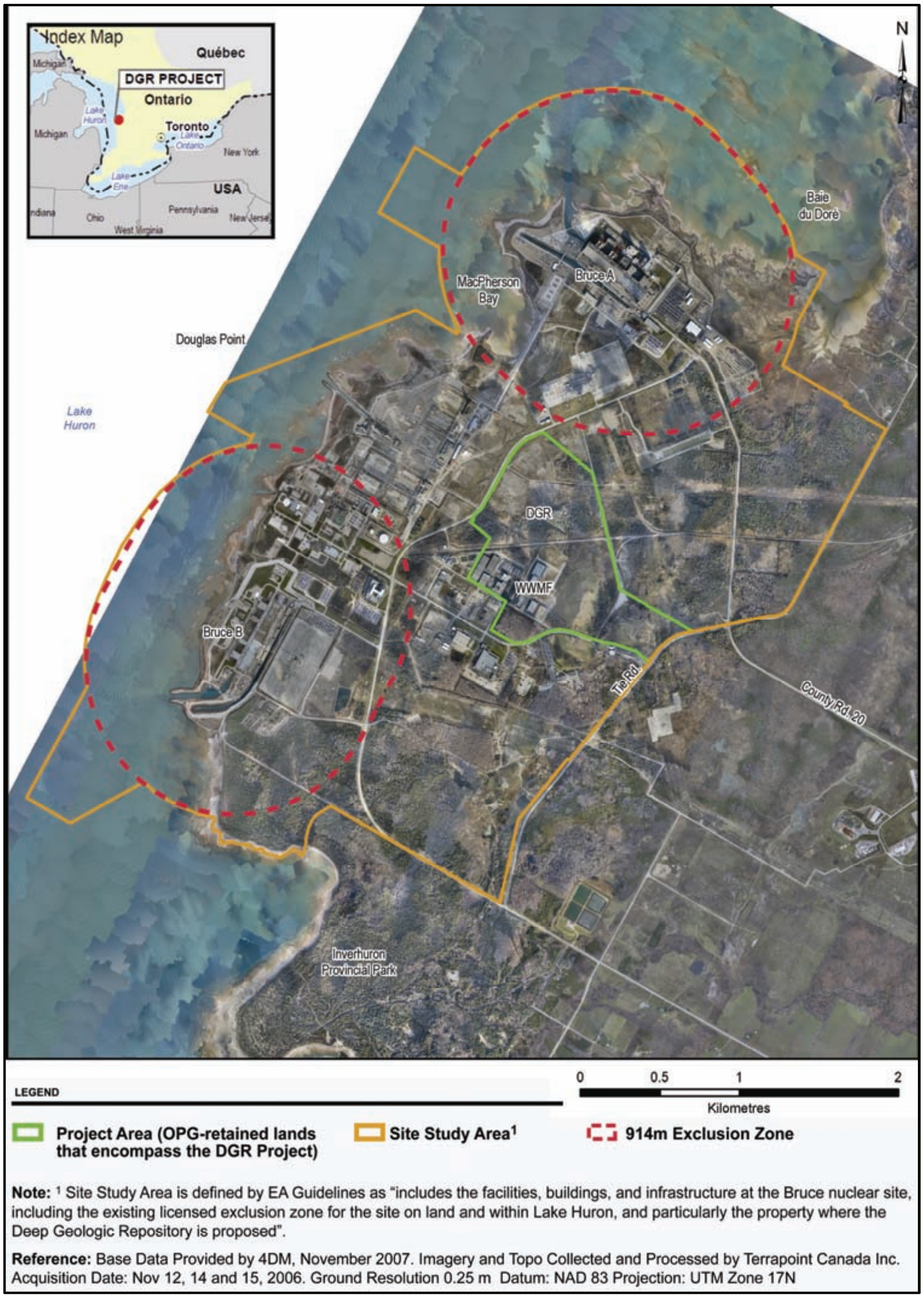


Figure 2-1: Location of DGR Site Relative to Bruce NGSs A and B Exclusion Zones

With respect to the geologic setting, the Bruce nuclear site is located in the eastern periphery of the Michigan Basin, on the northwestern flank of the Algonquin Arch (see Figure 4-1). The Arch represents a crystalline basement high within the North American craton, separating the Michigan and Appalachian sedimentary basins (ARMSTRONG10). The Algonquin Arch trends in a northeastern/southwestern direction and is covered with a thin sequence of Paleozoic sedimentary rocks, which dip away from the arch axis and thicken towards the adjacent basins. The Michigan Basin, where the DGR site is located, is classified as an intracratonic basin, displays a quasi-circular geometry with a diameter of approximately 400 km, and contains over 4 km of Paleozoic sediments at its deepest point in central Michigan (HOWELL99).

Water quality in shallow aquifers will be protected by the 200 m thick shale cap rock located directly above the DGR horizon. This layer hydrogeologically isolates the shallow water supply aquifer and protects it from the deep saline groundwater system. The deep groundwater is very saline and therefore has no potential as a source of potable water.

The area is seismically stable and is located in a region of very low seismic potential.

2.1.3 Bruce Nuclear Site Topography

The topography in the Bruce nuclear site area is classified as smooth to gently undulating, and the relief varies between elevations of 176 mASL (Lake Huron level) and 195 mASL within areas above the Nipissing Bluff. The Nipissing Bluff is a comparatively low, ancient beach and shoreline bluff eroded by post-glacial phases of Lake Huron at a recessional lake stage below that of the older Algonquin Bluff shoreline.

The Nipissing Bluff face occurs between elevations of approximately 185 and 190 mASL. During this post-glacial lake stage, the Bruce nuclear site was part of a point of land marked by the curving beach lines of the Nipissing Bluff extending to the north and south. Lake Huron subsequently continued to recede to its current level following the development of the Nipissing Bluff.

Although the Bruce nuclear site is located along the shore of Lake Huron, the DGR site is elevated a minimum of 3 m above the recorded highest instantaneous water level and therefore has low risk of flooding. The DGR Facility is located 1 km from shore. A flood hazard assessment concluded there is no potential for lake flooding (NWMO11b).

Within the immediate area of the Bruce nuclear site, the land is flat to gently sloping, with a gradual rise in the easterly or inland direction. Along the shore, there is a narrow strip of beach shingle and some sand.

2.1.4 Historical Context

Archaeological investigations conducted in and around the Bruce nuclear site since the 1950s reveal that the shorelines of Lake Huron and its ancestors have been the focus of intense cultural activity during the past 11,000 years. Burials have been encountered within the sandy landscapes around the Bruce nuclear site.

The OPG-retained lands on the Bruce nuclear site have been assessed several times for the potential to contain Aboriginal heritage resources. An archaeological assessment carried out in 2007 identified no evidence of habitation or burial sites activity on the DGR project site (FITZGERALD09). Two registered archaeological sites are recorded in the Ontario Ministry of Culture's Ontario Archaeological Sites Database within the Bruce nuclear site boundary. These sites are known as Upper Mackenzie (BbHj-6) and Dickie Lake (BbHj-12), and both sites are along the Nipissing Great Lakes strandline-sand dune complex. BbHj-12 has been assigned an Ojibway name, and is referred to as Jiibegmegoong (Spirit Place). Neither site is located in the OPG-retained lands that encompass the DGR.

Euro-Canadian presence on what is now the Bruce nuclear site was sparse between the 1850s and 1960 primarily because of the wetland conditions and physiography of the landscape. Early historical records indicate that in the late 19th and early 20th centuries the site was used for farming. Ruins of Euro-Canadian homesteads can be found on the site.

The ruin of a lime kiln is located approximately 200 m southwest of the Douglas Point NGS immediately above the active Lake Huron shoreline. Quicklime was an essential 19th century product used for building, disinfecting and in agriculture.

Prior to municipal amalgamation, the Bruce nuclear site was in the Township of Bruce. The interests of this township, settled in the 1850s, have been, and are, chiefly of an agricultural nature (ROBERTSON06). While rock salt has been mined continuously since 1959 at depths approaching 530 m near Goderich to the south (approximately 60 km from the DGR site), this salt layer has been dissolved and removed through natural geologic processes beneath the DGR site (NWMO11c). There have been no known mining related activities at or around the DGR site that could have an impact on the development of the DGR.

2.1.5 Bruce Nuclear Site Development

The 932 ha Bruce nuclear site has been undergoing development on a continuous basis since the initial clearing of land in 1960 for the building of the Douglas Point NGS. The Bruce property, with the exception of certain OPG-retained lands and lands used by Hydro One, was leased to Bruce Power by OPG in May 2001. The DGR site, as well as the WWMF, is located on a part of OPG-retained lands.

The entire Bruce property is fenced and access to the Bruce nuclear site is restricted and controlled by Bruce Power security personnel. There is a 914 m exclusion zone around each of the Bruce A and B NGSs. The DGR site does not intersect with either of these exclusion zones, as shown on Figure 2-1.

Within the Bruce nuclear site boundaries, existing land uses consist of activities, structures and transportation access required to operate and support Bruce NGSs A and B, Douglas Point NGS, and OPG's various waste operations. Douglas Point NGS, the first NGS built at the Bruce nuclear site, has ceased operations. Another former large facility on-site, the Bruce Heavy Water Plant (BHWP), has been shut down and decommissioned.

With the start-up of Douglas Point NGS, a small radioactive waste storage site, Radioactive Waste Operations Site 1, was established. With the expansion of the nuclear power program in Ontario in the 1970s, more storage space was required, land was reserved and a second larger storage site, Radioactive Waste Operations Site 2, was established in 1974. Additional storage structures and processing systems have been added over the years. In 2001, the name was changed from Radioactive Waste Operations Site 2 to WWMF.

OPG receives, processes by incineration or compaction where appropriate, and stores L&ILW at the WWMF. OPG also receives used fuel from the Bruce NGSs, and processes and stores the used fuel in dry storage at the WWMF, as described in more detail in Section 2.1.6.

The Bruce nuclear site is considered a "disturbed site", having been the site of construction activities and nuclear generating facilities for more than four decades. Structures on the site include a variety of low rise office, warehouse, maintenance, and storage buildings as well as structures designed specifically for the technical functions required for the generation and transmission of electricity. A fire fighting training area and a firing range used by security personnel for training are also located on the site. There is a provincially significant wetland on and immediately east of the site (i.e., Baie du Doré). About one half of the property remains covered with vegetation ranging from open fields to second growth woodland. Some wetland areas, beach communities, and a small alvar also occur within the Bruce nuclear site (NWMO11d). There are no other land uses within the Bruce nuclear site boundary. Inverhuron Provincial Park is located immediately south of the Bruce nuclear site.

The lands along the shoreline to the north and south of the Bruce nuclear site are designated primarily as shoreline development areas. The County of Bruce Official Plan identifies shoreline development areas as the principal areas for tourism and recreation in the County, while providing for limited permanent residential development.

2.1.6 Existing Waste Management Operations

2.1.6.1 WWMF

The WWMF is located adjacent to the DGR. This facility has been receiving radioactive waste for storage since 1974. It was licensed as a L&ILW storage facility until 2002, when the scope of its licence was expanded to include a Used Fuel Dry Storage (UFDS) facility constructed within the WWMF boundary. An aerial view of the WWMF is shown in Figure 2-2. The facility comprises two distinct areas within their own fences, the L&ILW storage area and the UFDS area, as described below.

2.1.6.2 L&ILW Storage Area

The L&ILW storage area consists of various structures such as the above ground low level storage buildings (LLSBs), Amenities Building, Waste Volume Reduction Building (WVRB), Transportation Package Maintenance Building, Refurbishment Waste Storage Building (RWSB), Quadricells, in-ground containers (ICs), trenches and tile holes. These structures are primarily used for storage of L&ILW from the OPG-owned or operated nuclear generating stations. Low level waste is processed at the WVRB.

Both LLSBs and RWSBs are single storey concrete structures with concrete floors. The RWSBs provide interim storage for radioactive wastes (e.g., steam generators, chopped up pressure and calandria tubes, and end fittings) from the refurbishment of NGSs.

2.1.6.3 UFDS Area

The UFDS area is a security-protected area located northeast of the L&ILW storage area, and consists of a Dry Storage Container (DSC) processing building and two DSC storage buildings. Two additional DSC storage buildings have also been approved by the CNSC for construction as required. An additional storage building is built approximately every five years.

2.1.7 Bruce Nuclear Site Environment

Bruce County is located within the Huron-Ontario section of the Great Lakes-St.Lawrence Forest Region. This physiographic region is generally characterized by sugar maple and beech climax forests, often in association with green ash, white ash, yellow birch, wild black cherry, American basswood, and red, white and bur oaks. Eastern hemlock, eastern white pine, and balsam fir are frequently located in drier or upland areas, while eastern white cedar is frequently recorded along swampy depressions.

The Bruce nuclear site occurs within the Alleghanian or Transition Life Zone, which aligns with the northern fringe of the deciduous forest zone. This zone supports fauna and flora from both northern and southern affinities. The local area includes the Huron Fringe woodland, which is a narrow stretch of woodland along the Lake Huron shoreline that features terraces created by glacial Lake Algonquin. Mature forests are a scarce resource near the Bruce nuclear site due to extensive farming. Remnant forests surrounding the Bruce nuclear site are primarily associated with the Lake Huron shoreline, valleys and areas with steep topography, and poorly drained sites.

The Douglas Point promontory, where the Bruce nuclear site is located, is a natural geographic transition point along the whole eastern Lake Huron shoreline. The shoreline configuration changes at Douglas Point from smooth shoreline (to the south) to rough (to the north). There are no major embayments along the whole eastern shoreline of Lake Huron to the south of Douglas Point. Baie du Doré is the first protected embayment, the next one being 40 km north (Chief's Point Bay) at the base of the Bruce Peninsula. The Baie du Doré wetland immediately adjacent to the Bruce nuclear site is a provincially significant wetland, which supports both provincially rare and endangered species, along with fish spawning and rearing habitat.

Using the Ecological Land Classification (ELC) system for southern Ontario (LEE98), the Bruce nuclear site includes 12 broad categories of communities, and 30 specific community-types. These broad categories of vegetation types include alvar, beach, cultural communities such as grasslands and meadows, forest, wetlands and open water, and industrial lands.

The land classifications for the Bruce nuclear site are provided in Figure 2-3. Based on the ELC, there are two wetland features within the OPG-retained lands that encompass the DGR. However, as shown on Figure 2-3, these wetland features are located outside the DGR project site. The swamp located within the eastern boundary of the Bruce nuclear site and extending further east to the former lake shoreline supports a higher diversity of vegetation and provides habitat for deer and waterfowl that utilize the Bruce nuclear site.

1. Low Level Storage Buildings
2. In-Ground Containers
3. Used Fuel Dry Storage Buildings
4. Refurbishment Waste Storage Buildings
5. DGR Site

Figure 2-2: WWMF

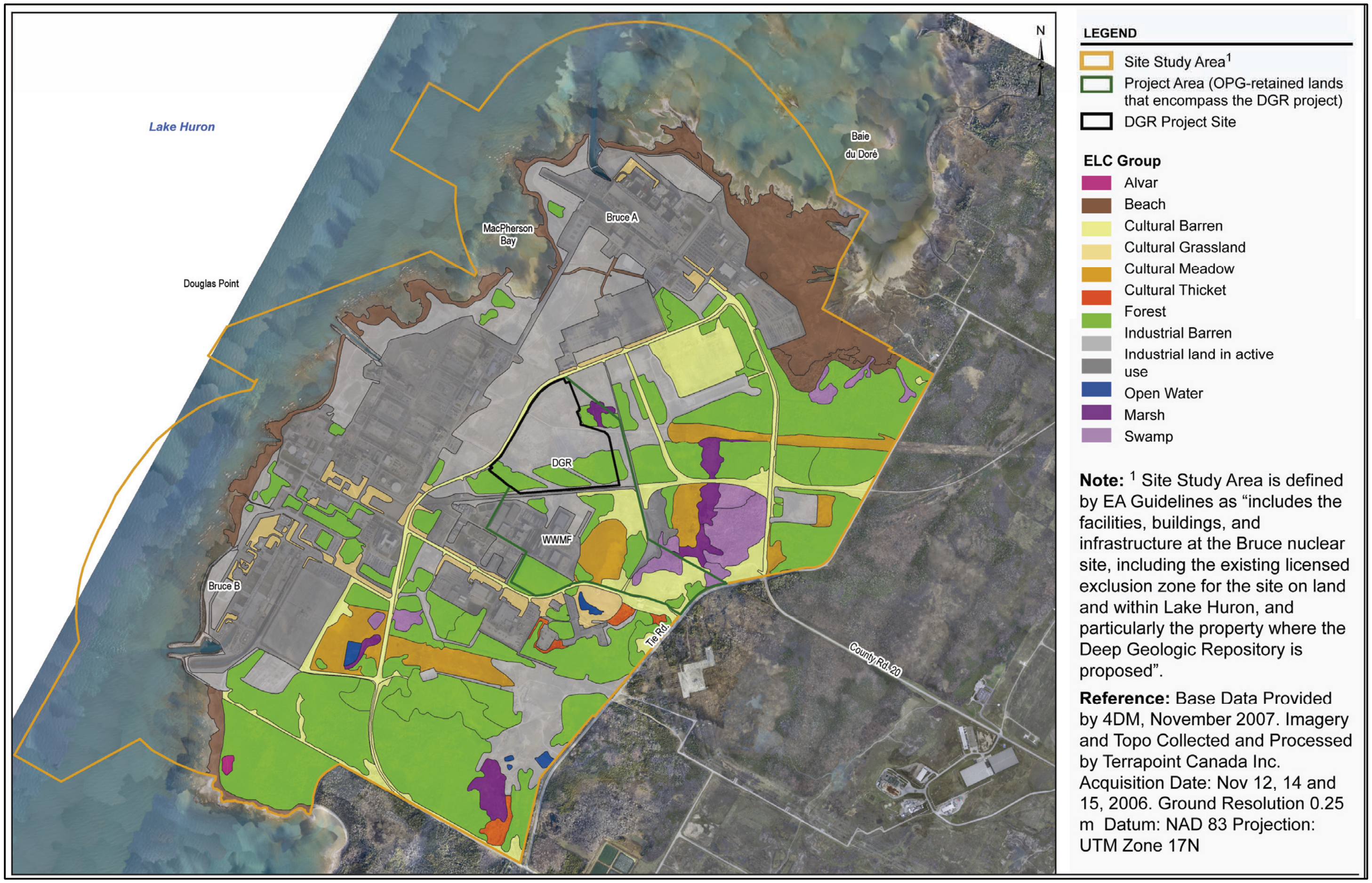


Figure 2-3: ELCs for the Bruce Nuclear Site

2.1.8 Site Accessibility

Tie Road is a two lane north-south rural road under the jurisdiction of the Municipality of Kincardine. It forms the eastern boundary of the property of the Bruce nuclear site and is the primary access to the site. Tie Road can be accessed from the east by either Bruce Concession 2 or Bruce County Road 20 and from the north by Bruce County Road 33 (formerly Lake Range Road). All are paved two lane roads.

Highway 21 is the major north-south highway and is under the jurisdiction of the Ontario Ministry of Transportation. It provides regional north-south access to the Bruce nuclear site from Port Elgin and Kincardine, as well as mid-western Ontario. There are two intersections at Bruce Concession 2 and Bruce County Road 20, both leading to Tie Road.

During construction of the original facilities at the Bruce nuclear site, a rail line was used for delivery of materials and goods. The rail line has been dismantled and is no longer in use.

Under existing conditions, there is no commercial delivery to the site by water. Although the Bruce nuclear site is located on Lake Huron and has potential for development of a deep water harbour, such a facility has not been developed.

Recreational fishing is a popular activity near the once-through cooling water channel; however, there is no public access to the Bruce nuclear site by water. Docks adjacent to the Bruce NGSs A and B are available for emergency access only.

2.2 On-Site Services

The Bruce nuclear site, including OPG facilities, is served by Bruce Power's own internal Emergency Response Team (ERT), medical aid and fire response facilities. In addition, a comprehensive on- and off-site emergency response plan is in place. Response teams have been trained and are equipped to respond to potential emergencies such as personal injury, fire or unplanned releases of radioactivity. The municipal fire department, the Regional Medical Officer of Health and Kincardine's health and safety service providers work co-operatively with Bruce Power to ensure that additional support and response capability is in place.

2.3 Site Security

Access to the Bruce nuclear site itself is strictly controlled by Bruce Power security personnel. A fence surrounds the perimeter of the Bruce nuclear site. The WWMF is surrounded by a separate fence. Access to the existing L&ILW facilities is restricted to qualified personnel and those escorted by qualified personnel. OPG contracts with Bruce Power to provide security for its facilities on the Bruce nuclear site.

Visitors to the site register with Bruce Power security personnel and their vehicles are subject to search prior to entering the site. Visitors who access zoned areas are escorted and must provide photo identification and pass monitoring ports before entering.

2.4 Environment Studies

2.4.1 Introduction

The natural environment of the Bruce nuclear site and the effects of operations on the natural environment have been extensively studied. Since 1997, there have been a number of EAs conducted for various activities related to the waste management operations (OH97, OH98, OPG00a, OPG01a, OPG03a, OPG04, OPG05a, OPG11a) and to activities related to power generation (BP02, BP04, BP05a). The Bruce B NGS Environmental Effects Report (OPG06a) and the Bruce Nuclear Power Development (BNPD) Ecological Effects Review (OPG00b) provide additional information about the effects of operations on the Bruce nuclear site. The BNPD Bioinventory Study (LGL02) provides a very detailed habitat and species summary. Additionally, a number of EA follow-up monitoring programs have been carried out over the past decade (e.g., KINECTRICS05a, GOLDER06, OPG07b, GOLDER10a, GOLDER10b), which contribute to the overall environmental data for the Bruce nuclear site.

In 2007 and 2009 additional baseline environmental monitoring studies were completed specific to the DGR site to update and improve the comprehensiveness of the information. This information is included in the EIS (OPG11a) and its supporting documents. References are provided to these documents, where appropriate, in this PSR.

The OPG-retained lands, including the DGR site and the existing WWMF area are largely developed, industrial lands. The following descriptions of the natural environment generally refer to the undeveloped parts of the OPG-retained lands, nearby areas and to the downstream drainage area.

2.4.2 Radiological and Environmental Monitoring

OPG has in place a number of programs focused on health and safety and environmental protection, which are further described in Chapter 10. The purpose of the programs is to ensure worker and public safety, and protection of the environment. The programs are based on the principles of loss control to manage risks and prevent foreseeable hazards that may result in personal injury, property or equipment damage, process loss, work environment damage, natural environment damage and regulatory non-compliance.

Safety elements include a radiation protection program, occupational radiological risks and occupational non-radiological (conventional) safety management.

Existing environmental monitoring programs at the Bruce nuclear site are conducted both by OPG and Bruce Power. These programs assess compliance with the Nuclear Safety and Control Act (NSCA97) and its associated regulations, applicable federal and provincial legislation and corporate requirements.

In all environmental monitoring programs, the media sampled, the locations, frequency of sampling and the analyses conducted are based on the following objectives:

- Demonstrate that releases of radioactive materials and chemical contaminants are within regulatory limits;
- Verify that assumptions concerning on-site release limits i.e., Derived Release Limits (DRLs) remain valid;
- Permit an estimate of doses to the public resulting from emissions; and
- Provide data to aid development and/or evaluation of models that describe the movement of radionuclides through the environment.

2.4.2.1 Bruce Nuclear Site Radiological Environmental Monitoring Program

The ongoing Radiological Environmental Monitoring Program (REMP) is conducted by Bruce Power to measure environmental radioactivity in the vicinity of the Bruce nuclear site from all site sources. Data from the REMP is used to assess off-site public dose consequences resulting from the operation of nuclear facilities at the Bruce nuclear site.

Radiological environmental monitoring is done at fixed locations surrounding the Bruce nuclear site facilities and at control areas 10 to 20 km from the Bruce nuclear site. Monitoring is conducted for radioactivity in the atmosphere, water, aquatic biota, sediments and terrestrial foodstuff. Monitoring results for 2001-2009 are provided in Table 2-1 based on data presented in the Radiation and Radioactivity Technical Support Document (NWMO11e). Results are less than 1% of the allowable limit.

Table 2-1: Doses from Radionuclides to Critical Groups of the Public

Year	Critical Group	Dose (μSv/y)	Percentage of Dose Limit (%)
2001	Infant at BR1[a]	2.0	0.20
2002	Infant at BR1[a]	2.26	0.23
2003	Infant at BR1[a]	2.08	0.21
2004	Infant at BR1[a]	1.58	0.16
2005	Infant at BR1[a]	1.98	0.20
2006	Infant at BR1[a]	2.45	0.25
2007	Adult at BF14[b]	2.07	0.21
2008	Adult at BR11[c]	2.70	0.27
2009	Adult at BF14[b]	4.41	0.44

Notes:
a. BR1 is represented by a non-farm resident, located on the lakeshore at Scott Point north of the Bruce nuclear site.
b. BF14 is represented by an agricultural, non-dairy farm resident located to the southeast of the Bruce nuclear site.
c. BR11 is represented by an agricultural, dairy farm resident located to the southeast of the Bruce nuclear site near Tiverton.
Radiation and Radioactivity Technical Support Document (NWMO11e)

2.5 Atmospheric Environment

2.5.1 Existing Air Quality

Suspended particulate matter is one measure of air quality. The source may be natural (e.g., wind-blown soil, forest fires) or anthropogenic (e.g., dust from construction or operations activities, stack emissions, transportation). Suspended particulate composition varies with source, location, and season, and normally includes soil particulate, organic matter, sulphur and nitrogen compounds, metals (e.g., lead), and carbon or higher molecular weight hydrocarbons formed by incomplete combustion of fuels. During a 1997 sampling period (OH97), total suspended particulate concentrations measured at the Bruce nuclear site (size range from 0.1 to 100 microns in diameter) were all below the Ontario 24-hour criterion. At an off-site monitoring location at Inverhuron, the suspended particulate criteria were exceeded due to some local activity and construction at the provincial park.

Total suspended particulate concentrations around the Bruce nuclear site were more recently measured during BHWP decommissioning work, both during the pre-demolition phase in 2003 (values ranged from 3.3 μg/m^3 to 67 μg/m^3) (KINECTRICS05b), and in the demolition phase in 2005 (values ranged from 23 μg/m^3 to 75 μg/m^3) (KINECTRICS05c). All values were below the Ministry of Environment (MOE) ambient air criterion (120 μg/m^3 – 24-hour average). Similarly, during December 2003, total dustfall ranged from 0.667 g/m^2 to 1.603 g/m^2, well below the MOE criteria for total dustfall (7.0 g/30 days) (KINECTRICS05b).

Small amounts of radiological emissions and non-radiological air pollutants are emitted to the atmosphere from the operations at the Bruce nuclear site. Sulphur dioxide and nitrogen oxides are also emitted to the atmosphere from the Bruce Power steam plant. Ozone may be generated in the atmosphere from photochemical reactions of nitrogen oxides. Dispersion of pollutants and their subsequent concentrations at the ground surface further depend on weather conditions. However, the concentrations of these pollutants are very small and their effects on local air quality are considered insignificant.

2.5.2 Existing Noise Environment

The nearest noise receptors within the vicinity of the DGR site, excluding on-site receptors, are recreational users of Inverhuron Provincial Park (2.5 km south) and Baie du Doré (2.4 km north). Inverhuron Provincial Park is used for both day access and overnight camping.

Continuous sound level monitoring was conducted at two receptor locations in August of 2001 for a period of about one week. The predominant sound sources at Baie du Doré included wave noise and other sounds of nature such as rustling leaves, insects and birds. The sound environment on the south side of Inverhuron Provincial Park was dominated by sounds of nature and local traffic noise. The lowest daytime one-hour L_{eq} (equivalent continuous noise level) sound levels measured at Baie du Doré and Inverhuron Provincial Park were 38 and 44 dBA, respectively (OPG05a).

The existing noise environment in the vicinity of the Bruce nuclear site has also been assessed using extended long-term noise monitoring. Noise monitoring was carried out at points of reception locations R1 and R2 (see description, below) between May 4 and 11, 2005, with acoustical parameters logged every hour over a continual 182 hours of monitoring. As part of the DGR project EA, long-term noise monitoring at R3 was carried out between May 8 and 22, 2007. The noise monitoring program is described in more detail in Section 5 of the Atmospheric Environment Technical Support Document (NWMO11f). The off-site noise points of reception are located as follows:

R1: Located on Albert Road adjacent to Inverhuron Provincial Park, approximately 3 km from the DGR site;

R2: Located on the north side of Baie du Doré, across from Bruce NGS A, approximately 2 km from the DGR site; and

R3: Located within Inverhuron Provincial Park at an existing camp site, approximately 2 km from Bruce NGS B and 3 km from the DGR site.

The monitoring data indicate that the existing off-site noise levels are reflective of a rural environment (i.e., sound levels are generally less than 50 dBA) and are characterized by sounds of nature (i.e., rustling leaves, waves on the shore of Lake Huron, and birds).

The sound environment at R1 is dominated by sounds of nature. Activities from the Bruce nuclear site were not audible during the daytime and night time site visits. At monitoring site R2, Bruce NGS A was barely audible during field monitoring. The dominant noise sources at this location were water noise on the shore of Lake Huron (i.e., breaking waves) other sounds of nature and traffic noise along Concession 6 and Tie Road. During the field studies, noise from operations at the Bruce nuclear site was barely audible at R3, and was not the dominant noise source at the monitoring location. The dominant noise sources were sounds of nature and water noise on the shore of Lake Huron. Based on measurements recorded during monitoring, it was determined that the minimum existing hourly L_{eq} at the three monitoring locations is between 35 and 37 dBA (NWMO11f).

2.5.3 Meteorology

2.5.3.1 Introduction

The meteorology in the vicinity of the Bruce nuclear site is affected by so-called mesoscale/synoptic and microscale factors. Mesoscale factors include the general circulation of air masses and the effects of the Great Lakes. Microscale factors include lake breeze effects (off-shore/on-shore winds for shoreline areas due to diurnal temperature changes), terrain and topography. These factors affect weather within 10 km (i.e., in the vicinity of the DGR). In the context of nuclear power plants, meteorology near the potential release point is more important.

Wind speed and direction at the Bruce nuclear site are measured continuously at two locations. A 50 m tower located on the Bruce nuclear site measures wind speed and direction at the 50 m and 20 m levels. A 10 m tower located along Concession 4 to the east of the Bruce Power Visitors' Centre measures wind speed and direction (BP09). A summary of only those meteorological parameters that are pertinent to dose calculations is provided, below. These include temperature, wind direction, wind speed and atmospheric stability.

Reported temperature data were measured at a height of 10 m on the 50 m tower for the period of 2005 to 2009. Wind data presented in Section 2.5.4.3 were also measured at a height of 10 m on the 50 m tower over the same period.

Atmospheric Dilution Factors (ADFs) are used to provide estimates of the amount of dilution experienced by a contaminant released into the atmosphere. Data used for the calculation of the ADFs were derived using a 5 year data set from 1998 through to 2002. Meteorological data were based on data collected from the 50 m tower on-site. For ground level releases, ADFs were based on the 10 m data. For fire scenarios with an elevated plume, ADFs were based on the 50 m data.

2.5.3.2 Temperature

The Bruce nuclear site is characterized by warm summers and cold winters. Temperatures recorded at the Bruce nuclear site over the 2005 to 2009 period are given in Table 2-2. For each month of the year, Table 2-2 presents: the lowest temperature recorded each month; the highest temperature recorded each month; and the monthly mean of the daily temperature recorded each month.

Over the 2005 to 2009 period, the records at the Bruce nuclear site indicated a mean annual temperature of 8.0°C, and the highest and lowest temperatures were 31.8°C and -21.0°C, respectively.

Table 2-2: Bruce Nuclear Site Temperature Information (2005-2009)

Month	Minimum Temperature (°C)	Maximum Temperature (°C)	Monthly Mean of the Daily Temperature (°C)
January	-21.0	17.2	-3.3
February	-19.4	10.4	-4.5
March	-18.6	19.8	-0.5
April	-7.8	28.4	6.6
May	-0.2	28.2	11.2
June	3.2	31.6	17.0
July	8.2	30.4	19.3
August	5.8	31.8	19.4
September	2.4	29.4	16.4
October	-1.6	27.0	10.5
November	-9.2	20.8	5.0
December	-14.2	15.8	-1.3
Mean Annual Temperature			**8.0**

2.5.3.3 Wind

The annual windrose, shown in Figure 2-4, illustrates a fairly even distribution of lower wind speeds (up to 10 km/hr) from all directions and a higher frequency of stronger wind speeds (>10 km/hr) from wind directions between south and north-northwest, as well as winds from the north-northeast. Wind speeds and directions from the south through the west-northwest dominate during the fall, winter and spring months (see Figure 2-4) which is consistent with the occurrence of frontogenesis (frontal system and storm formation) during these months. The strongest wind speeds occur during the fall and winter months, while the spring and summer months have more frequent winds from the northeast relative to the fall and winter.

Wind speeds and directions also vary by the time of day. The wind speeds during the night time hours are generally lower than during the daytime hours. There is a prevalence of weaker winds from the east-southeast through the south during the night time hours as opposed to stronger winds from the south through the west during the daytime hours. Table 2-3 provides a summary of wind speed distribution for the period 2005 to 2009.

2.5.3.4 Lake Effects

The proximity of the Bruce nuclear site to the lake affects the local environment and meteorology. In late spring and summer, under clear sky and light wind conditions, strong temperature gradients exist between the land and the lake during the morning and night time. During daytime, when land is generally warmer than water due to solar heating, air above the land rises and is replaced by cooler, more stable lake air causing lake breeze. This process reverses during the night (land breeze) as the land cools to below the water temperature.

In warm seasons, due to solar heating, the air over the land is often 10°C or warmer than that over water. When cold stable lake air flows over warmer land, the resulting upward heat flux gives rise to a Thermal Inversion Boundary Layer (TIBL) in which a stable atmospheric layer will form above an unstable atmosphere near the ground. A one-year study at the Bruce nuclear site (OPG06a) indicated that, on average, TIBLs occurred about 11% of the time.

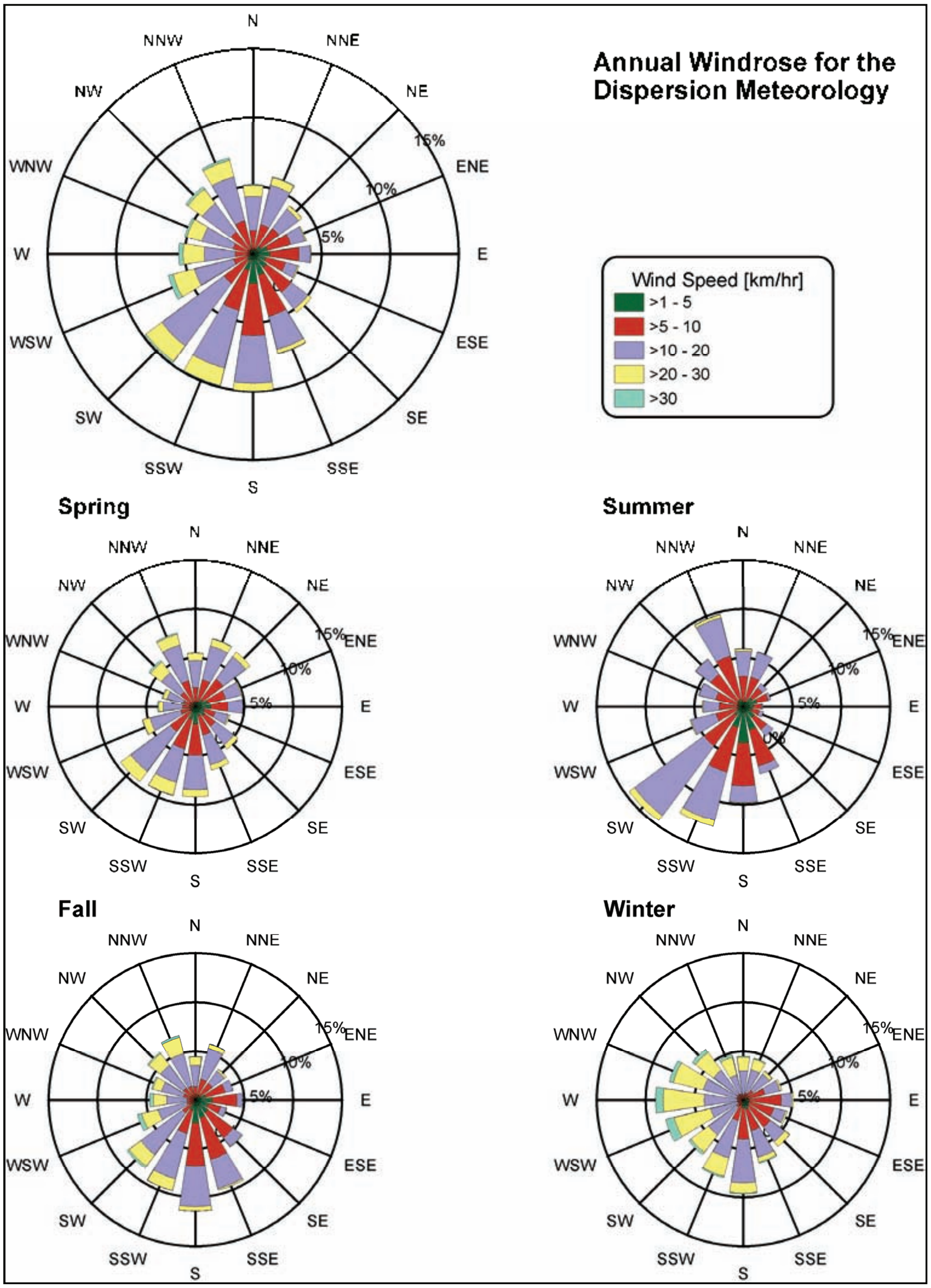


Figure 2-4: Annual and Seasonal Windroses (2005-2009 dataset)

Table 2-3: Wind Direction and Speed Frequencies (%) at Bruce Nuclear Site (2005-2009)

Wind Direction	Wind Speed Range (km/hr)							
	0-10	10-20	20-30	30-40	40-50	50-60	> 60	Total
N	1.67	2.38	0.77	0.05	0.00	0.00	0.00	4.87
NNE	2.14	2.83	0.56	0.04	0.00	0.00	0.00	5.57
NE	2.24	1.69	0.29	0.01	0.00	0.00	0.00	4.23
ENE	2.69	1.04	0.08	0.00	0.00	0.00	0.00	3.81
E	3.20	0.80	0.05	0.00	0.00	0.00	0.00	4.04
ESE	2.32	0.87	0.11	0.00	0.00	0.00	0.00	3.30
SE	3.29	1.47	0.29	0.00	0.00	0.00	0.00	5.05
SSE	4.45	2.37	0.30	0.03	0.00	0.00	0.00	7.15
S	5.65	3.34	0.55	0.02	0.01	0.00	0.00	9.57
SSW	4.03	4.14	1.25	0.05	0.00	0.00	0.00	9.48
SW	2.53	5.41	1.37	0.11	0.00	0.00	0.00	9.43
WSW	1.40	2.81	1.55	0.29	0.04	0.01	0.00	6.09
W	1.24	2.17	1.47	0.27	0.04	0.00	0.00	5.20
WNW	1.45	2.23	1.09	0.15	0.01	0.00	0.00	4.92
NW	1.96	2.46	1.27	0.18	0.00	0.00	0.00	5.86
NNW	2.41	3.15	1.21	0.16	0.00	0.00	0.00	6.93
Total	42.67	39.17	12.20	1.34	0.11	0.01	0.00	95.51

TIBLs restrict the amount of vertical mixing. The one-hour average TIBL heights measured approximately 1 km from the shoreline ranged from 50 m to 315 m (TAM86).

For elevated sources, the plumes initially emitted into the stable on-shore air above the TIBL are entrained into the TIBL upon intersecting the growing TIBL interface and consequently fumigate downward. For ground level sources, the emissions in the unstable boundary layer would often result in higher than expected ground level concentrations during on-shore flows because the stable layer aloft would limit vertical diffusion. However, in safety analyses, it is conservatively assumed that the surface layer is under stable atmospheric conditions so that the ground level emissions will produce high concentrations with no credit given to plume rise except in a fire situation.

2.5.3.5 Atmospheric Stability

The stability of the atmosphere can be described as its tendency to resist or enhance vertical motion in the boundary layer. Three states of atmospheric stability are distinguished according to the vertical temperature profile or “lapse rate”, namely: unstable, neutral and stable atmospheric conditions. Vertical movement is greatest under unstable atmospheric conditions, where the temperature decrease with height is greater than the adiabatic lapse rate of 0.98°C/100 m. An air parcel, which is forced to rise in an unstable atmosphere, will cool adiabatically, and hence remain warmer than the surrounding atmosphere and continue to rise. Similarly, if the parcel is forced downwards, the parcel of air will continue to fall, since it will cool faster than the atmosphere. Unstable conditions tend to enhance the vertical growth of the plume, causing an elevated plume to intersect the ground more rapidly. Unstable conditions are primarily associated with daytime heating conditions, which result in enhanced turbulence levels and enhanced dispersion. Stable conditions are primarily associated with night time cooling conditions, which result in suppressed turbulence levels (poorer dispersion). Neutral conditions are primarily associated with higher wind speeds or overcast conditions (BOUBEL94).

Very unstable conditions occur infrequently (approximately 1% of the time), and correspond with strong convective and thunderstorm activity. Unstable and slightly unstable conditions occur approximately 22% of the time (combined) and include lake effect phenomena such as rain-showers and snow-showers. Neutral stability conditions occur approximately 54% of the time. Stable and very stable conditions account for the remaining 23% of the time.

2.5.3.6 Mixing Layer

The depth of the surface mixing layer is another important dispersion parameter, and determines the region of the lower atmosphere where pollutants can be dispersed vertically. The average convective mixing height is approximately 650 m. Typically, convective activity associated with frontal systems and thunderstorms exhibit higher convective mixing heights, while lake-effect precipitation is characterized by lower convective mixing heights. The absence of night time convective mixing heights is reflective of the rural nature of the land around the Bruce nuclear site. After sunrise, the mixing height continually increases during the day before drastically dropping at sunset. Night time convective mixing depths are usually restricted to urban areas where radiant heating can generate convective mixing.

The high frequency of occurrence of low mechanical mixing heights is indicative of increased frequency of lower wind speeds during the night time hours, an observation supported by the daytime/night time windroses. The frequency of occurrence of the daytime mechanical mixing height increases from the surface to about 200 m and then decreases with altitude. This maximum at 200 m indicates the average height of the surface roughness, a measure of the variations in the height of topographical features and average wind speed.

2.5.3.7 Atmospheric Dilution Factors

ADFs are used to provide estimates of the amount of dilution experienced by a contaminant released into the atmosphere between the emission point and the receptor location. The ADFs are calculated from the measured ambient concentrations and the source emission rates, together with the measured hourly values of wind direction, wind speed, and standard deviation of wind direction.

For the Bruce nuclear site, the ADFs have been calculated for the distance (approximately 750 m) from the WWMF to the Bruce nuclear site main guardhouse where a member of the public may be located. These ADFs apply to the DGR due to its proximity to the WWMF. Conservative assumptions are made on local surface roughness and TIBL conditions for all the calculations.

Pollutant concentrations directly downwind from a source decrease with increased sampling time because of increased meander of wind directions. Thus, the derived ADFs follow this same pattern for constant emission rates. For the Bruce nuclear site, the calculated ADFs based on five years (1998-2002) hourly meteorological data measurements, are grouped into three averaging sampling periods:

Short-term: for the first hour period, defined as the 90th percentile value of the cumulative frequency distribution of the calculated ADFs.

Prolonged-term: between 1 and 24 hour period, defined as the 90th percentile value of cumulative frequency distribution of the calculated ADFs for the worst wind sector, based on 24 consecutive hours of meteorological measurements.

Long-term: for more than 24 hour period, takes into account joint frequency of wind speed, stability class and wind direction over the time period of interest.

The calculated ADFs for the WWMF site, which also apply to the DGR, are shown below. ADFs are calculated using the model suggested by the Canadian Standards Association (CSA) N288.2 (CSA91):

Short-term:		1.6×10^{-4} s/m^3
Prolonged-term:	First hour	1.6×10^{-4} s/m^3
	Between 1 and 24 hours	1.5×10^{-5} s/m^3
Long-term:		4.4×10^{-6} s/m^3

Accident analyses for the DGR included an above ground and an underground fire situation. The ADFs used for the present DGR accident assessment are based on WWMF values (OPG06a).

Above Ground

Short-term	First hour	4.3×10^{-6} s/m^3
Prolonged-term	Next 6 hours	1.2×10^{-5} s/m^3

Underground

Short-term	First hour	1.6×10^{-4} s/m^3
Prolonged-term	Next 6 hours	1.5×10^{-5} s/m^3

2.5.3.8 Severe Weather

Severe weather events in the region generally include thunderstorms and lightning, icing storms, windstorms, extreme heavy precipitation and fog.

Thunderstorms require low level, warm, moist air, which, when lifted, will release sufficient latent heat to provide the buoyancy necessary to maintain its upward movement in an extremely unstable atmosphere. Thunderstorms produce lightning and on occasion, a tornado. In southern Ontario, thunderstorms normally occur 20 to 25 days a year (PHILLIPS90). The frequency of thunderstorm occurrence at the Bruce nuclear site is expected to be similar to that at Wiarton Airport, the location of the nearest meteorological station that records thunderstorms. For the period 1961-1990, Wiarton Airport averaged 28 thunderstorms per year (EC10a). The period from 1961-1990 represents the most recent timeframe for which published data is available at the time of writing.

Environment Canada has developed a flash density map indicating the number of lightning flashes per square kilometre per year. Extreme southwestern Ontario experiences a large area of increased lightning activity (3.0 to 5.0 flashes per square kilometre). A second maximum is located along a line from the southern tip of Georgian Bay to southeast of Barrie (2.5 to 4.5 flashes per square kilometre). The Bruce nuclear site has an average of 2.0 to 3.0 flashes per square kilometre for the period 1999-2008 (EC10b).

Icing storms, including freezing rain and ice pellets, are associated with an elevated inversion with a maximum temperature above 0°C overriding lower subfreezing air. Freezing rains occur in southern Ontario, on average, 25 to 50 hours per year (PHILLIPS90). They are usually accompanied or followed by precipitation such as snow, wet snow, ice pellets, rain and fog. For the 1961-1990 period, freezing precipitation occurred 9 days per year on average at Wiarton Airport (EC10a).

The most severe windstorm is a tornado. Tornadoes most often occur along squall lines of a tropical cyclone (low pressure centre), in conjunction with cumulonimbus clouds and severe thunderstorms. The average tornado in southern Ontario has a diameter of between 150 to 600 m, and typically travels at a speed of 50 to 70 km/hr in a southwest to northeast direction (PHILLIPS90). Tornadoes normally touch ground for less than 20 minutes. The Bruce nuclear site lies north of the main tornado corridor in southern Ontario. One to two tornadoes per 10,000 km^2 can be expected annually in the southwestern Ontario region that includes the Bruce nuclear site (PHILLIPS90).

Heavy precipitation is usually associated with extra-tropical cyclones, intense convection and thunderstorms. Heavy precipitation is the main cause of flooding and landslide. Extreme rainfall events in Ontario have produced rainfall amounts ranging from 250 mm over a 9 hour period in Peterborough to 450 mm over a 30 hour period in

Harrow. The Probable Maximum Precipitation (PMP) is defined as the greatest depth of precipitation for a given duration that is physically possible over a given size storm area at a particular geographical location at a particular time of year (WMO86). The PMP for the DGR site area was estimated using several methods. In the event of a PMP event occurring across the DGR site, there is potential to generate a flood level in excess of 186 m, and the maximum water surface elevation was estimated to be about 186.86 m (i.e., maximum 86 cm Probable Maximum Flood (PMF) level) at a number of locations around the DGR site (NWMO11b).

High water level wave setup and wave uprush scenarios were used to assess the potential nearshore wave propagation. Considering potential maximum inundation or horizontal extent, the extreme prediction results in wave influx to a distance of approximately 500 to 550 m inland. This is well-removed from the DGR site. It is concluded there is no potential for lake flooding (NWMO11b).

Tsunamis are long period gravity waves generated by seismic disturbances of the sea bottom or shore, or landslides resulting in a sudden displacement of the water surface with the resulting wave energy spreading outwards across the ocean or lake at high speed. Tsunami occurrences in Canada are rare, with the Pacific coast at greatest risk due to the high occurrence of earthquake and landslide activity. No probable or definite tsunamis have been recorded for Lake Huron (NWMO11b).

2.6 Aquatic Environment

2.6.1 Lake Huron

Lake Huron is the major water body near the Bruce nuclear site. The lake is the second largest of the Great Lakes, with a surface area of 59,596 km^2 and a shoreline length of 6,157 km. The surface of Lake Huron is nominally 176 mASL. The average depth is 59 m, while the maximum depth is 229 m. Approximately 40% of Lake Huron's waters are less than or equal to 40 m deep, and are located in the shallows of Georgian Bay, North Channel in the north, Saginaw Bay in the south and a narrow band along the entire perimeter of the lake.

The Great Lakes water levels have fluctuated throughout their history. Levels of Lakes Michigan and Huron, for example, reached record highs in both 1886 and 1986. Monthly mean lake levels range from 176.3 to 176.6 m or 0.3 to 0.6 m above the chart datum of 176 m based on the International Great Lakes Datum 1985. The historical maximum (October 1986) of 177.5 m is 1.5 m above chart datum. The maximum over the past 10 years (July and August 2009) of 176.44 m is 0.44 m above chart datum. The minimum over the past 10 years (July and August 2009) of 175.68 m is 0.32 m below chart datum. Lakes Michigan and Huron's record low water levels coincided with climatic events such as the Dust Bowl of the 1930s, a multi-continental severe drought of 1964 (which is the record low for the two lakes), and the most recent and strongest El Niño on record of 1997 (NOAA08). Future lake levels are uncertain.

Although there are extensive networks of small rivers and creeks feeding into Lake Huron in the region, there are no major rivers near the Bruce nuclear site. The nearest river is the Little Sauble, a small river. There are two small east to west drainage courses entering the lake adjacent to the Bruce nuclear site. Underwood Creek empties into the Baie du Doré to the north. The Little Sauble River, which forms the southern boundary of Inverhuron Provincial Park, empties into Inverhuron Bay to the south. To the west and northwest, Lake Huron stretches uninterrupted for approximately 128 km. The nearest land across the lake is Port Hope, Michigan, United States, approximately 98 km southwest of the Bruce nuclear site.

Lake Huron is used locally for sport and commercial fishing, as well as recreational swimming and boating. The modestly warmer waters from the once-through cooling water discharges from the Bruce NGSs A and B provide year round sport fishing opportunities. The Baie du Doré wetland adjacent to the Bruce nuclear site provides habitat suitable for warm-water and cool-water fish spawning and rearing.

In general, water depths in the nearshore zone of the lake range from 6 to 20 m, except in Baie du Doré where depths do not exceed 5 m. Bedrock substrate predominates in the shallow areas of the open shoreline, grading to a mixture of pebble, cobble and boulder in the 7 to 12 m depth range.

Nearshore currents in Lake Huron have been measured during the ice-free period since the early 1970s. Current direction in the region is predominantly parallel to the shoreline with a northeastern direction being the most common. Currents to the southwest also occur but on a less frequent basis (BP05b).

2.6.2 Lake Huron Ice Conditions

Ice normally begins to form in harbours and shallow water areas in early December with ice fields and concentrated brash forming in early January. The central part of Lake Huron is mainly an open water area, but drifting patches of thin ice may be present from early February until mid-March. Annual maximum ice coverage ranges from 45 to 79% (ASSEL03). The shallow areas of the lake (less than 40 m deep) typically have extensive ice cover every winter.

2.6.3 Bruce Nuclear Site Surface Water

The Bruce nuclear site is located within a small local watershed (Stream C) bounded by the Little Sauble River watershed to the south and the Underwood Creek watershed to the north. Natural drainage enters the Bruce nuclear site via Stream C, a former tributary of the Little Sauble River that was diverted to Baie du Doré during the initial development of the Bruce nuclear site in the 1960s. All of the surface drainage from the Bruce nuclear site is directed to Lake Huron through several constructed outfalls and drainage ditches.

Drainage from most of the OPG-retained lands in area of the DGR site is carried through a ditch to Lake Huron via MacPherson Bay. A small portion of this area drains to Baie du Doré via the railway ditch on the north side of the abandoned rail bed (north railway ditch) and Stream C. Another railway ditch, located on the south side of the abandoned rail bed (south railway ditch), drains the WWMF site. Both ditched channels along the abandoned rail bed run from west to east, carrying drainage in an easterly direction, parallel to the abandoned rail bed. Both ditches drain into Stream C approximately 500 m west of the DGR site. The site drainage is shown on Figure 2-5. Stream C drains to Baie du Doré.

The on-site drainage ditches have become naturalized over time, with cattails dominating most of the length. The root structure of the cattails provides highly stable ditch beds. The physical presence of the cattails also serves to reduce water velocity, which minimizes ditch bed erosion and increases the rate of settling for suspended sediments that may enter the ditch system. Natural herbaceous vegetation, trees and shrubs that have established themselves over the years have stabilized the side slopes of the ditches.

There is a small marsh located to the northeast of the DGR site, and a seasonal swamp is located to the southeast of the WWMF and the DGR site. These features were introduced in Section 2.1.7, and are shown on Figure 2-3.

2.6.4 Aquatic Habitat and Biota

Aquatic habitat reconnaissance and fish identification was conducted for the DGR site in 2007 and 2009. The results of these studies are described in detail in Section 5 of the Aquatic Environment Technical Support Document (NWMO11g).

Stream C is designated by Fisheries and Oceans Canada as coldwater fish habitat, and supports a fish community composed of an assemblage of coldwater and warm-water species including brook trout, rainbow trout, brown trout, various sucker species, and cyprinid species including spottail shiner (NWMO11g).

As described above, the north and south railway ditches are located adjacent to the DGR site, and both flow toward Stream C. The dominant macrophyte that occurs in these ditches is cattail. Other macrophytes occur along the edges of these ditches, and in areas where dredging has recently been carried out.

Fish community investigations conducted in the south railway ditch in June 2000 indicated that it supports a warm-water baitfish community including bluntnose minnow, fathead minnow, northern redbelly dace, central mudminnow, brassy minnow and brook stickleback (OPG05a). The 2007 and 2009 DGR field program confirmed these earlier observations. These fish represent a mix of species that are typical of warm-water creeks and wetlands, and are known to be tolerant of a wide range of environment conditions (SCOTT98). These species are common and widespread throughout central and southern Ontario. Aquatic invertebrate life in the south railway ditch includes leeches and snails, as well as aquatic crayfish (NWMO11g). Semi-terrestrial burrowing crayfish have also been identified in wetted areas that occur on the DGR site. These crustaceans are discussed further in Section 2.7.3.2.

It was noted during field investigations in 2007 and 2009 that the north railway ditch does not contain enough water for any length of time to support fish or fish habitat. Previous studies by the Saugeen Valley Conservation Authority (SVCA) did not classify this ditch as fish habitat. The south railway ditch has, however, been classified by the SVCA as fish habitat since some areas of open water occur during low flow/dry conditions.

2.7 Terrestrial Environment

The 932 ha Bruce nuclear site contains cleared service areas with fenced facilities and slightly rolling undisturbed areas with second growth woodland outside the fences.

2.7.1 Vegetation Communities and Species

A large portion of the DGR site was used as a construction laydown area during the construction of the original Bruce NGSs. More recently, it has been used as a site for storing clean fill material from excavation at other locations on the Bruce nuclear site.

An ecological land classification, using the ELC system for southern Ontario, was conducted in 2001 for the Bruce nuclear site (LGL02). The ecological land classification mapping was updated in 2007. The results of this mapping are shown on Figure 2-3, and are described in more detail in Section 5 of the Terrestrial Environment Technical Support Document (NWMO11d). The broad categories of vegetation found within the Bruce nuclear site include alvar, beach, cultural barren, cultural grassland, cultural meadow, cultural thicket, forest, industrial barren, industrial lands, marsh, swamp, and open water.

Note: [1] Site Study Area is defined by EA Guidelines as "includes the facilities, buildings, and infrastructure at the Bruce nuclear site, including the existing licensed exclusion zone for the site on land and within Lake Huron, and particularly the property where the Deep Geologic Repository is proposed".

Reference: Base Data Provided by 4DM, November 2007. Imagery and Topo Collected and Processed by Terrapoint Canada Inc. Acquisition Date: Nov 12, 14 and 15, 2006. Ground Resolution 0.25 m Datum: NAD 83 Projection: UTM Zone 17N

Figure 2-5: Site Drainage

Cultural and industrial communities predominate the lands in the area of the DGR. Just less than 63% of the area is in active industrial use or in barrens that have been created by past clearing and/or grading and filling. The extent of anthropogenic activity is considerable and even the naturally occurring vegetation has been affected by past human activity. Fill has been placed in some areas and mounded in others. For example, the old-field type meadow immediately east of the WWMF is established on a closed landfill site. At this location, the plant community represents a combination of postclosure seeding with a cover-crop mix of agricultural species and invasion of the area by colonizing species suited to the local soil, moisture and climatic conditions. The small marsh on the north side of the DGR site appears to be established in ditches that may have had some past drainage function but are presently isolated by fill surrounding the trenches and adjacent low-lying lands.

For OPG-retained lands that encompass the DGR that are not under industrial use or an industrial barren, just over 20 ha (43%) are occupied by cultural plant community types and just under 28 ha (57%) support naturally-occurring plant community types. Approximately 24 ha (86%) of the naturally-occurring vegetation is forest. As noted, small marsh and swamp areas are also present.

The woodlands are relatively young and are fragmented into 12 separate units. They are strongly influenced by white-tailed deer browse. Woodlands represent a total of nearly 25% of the OPG lands that encompass the DGR. Most of the woodlands on the site are dominated by eastern white cedar. Minor components of balsam fir and white birch are also present. Trembling aspen and red maple occur as scattered trees or small patches at the woodland edges. The understory is relatively sparse and patchy. Ground cover is sparse and varies greatly from stand to stand. Few plants are present where the cedar canopy is dense.

The various industrial and cultural barrens that occupy most of the OPG-retained lands that encompass the DGR appear to be areas in which some historical grading and movement of fill has occurred. Bare ground is prevalent and plants occur as sparse, scattered individuals or as small clusters. Few scattered tree stems of white birch, white spruce, white pine, balsam poplar and eastern white cedar occur. The vast majority of the shrubs and herbs that are present are colonizing species including some that are typical of shoreline colonizing species (e.g., silverweed) since the drainage is bedrock controlled.

Although more than 500 species of vascular plants occur in the vicinity of the Bruce nuclear site, only a modest subset of that number occurs in the area of the DGR site. The OPG-retained lands that encompass the DGR have been affected by anthropogenic factors and as a result, fewer habitats are present. These habitats are also smaller in size (area) than the habitats that have been documented within the larger Bruce nuclear site. For the lands that encompass the DGR, a total of 181 taxa of vascular plants have been identified, including 16 species of trees, 19 species of shrubs and woody vines, 5 species of ferns and fern allies, 50 graminoids (plants with grass-like leaves) and 91 forbs (all herbaceous flowering plants, excluding graminoids). The non-native component of the local flora is just over 34%, a value that is slightly above the provincial average of 28.3%, reflecting, in part, the anthropogenic disturbance that has occurred on the site.

2.7.2 Wildlife Habitat

The wildlife habitat functions of the remnant woodland units within the Bruce nuclear site are limited by their small size, high degree of fragmentation, and disturbed nature. These areas are capable of supporting wildlife species that are not dependent on forest interior; however, they may be part of habitat areas used by wildlife with larger territorial ranges (e.g., wild turkey and white-tailed deer). The lands that encompass the DGR have been extensively modified through the placement of fill, limiting the availability of topsoil. The site does not provide good habitat for burrowing mammal species, and the stony nature of the soils limits the growth of herbaceous groundcover in some of the more open habitats. Networks of small naturalized ditches that are intermittently wet provide corridors for wildlife movement.

The railway ditches, which traverse the OPG-retained lands that encompass the DGR adjacent to the north side of the WWMF, provide one the largest of these naturalized corridors. Although riparian vegetation is limited along the length of the railway ditches, they are populated by a variety of typical emergent and submergent vegetation, dominated by cattails. A variety of herptiles (e.g., green frog and northern watersnake) and small mammals (e.g., muskrat) are regularly recorded using these areas (NWMO11d). Additionally, vernal ponds within the Bruce nuclear site provide habitats that are used by amphibians during various life history phases. Within the OPG-retained lands that encompass the DGR, northern leopard frog egg masses have been recorded (KINECTRICS05a).

2.7.3 Fauna

2.7.3.1 Amphibians and Reptiles

Amphibian surveys undertaken during May 2001 on the Bruce nuclear site and associated lands recorded a total of 9 amphibian and 8 reptile species (LGL02). Spring peeper and American toad are the most commonly recorded amphibian species found on the Bruce nuclear site. Within the DGR site and WWMF area, green frog and northern leopard frog were observed (OPG05a).

Field studies undertaken at 13 locations on the Bruce nuclear site in spring 2007 and 2009 reinforced the historical findings. Spring peeper, northern leopard frog, chorus frog, gray treefrog, American toad, and green frog were identified as actively breeding within the Bruce nuclear site. Spring peeper and chorus frog were identified as actively breeding within the lands that encompass the DGR. A new species for the Bruce nuclear site, western chorus frog, was added to the species list in 2009. Breeding activity was found to be most common in wetland areas within the Bruce nuclear site with the greatest amount of surface water.

Basking turtle surveys, based on reconnaissance of areas of open water and wetlands within the Bruce nuclear site, were completed in 2009. Thirty individuals were recorded over the course of the field season: 29 midland paint turtles and one common snapping turtle. The surveys indicated the preferred basking turtle habitat on the Bruce nuclear site occurs in the pond located near the landfill in the southeastern corner of the site. Few basking turtles were recorded using the OPG-retained lands that encompass the DGR.

2.7.3.2 Burrowing Crayfish

Chimneys of terrestrial burrowing crayfish have been documented on the Bruce nuclear site and in the area of the DGR. In 2006, the presence of two burrowing crayfish species was documented at the Bruce nuclear site, Baie du Doré and MacGregor Point Provincial Park (GOLDER06). During field studies in 2007, these results were confirmed as chimneys of burrowing crayfish were documented within the railway ditches as well as other areas of the OPG-retained lands that encompass the DGR. The burrows of these species of crayfish are found in wetlands, roadside ditches, and creek banks where moist soils with clay content occur.

2.7.3.3 Mammals

A Bioinventory Study of the Bruce nuclear site was undertaken in 2000-2001, and identified 15 mammal species (LGL02). With the exception of the Virginia opossum (which is a species found in southern areas of Ontario and in the United States), observations did not reveal the presence of unusual or significant wildlife species. Evidence of star-nosed mole, groundhog, eastern chipmunk, racoon, and white-tailed deer, beaver, muskrat, and water shrew have also been recorded in various historical studies (OPG05a).

Incidental observations of mammals within the Bruce nuclear site as part of field studies undertaken in 2007 included beaver, eastern cottontail rabbit, coyote, grey squirrel, snowshoe hare, striped skunk, weasel and white-tailed deer. Most mammals were observed in the wooded area at the southwest corner of the Bruce nuclear site, adjoining Inverhuron Provincial Park. Incidental observations within the OPG-retained lands that encompass the DGR included several occurrences of white-tailed deer in the wooded areas north of the railway ditches and southwest of the WWMF toward Tie Road, and two striped skunks south of the WWMF (NWMO11d). In recent years (e.g., 2006, 2009), transient individual American black bears have been observed on the Bruce nuclear site.

Surveys were undertaken in 2009 that specifically focused on the use of the Bruce nuclear site by white-tailed deer. A late fall aerial survey was undertaken, which recorded a single buck within the perimeter fence of the Bruce nuclear site. Additional individuals were recorded in nearby agricultural fields, suggesting there are corridors throughout the Huron Fringe, including the Bruce nuclear site, used by this species.

A meadow vole trapping program was also established in 2009 on the OPG-retained lands that encompass the DGR to better understand the use of potentially suitable habitat units by this small rodent, and possibly other similar mammals. No meadow voles were captured; however, both northern short-tailed shrews and deer mice were found in the traps. All captured specimens were adults.

2.7.3.4 Avian Species

The bioinventory of the site carried out in 2000-2001 identified 83 species of birds as having potential for breeding within the Bruce nuclear site (LGL02). Approximately 40 species were identified as having breeding potential within the OPG-retained lands that encompass the DGR, including one species confirmed as breeding. Mainly forest species, such as red-eyed vireo, blue jay, and black-capped chickadee, were recorded. Twenty five bird species were identified in a field study within the immediate area of the DGR site in 2004 (OPG05a). In that study, four species were confirmed breeders; northern flicker, chipping sparrow, American robin and black-capped chickadee.

A breeding bird survey carried out in 2007 observed 37 birds, representing 21 different species exhibiting breeding behaviour, in the OPG-retained lands that encompass the DGR. American redstart was the most commonly observed bird species within this area, followed by eastern wood-pewee and red-eyed vireo. The breeding bird survey was updated in 2009. Species at risk identified during the surveys were limited to two black-crowned night herons observed flying over the site in 2009 and a common nighthawk observed in Inverhuron Provincial Park.

A wild turkey habitat use and suitability survey conducted in February 2007 revealed that at least two distinct flocks of 20 to 30 birds occur on the Bruce nuclear site. Turkey roosting on the site is habitat-specific, with a preference for a combination of open field areas edged by a mix of larger deciduous and coniferous tree stands. No roosts were identified within the lands that encompass the DGR.

The exposed Lake Huron shoreline surrounding the Bruce nuclear site supports loafing habitat for waterfowl, herons, and gulls. Some shorebird species that have been noted along the exposed shoreline include spotted sandpiper, great blue heron, and black-crowned night-heron. The wetland habitat of Baie du Doré and the surrounding area continues to be important habitat, including for over-wintering, for bald eagle.

2.8 Social and Economic Environment

2.8.1 Land Use

Municipality of Kincardine zoning bylaws identify the Bruce nuclear site as "general industrial" and permit a variety of land uses related to electrical and heat energy production, transmission, and distribution. Land use adjacent to the Bruce nuclear site is consistent with rural development within the township, including agriculture, recreation and rural residential development. OPG owns a considerable amount of land adjacent to the Bruce nuclear site and outside the Bruce nuclear site boundary, creating a non-resident buffer consisting mainly of unoccupied forest and/or swamp and, to the south, Inverhuron Provincial Park.

Inverhuron Provincial Park was operated as a day-use park for more than 30 years. In 2005 it began to operate again as an overnight camping park, as well a day-use park. OPG has title to Inverhuron Provincial Park, which adjoins the southern boundary of the Bruce nuclear site. The park is leased back to the Ontario Ministry of Natural Resources, who manage the park.

The Bruce ECO-Industrial Park is a 485 ha serviced industrial park located southeast of the Bruce nuclear site. Within the Bruce ECO-Industrial Park, the majority of the land is designated as either industrial, or natural environment, and a small portion is designated open space. One of the objectives of the Industrial designation listed in the Municipality of Kincardine's Official Plan is to encourage secondary industries related to the Bruce nuclear site to locate in the Bruce ECO-Industrial Park (NWMO11h).

Within a 50 km radius of the Bruce nuclear site, there are 250,000 ha of arable farmland in Bruce County. More than 60% of the County's land area is dedicated to the agricultural industry.

Structures in the vicinity of the Bruce nuclear site include seasonal and permanent year-round dwellings, and agricultural buildings.

Lake Huron provides the water supply for adjacent municipalities as well as the Bruce nuclear site. The Municipality of Kincardine has two separate water systems; one for the community of Tiverton and one in the former Town of Kincardine from which a trunk water main has been extended to service shoreline developments and the community of Inverhuron.

2.8.2 Population and Community Profile

As described previously, the DGR site is located on the east shore of Lake Huron in the Municipality of Kincardine in the southern portion of the County of Bruce. In 1999, as a result of municipal restructuring, Tiverton, Bruce Township, Kincardine and Kincardine

Township were amalgamated to become the Municipality of Kincardine. In addition, as of January 1 1999, Port Elgin, Saugeen Township and Southampton were amalgamated to become the Town of Saugeen Shores.

The level and distribution of population across the region has not changed dramatically since 1996. Overall, the regional population declined by 4.9% from 1996 to 2001 but recovered from 2001 to 2006 with an overall increase of 1.6% (NWMO11h). According to the most recent Census (2006), the Municipality of Kincardine population is 11,173 people, and the population base of Saugeen Shores is 11,720 people (STATSCAN07a).

The Bruce nuclear site is one of the largest centres of energy production in the world. With the restart of two units of Bruce NGS A in 2003 and 2004, and the refurbishment of the remaining two units in progress, local employment has increased. The operations on the Bruce nuclear site remain the major economic influence in the area. The Bruce nuclear site is Bruce County's single largest employer. In 2009, employment at the Bruce nuclear site included approximately 4000 Bruce Power employees, 400 refurbishment contractors, and 306 OPG and Atomic Energy of Canada Limited (AECL) employees. Approximately 90% of the workers employed at the Bruce nuclear site live in Bruce County, and more than 75% of employees reside either in the Municipality of Kincardine or the Town of Saugeen Shores.

The economy of Bruce County is diverse, and includes agriculture, tourism, recreation, services, small manufacturing, and some resource extraction. A consortium of private companies together with OPG and the Ontario Energy Corporation has developed an industrial and agricultural park, known as the Bruce ECO-Industrial Park, just outside the Bruce nuclear site as described in Section 2.8.1. About 100 people are employed in a number of small industries at the Bruce ECO-Industrial Park.

Agriculture is an important component of the local economy. The area has over 3750 farm operators that generate over 225 million dollars in gross sales annually (NWMO11h). Approximately 63% of all Bruce County farms are family-owned and operated. The County is ranked first in Ontario for total cattle production, with 51% of the farms dedicated to the production of beef cattle. The County is ranked third in Ontario for sheep production. Bruce County is also the top producer of oats and the second largest producer of canola, barley and hay in Ontario. With this agricultural activity also comes a variety of supporting and processing industries related to the production of food, animal breeding, and horse boarding.

In 2007, Bruce County attracted over 1.3 million visitors from Canada, the United States and overseas, who spent over 187 million dollars on tourism-related expenditures including food and beverages, transportation, accommodation, retail and entertainment (MTO09). The tourism industry in Bruce County employs more than one in seven of the working population (PKF01). Local service clubs, agriculture societies and community non-profit groups organize over 700 events annually.

The waters of Lake Huron are used for sport and commercial fishing. Sport fishing, in the lake itself as well as the tributary streams and lake bays, is increasing with the growth in tourist activity and the improvement of beach facilities. The commercial fishery production varies from year to year, and the majority of the catch is exported to the northeastern United States.

Cottages, resorts, beaches and marinas are located along the shoreline focused around the communities of Kincardine and Port Elgin. Within a 5 km radius of the Bruce nuclear site, there are approximately 60 homes (permanent and seasonal cottages) located around the Scott Point area, and approximately 450 permanent and seasonal residences (only about 200 are permanent) located in Inverhuron. Farm and non-farm residents are also dispersed along concession roads.

The South Bruce County area enjoys a full range of services and facilities, including health and education facilities. The South Grey Bruce Health Centre (Kincardine Hospital) provides in- and out-patient services. Along with a number of public secondary and elementary schools, the area has one of two nuclear teaching facilities in Ontario, which trains Bruce Power staff in the operation, maintenance and safety aspects of Canada Deuterium Uranium (CANDU) reactors.

2.8.3 Aboriginal Communities

The traditional territory of the Ojibway in the Saugeen region covers the watersheds bounded by the Maitland River to the south and the Nottawasaga River east of Collingwood on Georgian Bay. The area includes the Bruce Peninsula, all of Grey and Bruce Counties, and parts of Huron, Dufferin, Wellington and Simcoe Counties (TOTTMAN94). The Bruce nuclear site is located within this traditional territory.

The Saugeen Ojibway Nation (SON) is the collective name for the two First Nations communities with reserve lands in the area. The Chippewas of Saugeen First Nation and the Chippewas of Nawash Unceded First Nation share the same Aboriginal and treaty rights, including rights to fish commercially in the waters around the Bruce Peninsula.

The Chippewas of Saugeen First Nations Reserve No. 29 is located adjacent to the town of Southampton, about 30 km north of the Bruce nuclear site. The 2006 census estimated the on-reserve population of this reserve to be 760 (NWMO11j). The Department of Indian and Northern Affairs estimated that 836 members live off-reserve, many within the traditional territory in Bruce County (OPG05a).

The Community of Chippewas of Nawash Unceded First Nation is located at the Cape Croker Reserve No. 27 on the east shore of the Bruce Peninsula, north of the Town of Wiarton. The population on this reserve was estimated to be 591 in the 2006 census (NWMO11j), with 1338 members living off-reserve, many within the traditional territory in Bruce County (OPG05a).

Métis peoples in Ontario are a distinct Aboriginal people with a unique history, culture, language and territory that includes the waterways of Ontario, surrounds the Great Lakes and spans what was known as the historic Northwest. The Métis people do not comprise one settlement; rather they are mobile regional communities that are not tied to a land base.

The Métis people who most likely have an interest in the DGR project are those that have traditionally lived alongside the SON, hunting, fishing, harvesting and trading. Specific community activities were affiliated with the Hudson's Bay Company post at the mouth of the Saugeen River, the Owen Sound area, and the historic Bruce Peninsula portage route that facilitated travel between the main basin of Lake Huron and Georgian Bay. These Métis people may be represented by either a local Métis Nation of Ontario (MNO) council or by the Historic Saugeen Métis, which is not affiliated with the MNO. According to the 2006 Census information from Statistics Canada (STATSCAN07b), 360 Métis persons reside in Bruce County and 825 reside in Grey County. The Métis people participate fully in the community, and are integrated into the regional population.

The First Nations and the Métis make use of Lake Huron for traditional and commercial harvesting of fish. The First Nations' economies also rely on tourism, agriculture, construction, cottage rental and native craft manufacture and sale. Both the Saugeen and Nawash First Nations have developed a wide range of community services. They obtain water from on-reserve wells, from the lake or nearby communities. Their ongoing use of their traditional lands and waters includes personal and communal commercial harvesting of traditional foods and medicines.

3. SITE EVALUATION AND CHARACTERIZATION

As part of early feasibility studies (2002-2003), the regional geologic framework in which the Bruce nuclear site resides was investigated to assess the potential of the site to host a DGR for the long-term management of L&ILW. The results of these studies, reported in the Geotechnical Feasibility Study (GOLDER03), led to the development of seven hypotheses (see Section 3.3) specifying geoscientific attributes and characteristics of the Bruce nuclear site that are favourable for the safe implementation of the DGR concept. These hypotheses served as a basis to develop the site-specific and regional geoscientific characterization plans necessary to gather data and information to test the hypotheses and provide evidence supporting the DGR safety case. These plans, and details surrounding their execution, are described in the following sections.

3.1 Geoscientific Site Characterization

A GSCP was prepared to collect the necessary geoscientific information to support the development of a Descriptive Geosphere Site Model (DGSM) (OPG06b, OPG08a), and to provide the necessary information to support development of a safety case for the DGR. The GSCP provides a technical description of the selection and application of preferred tools and methods for site-specific geoscientific characterization of the deep sedimentary bedrock formations underlying the Bruce nuclear site. The GSCP also describes regional scale geoscientific studies considered necessary for the development of a Geosynthesis, which examine the past, present and future state of the geosphere relevant to the DGR concept. A GRG[1] provided independent oversight and peer review of the DGR geoscience work program.

The purpose of the GSCP was to yield:

- A DGSM (NWMO11k), which represents an integrated, multidisciplinary, geoscientific description and explanation of the undisturbed subsurface environment as it relates to site-specific geologic, hydrogeochemical, hydrogeologic and geomechanical characteristics and attributes; and

- A Geosynthesis (NWMO11c), which is a geoscientific explanation of the overall understanding of site characteristics, attributes and evolution as they relate to demonstrating long-term DGR performance and safety. The Geosynthesis

[1] The GRG is composed of internationally renowned scientists and engineers, who provide guidance, advice and lessons learned from similar international geoscience work programs.

includes various regional data, the geological, geomechanical, hydrogeological settings and frameworks, as well as seismicity.

The GSCP comprised a four-year, three-phase work program designed to allow iterative development, testing and refinement of a site-specific DGSM, consisting of individual geologic, hydrogeologic and geomechanical conceptual models. The document was released in April 2006 (OPG06b) and revised for Phase 2A and 2B based on knowledge gained during Phase 1 in April 2008 (OPG08a).

Site-specific Phase 1 work began in August 2006 with a two-dimensional (2D) seismic survey and was followed by the drilling of two vertical boreholes, DGR-1 and DGR-2, to depths of approximately 462 and 862 mBGS, respectively. The drilling included continuous rock coring (core diameter of 76 mm and borehole diameter of 159 mm). The boreholes were drilled approximately 40 m apart, with each exploring different stratigraphic horizons. Drilling of two separate boreholes for DGR-1 and DGR-2 was designed to minimize vertical cross-connection and cross contamination of groundwater between the shallow and deep hydrogeologic environments with suspected distinctly different groundwater chemistry and to minimize the risk of borehole loss should caving or other poor drilling conditions be encountered. These boreholes: (1) confirmed the Paleozoic bedrock stratigraphy at the site down to the Precambrian basement; (2) provided core for laboratory testing (petrographic, geochemical, hydrogeological, and geomechanical); and (3) provided access for borehole geophysical and borehole hydraulic testing, as well as the installation of multi-level sampling and monitoring equipment. Phase 1 also included the refurbishment, monitoring and sampling of the site US-series wells in the shallow bedrock aquifer; the installation of three borehole seismograph stations within 50 km of the Bruce nuclear site; and the completion of a 2D seismic reflection study. Phase 1 investigations were conducted between August 2006 and December 2007.

Phase 2A consisted of two continuously cored vertical boreholes (DGR-3 and DGR-4) drilled to approximately 869 and 857 mBGS, respectively, and terminating in the Cambrian sandstone. Phase 2A drilling was carried out from April to October 2008. The results from DGR-3 and DGR-4 complement the information gathered from Phase 1. The Phase 1 and 2A boreholes (DGR-1, DGR-2, DGR-3 and DGR-4) are spaced approximately 1047 to 1318 m from each other and are situated in a triangular arrangement to determine stratigraphic continuity, uniformity of bedrock thickness, as well as the strike and dip of the various sedimentary strata at the DGR location. Additional rock core samples were tested in the laboratory to expand the site-specific database collected during Phase 1.

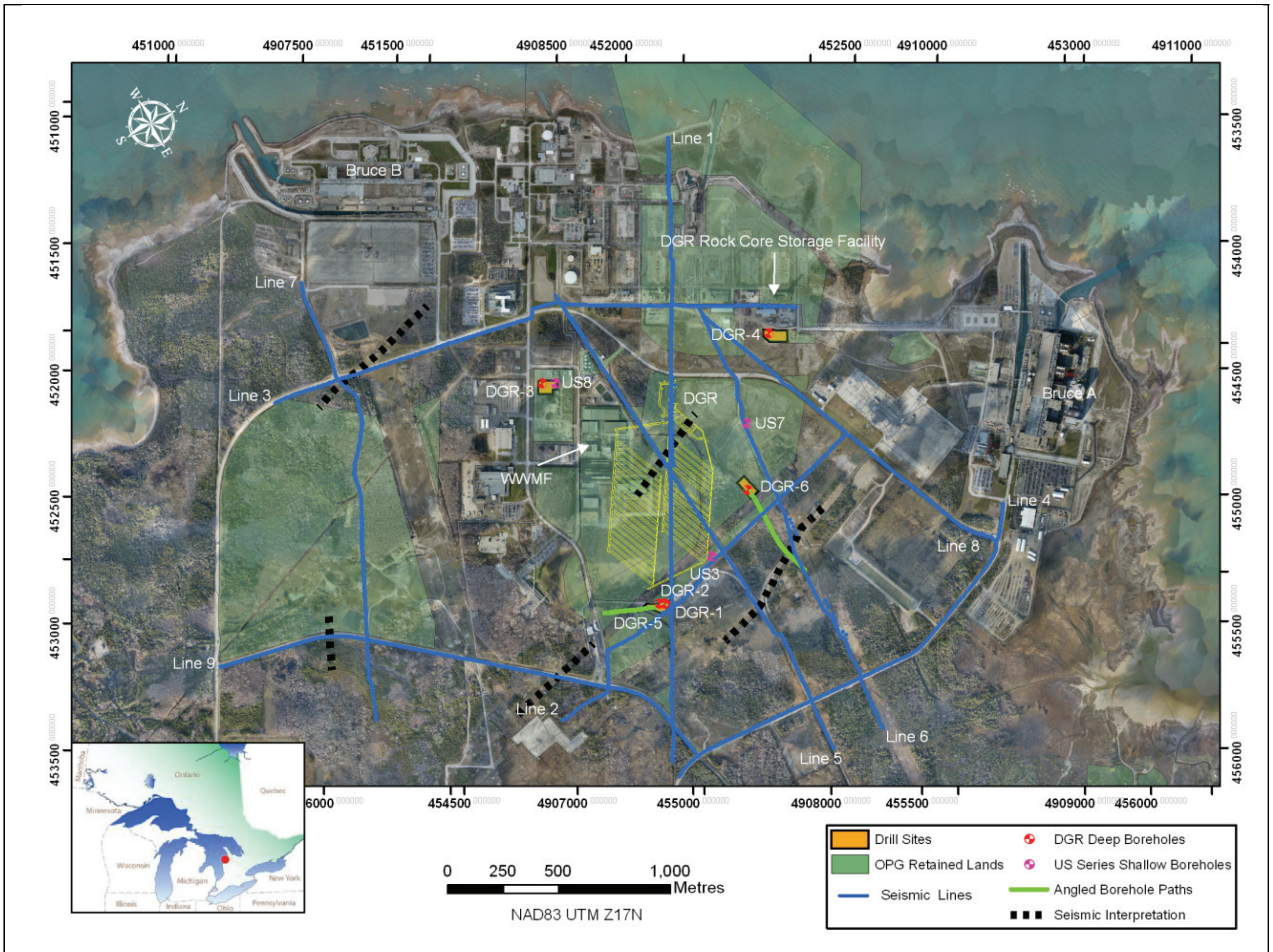
Phase 2B included the drilling of two inclined boreholes, DGR-5 and DGR-6, oriented orthogonal to one another to intersect all possible rock joint sets at site. Each borehole is also located to intersect potential fault locations interpreted from the 2D seismic survey. The Devonian and Upper Silurian rocks (approximately the upper 180 m) were not cored in the inclined boreholes. Borehole DGR-5 was rotary drilled to 188 mLBGS to allow for installation of intermediate steel casing, and then continuously cored to a target depth of 807 mLBGS and terminating near the bottom of the Kirkfield Formation. Starting and final azimuth/plunge of DGR-5 were 190°/65° and 201°/78°, respectively. Borehole DGR-6 was rotary drilled to 215 mLBGS to allow for installation of intermediate steel casing, and then continuously cored to a target depth of 903 mLBGS and terminating within the top of the Gull River Formation. Starting and final azimuth/plunge of DGR-6 were 80°/60° and 73°/57°, respectively. Laboratory testing was also carried out on the rock core.

The location of the deep DGR-series boreholes and shallow bedrock US-series boreholes are shown in Figure 3-1. All boreholes were drilled at least 100 m from the footprint of the DGR.

The drilling during Phase 1 and 2 consisted of several main work program elements, as listed below:

- Borehole drilling and coring;
- Drill water tracing;
- Opportunistic groundwater sampling;
- Borehole orientation survey and directional drilling (where required);
- Geologic core logging, photography and sample preservation;
- A full suite of borehole geophysical logs;
- In-situ straddle packer hydraulic testing;
- Laboratory petrologic, mineralogical, geochemical/isotopic and geomechanical testing of core;
- Installation of a multi-level groundwater sampling and monitoring system (Westbay); and
- Refinements of Descriptive Geologic Site Model and other models.

To define and ensure consistency of the stratigraphic nomenclature and identify stratigraphic contacts, three workshops were held to seek input from members of the Ontario Ministry of Natural Resources Petroleum Resources Centre, the Geological Survey of Canada, the Ontario Geological Survey, and various universities. The first workshop was held in September 2007, when DGR-1 and DGR-2 cores were examined. The second workshop was held in November 2008 and focused on the DGR-3 and DGR-4 cores. The third workshop was held in May 2010 to examine the rock cores from inclined boreholes DGR-5 and DGR-6.



Note: Figure indicates the position of the proposed DGR footprint, all DGR- and US-series boreholes, the WWMF, and the 9 Line seismic survey array.

Figure 3-1: Borehole and Geophysical Investigations at the Bruce Nuclear Site

3.2 Descriptive Geosphere Site Model

The DGSM was developed from the site-specific studies through the description of conceptual geological, hydrogeological and geomechanical models (NWMO11k). It provides a summary compilation, description, assessment, and interpretation of geoscientific data collected as part of a series of investigations, which are described in a set of 69 technical reports. Technical reports serve as primary data sources that provide limited interpretation. All technical reports were completed in accordance with approved test plans.

3.2.1 Descriptive Geological Site Model

The geological site model describes the occurrence and the lithological and structural characteristics of 35 distinct bedrock stratigraphic formations (Paleozoic and one Precambrian), members or units extending from ground surface to a depth of about 860 m based on core retrieved from the deep DGR-series boreholes and data from the shallow bedrock US-series boreholes (maximum 200 m depth). The entire Paleozoic sedimentary sequence at the Bruce nuclear site, ranging in age from Middle Devonian to Cambrian, is comprised of near horizontally layered limestones, dolostones, shales and some sandstone at the base. The thickness and orientation of the Paleozoic strata are remarkably uniform between the DGR boreholes. The average strike and dip (N20°W/0.6°SW) of the deeper Silurian and the Ordovician formations at the Bruce nuclear site are consistent with the regional geological framework.

The Devonian and Upper Silurian dolostones are moderately to highly fractured and of poor to fair rock quality, whereas the deeper Silurian formations and the Ordovician shales and limestones (including the DGR host formation, the Cobourg Formation limestone) are very sparsely fractured to unfractured with excellent rock quality. Occasional natural fractures, which were commonly sealed and tight, were also identified within the deeper Silurian and Ordovician formations. Natural fracture frequency was greater in the thin Cambrian sandstone where fractures are open and permeable.

Analysis of the identified inclined fractures in the Ordovician shales and limestones suggests they preferentially strike in northeast and southeast directions. A lack of measurable offset along these fractures indicates that they can be classified as joints where unfilled with secondary minerals, and as veins where filled. Both joints and veins are found at the DGR site.

The mineralogy and geochemistry of rock cores were obtained using laboratory test methods including thin section petrography with electron microscope analyses, whole rock and clay fraction X-Ray Diffraction (XRD) testing, Scanning Electron Microscope/Energy Dispersive Spectral (SEM/EDS) analyses, trace element Inductively Coupled Plasma (ICP) analyses, elemental oxide analyses by ICP optical emission spectrometry, carbon and sulphur infrared spectroscopy analyses, and chloride by instrumental neutron activation analyses. These detailed analyses generally confirm the strata mineralogy as defined regionally. The organic geochemistry of the Ordovician shales was characterized by standard source rock evaluation methods ('Rock-Eval' pyrolysis) and by measurement of the Total Organic Carbon (TOC) in weight percent.

The presence of fracture infill, vein and secondary mineralogy was identified in DGR cores and consist of chert, quartz, calcite, pyrite, anhydrite, gypsum, halite, celestite, illite, chlorite, marcasite and Fe oxide/hydroxide. Several of these minerals (e.g., halite, gypsum, anhydrite and celestite) are soluble and their occurrence is important to interpretation of porewater analyses and as an indicator of the absence of advective groundwater migration through these discontinuities.

Surface 2D seismic reflection surveys were completed over 19.7 km, on nine survey lines (Figure 3-1), which identified the possible presence of five seismic discontinuities that may represent vertical to sub-vertical faults within the Ordovician formations. These possible faults trend north-northwest to northwest. Two of these possible structures were investigated through inclined drilling of boreholes DGR-5 and DGR-6. These drilling investigations did not identify faults at the target locations identified in the 2D seismic surveys.

3.2.2 Descriptive Hydrogeological Site Model

The hydrogeological site model describes the hydrogeologic properties and hydrostratigraphic units within the Paleozoic sedimentary sequence. The descriptive model, based on detailed field and laboratory testing, provides information necessary to understand groundwater migration and properties governing solute transport. Further, the model is divided into physical and geochemical aspects, with the former establishing properties such as porosity, fluid saturations, surface area, permeability, hydraulic head, diffusion coefficients and two-phase gas flow, and the latter, the spatial distribution of groundwater and pore fluid geochemical and isotopic characteristics within the sedimentary sequence.

Laboratory testing of DGR core samples was undertaken to quantify intact rock physical properties including bulk and grain density, physical and water loss porosity, residual fluid saturations, rock permeability to gas, pore-size distribution, gas entry pressure, specific surface area, gas-brine flow properties, effective diffusion coefficients, and diffusion accessible porosity. Bulk and grain densities were measured by three different laboratories with comparable results that were in accordance with expectations based on formation mineralogy and porosity. Total and liquid porosities were also measured on DGR core samples by four independent laboratories. Total porosity was measured by helium gas expansion and from bulk dry and grain density data. Liquid porosity was measured by vacuum distillation and oven drying.

The fluid saturations or fractions of brine, oil and gas within rock pore volumes were determined. It is possible that a discontinuous gas phase is present within the pore space of the Ordovician shales and limestones.

Vertical and horizontal permeabilities were determined on DGR core samples using gas and brine pulse decay methods prior to and following pore fluid extractions. The results suggest irrecoverable damage to the cores due to stress relief and microcrack formation resulting in an overestimation of the actual in situ permeability by a factor of 10 to 100 for Ordovician limestones and 100 to 1000 for Ordovician shales.

Mercury injection porosimetry data, fluid saturation data, and gas pulse permeability data were used to calculate van Genuchten characteristic gas-brine flow parameters. The fitted relative permeability saturation curves indicate that the apparent small gas saturations could result in reductions of brine permeability by upwards of one order of magnitude.

Diffusion properties of intact DGR cores were measured by conventional steady-state laboratory through-diffusion methods using iodide, tritium and ^{125}I as tracers to estimate vertical properties, and X-ray radiography with an iodide tracer to estimate vertical and horizontal properties. Effective diffusion coefficients for iodide within the Ordovician shales and limestones were found to be low with values of approximately 1×10^{-12} m^2/s for shales and 4×10^{-13} m^2/s for the limestones.

Groundwater and pore fluids in the sedimentary rock were characterized for pH and Eh, major and trace elements, environmental isotopes, radioisotopes, and some gases.

Porewaters were extracted from crushed DGR core samples at the University of Ottawa by high-temperature vacuum distillation (150°C) for dissolved gases and isotopes, followed by deionized water leaching of crushed (2-4 mm grain size) rock samples for major dissolved ions. These porewater analyses, supported by the available shallow and deep groundwater analyses completed by the University of Bern, Switzerland, and the University of New Brunswick using crush and leach methods, were used to generate water chemistry profiles for the Paleozoic bedrock at the Bruce nuclear site.

Historical hydraulic testing of the upper 100 m of bedrock, in combination with the results of pulse, slug and drill-stem hydraulic testing of DGR boreholes using a custom-built straddle-packer testing tool, were used to quantify formation horizontal hydraulic conductivity. Borehole straddle-packer tests were analyzed using the Sandia National Laboratories numerical hydraulic-test simulator (nSIGHTS).

Upon completion, boreholes US-3, US-7, US-8, DGR-1, DGR-2, DGR-3 and DGR-4 were equipped with Westbay multiport groundwater monitoring systems. Stable pressure profiles measured in US-3, US-7 and US-8 show slight upward hydraulic gradients in the upper 200 m of dolostone bedrock, with lateral flow toward Lake Huron. Monitoring of formation pressures in 42 (each in DGR-3 and DGR-4) to 46 (DGR 1 and DGR-2) packer-isolated intervals in the DGR boreholes, over periods of months to a year, shows the presence of moderate overpressures in Salina A1 and A0 units, and the Goat Island, Gasport and Fossil Hill formations; significant stable overpressure in the Cambrian sandstone; and significant transient underpressures throughout most of the Ordovician shale and limestone. The occurrence of such significant underpressures implies that the formations in which they exist are of extremely low permeability in order for them to persist.

3.2.3 Descriptive Geomechanical Site Model

The geomechanical site model describes and summarizes the current understanding of the principal geomechanical properties of the rock materials and rock mass beneath the Bruce nuclear site. The geomechanical site model focuses on presentation of quantitative estimated physical properties that will control the geomechanical behaviour of the rock mass beneath the site during and after construction of the sub-surface DGR infrastructure. Representative values are based on combining the specific quantitative values of various parameters derived from field and laboratory testing and with up-scaling for rock mass properties, where appropriate.

Rock material geomechanical characteristics include, where available, information on short and long-term uniaxial compression strengths, triaxial compression strength, indirect tensile strength, direct shear strength, slake durability, free swell behaviour, abrasiveness, and dynamic properties (elastic and shear moduli, Poisson's ratio) based on the testing of intact cores. Rock mass geomechanical characteristics include, information on Rock Quality Designation (RQD), natural fracture frequency, and bulk properties from borehole geophysical logging (dynamic elastic and shear moduli).

Five mechanostratigraphic units are developed for describing the geomechanical properties of the bedrock formations at the Bruce nuclear site and are described in detail in Chapter 4.

An expanded seismic monitoring network was installed in 2007 to gain an understanding of the contemporary microseismic activity within approximately 40 km of the Bruce nuclear site. This monitoring, in addition to other information, has been used to conduct a seismic hazard assessment for the DGR and to provide information on the contemporary seismicity and microseismicity that can be used in identification of seismogenic features in proximity to the Bruce nuclear site. The current and historical monitoring data show that the Bruce nuclear site is located in a seismically quiet area.

The magnitude and direction of in-situ stresses at the elevation of the DGR (approximately 680 mBGS) was estimated based on a review of the regional stress measurements and the observation of a lack of borehole breakouts/deformation in DGR boreholes for up to 2 years following completion of drilling.

3.3 Geosynthesis

The Geosynthesis (NWMO11c) summarizes site-specific and regional geoscientific studies conducted to understand the predictability, stability and geologic isolation provided by the geologic setting at the Bruce nuclear site. A key aspect of this approach has been the coordination of multi-disciplinary investigations and research conducted by NWMO and its contractors, including about 20 universities and specialized laboratories and consulting groups.

An important element of the DGR safety case is a demonstrated basis to understand and justify the characteristics and behaviour of the rock mass that will enclose the DGR and contribute to the long-term containment and isolation of the L&ILW. The Geosynthesis brings together and summarizes in quantitative detail the physical attributes of the Middle Ordovician Cobourg Formation that support its suitability as the host formation for the DGR. Similarly, the Middle Ordovician lower bounding carbonate units and the overlying Upper Ordovician shale cap rocks are discussed in support of the argument that they will provide long-term isolation of the L&ILW and act as an integral barrier to solute migration.

Based on an initial understanding of the geologic and hydrogeologic setting in the sedimentary sequence beneath the Bruce nuclear site, the Geosynthesis introduced seven hypotheses that specified geoscientific site attributes and characteristics favourable for safe implementation of the DGR concept. These hypotheses are described below.

Predictable: The Paleozoic sequence that would host and enclose the DGR comprises near horizontally layered, undeformed sedimentary shale and limestone formations of large lateral extent.

Seismically Quiet: The Bruce nuclear site exists in a region where the seismic hazard is low. Seismicity is comparable to a stable Canadian Shield setting.

Multiple Natural Barriers: The sedimentary sequence consists of multiple low-permeability bedrock formations that enclose and overlie the DGR.

Shallow Groundwater Resources are Isolated: The near-surface fresh water aquifers are isolated from the deep, low permeability, saline groundwater regime in which the DGR would reside.

Geomechanically Stable: The host and enclosing rock formations will provide stable, virtually dry openings for safe implementation of the DGR concept during operation and postclosure phases.

Contaminant Transport is Diffusion-Dominated: The deep groundwater system in which the DGR would be constructed is ancient with no evidence of glacial perturbation or cross-formational flow. Solute transport processes are dominated by diffusion over geologic time periods.

Natural Resource Potential is Low: The potential for hydrocarbon, base mineral or sustainable groundwater resources at, and in the vicinity of, the Bruce nuclear site is extremely low.

A key role of the Geosynthesis was to conduct studies and assemble information from the site specific site characterization work program to assess the validity of the hypotheses. In this regard, notable supporting studies summarized in the Geosynthesis report (NWMO11c) are documented in reports listed below.

- **Regional Geology – Southern Ontario** (NWMO11m): Presents the regional geologic setting in the context of stratigraphy, lithology, structural geology, tectonics, basin history, sedimentology, formation thermochronology, depth of burial, economic resources, and glacial history. Provides a basis for geoscientific understanding of the current geological framework of the Bruce nuclear site, its past evolution, and likely future natural evolution.

- **Regional Geomechanics – Southern Ontario** (NWMO11n): Provides a synthesis of our current understanding of regional intact rock strength and rock mass characteristics, rock joint sets, magnitude and direction of in-situ stresses, and sub-surface excavation experience in similar rock formations. Compiled from available existing regional data.

- **Hydrogeologic Modelling** (NWMO11p): Presents hydrogeologic modelling and analyses at regional- (18,000 km^2) and site- (400 km^2) scales using the mathematical models FRAC3DVS-OPG and TOUGH2-MP. Provides an assessment of groundwater system behaviour through a sequence of analyses that explore the influence of hydrostratigraphy, physical and groundwater properties, boundary conditions, and external perturbations on groundwater system evolution. Also provides quantitative evidence supporting a conclusion that the deep groundwater regime has remained diffusion dominant at time scale relevant to demonstrating DGR safety.

- **Regional Hydrochemistry – Southern Ontario** (NWMO11q): Presents information on the hydrogeochemical and isotopic characteristics of groundwaters within the Paleozoic sedimentary sequence underlying southwestern Ontario, including their compositions and spatial distribution; and the scientific understanding of the origin, evolution and timing of emplacement of these groundwaters within the sedimentary sequence.

- **Long-Term Climate Change** (NWMO11r): Provides detailed estimates of long-term climate change and ice-sheet history as it has and, in the future, may influence the Bruce nuclear site if the Canadian land mass were to be re-glaciated. The results support other work activities, for example, analyses of long-term DGR geomechanical stability and integrity of the overlying rock mass, coupled hydro-mechanical analyses to explore the response of the groundwater system to ice-sheet loading, estimates of glacial erosion and near-surface groundwater geochemistry.

- **Long-Term Geomechanical Stability Analysis** (NWMO11t): Assessments of DGR cavern, pillar and shaft stability, and the evolution of damage and deformation of the surrounding rock mass in response to excavation activities and the long-term dynamic geological conditions expected at the site. Presents the results of deformation, damage, and stability analyses due to long-term rock strength degradation, gas and pore pressure changes, seismic ground shaking and glacial loading/unloading cycles.

- **Glacial Erosion Assessment** (NWMO11u): Provides an understanding of glacial erosion processes and erosion rates that could influence the Bruce nuclear site.

- **Neotectonic Features and Landform Assessment** (NWMO11v): A field study examining evidence within 50 km of the Bruce nuclear site of paleoseismicity in the form of soft sediment disturbance, paleoliquefaction and other features such as offset beaches. The intent is to assess evidence for significant seismic events following the last glacial ice-sheet retreat.

- **Seismic Hazard Assessment** (NWMO11w): Provides a seismic hazard assessment for the Bruce nuclear site and DGR considering return times of 100,000 and 1,000,000 years. Specifically, the report provides estimates of uniform hazard response spectra at the surface for a reference hard rock site; develops a probabilistic model for site response utilizing measured dynamic properties of the site geologic units; and develops design ground motion time histories for the repository and selected sub-surface levels.

- **Excavation Damaged Zone Assessment** (NWMO11x): Provides a summary of international geoscientific studies and experimental work related to the characterization, nature and distribution of properties within the Excavation Damaged Zone (EDZ) as relevant to sedimentary rock. Also provides a summary of the treatment of the EDZ in Safety Assessment analyses.

- **Analogue Study of Shale Cap Rock Barrier Integrity** (NWMO11y): Presents evidence examining the long-term integrity of shale cap rock as a barrier and structural trap for oil and gas reservoirs in the Appalachian and Michigan basins, and assessed a suitable analogue to illustrate the long-term barrier integrity of the Ordovician shale sequence above the DGR.

- **Karst Assessment** (NWMO11z): Presents an understanding of karst ground conditions and its occurrence in southern Ontario as it relates to understanding groundwater system evolution and the long-term performance of the DGR.

- **Three-Dimensional Geological Framework Model** (NWMO11aa): Provides a description of the development of a 3-dimensional regional scale (34,000 km^2) stratigraphic model centered on the Bruce nuclear site. This model involved the re-interpretation and assembly of stratigraphic information from Ministry of Natural Resources oil and gas well records. The regional study area contained at total of 341 wells, from which 299 wells were determined useful through a data validation process. This stratigraphic model is used explicitly to establish the hydrostratigraphy in numerical simulations of the groundwater system.

- **Outcrop Fracture Mapping** (NWMO11ab): Presents the results of detailed structural mapping of the bedrock exposures along the Lake Huron shoreline near the Bruce nuclear site. The report provides an assessment of mechanisms responsible for structural discontinuity timing and formation. No evidence of post-Paleozoic fracture development was recognized.

A detailed description of the site characterization and Geosynthesis work program results is provided in Chapter 4. This geoscience information is discussed further in Chapter 14, as it supports the DGR safety case and conclusions of long-term geosphere stability that contribute to the ability of the far-field bedrock to safely contain and isolate the L&ILW.

3.4 Additional Geoscientific Investigations

As part of activities associated with the geoscientific characterization of the Bruce nuclear site, a shaft pilot investigation at the location of the main and ventilation shafts is planned in 2011. The purpose of this work program is to conduct geoscientific and geotechnical investigations at the locations of the DGR shafts in support of: i) DGR geoscientific verification studies; ii) a demonstration shaft ground improvement study; and iii) detailed design and construction of the two DGR shafts. The program would involve two cored boreholes, one each at the center of the vent (borehole depth ≈180 m) and main (borehole depth ≈720 m) DGR shafts. Because of the higher permeability of the upper 180 m and the need for groundwater inflow control (grouting/freezing) a separate borehole is required at each shaft location. Below this depth, only one borehole is required, because virtually no change in the rock properties at the two locations is expected. The investigation, under development, will include a program of field in-situ borehole and laboratory testing necessary to assess and verify geologic, hydrogeologic and geomechanical conditions along the shaft alignments. The completion of field studies and reporting is scheduled for late 2011.

In the event a DGR site preparation and construction licence is granted, further geoscience investigations are proposed to verify sub-surface conditions. These proposed investigations are described in a Geoscience Verification Plan (NWMO11ar) and would be conducted during both vertical and horizontal DGR development. The investigations focus on gathering information necessary to further assess and demonstrate DGR safety and confirm DGR engineering design and layout. The proposed investigations would include the characterization of the EDZ, in-situ rock stress measurements, bedrock formation permeabilities, diffusion properties, and hydrogeochemical and microbiological conditions. These investigations studies are primarily focused on the collection of geoscientific data to: i) support engineering decisions and DGR design, and ii) support the DGR safety case prepared as part of regulatory approvals seeking an operating licence for the DGR. Verification activities proposed in this plan may need to be revised based on experience gained in the execution of the plan.

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4. GEOSCIENCE

The geoscientific characterization of the Bruce nuclear site, as described in the preceding chapter, involved two key tasks: i) a step-wise multi-year site-specific geoscientific investigation which is documented in the DGSM (NWMO11k); and ii) the development of a Geosynthesis (NWMO11c), which examines the past, present and future evolution of the site at scales relevant to understanding performance and safety of the proposed DGR. The results of this work, which support DGR engineering design and safety assessment activities, are summarized below. Key terms used are provided in the OPG DGR project glossary (NWMO10a).

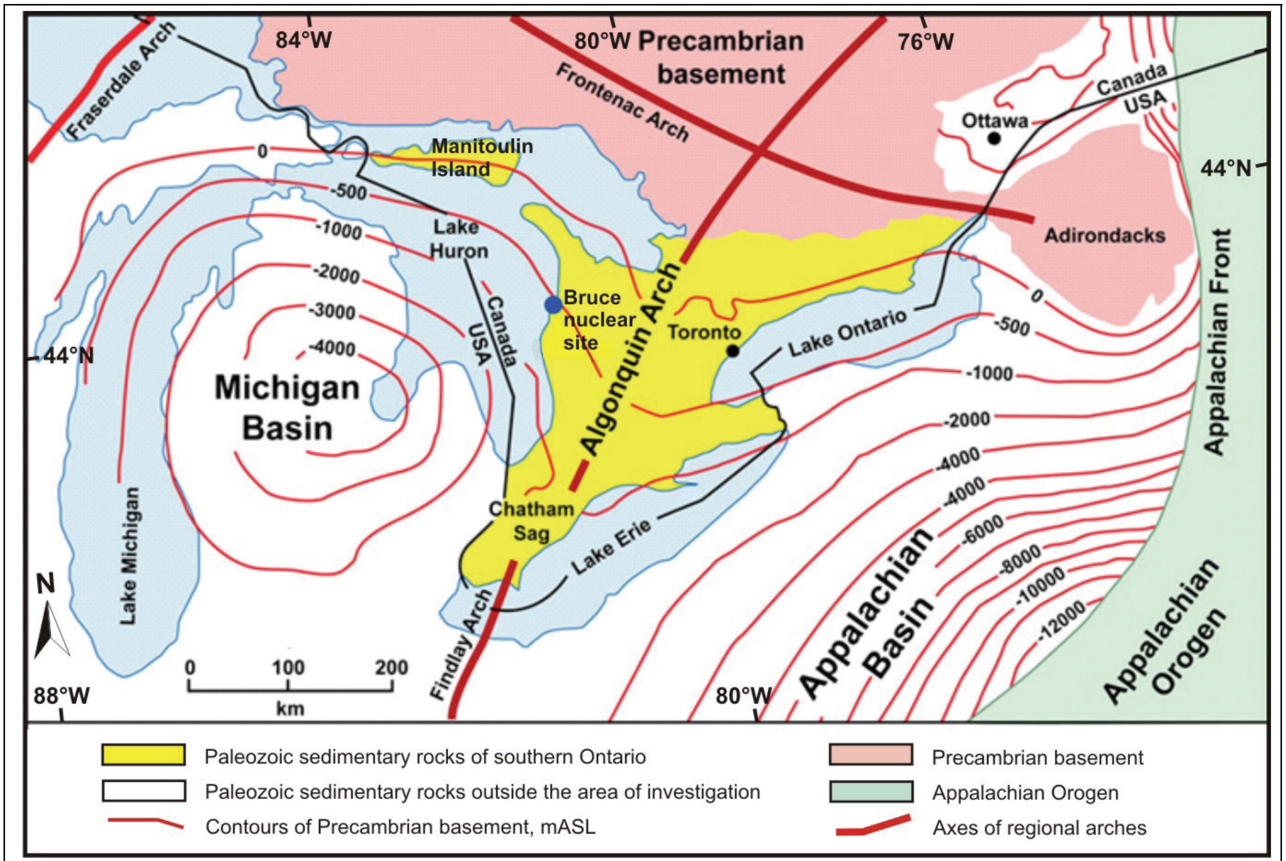
This chapter describes the geological (Section 4.1), geomechanical (Section 4.2), hydrogeochemical (Section 4.3) and physical hydrogeological (Section 4.4) settings of the Bruce nuclear site, and the predicted future evolution of the site once perturbed by DGR construction (Section 4.5). This information is then summarized in Section 4.6 in terms of the multiple lines of evidence gathered during the four years of investigation that are used to test the seven geoscientific hypotheses introduced in Chapter 3. It will be shown below, based on the results of the geoscientific characterization activities, that the Paleozoic succession beneath the Bruce nuclear site possesses the attributes necessary to safely contain and isolate the L&ILW.

4.1 Geology

The following sections discuss the geological setting of the Bruce nuclear site. Firstly, the stratigraphy, tectonic history, and structural framework are described at the regional-scale. Secondly, the stratigraphy and structural framework are described at the site-scale. These descriptions are presented as a basis for understanding the site stratigraphy in terms of its lateral predictability and structurally simple character. This site-scale description includes results pertaining to:

- A detailed fracture mapping analysis (NWMO11ab);
- An analogue study that looked at shale cap rock barrier integrity (NWMO11y); and
- An assessment of the occurrence of karst and paleokarst at the site (NWMO11z).

Aspects of the geological setting which relate to the future evolution of the Bruce nuclear site include a description of Quaternary glacial history and other geological disturbances such as seismicity (including a neotectonic features and landforms assessment), volcanism, and fault reactivation/rupture, and the distribution of natural resources as it pertains to potential future human intrusion. These will be discussed in detail in Section 4.5.



Note: Modified from JOHNSON92.

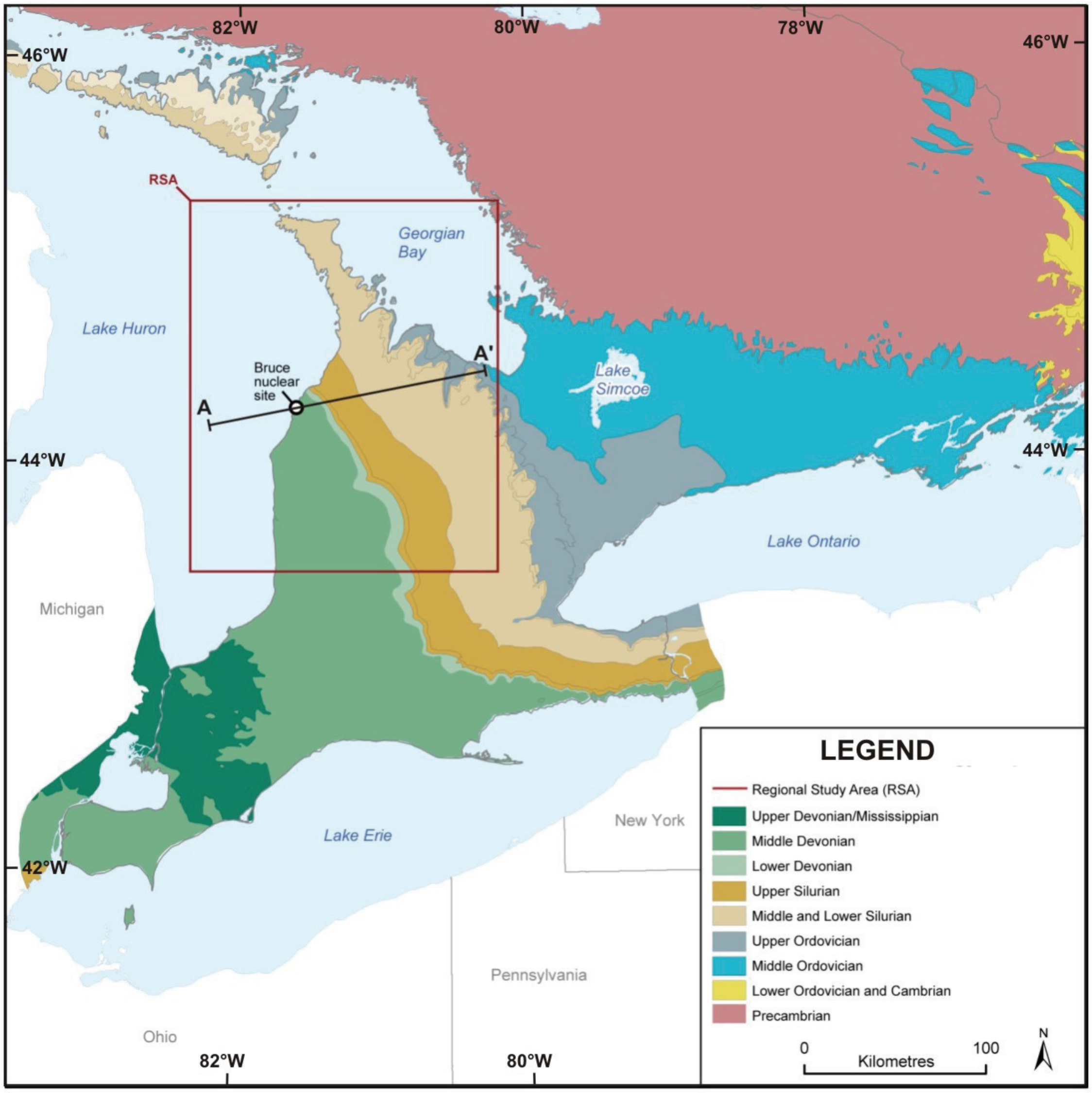
Figure 4-1: Geological Features of Southern Ontario

The information presented below is summarized from a comprehensive description of the regional geology of southern Ontario (NWMO11m), from the DGSM (NWMO11k), from publicly available literature and from elements of the Geosynthesis work program as documented therein (NWMO11c). The reader is directed to these other sources for supplemental information where indicated.

4.1.1 Regional Geological Setting

Southern Ontario is underlain by Upper Cambrian (~510 Ma) to Devonian/Mississippian (~359 Ma) sedimentary rocks (yellow fill on Figure 4-1) unconformably overlying Precambrian basement (ca. 1600 - 542 Ma) of gneisses and metamorphic rocks (pink fill on Figure 4-1) of the Canadian Shield. The regional study area, which is centered on the Bruce nuclear site, is situated on the northeastern margin of the Michigan Basin (Figure 4-1 and Figure 4-2). This area forms part of the northwestern flank of the

Algonquin Arch (Figure 4-1), which is a subsurface basement high overlain by these Paleozoic sediments (e.g., CARTER90a).



Notes: Section along line A-A' is shown in Figure 4-3. See Figure 4-5 for detailed stratigraphic nomenclature of the mapped region. Modified from the Ontario Geological Survey bedrock geology map (OGS91).

Figure 4-2: Generalized Paleozoic Bedrock Geology Map of Southern Ontario

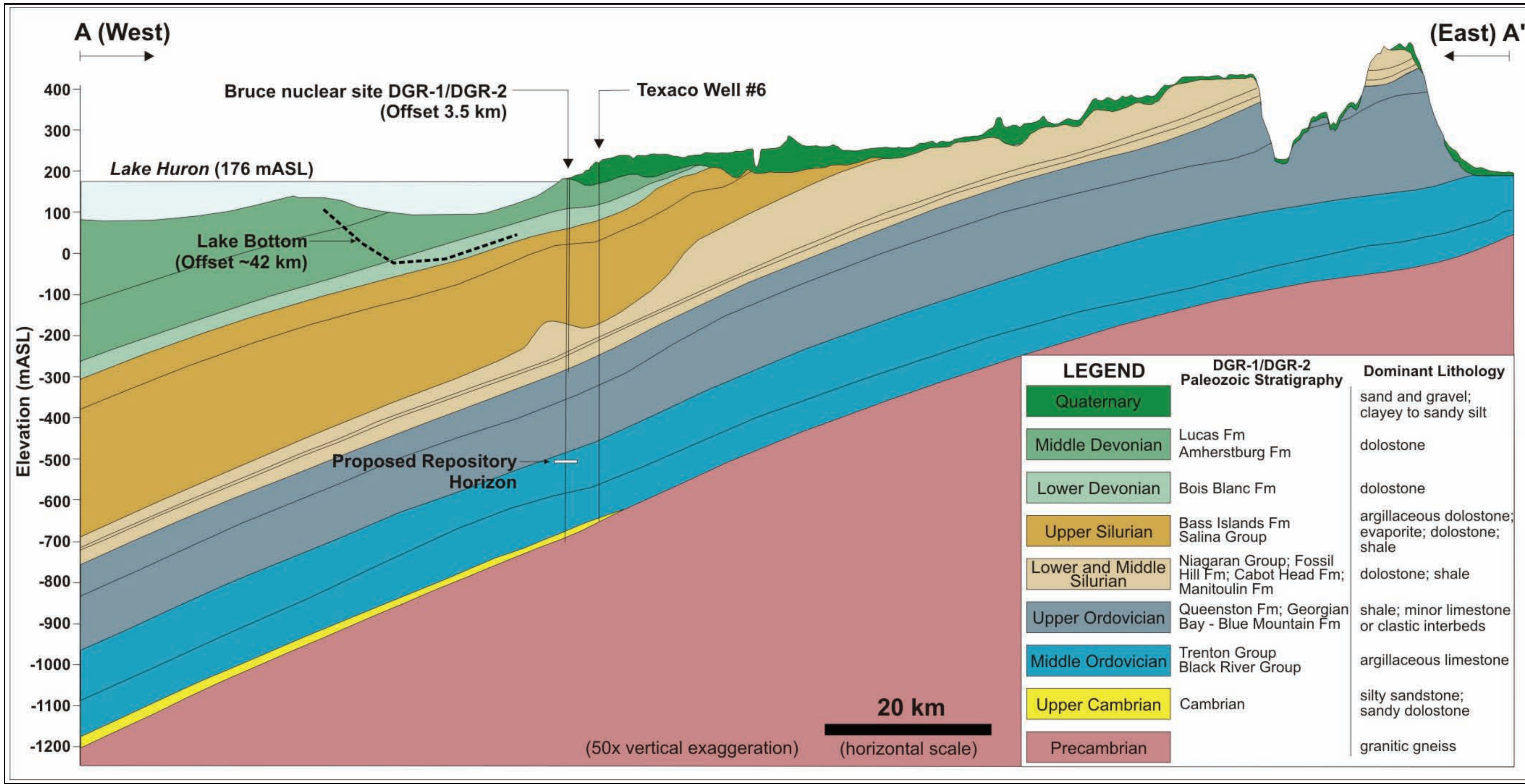
The Paleozoic succession thins from a maximum of approximately 4,800 m at the centre of the Michigan Basin to approximately 850 m at the Bruce nuclear site on the flank of the Algonquin Arch. In general, the strata dip gently from all margins at between 4 and 17.5 m/km, or 0.23° to 1° toward the basin depocentre in central Michigan (LIBERTY71, WATTS09, WIGSTON09). Bedding dips reported from the southern Bruce Peninsula and formation top dips beneath the Bruce nuclear site all fall within this range (ARMSTRONG93, NWMO11k). Figure 4-3 presents a geological cross-section through the Bruce nuclear site.

The regional study area is underlain by low to moderate relief basement rocks of the Huron Domain of the Central Gneiss Belt (Figure 4-4) and is located southeast of the surface trace of the Grenville Front Tectonic Zone (GFTZ) (CARTER90a, MELCHIN94, CARTER96, ANDJELKOVIC98). The basement geology is understood by extrapolation of inferred basement structural boundaries beneath the Paleozoic cover (Figure 4-4). This process is aided by seismic, aeromagnetic, and gravity map interpretation (WALLACH98, BOYCE02), and by geochemical, geochronological, and petrographic analyses of samples recovered from drill cuttings and core (CARTER90b, EASTON95, CARTER96).

4.1.1.1 Stratigraphy

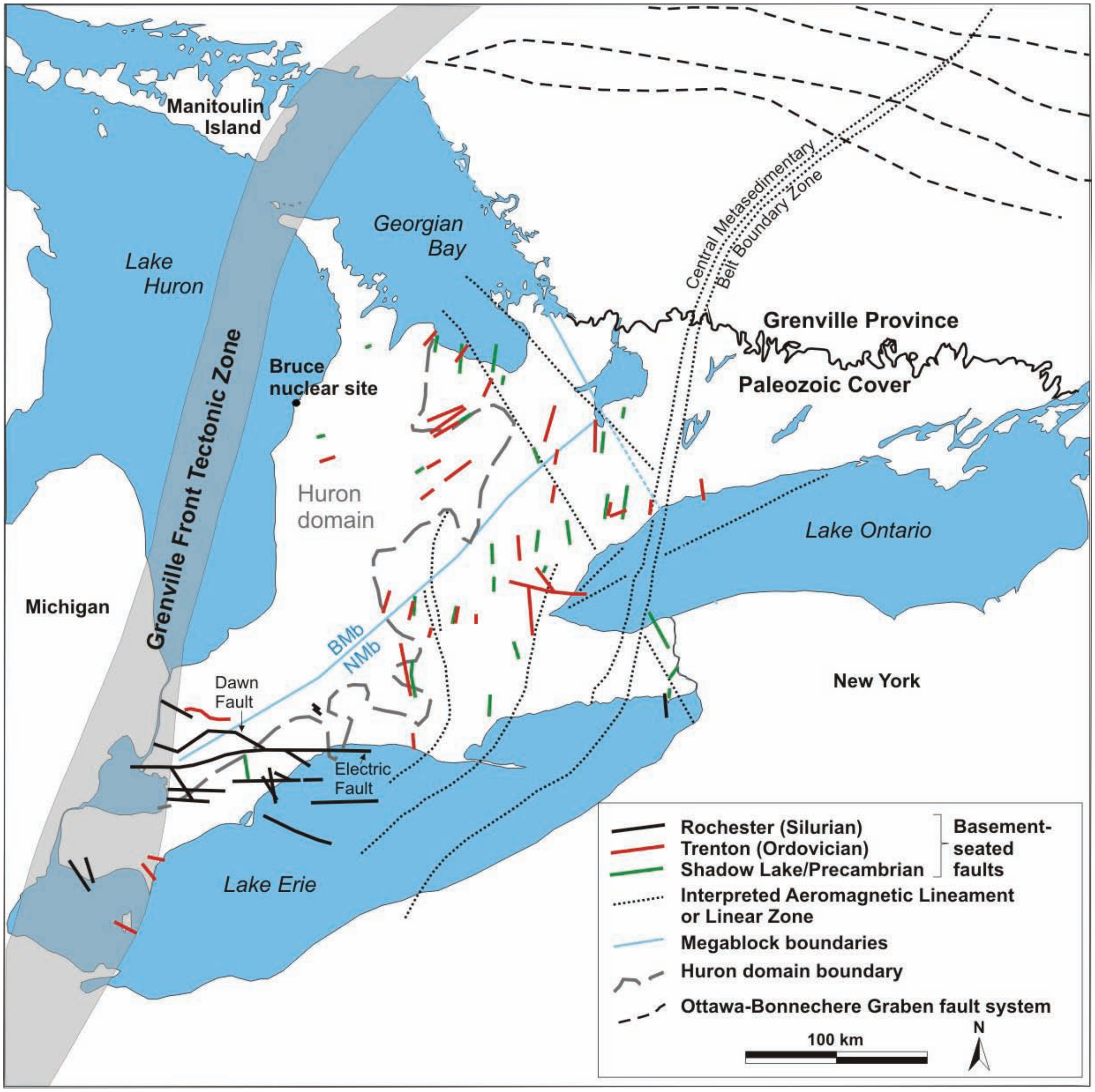
The nearly flat-lying Paleozoic succession was deposited over a broad carbonate and clastic shelf and platform setting that extended from the eastern margin of the Appalachian Basin to beyond the western margin of the Michigan Basin (Figure 4-1). The central column on Figure 4-5 shows the Paleozoic stratigraphy that is encountered beneath the Bruce nuclear site region (ARMSTRONG06). Importantly, this group- and formation-scale stratigraphy is traceable from the Michigan Basin in southwestern Ontario (left column in Figure 4-5) across the arch and into the Appalachian Basin (right column in Figure 4-5). This is to be expected since depositional environments that controlled lithofacies associations evolved at a scale much larger than the regional study area (BROOKFIELD88, NWMO11m, Figure 2.9 of NWMO11c). It follows therefore that the stratigraphy throughout the regional study area is generally predictable across large distances.

A Three-Dimensional Geological Framework (3DGF) model was constructed for the regional study area in a 35,000 km^2 region surrounding the Bruce nuclear site as shown in Figure 4-6. The purpose of the 3DGF model was to better define the stratigraphic and spatial continuity of the Paleozoic succession across this region of southern Ontario (NWMO11aa). The model is based on observation and re-interpretation of Ontario Ministry of Natural Resources well records. The primary data source for the model construction was the Oil, Gas, and Salt Resources Library (OGSR) Petroleum Wells Subsurface Database (OGSR04, OGSR06). At the time of model development, the regional study area contained a total of 341 wells, from which 299 wells were determined useful through a data validation process (NWMO11aa).



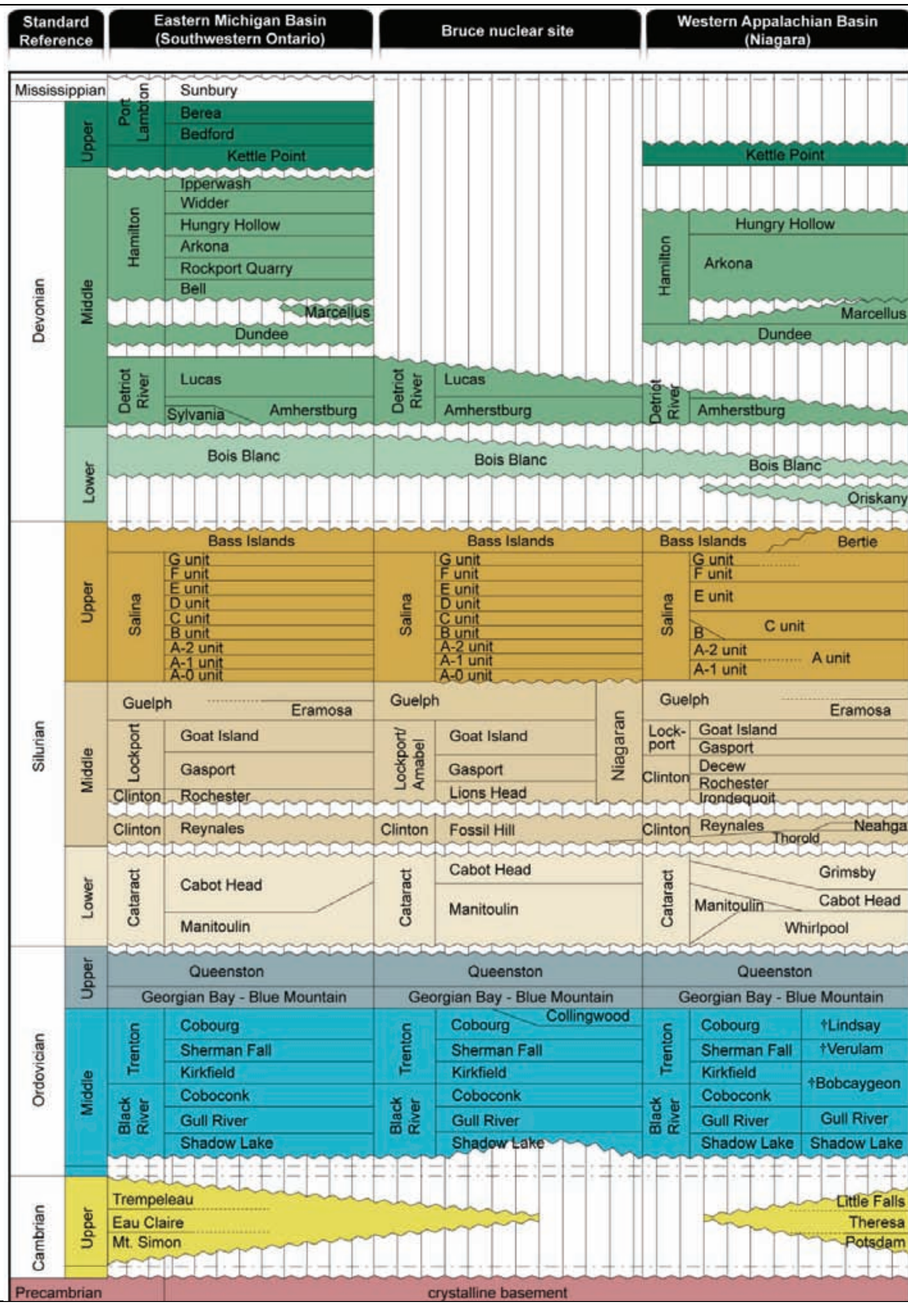
Notes: Cross-section line is shown in Figure 4-2. The subsurface trace of boreholes DGR-1/DGR-2 and Texaco #6 have been projected onto the cross-section. Simplified lithological descriptions are from the DGSM (NWMO11k). Detailed stratigraphic nomenclature for the cross-section is shown in Figure 4-5. Dashed line indicates maximum depth of lake bottom ~42 km north of the site. Fm = Formation; mASL = metres above sea level. Modified from Figure 2.23 of the Geosynthesis (NWMO11c).

Figure 4-3: Geological Cross-Section Through the Regional Study Area



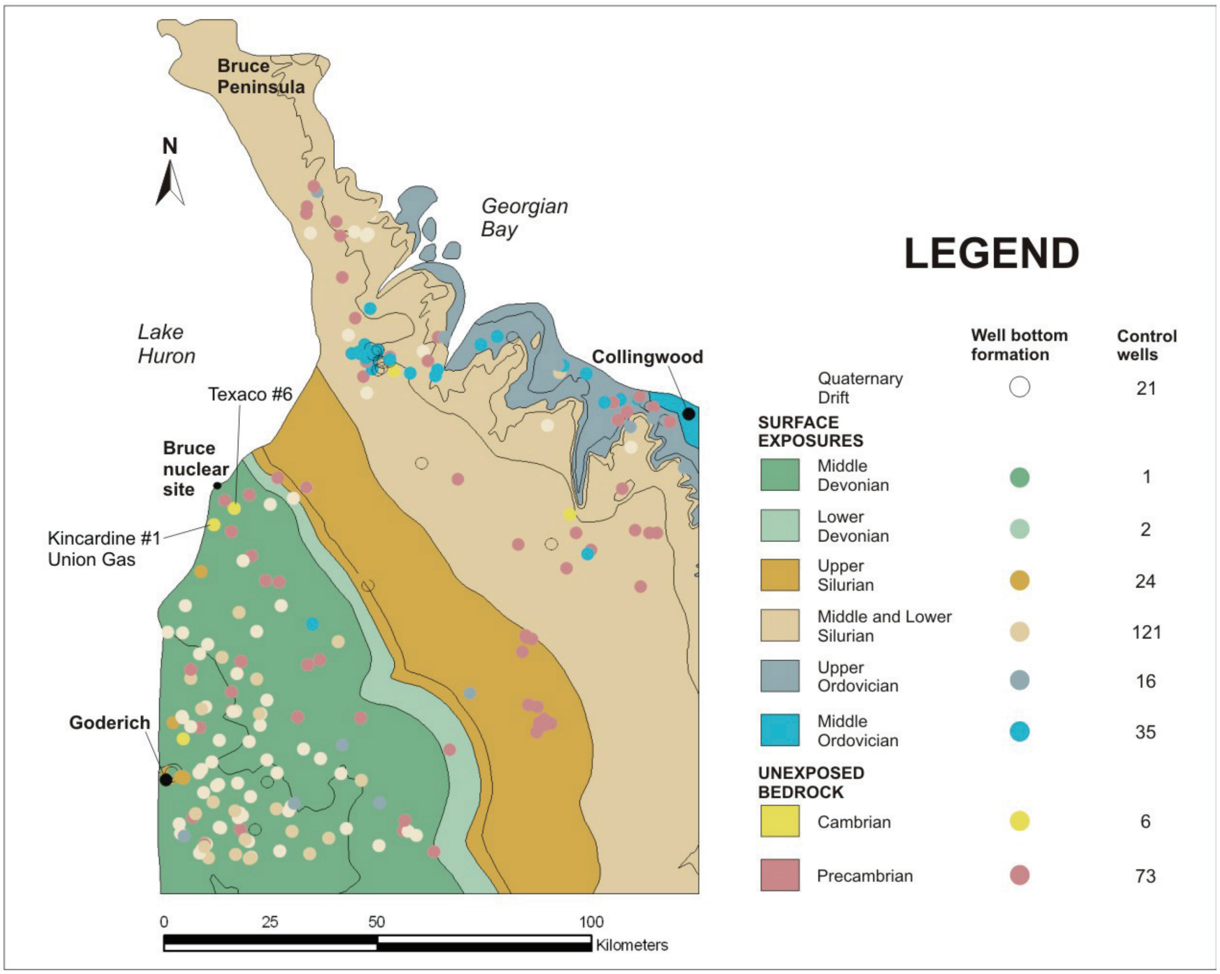
Notes: Contacts are based on field mapping and interpretations aided by subsurface drilling, borehole stratigraphic correlation, and regional compilations (LIBERTY71, BRIGHAM71, BAILEY84a, BAILEY84b, SANFORD85, SAGE91, JACOBI93, CARTER90a, EASTON95, CARTER96, WALLACH98, KETCHUM00, BOYCE02, ARMSTRONG10). BMb – Bruce Megablock; NMb – Niagara Megablock. See text for further discussion. Modified from Figure 2.5 of the Geosynthesis (NWMO11c).

Figure 4-4: Interpreted Boundaries and Fault Traces in Southern Ontario



Notes: Includes nomenclature from locations in the Michigan Basin (left), Bruce nuclear site (centre), and Appalachian Basin. † indicates outcrop nomenclature for southern and eastern Ontario. Modified from Armstrong and Carter (ARMSTRONG06).

Figure 4-5: Paleozoic Stratigraphic Nomenclature of Southwestern Ontario



Notes: Borehole control points are colour-coded to indicate the lowermost stratigraphic unit encountered in each well. The legend indicates how many of the 299 total number of wells bottom out within each stratigraphic unit. Boreholes Kincardine #1 – Union Gas and Texaco #6, proximal to the site, are mentioned in the text. Modified from Figure 1.2 of the 3DGF modelling report (NWMO11aa).

Figure 4-6: Regional Study Area Geology and Well-Control for the 3DGF Model

Each of these 299 wells is colour-coded by well bottom formation to indicate the spatial stratigraphic control in the model (Figure 4-6). The 3DGF model accurately reproduced regional stratigraphic relationships using these documented formation contact elevations and thicknesses. The final 3DGF model geometry is consistent with the regional geological framework based on published literature, maps and cross-sections of the region (ARMSTRONG06, ARMSTRONG10).

Armstrong and Carter (ARMSTRONG06) describe the occurrence of 31 formations, members or units within the Paleozoic succession from its Cambrian base to the Devonian Lucas Formation, the youngest exposed bedrock in the regional study area (Figure 4-5). When the Salina A-1, A-2, and B units are further divided into evaporite and carbonate sub-units, this totals 34 recognizable stratigraphic entities.

A recently published update of the Paleozoic stratigraphy of southern Ontario includes minor modifications to the stratigraphic nomenclature shown in Figure 4-5 (ARMSTRONG10). The middle Silurian designation has been removed and now the Upper and Lower Silurian are separated at the top of the Eramosa Member of the Guelph Formation. In addition, the Black River and Trenton groups are now both included in the Upper Ordovician Period. Acknowledging these recent re-interpretations, the stratigraphy at the Bruce nuclear site is organized according to the original framework shown in Figure 4-5 (ARMSTRONG06).

4.1.1.2 Tectonic History and Diagenesis

The tectonic evolution of southern Ontario has occurred over the last, approximately, 1210 Ma, as summarized in Table 4-1. The first half of this period involved the formation of the Precambrian Grenville basement beneath southern Ontario during the development and subsequent collapse of the Grenville Orogen (e.g., CARR00). This part of the tectonic history is discussed in Section 2.2.3.1 of the Geosynthesis (NWMO11c) and will not be discussed in detail here. It is sufficient to point out that the record of this Precambrian tectonism is preserved in the form of ancient boundary zones which are traced beneath the Phanerozoic cover of southern Ontario as shown in Figure 4-4 and discussed in Section 4.1.1 above.

At the basin-scale, the basement has remained relatively stable since at least the end of the Paleozoic (e.g., MILKEREIT92, PARK94, VANDERPLUIJM04) and apart from localized low-level seismicity near the subsurface trace of the Central Metasedimentary Belt Boundary Zone (CMBBZ), shown in Figure 4-4, there is no evidence for significant neotectonic activity localized along these ancient boundaries in southern Ontario (PERCIVAL07). This interpretation is consistent with the recognition that the Bruce nuclear site is situated within an area of low, diffuse seismicity with no identified active faults (NWMO11w) or evidence of neotectonism (NWMO11v).

Table 4-1: Timetable of Major Tectonic Events in Southern Ontario

Time Interval (MaBP)	Tectonic Activity	Reference
1210 – 1180	Regional metamorphism in CMBBZ (proto-Grenville)	EASTON92, LUMBERS90, HANMER92
1109 – 1087	Magmatism and formation of Midcontinent Rift	VANSCHMUS92
1030 – 970	Main phase of Grenville Orogeny	CARR00, WHITE00
970 – 530	Extensional rifting and opening of the Iapetus Ocean	THOMAS06
530 – 320	Subsidence of Michigan Basin and Uplift of Frontenac and Algonquin Arches (episodic)	HOWELL99 SANFORD85
470 – 440	Taconic Orogeny E-W to NW-SE compression, uplift (Frontenac and Algonquin Arches)	QUINLAN84, SLOSS82 MCWILLIAMS07
410 – 320	Caledonian/Acadian Orogeny E-W to NW-SE compression, uplift(Frontenac and Algonquin Arches)	GROSS92, MARSHAK89 SUTTER85, KESLER02
300 – 250	Alleghenian Orogeny E-W to NW-SE compression Peak burial conditions	GROSS92, ENGELDER80
200 – 50	Opening of the Atlantic Ocean St. Lawrence rift system created Reactivated Ottawa-Bonnechere Graben NE-SW extension Uplift	KUMARAPELI76 KUMARAPELI85
50 – Present	NE-SW compression (from ridge push) Post-glacial uplift	BARNETT92
Notes: Time interval ranges are approximate. See accompanying references for detailed descriptions. MaBP – Million years Before Present. Modified from Table 2-2 of the Geosynthesis (NWMO11c).		

The Phanerozoic (Cambrian to present) history of southern Ontario can be explained in terms of two protracted tectonic cycles (Figure 4-7a). Tectonic Cycle I reflects the complex interaction between regional-scale tectonic forces, sedimentation, and eustatic sea level fluctuations associated with the Appalachian-Caledonian Orogen (SANFORD85, HOWELL90, HOWELL99, COAKLEY95). This cycle includes an initial passive phase which correlates with an initial episode of subsidence and deposition within the Michigan Basin (SANFORD85). Early Middle Ordovician uplift of the arch eroded away much of the rock above it preserving a regional unconformity. In the regional study area, the unconformity is overlain by rocks of Cambrian age, where they are preserved, or rocks of the early Middle Ordovician Black River Group where the Cambrian is absent (ARMSTRONG06).

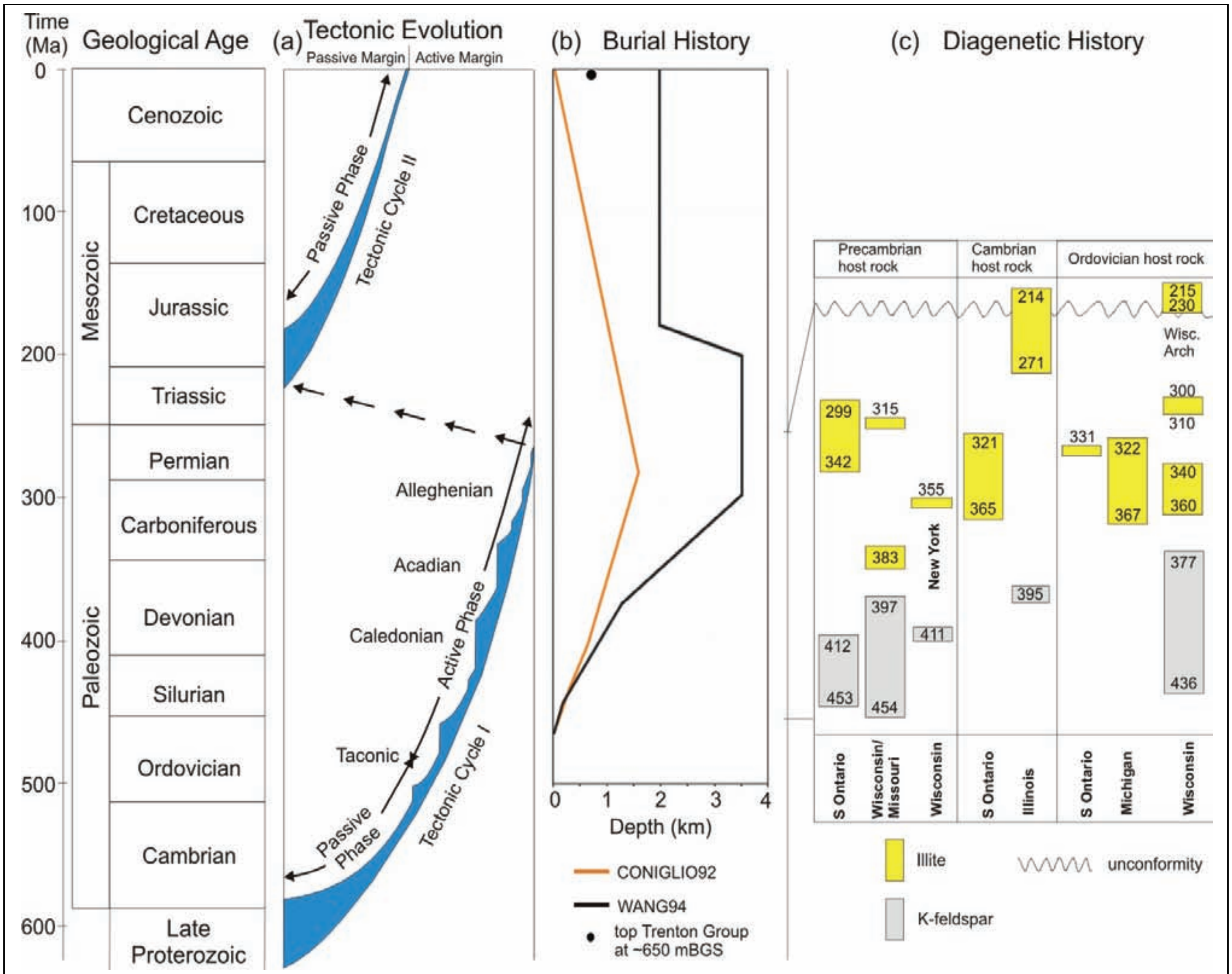
The active second phase of Tectonic Cycle I is characterized by several pulses of tectonic activity, including the Taconic (Ordovician), Caledonian/Acadian (Silurian-Devonian) and Alleghenian (Carboniferous-Permian) orogenies (Figure 4-7a). These tectonic events controlled the deposition of the Middle Ordovician to Devonian sedimentary succession as described previously (JOHNSON92, HOWELL90, HOWELL99, ARMSTRONG06), and are interpreted to have played an important role in fluid migration and diagenesis at the basin-scale (e.g., BETHKE90). The temporal association between tectonism, burial and diagenesis is shown in Figure 4-7.

Tectonic Cycle II comprises the Mesozoic evolution of the region and is characterized by the transition to a passive tectonic (extensional margin) cycle when the Atlantic Ocean began to open at the end of the Triassic Period, approximately 200 million years before present (MaBP) (Figure 4-7a). Much of the resulting tectonic activity was concentrated near the continental margin, where Triassic and Lower Jurassic rift basin deposits record the onset of continent break-up (e.g., LINDHOLM78). Further inland, the majority of rift-related deformation occurred in proximity to the trace of the Appalachian thrust front (WHEELER95). Pre-existing faults, including those of the Neoproterozoic to Early Cambrian (Iapetan) St. Lawrence rift system, and the Ottawa-Bonnechere Graben structure (Figure 4-4), were re-activated as a system of NE-striking extensional normal faults and WNW-striking transfer faults (THOMAS06). These areas of re-activation, all farther than 150 km from the Bruce nuclear site, remain seismically active to the present-day (KUMARAPELI66, ADAMS91). Mesozoic volcanic activity, evidence for which is also found only at a considerable distance (>150 km) from the Bruce nuclear site, will be discussed in detail in Section 4.5.2.

The following sections discuss the temporal relationship between the active phase of Tectonic Cycle I (Figure 4-7a), sediment burial and thermal history (Figure 4-7b), and diagenesis (Figure 4-7c). This information provides evidence to support the conclusion that tectonically-related perturbations to the Paleozoic sedimentary succession ceased in importance by the end of the Paleozoic or earliest Mesozoic. The reader is also referred to Section 2.2.5.3 of the Geosynthesis (NWMO11c) for a more detailed treatment of the information presented below.

Burial and Thermal History

Two independent estimates of burial depth and timing are shown in Figure 4-7b. The orange curve, based on a study of Ordovician diagenesis from Manitoulin Island (CONIGLIO92), and the black curve, based on a basin-scale analysis of apatite fission track dates (WANG94), both indicate a late Carboniferous to early Permian timing for peak burial. These studies were undertaken near the margin (orange curve), and closer towards the centre (black curve), of the Michigan Basin, thus explaining the differences in total burial depth of 1500 m and 3500 m, respectively. They both suggest that approximately 1500 m of sediment has since been eroded (Figure 4-7b).



Notes: (a) Tectonic Evolution: Band widths represent relative tectonic intensity (modified from SANFORD85).
(b) Burial History: The orange curve (CONIGLIO92) and black curve (WANG94) provide burial duration and magnitude estimates for locations in the Michigan Basin. The • indicates the present-day burial depth of approximately 675 mBGS for the top of the Middle Ordovician Trenton Group at the Bruce nuclear site (modified from CONIGLIO92, WANG94).
(c) Diagenetic History: Duration of secondary mineralization diagenesis for southern Ontario and the region around the Michigan Basin. Documented ages (~454 to 214 Ma) are indicated by number(s) within boxes (grey fill – K-feldspar; yellow fill – illite). These ages coincide with the main pulses of Paleozoic orogenesis during Tectonic Cycle I. Diagenesis schematic has been enlarged for clarity. Lines extending beyond left margin in (c) indicate the approximate time interval relative to (a) and (b) (modified from ZIEGLER00a). See text for further discussion.

Figure 4-7: Phanerozoic Tectonic Cycles and Burial and Diagenetic History for the Michigan Basin

Given that the top of the Ordovician succession exposed at Manitoulin Island is encountered at approximately 450 mBGS beneath the Bruce nuclear site (NWMO11k), and the Bruce nuclear site is located slightly closer to the basin centre, it is reasonably estimated that approximately 1000 m of sediment has been eroded from above the existing Paleozoic succession at the site (NWMO11c).

Based on this total erosion estimate, an approximate peak burial in-situ temperature of 70 °C was calculated for the top of the Trenton Group limestones (Collingwood Member) which is encountered at approximately 650 mBGS beneath the site (NWMO11k; also see discussion in Section 2.2.5.3 of NWMO11c). The estimated peak temperature is consistent with the interpretation that the Upper Ordovician shales directly above the Collingwood Member barely reached the lower threshold of the oil window in terms of thermal maturation (e.g., Section 3.7.4.2 in NWMO11k). At the regional-scale, the conodont alteration index designation of Legall et al. (LEGALL81) indicates a very limited potential for in-situ petroleum generation in rocks as deep as the Middle Ordovician Trenton Group in southern Ontario (POWELL84). This interpretation is also consistent with the above temperature estimate.

Diagenesis

Diagenetic processes that have influenced the Paleozoic rocks within the Michigan Basin include clay alteration, dolomitization, Mississippi Valley Type (MVT) mineralization, salt dissolution, precipitation of late stage cements and oil and gas generation and migration. Some studies of the Michigan Basin document fluid inclusion homogenization temperatures and degree of organic maturation that cannot be explained by burial history alone and therefore requires the influence of additional heat sources (e.g., CONIGLIO92). These same heat sources provide the mechanisms for diagenetic fluid flow. Important features of the diagenetic history of the Michigan Basin are described briefly below.

Two stages of diagenetic secondary mineral growth have produced clay mineral alteration products along the unconformable contact between the Precambrian basement and overlying Paleozoic cover as shown in Figure 4-7c (ZIEGLER00a). Based on this observation, a conceptual model was suggested whereby regional brine migration was focused along the unconformity in response to hydraulic gradients and crustal motion related to Appalachian orogenesis (ZIEGLER00a). The distribution of secondary mineral ages for the Michigan Basin and surrounding regions, based on the radiogenic (Potassium-Argon) dating of secondary illite (yellow fill) and K-feldspar (grey fill), are shown in Figure 4-7c. As can be seen, the range of ages spans the entire active phase of Tectonic Cycle II (Figure 4-7a) and was concurrent with deposition and burial of the Paleozoic succession (Figure 4-7b). K-feldspar alteration was initiated early during the Taconic Orogeny and continued through to the end of Caledonian-Acadian Orogeny (Figure 4-7c). Illite alteration was contemporaneous with the Acadian and Alleghenian orogenies as shown in Figure 4-7c (HARPER95, ZIEGLER00a, ZIEGLER00b). In the broader region, illitization is interpreted to have continued until approximately 214 MaBP (Figure 4-7c).

Hydrothermal dolomitization selectively altered the Paleozoic rocks along, and adjacent to, discrete fracture systems which in turn appear to be controlled by basement-seated faults. The timing of dolomitization events ranged from during or shortly after marine carbonate deposition during the Ordovician, to the Late Paleozoic or Early Mesozoic in correspondence with the timing of peak burial compaction. The conditions that led to dolomitization within the regional study area of the Michigan Basin (i.e., basinal groundwater flow, fracture-related tectonically driven flow, and hydrothermal dolomitization) have not existed since the late Paleozoic or early Mesozoic (e.g., CONIGLIO92).

The key post-dolomitization diagenetic phases are all volumetrically minor and include late stage calcite cements, MVT mineralization, and late stage anhydrite and gypsum (BUDAI91, CONIGLIO94). Consistent with the range of secondary mineralization ages, the conditions that led to dolomitization within the regional study area of the Michigan Basin (i.e., basinal groundwater flow, fracture-related tectonically driven flow, and hydrothermal dolomitization) have not existed for approximately 200-250 Ma (e.g., CONIGLIO92), since the time of peak burial.

Salt dissolution is typically identified at the margin of the Michigan Basin in a zone extending from the Bruce Peninsula south along Lake Huron and into southwestern Ontario. The process of dissolution is interpreted to have occurred via fluid migration through regional fractures and faults and the affected zones are brecciated and characterized by an evaporite cement filling (gypsum and/or anhydrite) enclosing dolostone clasts (SANFORD85). At the Bruce nuclear site, salt dissolution has occurred throughout the middle to lower Salina Group units. Pervasive cementation and fracture infilling has resulted in very low measured hydraulic conductivities in the Silurian rocks beneath the Bruce nuclear site (NWMO11k). Salt dissolution occurred primarily during the Late Silurian to Devonian Caledonian Orogeny. A second major salt dissolution event occurred during the Late Devonian-Mississippian Acadian Orogeny (SANFORD85).

4.1.1.3 Structural Overview

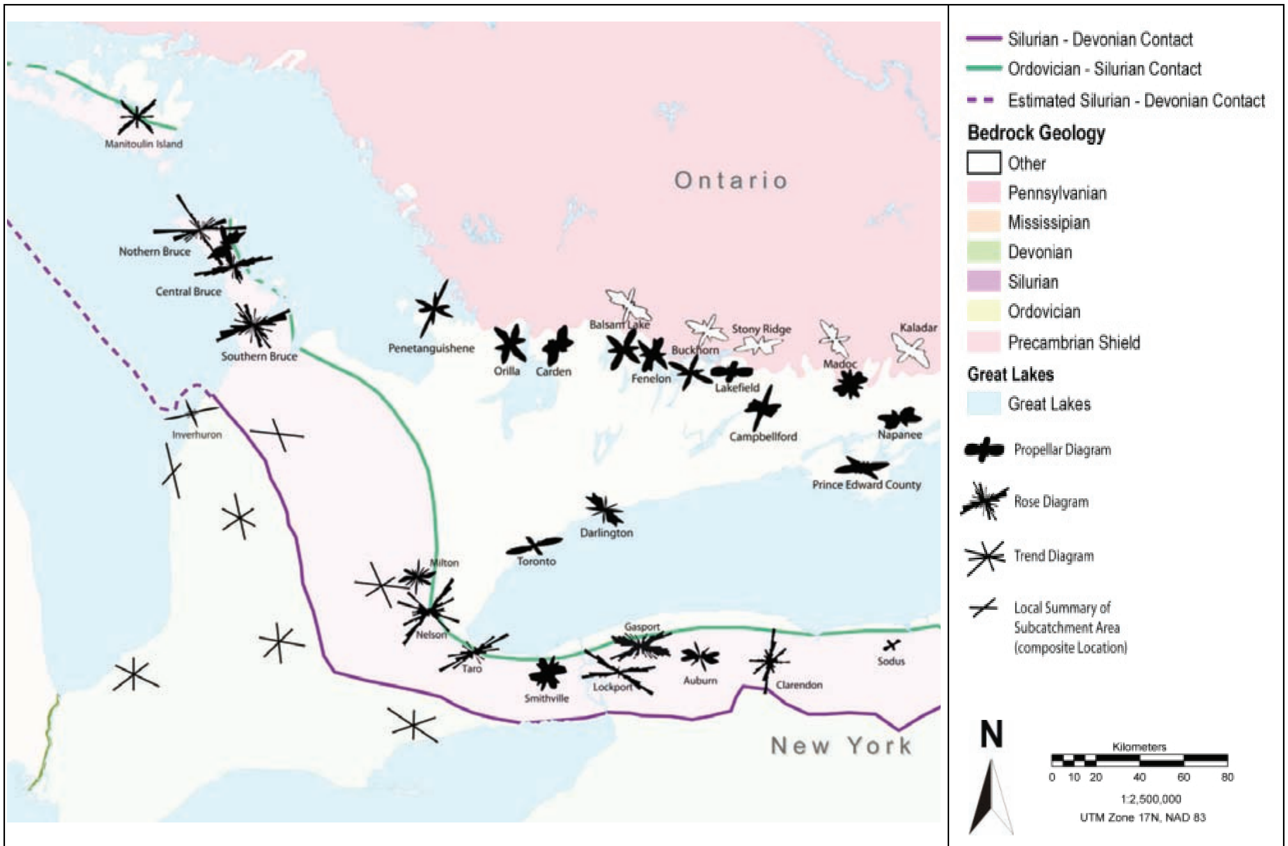
Figure 4-4 shows all faults known to displace the Proterozoic/Paleozoic unconformity in southwestern Ontario (CARTER96, ARMSTRONG10). This analysis is based on geophysical and borehole data, and regional compilations (BRIGHAM71, BAILEY84a, BAILEY84b, CARTER96). Within southeastern Ontario where there is an abundance of subsurface data available, the faults have been mapped with a high degree of confidence. The faults shown in Figure 4-4 are grouped based on observation of the youngest stratigraphic unit that is offset (ARMSTRONG10). The oldest faults, indicated by the green lines on Figure 4-4, only offset Cambrian strata and rocks of the immediately overlying Ordovician Shadow Lake Formation. Another group of faults, indicated by the red lines on Figure 4-4, offset rocks as young as the Ordovician Trenton Group limestones. The youngest mapped faults in southern Ontario, indicated by the solid black lines on Figure 4-4, offset rocks of the Silurian Rochester (Lions Head equivalent) Formation (ARMSTRONG10).

Within the regional study area, where subsurface data are sparse, these features are inferred by subsurface structure contouring and isopach mapping, with limited well-control, or through seismic interpretation. As a result, these faults are poorly constrained in terms of location and movement history and are mapped with a low degree of confidence. Regardless, the closest interpreted fault structure is >25 km away from the proposed DGR footprint (Figure 4-4) and it is overlain by undisturbed Ordovician strata. As well, no mapped faults within the regional study area are interpreted to be younger than the limestones of the Ordovician Trenton Group (ARMSTRONG10).

In a conceptual tectonic model for southern Ontario, a megablock model was proposed in which the Bruce Megablock was distinguished as a distinct tectonic unit with a simple ESE-trending fracture network from a complexly fractured Niagara Megablock to the south (SANFORD85). This model was based on satellite lineament mapping of the Precambrian shield in conjunction with interpretation of subsurface data from southern Ontario with the fracture networks thought to be controlled by Paleozoic re-activation of pre-existing basement-seated faults (SANFORD85). The blue lines on Figure 4-4 outline these two interpreted megablock regions. While the distribution of mapped faults in southwestern Ontario appear to agree with the complex interpretation of Sanford et al. (SANFORD85) for the Niagara Megablock, the sparse faults mapped within the Bruce Megablock show no clear relationship with their tectonic interpretation.

Regional Fracture Patterns

Perhaps the best gauge of the history of tectonic forces in southern Ontario are the regionally consistent, systematic fractures, which have formed in response to loading or unloading of the rock mass. The majority of fractures observed in southern Ontario exhibit no measureable slip or dilation at the scale of observation and are therefore classified as joints (e.g., HANCOCK85). The Regional Geomechanics - Southern Ontario report (NWMO11n) provides a review of the literature with respect to joint orientation and location both regionally and through geologic time. The following section summarizes the fracture network of the regional study area. The distribution of documented joint orientations within and surrounding the regional study area is shown in Figure 4-8.



Notes: Joint orientations are plotted as Gaussian contoured and smoothed propeller, rose and trend diagrams. Inverhuron is immediately south of the Bruce nuclear site. Modified from Figure 2.4 of the Regional Geomechanics report (NWMO11n).

Figure 4-8: Joint Orientations In and Around Southern Ontario for the Paleozoic Cover and Precambrian Basement

Fracture orientations in the regional study area cluster into two major sets trending approximately NW to NNW and NE to ENE. A third set trends approximately ESE (Figure 4-8). A westerly rotation of the Set 1 fractures towards the north and west, based on interpretation of measured fracture orientations in Paleozoic strata from the northern and northwestern flanks of the Michigan Basin (HOLST82), suggests that they are part of a basin-concentric fracture pattern which may have been formed due to radial tensile stresses generated during middle to late Paleozoic basin-centred subsidence (HOWELL99, NWMO11ab).

4.1.2 Bruce Nuclear Site Geological Setting

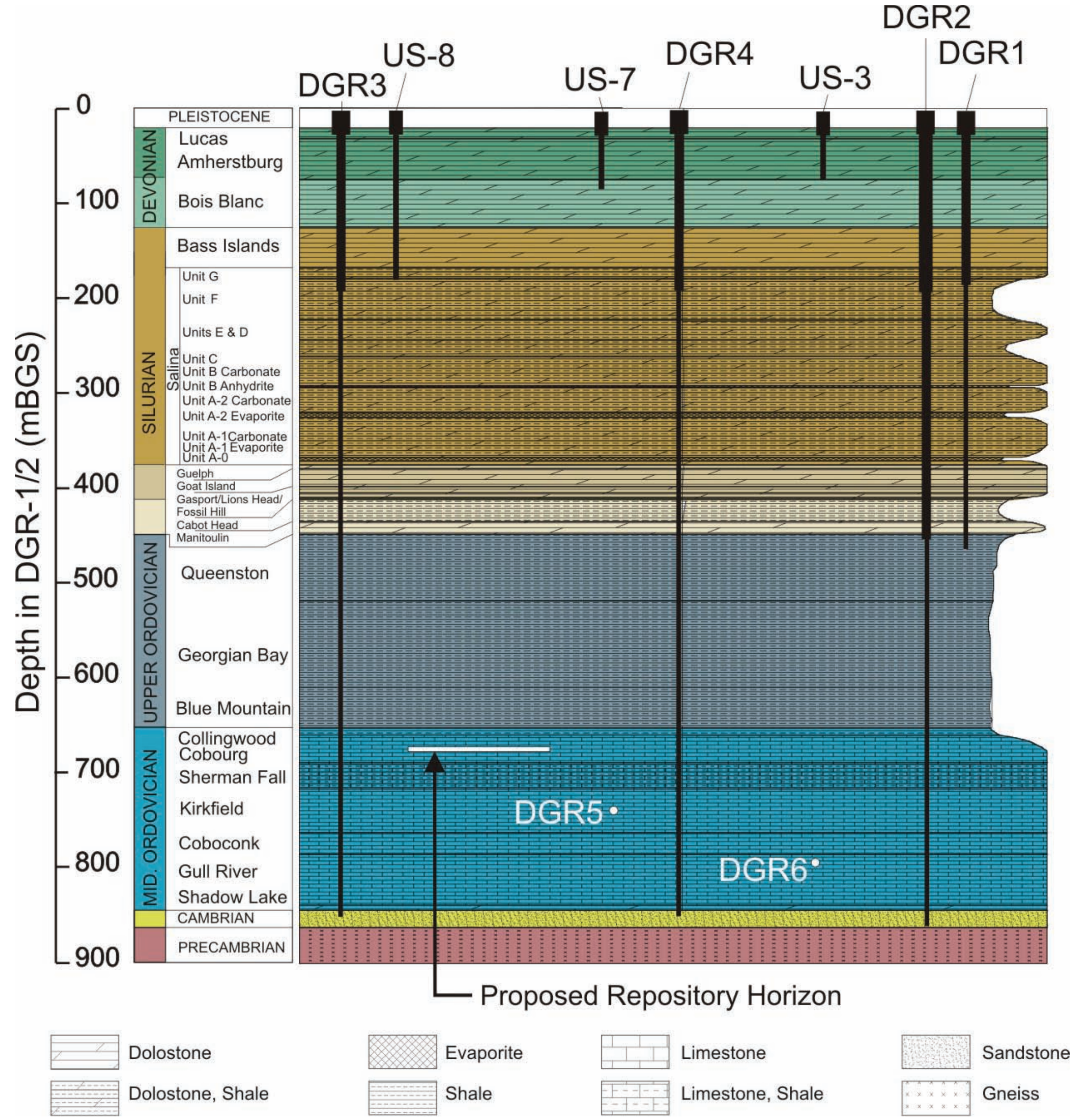
The following sections provide an overview description of the geological setting of the Bruce nuclear site. This includes a brief introduction to the stratigraphy encountered beneath the site, and then follows with a more detailed discussion of the predictable nature of this stratigraphic succession, based primarily on the results of the site characterization activities described in the DGSM (NWMO11k). The geological overview also includes a detailed discussion of the site-scale structural setting.

4.1.2.1 Stratigraphy

Drilling, logging, and testing of boreholes DGR-1 to DGR-6 at the Bruce nuclear site led to the identification of 34 distinct Paleozoic bedrock formations, members, or units of approximately 840 metres cumulative thickness beneath a thin veneer (7 to 20 m) of Pleistocene overburden and unconformably overlying Precambrian granitic gneiss (Figure 4-9; NWMO11k). The reference Paleozoic sequence, based on core logging of the DGR-1 and DGR-2 boreholes, comprises 104.0 m of Devonian dolostone, 323.7 m of Silurian dolostone, argillaceous dolostone, shale and evaporite, 211.8 m of Upper Ordovician shale, 179.1 m of Middle Ordovician argillaceous limestone, 5.2 m of Ordovician siltstone and sandstone, and 16.9 m of Cambrian sandstone (Figure 4-9). A total of 1.55 m of the Precambrian basement was sampled at the bottom of DGR-2 (NWMO11k). The proposed DGR underground facilities will be located within argillaceous limestone of the Middle Ordovician Cobourg Formation and situated beneath a thick (>200 m) Upper Ordovician shale-dominated sequence (Figure 4-9). The following is a brief summary of the rock units encountered based on the detailed borehole logging descriptions (NWMO11k).

The Pleistocene overburden typically comprises 1 to 3 m of surficial fill, and/or sand and gravel overlying 5 to 21 m Elma-Catfish Creek till, a clayey silt to sandy silt glacial deposit (SHARPE79). The till is underlain by 0 to 2 m of basal gravel deposited at the weathered bedrock surface.

The Devonian dolostone interval includes the highly permeable rocks of the Lucas, Amherstburg, and Bois Blanc formations. The Lucas Formation is a thin- to medium-bedded, light to grey-brown, finely-crystalline, dolostone with stromatolitic laminations and abundant calcite-filled fractures and vugs. The Amherstburg Formation is a tan to grey-brown, fine- to coarse-grained, fossiliferous dolostone, which is extensively fractured and vuggy. The Bois Blanc Formation is a light grey to brown cherty dolostone with wavy argillaceous laminae throughout. A major erosional unconformity occurs at the base of the Devonian interval.



Notes: Vertical boreholes penetration depths are indicated by vertical black lines. White dots indicate approximate depth of penetration for angled boreholes DGR-5 and DGR-6. Figure was developed based on information from the DGSM (NWMO11k) and modified from Figure 2.25 of the Geosynthesis (NWMO11c).

Figure 4-9: Stratigraphic Sequence Beneath the Bruce Nuclear Site

The Silurian interval includes the Bass Islands Formation, the Salina Group of 12 Units, and the underlying Guelph, Goat Island, Gasport, Lions Head, Fossil Hill, Cabot Head, and Manitoulin formations and members (Figure 4-5). The Lions Head Member of the Amabel Formation, the Gasport and Goat Island members of the Lockpot Formation, and the Guelph Formation have been grouped collectively as the Niagaran within the 3DGF model (Figure 4-5; NWMO11aa). A similar grouping of formations is employed for the regional- and site-scale hydrogeologic modelling (NWMO11p) as discussed in Section 4.4.

The Bass Islands Formation is a light brown to tan-grey, variably laminated, very fine- to fine-grained argillaceous dolostone. It exhibits a high degree of natural fractures, which are either open or calcite infilled. The Salina Group includes a succession of evaporites and evaporite-related carbonate sediments subdivided into units A through G. They comprise tan to brown to grey, thin- to medium-bedded, dolostones to argillaceous dolostones, with shale and anhydrite interbeds, and with locally abundant gypsum and anhydrite veins. Brecciation is evident in the middle and lower part of the interval owing to salt dissolution.

The A-1 Carbonate has open vuggy porosity and permeability at its top and shows oil hydrocarbons seeping from its base. The Guelph Formation is a porous and permeable, vuggy, sucrosic dolostone with abundant halite infilled veins, minor disseminated pyrite, and minor seeps of oil hydrocarbon. The Goat Island Member is a light brown-grey, very fine-grained, moderately fossiliferous, thin- to medium-bedded dolostone with minor chert and microstylolites. The Gasport Member is a blue-grey to white, fine- to coarse-grained, dolomitic limestone with bituminous laminations and stylolites throughout. The Lions Head Member is a grey-brown, fine-grained, dolostone with sparse fossils and locally abundant chert nodules. The Fossil Hill Formation is a light brown-grey, coarse-grained, thin- to medium-bedded, fossiliferous dolostone. The Cabot Head Formation is a green-grey to red massive shale with grey carbonate interbeds and, near its base, black fossiliferous shale. The Manitoulin Formation is a grey, fine- to medium-grained, locally cherty, dolostone with minor interbeds of grey-green non-calcareous shale. Its base marks a major erosional unconformity with the underlying Ordovician shales.

The Ordovician rocks encountered are sparsely fractured and generally of very low permeability and porosity. The Upper Ordovician interval includes the shale-dominated Queenston, Georgian Bay, and Blue Mountain formations. The Queenston Formation is a massively bedded, red-maroon to locally grey-green calcareous shale with abundant halite near its top and minor limestone interbeds near its base (Figure 4-10). Through the middle of the unit is an interval rich in green shale with medium- to coarse-grained, grey fossiliferous, limestone interbeds.

Note: From Figure 2.29 of the Geosynthesis (NWMO11c).

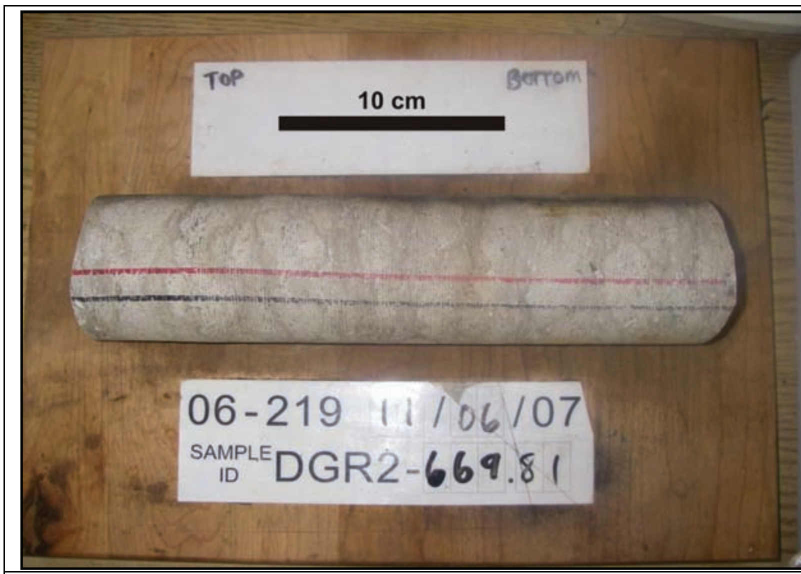
Figure 4-10: Core Sample of Green and Red Calcareous Shale, Upper Ordovician Queenston Formation, 454.82 mBGS, DGR-1

Note: From Figure 2.28 of the Geosynthesis (NWMO11c).

Figure 4-11: Core Sample of Interbedded Shale and Limestone, Georgian Bay Formation, 542.25 mBGS, DGR-2

The Georgian Bay Formation is a dark grey-green shale with grey, fine- to medium-grained, limestone, siltstone, and/or sandstone interbeds whose frequency decreases with depth (Figure 4-11). Minor halite-infilled fractures and a petroliferous odour are evident towards its base. The underlying Blue Mountain Formation is predominantly dark greenish-grey shale with grey siliceous siltstone and sandstone, and fossiliferous limestone, and transitioning into dark grey calcareous shale at its base. It exhibits a petroliferous odour throughout.

The Middle Ordovician interval includes sparsely fractured low-permeability and low-porosity argillaceous limestones of the Trenton and underlying Black River groups. The Trenton Group includes the Cobourg, Sherman Fall, and Kirkfield formations. The Cobourg Formation is further subdivided based on lithology into upper and lower members. The upper Collingwood Member comprises a dark grey to black, organic-rich, calcareous shale with thin fossiliferous interbeds. It is distinctive, regionally, based on an increase in organic content but still with a predominantly carbonate composition (ARMSTRONG06). It also has a petroliferous odour throughout and shows minor oil hydrocarbon seeps. The underlying Lower Member is characterized by coarse-grained, fossiliferous, bluish-grey to grey-brown limestone and argillaceous limestone (Figure 4-12). Unless otherwise indicated, reference to the Cobourg Formation, or simply Cobourg, throughout the rest of this chapter implies reference to the Lower Member of the Cobourg Formation.



Note: From Figure 2.27 of the Geosynthesis (NWMO11c).

Figure 4-12: Core Sample of Argillaceous Limestone from the Repository Horizon Depth, Cobourg Formation, 669.81 mBGS, DGR-2

The Sherman Fall Formation is a grey-brown, coarse-grained, argillaceous limestone interbedded with calcareous shale near its base. The Kirkfield Formation is a tan to dark grey, fine-grained, irregular-bedded, fossiliferous and argillaceous limestone with dark grey-green shale interbeds. It emits a petroliferous odour and has minor oil hydrocarbon seeps near its base.

The Black River Group, in comparison to the Trenton Group, has a lower argillaceous content overall and has a prevalent petroliferous odour with minor oil hydrocarbon seeps throughout. It comprises the Coboconk, Gull River, and Shadow Lake formations. The Coboconk Formation is a light- to medium-grey, very fine-grained, bioturbated limestone with minor dark grey-green shale interbeds and a characteristic mottled texture. Minor seeping oil hydrocarbon is observed below its mid-point, and an approximately 10 cm thick bentonite bed, interpreted as a volcanic ash layer (e.g., KOLATA98), is observed at approximately 7 m below its upper contact. The Gull River Formation is a medium grey, fine- to very fine-grained, fossiliferous limestone with thin dark grey shale interbeds. A 60 cm thick tan dolostone horizon is traceable through the mid-point of this formation. The Shadow Lake Formation is a dolomitized silty limestone with sandy mudstone and coarser sandstone layering. The base of this unit marks an unconformity with the underlying Cambrian.

The Cambrian is a tan to orange-grey, fine- to medium-grained, silty sandstone and sandy dolostone with clasts of the underlying granitic basement, abundant calcite infilled veins and vugs, and glauconite stringers. Its base is a quartzose sandstone and its upper portion is up to 100% dolomitized. Only a very small portion of the underlying Precambrian basement was intersected during drilling. It is described as a pink to grey, fine- to medium-grained, felsic granitic gneiss with extensive alteration along its upper contact and has a well-defined tectonic foliation marking an erosional unconformity with the overlying Cambrian. The Cambrian unit pinches out to the east of the Bruce nuclear site along the flank of the Algonquin Arch (e.g., BAILEY84a).

4.1.2.2 Predictability of the Ordovician Sedimentary Rocks

This section builds a case for site-scale predictability of the Ordovician stratigraphy beneath the Bruce nuclear site. The assessment is based on the recognition that data collected during core logging and subsequent analysis of this data shows a marked consistency between individual DGR boreholes. Predictability will be discussed in terms of the following characteristics:

- Stratigraphic thicknesses;
- Vertical and horizontal lithofacies distribution;
- Marker bed traceability;

- Formation-scale major mineralogy and geochemistry;
- Fracture infilling; and
- Presence of hydrocarbons.

Stratigraphic Thicknesses

Intersection of the Ordovician stratigraphy by the DGR boreholes, except for the deepest formations in DGR-5 and DGR-6, allows for an assessment of the uniformity in thicknesses across the site (Table 4-2). Individual formations vary in thickness by a few metres at most, but the similarity in total thickness suggests that these variations are minor in importance. Two of the nearest boreholes beyond the boundaries of the Bruce nuclear site, Kincardine #1 - Union Gas and Texaco #6 (Figure 4-6), yielded very similar total Ordovician thicknesses of 393.5 and 393.1 metres, respectively. Table 4-2 also reports the strike and dip value for each formation. Not surprisingly these values are also consistent through the entire interval.

Table 4-2: Summary of Strike, True Dip, and Thicknesses of Ordovician Formations and Members Encountered in the DGR Boreholes

Ordovician Formation/Member	Strike	Dip	Thickness (m)				
			DGR-2	DGR-3	DGR-4	DGR-5	DGR-6
Queenston	N24ºW	0.41ºSW	70.3	74.4	73.0	70.3	69.3
Georgian Bay	N17ºW	0.61ºSW	90.9	88.7	88.7	88.6	88.2
Blue Mountain	N23ºW	0.51ºSW	42.7	44.1	45.1	45.1	45.0
Collingwood Member	N14ºW	0.56ºSW	7.9	8.7	8.4	8.6	6.5
Cobourg	N14ºW	0.60ºSW	28.6	27.8	27.5	27.1	28.5
Sherman Fall	N17ºW	0.57ºSW	28.0	28.9	28.3	29.3	28.8
Kirkfield	N18ºW	0.63ºSW	45.9	45.8	45.7	-	46.8
Coboconk	N19ºW	0.63ºSW	23.0	23.7	23.8	-	22.4
Gull River	N16ºW	0.66ºSW	53.6	51.7	52.2	-	-
Shadow Lake	N19ºW	0.56ºSW	5.2	4.5	5.1	-	-
Total Ordovician Thickness			396.1	398.3	397.8	-	-
Notes: Strike and true dip values are based only on data from the vertical boreholes, DGR-2 to DGR-4 (DGR-1 only intersected the top of the Queenston Fm and therefore is not included in this analysis). Dashes (-) are required where DGR-5 and DGR-6 did not intersect the entire Ordovician interval, as illustrated in Figure 4-9. Data are from Table 3.1 and Table 3.2 of the DGSM (NWMO11k).							

Lithofacies Analysis

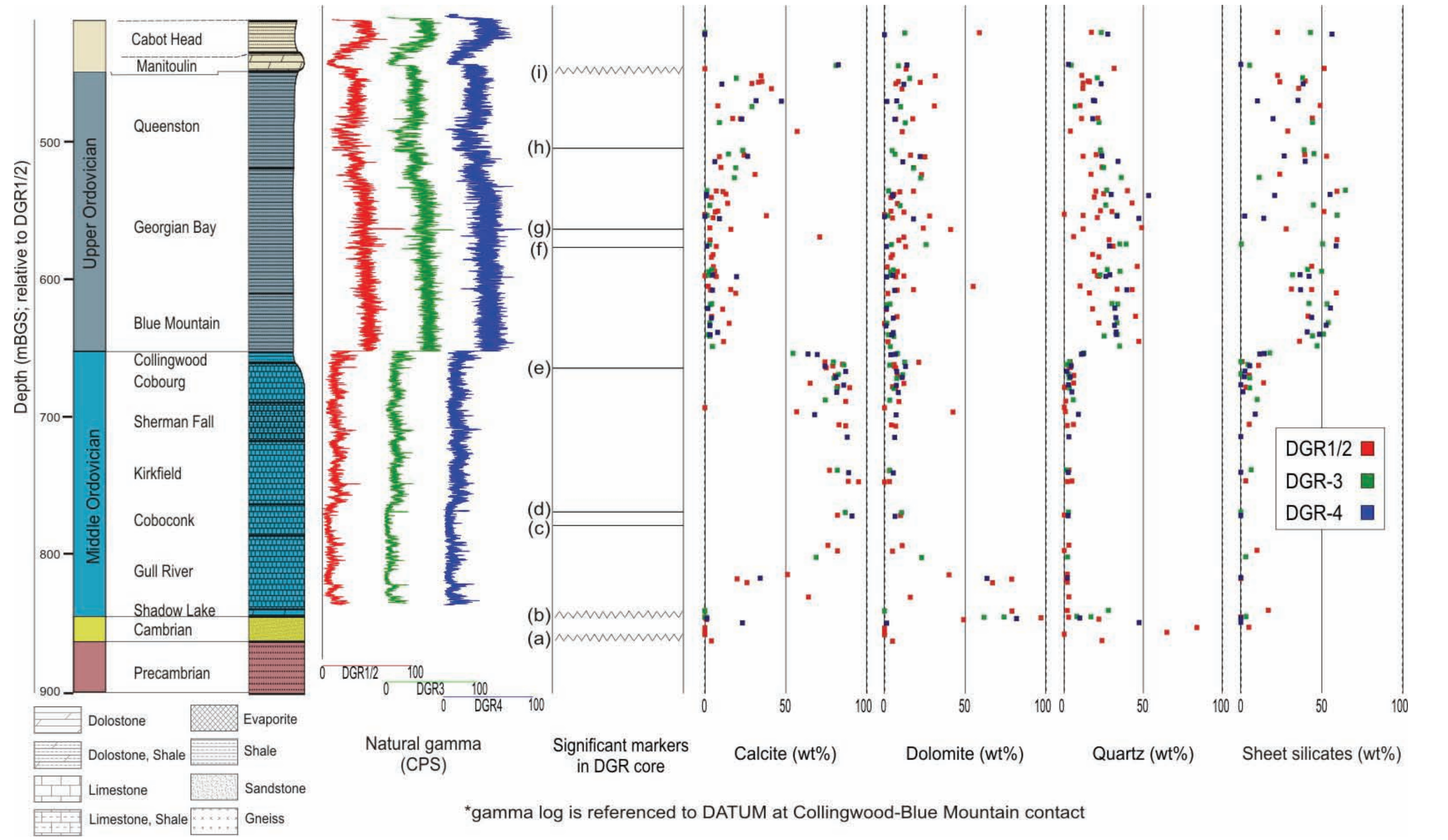
In order to fully assess the degree of predictability of individual lithofacies at the site-scale, an evaluation of the lateral (horizontal) homogeneity and vertical variation of lithofacies within key Ordovician intervals was conducted (see also NWMO11c). Vertical borehole coverage (DGR-2, DGR-3, and DGR-4) around the periphery of the proposed DGR footprint provides the data control for this analysis (Figure 3-1). Facies variations are caused by the changing dynamics of the depositional environment, and can potentially alter the hydrogeological and mechanical properties of the rock mass. If sufficient homogeneity exists, then the important geophysical, geomechanical, and hydrogeological datasets may be associated to specific lithologies. A positive correlation of intraformational facies changes between the boreholes would, therefore, allow interpolation of the lithostratigraphy across the proposed DGR footprint. The specific targets for this analysis were portions of the cap rock shales (Queenston and Georgian Bay formations) and the host rock (Cobourg Formation) for the proposed DGR (Figure 4-13). The reader is directed to Section 2.3.4.1 of the Geosynthesis for a complete description of the lithofacies analysis (NWMO11c). Important conclusions based on this work are discussed below.

- The natural gamma ray profiles for the Ordovician section from each of boreholes DGR-1/2, -3, and -4 as plotted in Figure 4-13 show a consistent bimodal distribution of Counts Per Second (CPS) values. A high CPS count in the upper interval highlights the > 200 m thick shale-dominated Upper Ordovician rock sequence, which represent the primary cap rock to the proposed DGR, above the low CPS count and carbonate-rich Middle Ordovician sequence.

- Consistency in natural gamma profiles, as shown in Figure 4-13, supports the assessment of uniform unit thicknesses and a structurally simple geometry across the site.

- Lithological variation is likely to occur as minor, dm- to cm-scale typically, conformable changes in quantities of shale, siltstone, or limestone of mm- to cm-thick beds as evidenced by minor variation of the gamma ray profiles between boreholes (Figure 4-13).

This last point is not unexpected given the nature of the carbonate shelf depositional environments characteristic of the Middle Ordovician (e.g., LEHMANN95) and the clastic-dominated shallow prograding coastal plain and deltaic depositional environment characteristic of the Upper Ordovician (BROGLY98).



Notes: Marker units (a to i) are discussed in the text. Jagged lines at (a), (b), and (i) indicate stratigraphic unconformities recognized in all DGR boreholes. Figure is based on data from the DGSM (NWMO11k) and modified from Figure 2.30 of the Geosynthesis (NWMO11c).

Figure 4-13: Ordovician Lithostratigraphy, Natural Gamma Profiles, Marker Units and Major Mineralogy

Therefore, the Ordovician stratigraphy at the Bruce nuclear site is considered to be laterally homogeneous and predictable at the dm- to m-scale and the lithostratigraphy is considered to be consistent and predictable at the site-scale. As will be discussed in Section 4.4, hydraulic testing of these formations demonstrates that regardless of the small-scale vertical lithofacies variation, the hydraulic conductivities remain extremely low throughout the entire Ordovician interval.

Marker Beds

Several laterally continuous marker beds, (c), (d), (f) and (h) in Figure 4-13, were identified during DGR core logging activities and provide a further indication of formation lateral continuity at the site-scale as shown in Table 4-3 (WIGSTON09, STERLING10). These marker beds are typically 10-20 cm thick beds with visually identifiable lithofacies features and/or borehole geophysical logging signatures that are distinct from the surrounding rocks. Figure 4-14 shows the Georgian Bay fossiliferous limestone bed as an example marker bed observed in the recovered core (NWMO11k).

The lithofacies analysis discussed in the previous section indentified other marker beds during a more detailed examination of the Ordovician units. One, (g) in Figure 4-13, corresponds to a marked CPS spike in the middle of the gamma profile at the same stratigraphic depth in the Georgian Bay Formation in all boreholes. Visual core inspection confirmed that this spike is lithologically controlled and defined by the sharp transition from a distinct 3 to 15 cm thick fossiliferous limestone bed into underlying dark shale. A distinct 3 to 4 cm thick shale marker again with a distinct CPS spike, (e) in Figure 4-13, was also identified within the Cobourg Formation. Other stratigraphic features which are traceable across the site include regionally recognized unconformable horizons at the base and top of the Cambrian, (a) and (b) in Figure 4-13 respectively, and the top of the Queenston Formation, (i) in Figure 4-13.

Table 4-3: Summary of Marker Bed Descriptions, Depths and Orientations Determined from Core Logging

Formation	Marker Bed or Horizon	Depth (mLBGS)					Orientation	
		DGR-1/2	DGR-3	DGR-4	DGR-5	DGR-6	Strike	Dip
Salina F Unit	brown dolostone bed in grey shale	182.0	200.7	181.5	--	--	N32ºW	0.98ºSW
Queenston (h)	limestone bed in shale	504.3	517.7	505.6	546.0	568.6	N17ºW	0.61ºSW
Georgian Bay (f)	fossiliferous limestone bed in grey shale	576.5	589.2	577.9	622.3	649.6	N14ºW	0.56ºSW

Formation	Marker Bed or Horizon	Depth (mLBGS)					Orientation	
		DGR-1/2	DGR-3	DGR-4	DGR-5	DGR-6	Strike	Dip
Coboconk (d)	dark grey volcanic ash bed in grey limestone	768.8	781.0	769.0	--	876.7	N19ºW	0.55ºSW
Coboconk (c)	tan dolostone bed in grey limestone	778.7	790.5	778.3	--	888.0	N22ºW	0.54ºSW

Notes: Lowercase letters in parentheses in first column on left refer to specific marker beds indicated on Figure 4-13. Data is from Table 3.12 of the DGSM (NWMO11k).

Rock Mineralogy and Geochemistry

Samples of core recovered from the DGR-series of boreholes were subjected to a suite of laboratory tests to determine the rock mineralogy and lithogeochemistry, as well as for comparison wtih the stratigraphic and lithologic descriptions (ARMSTRONG06, ARMSTRONG10). Results for the Ordovican interval are shown in Figure 4-13 and discussed below (see also Section 2.3.5 in NWMO11c).

- The Upper Ordovician shales are dominated by sheet silicates, with increasing amounts of quartz with depth. Moderate amounts of calcite and dolomite are noted, particularly in the Queenston Formation, and decreasing in percentage with depth. Predictably, the Middle Ordovician limestone formations consist of typically greater than 80% calcite, with the remainder comprising sheet silicates, dolomite, and quartz.

- Dolomitization is evident in varying proportions in parts of the Queenston, Georgian Bay, Blue Mountain, Collingwood, Shadow Lake, and lower Gull River formations.

- Sheet silicate content ranges between 25 to 70% within the Ordovician shales of the Queenston, Georgian Bay, and Blue Mountain formations. Subdivisions include Illite + mica (> 50 %), chlorite (20 to 45 %), minor kaolinite and interstratified illite-smectite. The interstratified illite-smectite is predominantly illite, with only 5-10% smectite layers (JACKSON09). In all cases, illite and chlorite are the major and minor phases, respectively (NWMO11k). The sheet silicate content of the Ordovician limestones is typically less than 20%.

- Pyrite is the principal iron mineral throughout the entire Ordovician interval, although hematite is observed in the Queenston Formation.

These results highlight the consistency in formation-scale mineralogical associations or trends across the site and further support the conclusion that lithofacies are predictable at the site-scale.

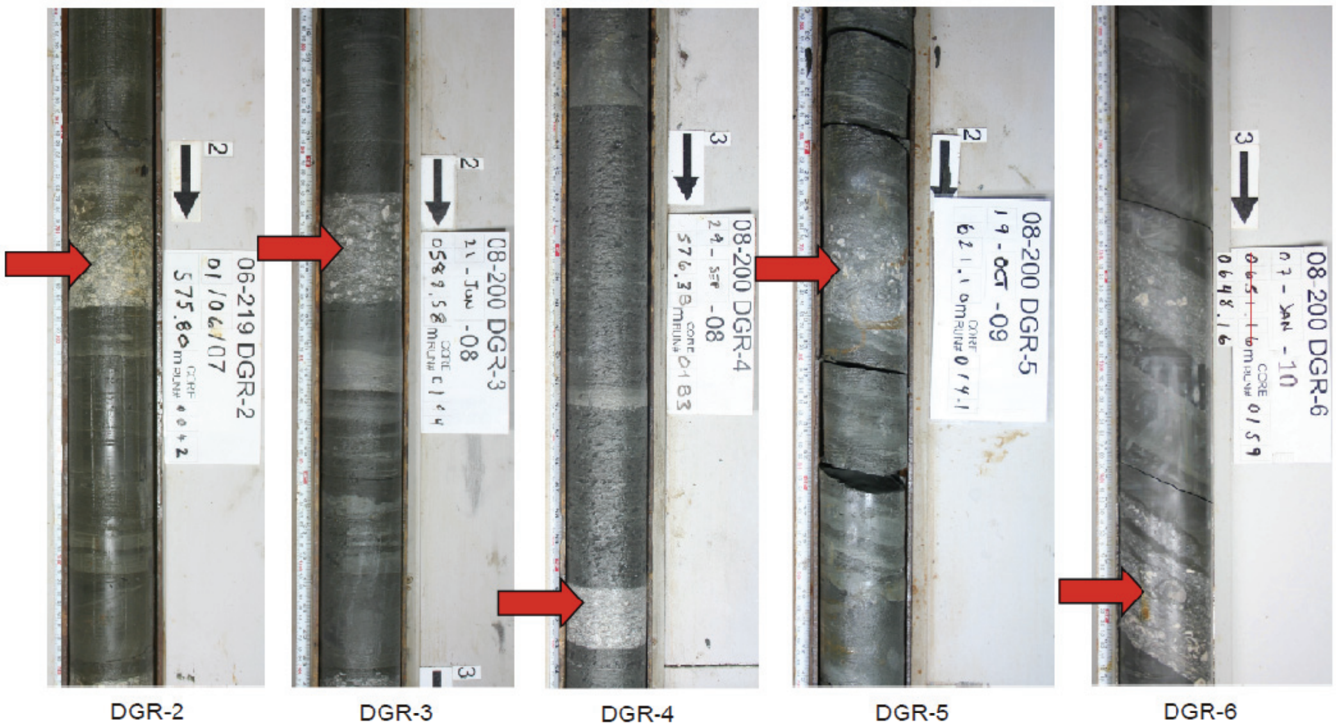
Fracture Filling

Self-sealing by a precipitating mineral phase is a naturally occurring time-dependent process that leads to a reduction in the hydraulic transmissivity of a fracture. When fully self-sealed, the fracture is not a preferential pathway for fluid migration. If partially self-sealed, the fracture may act as a pathway but at a lower transmissivity than when it was open.

Infilled fractures observed during core logging and by petrographic analysis may be of hydrothermal origin or result from mineral precipitation during diagenesis. The vast majority of these secondary mineral phases occur within healed discontinuities in the otherwise intact host rock (e.g., Figure 4-15). The infilling mineral phases include quartz, calcite, pyrite, anhydrite, Fe oxide/hydroxide, clay, halite, and gypsum. Anhydrite is frequently observed from the Bass Islands Formation to the Coboconk Formation. Gypsum was observed in the Salina G Unit to A2 Unit interval. Both anhydrite and gypsum are present in many samples. They are differentiated in the field based on hardness and colour. Calcite and pyrite are observed from the Amherstburg Formation to the Shadow Lake Formation. Halite distribution will be discussed in more detail below.

Sphalerite (Lucas and Georgian Bay formations), marcasite (Kirkfield Formation and Cambrian), and pyrite (entire Paleozoic interval) are present in trace amounts within the host rock and secondary vein infillings. These occurrences are not associated with any commercially exploitable base metal accumulations, as discussed in the DGSM (NWMO11k).

Shales from the upper Queenston Formation contain prominent millimetre thick halite-filled fractures bounded by a carbonate mineral lining the fracture wall (see halite discussion below). The Queenston Formation also displays calcite, anhydrite, celestite, and gypsum veins. Georgian Bay Formation shales include illite and calcite-filled veins and one ~0.15 mm thick halite vein was observed in thin section. Pyrite and illite veins are observed in shales of the Blue Mountain Formation. Middle Ordovician limestones exhibit dolomite veins and other infill material including iron oxide, pyrite, calcite, anhydrite, and occasionally halite (NWMO11k).



Note: Red arrows define the limestone marker bed. From Figure 3.48 of the DGSM (NWMO11k).

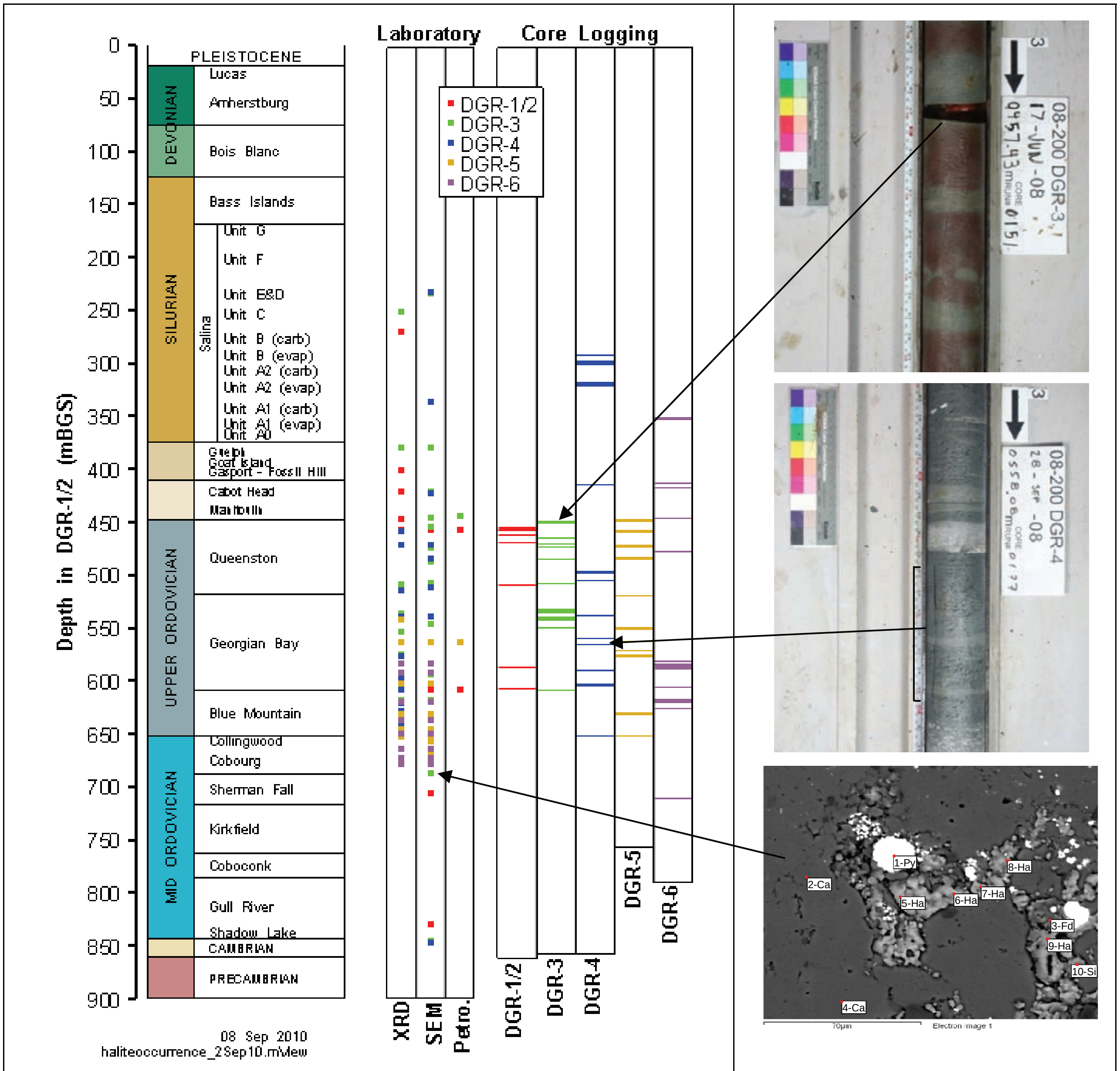
Figure 4-14: Fossiliferous Limestone Marker Bed within Georgian Bay Formation Shale in DGR Boreholes

Halite Occurrences

Halite was specifically targeted for identification and distribution analysis because of its high solubility (~6000 mmol/kgw) and its role as a groundwater tracer (NWMO11k). The presence of halite within a formation or group of formations is a strong indicator that there has been no flow of fresh, or halite-undersaturated, water through that rock sequence since the halite was precipitated (NWMO11k).

Halite was detected visually during core logging, and via optical microscope, XRD, and SEM/EDS analyses (Figure 4-15; KOROLEVA09, NWMO11k). Halite occurrences include: mineral infilling of subhorizontal and steeply dipping fractures; voids and cavities; a grain-boundary mineral phase within a matrix dominated by gypsum, dolomite, calcite, or silicate minerals; and, as disseminated grains and irregular, discontinuous stringers. Halite was found in abundance throughout the Upper Ordovician shales, as a minor mineral phase throughout the Cobourg, Sherman Fall, and Gull River formations, and the Cambrian (Figure 4-15; HERWEGH08, KOROLEVA09, NWMO11k). Whole-rock and clay-mineral XRD analyses yielded average halite concentrations of 0.7 wt % and 0.6 wt % in DGR-3 and DGR-4, respectively. Maximum halite concentrations were recorded in the Blue Mountain Formation with concentrations ranging from 0.5 to 1.4 wt %.

Halite was most commonly observed infilling mm-scale to hairline thickness fractures throughout the Upper Ordovician shales (e.g., top and middle right photographs in Figure 4-15). There is visual evidence that drilling fluids locally dissolved some of the vein halite (e.g., top right photograph of Figure 4-15), but where this occurred there was generally enough preserved for positive identification. In the deeper limestones, including the Cobourg Formation, a lack of open fractures is consistent with halite only being recognized as a mineral phase at the micron-scale. In these instances, it was commonly observed within networks of irregular cavities between larger calcite grains (e.g., bottom right photograph of Figure 4-15).



Notes: Observed halite distributions based on core logging descriptions, and XRD, SEM, and petrographic (laboratory) analyses of DGR cores (NWMO11k and references therein). Top right: Subhorizontal halite-filled fracture in the Queenston Formation. Middle right: Subvertical halite-filled fracture in the Georgian Bay Formation. Bottom right: SEM backscatter image of pore-filling halite in the Cobourg Formation (DGR-3 699.6 mBGS) with spot mineral analyses indicated by red dots. From Figure 3.9 of the DGSM (NWMO11k).

Figure 4-15: Summary of Observations of Halite Presence in the DGR Cores

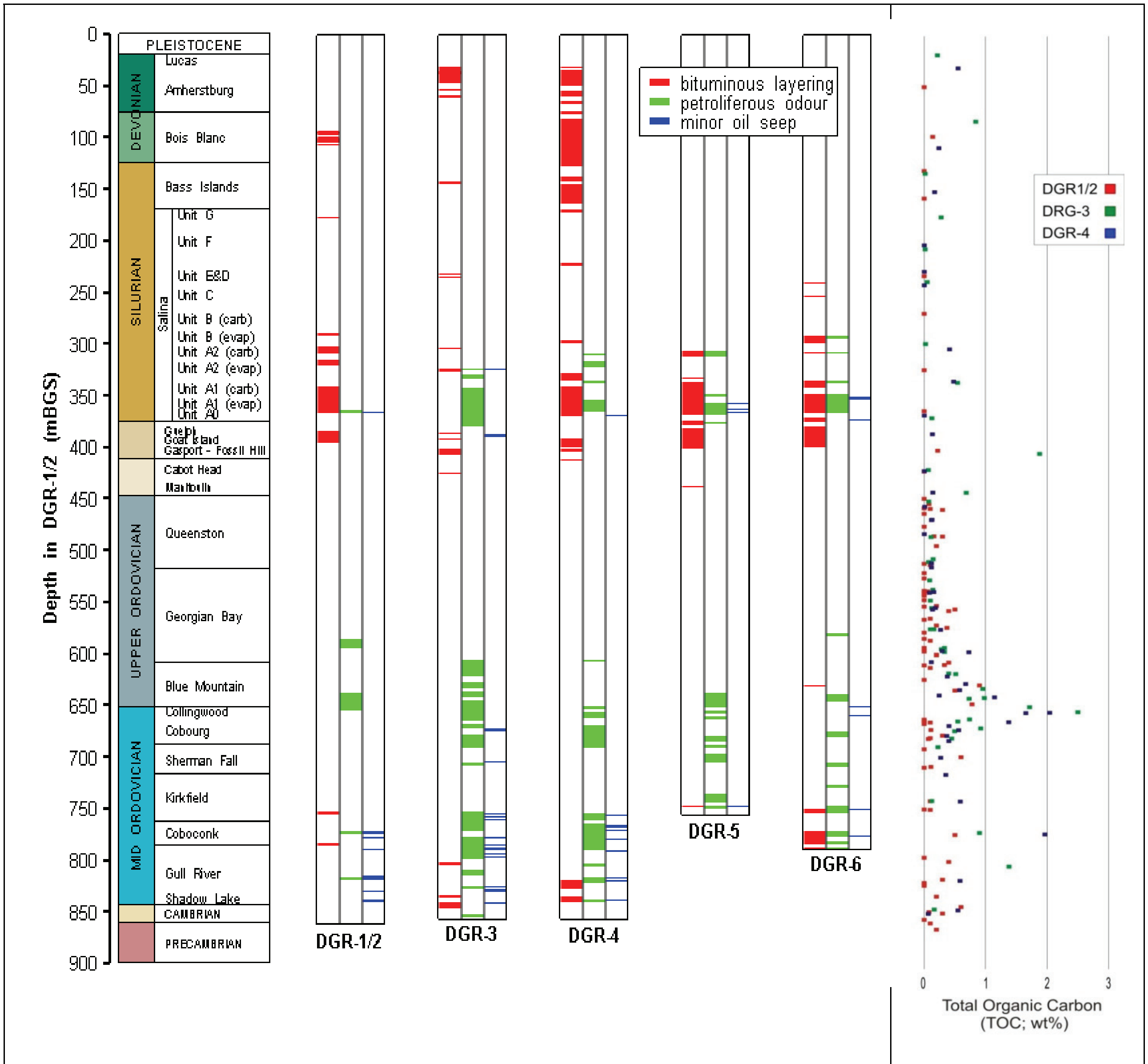
Hydrocarbon Occurrences at the Bruce Nuclear Site

The site characterization activities found no evidence for any economical accumulation of hydrocarbon resources beneath the Bruce nuclear site (Section 3.7.4.2 of NWMO11k). This is consistent with the observed lack of either structural or stratigraphic evidence for voluminous hydrothermal dolomitization or its attendant fault system, as will be discussed in Section 4.1.2.3. Detailed core logging and laboratory analyses provide an understanding of the minor hydrocarbons occurrences beneath the site (Figure 4-16) (JACKSON09, and Sections 3.7.4 and 3.7.5 of NWMO11k).

Hydrocarbons are observed in the DGR cores primarily as thin bituminous layering, indirectly as a prominent petroliferous odour, and as minor seeping or oozing of oil from vugs, fractures, and dolomitized sedimentary horizons. The hydrocarbon-bearing intervals are concentrated into three main horizons which correspond in general to zones of elevated TOC within the subsurface stratigraphic sequence (Figure 4-16). A shallow interval of prominent petroliferous odour and minor oil seeping is observed at the top of the Silurian Guelph Formation and into the overlying basal Salina units (Figure 4-16). An intermediate interval corresponds to the base of the Upper Ordovician shales which, in general, exhibit average TOC values of less than 1.0 wt% (Figure 4-16). A deep interval comprises isolated hydrocarbon occurrences throughout the Black River Group including the base of the Kirkfield Formation of the overlying Trenton Group (Figure 4-16).

DGR core samples from locations within the Upper Ordovician shales were also evaluated by Rock-Eval pyrolysis in order to characterize their thermal maturity and kerogen source (e.g., JACKSON09). It was determined that shales from the Collingwood Member and the Blue Mountain Formation are considered to be near the lower threshold of thermal maturity and of marine origin, tending to form oil rather than gas. Most shales of the Georgian Bay and Queenston formations contain kerogen derived from a terrestrial source and are more gas prone. From the limited extent of visible oil in the cores, and based on the burial history of the regional study area discussed in Section 4.1.1.2, it was estimated that the peak temperature during maturation was approximately 70 °C at the top of the Collingwood Member (top of Trenton Group).

Lateral traceability between the Bruce nuclear site boreholes and other proximal dry wells (Union Gas #1, Texaco #4 and Texaco #6) demonstrates that locally around the Bruce nuclear site (~7km radius), no pockets of oil or gas hydrocarbons are likely to exist (see Chapter 7 of NWMO11c).



Notes: As indicated, the hydrocarbon occurrences are based on core logging observation of bituminous layering, petroliferous odour, and visible liquid hydrocarbon seeping from the core. Compiled from Figures 3.15 and 3.16 of the DGSM (NWMO11k).

Figure 4-16: Summary of Observations of Hydrocarbon Presence in DGR Cores

Karst Occurrences

Based on the recognition that karst occurrences are common in exposed carbonate bedrock throughout southern Ontario (BRUNTON08; see Section 2.2.5.5 of NWMO11c), an evaluation of the distribution of karst beneath the Bruce nuclear site was undertaken (NWMO11z). Karstification is the process by which the flux of chemically undersaturated water through an aquifer preferentially dissolves rocks of carbonate or evaporitic composition. A key property of karst aquifers, and important to understanding the shallow groundwater system at the Bruce nuclear site, is that highly permeable channels resulting from the karstification process become interconnected to form a network in the shallow subsurface (NWMO11z).

Paleokarst refers to karst that was formed at an earlier time and subsequently buried and rendered inactive by later deposition of sediments or by changes in groundwater flow conditions. Paleokarst is therefore ancient and most likely to have been most extensive at the largest unconformable breaks in the sedimentary record. At the site, and regionally, these breaks are recognized at the top of the Bass Islands Formation (e.g., BRUNTON08), below and above the Reynales/Fossil Hill formations, below and above the Detroit River Group and at the top of the Guelph Formation. The lateral extent of these high-permeability zones were a few kilometres at most (NWMO11z), consequently, this localized karstification is unlikely to contribute significantly to modern regional groundwater flow. Though karst features are preserved at such paleokarst horizons, subsequent deposition and diagenesis would have occluded much of the karstic function (i.e., enhanced permeability) of such strata.

The pertinent results of the karst study are summarized below. These conclusions are based on the interpretation of several independent data sets collected during the site characterization and compiled in Figure 4.1 of the Karst Assessment report (NWMO11z).

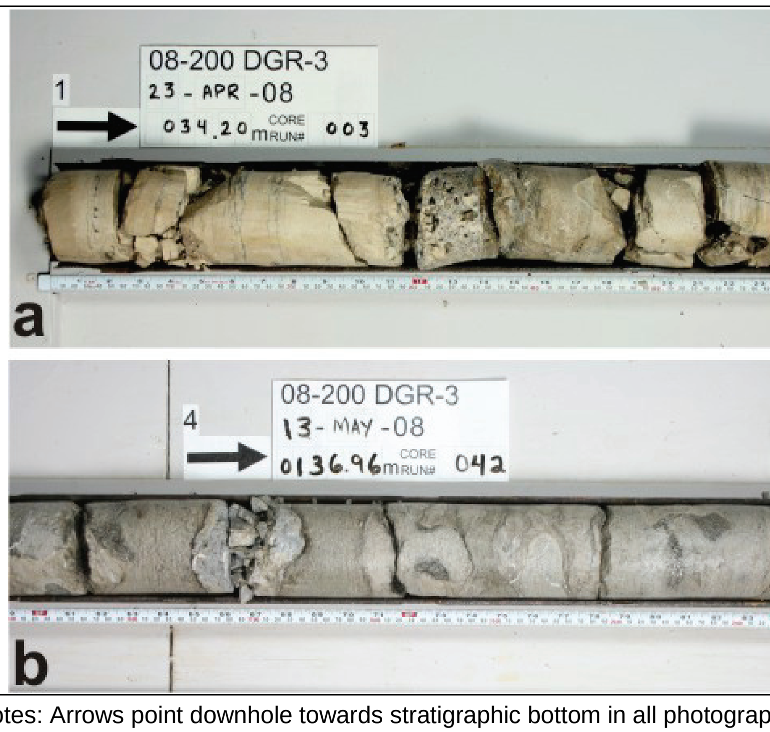
- The top approximately 180 mBGS of bedrock at the Bruce nuclear site down to the Salina G Unit is recognized as a zone of modern karst development. This zone is characterized by higher hydraulic conductivity than is found in the deeper units, and groundwaters that range in Total Dissolved Solids (TDS) from fresh (< 0.5 g/L) to brackish (approximately 5.0 g/L) near the bottom of this groundwater zone.

- Higher hydraulic conductivity intervals at depths of about 326 to 329 mBGS (Salina A1 dolostone) and 375 to 379 mBGS (Guelph Formation) also show isolated evidence of potential karstification. However, these zones are characterized by Na-Cl waters with TDS values of 29 g/L and 371 g/l, respectively.

- The Ordovician carbonates are unaffected by modern karstification processes. This conclusion is supported by the results of the hydraulic testing which indicate uniformly very low hydraulic conductivities throughout the deep Ordovician interval.

The site-scale distribution of TDS, and formation-scale hydraulic conductivities are discussed in more detail in Section 4.3.2 (Figure 4-34) and Section 4.4.1, respectively.

Examples of modern karst and potential paleokarst from beneath the Bruce nuclear site are shown in Figure 4-17. Shallow Devonian carbonates are characterized by modern karst features such as solution-enhanced joints and stained/weathered fractures (Figure 4-17a). Groundwater in the shallow bedrock system may preferentially flow along paleokarst horizons where modern karstification has dissolved cement infilling. An example is observed near the bottom of the Bois Blanc Formation (Figure 4-17b) which overlies the unconformity at the top of the Bass Islands Formation.



Notes: Arrows point downhole towards stratigraphic bottom in all photographs. (a) Core photo from shallow Devonian Lucas Formation carbonates. This interval is characterized by karst features such as solution-enhanced joints and stained/weathered fractures. (b) Core photo from a section of the Devonian Bois Blanc Formation where present-day groundwater flow may be concentrated along a remnant paleokarst horizon near the top of the Bass Islands Formation. From Figure 2.34 of the Geosynthesis (NWMO11c).

Figure 4-17: Karst and Paleokarst Intervals Beneath the Bruce Nuclear Site

4.1.2.3 Site-Scale Structural Geology

Studies undertaken as part of the Geosynthesis work program which focused on understanding the structural geological framework of the Bruce nuclear site included:

- A 2D seismic reflection survey across the site (WATTS09);
- A detailed fracture mapping exercise (NWMO11ab) on the exposed shoreline proximal to the site; and
- Core logging activities undertaken during site characterization (NWMO11k).

The important results of each of these studies are summarized below. The predictable nature of the Ordovician interval, as discussed above in Section 4.1.2.2, provides additional evidence to support the assertion that the Bruce nuclear site is structurally simple and relatively undeformed.

2D Seismic Reflection Survey

A 2D seismic survey, including nine survey lines totalling 19.7 km as shown in Figure 3-1, was conducted on the Bruce nuclear site as part of the Geosynthesis work program (WATTS09). The purpose of this 2D seismic survey was to obtain deep bedrock geological, stratigraphic, and structural information for the Bruce nuclear site and to assess the predictability and continuity of the host rock for the DGR (Cobourg Formation) and the "potential" location of faults and fault zones in the subsurface within the Paleozoic bedrock. The bedrock units of primary interest were the shales and argillaceous limestones at depths of about 400 to 800 m. These strata include the Middle Ordovician limestones (Cobourg, Sherman Fall, Kirkfield, Coboconk, and Gull River formations) and overlying Ordovician shales (Queenston, Georgian Bay, and Blue Mountain formations), as well as the intervening Collingwood Member.

Interpretation of the seismic dataset indicates the possible presence of some key structural features.

- Within the proposed DGR footprint, an apparent north-trending basement structural high with as much as 10 m of relief is imaged in Line 1 (Figure 4-18) as well as lines 5 and 6 (Figure 3-1). The basement high is interpreted to be bounded on its eastern flank by a steeply dipping normal fault and on its western flank by several distinct elevation lows within the Ordovician succession which may represent a graben-type structure.
- Another basement high, which may be an extension of the feature interpreted from lines 1, 5, and 6, is bounded by a steeply dipping NNW-trending fault that crosses Line 9 (Figure 3-1).

- A NNW-trending and steeply dipping fault structure, possibly with normal-sense displacement, crosses Line 7 approximately 1.25 km southwest of the proposed DGR footprint (Figure 3-1). This interpreted fault bounds a basement high to the east and is interpreted to terminate within the Ordovician shales. It is therefore constrained to a pre-Silurian movement history, if it exists.

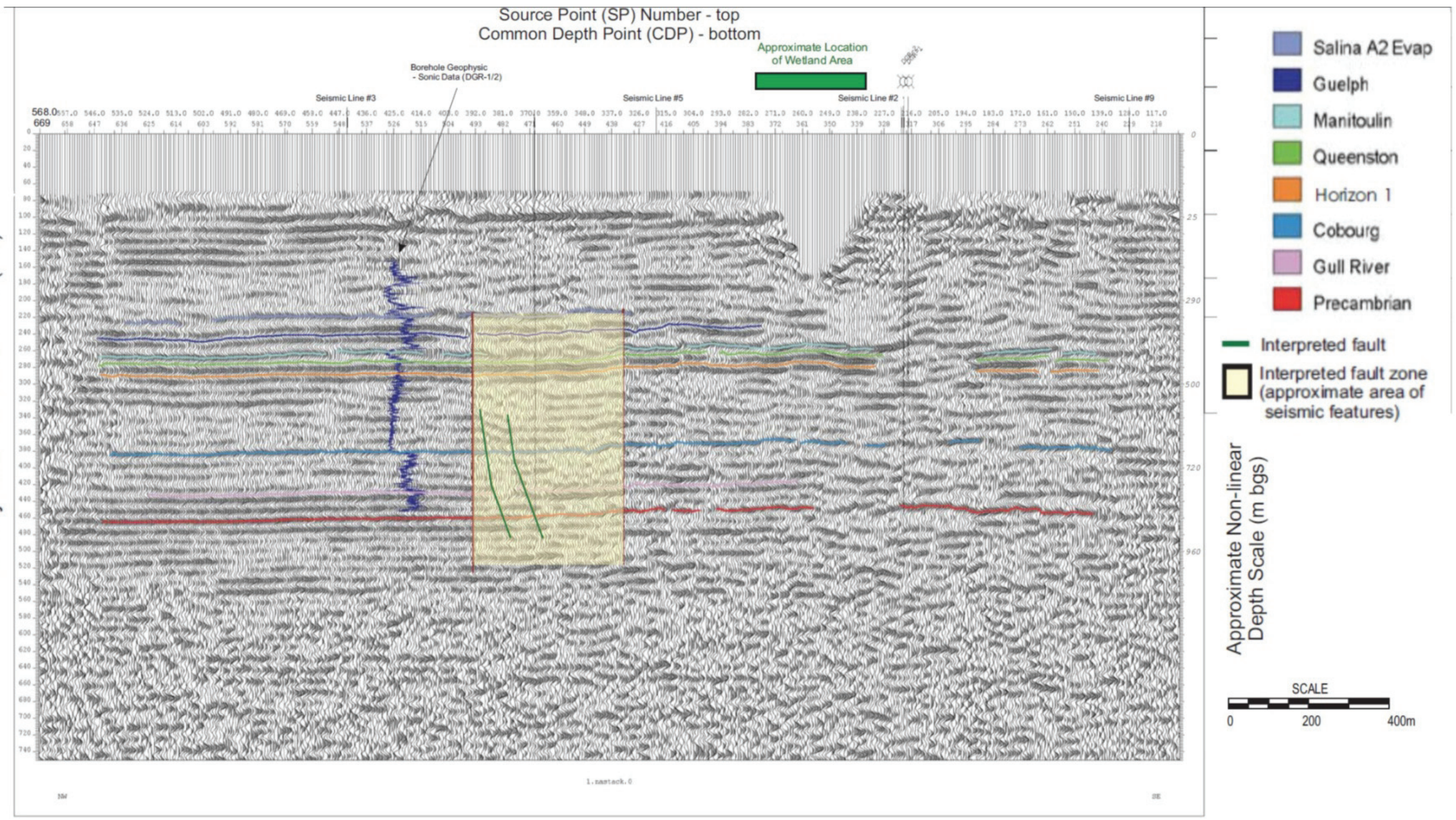
Important results of the seismic analysis are included below.

- In general, the seismic survey imaged horizontal reflections interpreted to represent traceable bedrock stratigraphy across the proposed DGR footprint (WATTS09).
- The seismically interpreted faults within and proximal to the proposed DGR are not consistent with known geometry, size, and seismic profiles of faults normally associated with a Hydrothermal Dolomite (HTD) reservoir (NWMO11k, NWMO11y). HTD-related sag structures are related to transtensional (strike slip and extensional) fault zones, with the structural lows being the expression of negative flower structures where strata have been faulted downward.
- An interpreted basement high of 10 m as shown in Line 1 (Figure 4-18) and an equivalent fault offset in the overlying stratigraphy are not supported by the marked consistency in formation thicknesses, strike and dip across the proposed DGR footprint as discussed in Section 4.1.2.2 (see also NWMO11k).
- No seismically imaged faults are interpreted to have breached the Upper Ordovician shale-dominated sedimentary package (e.g., Figure 4-18).

Importantly, these conclusions aid in our understanding of the barrier integrity of the Upper Ordovician shales which will serve as the primary cap rock for the proposed DGR. This attribute will be discussed in more detail at the end of this sub-section.

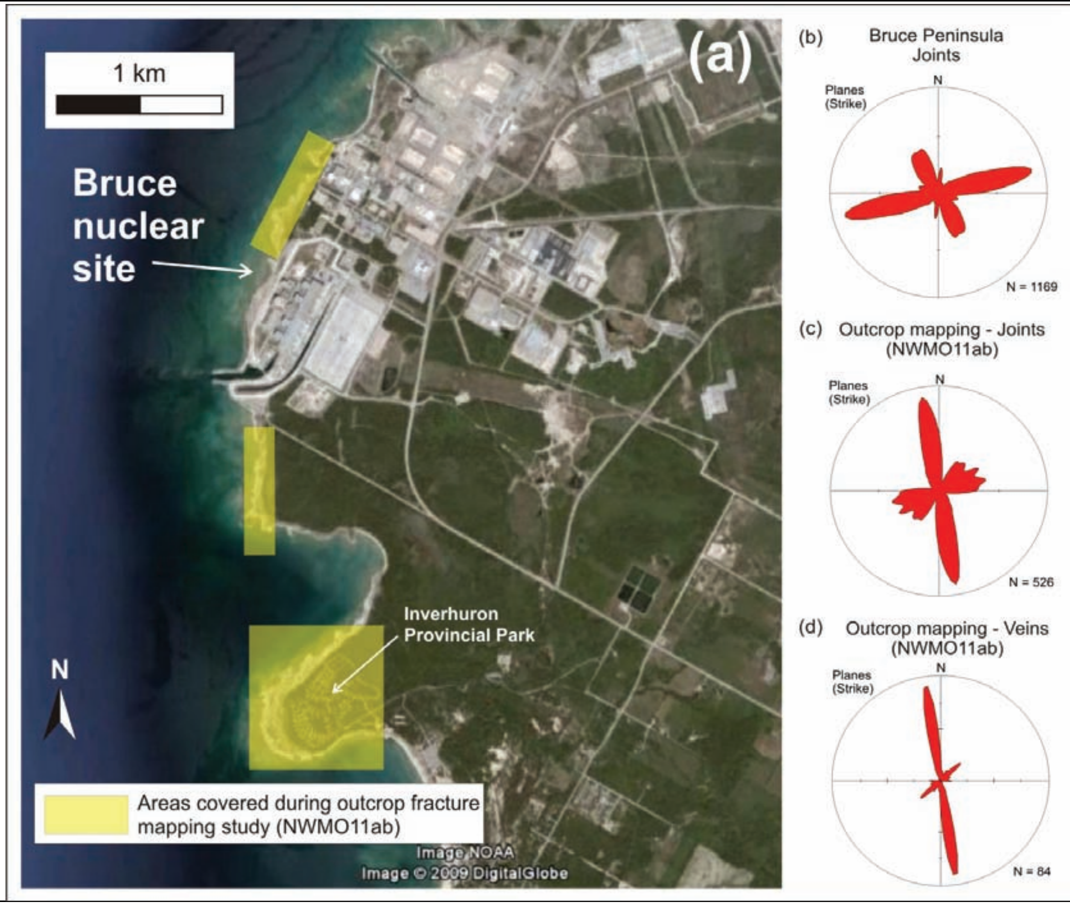
Fracture Analysis

The results of a detailed fracture mapping study undertaken near the Bruce nuclear site, and with the objective of collecting brittle fracture orientation data, including a systematic examination of joint, vein, and fault features, are discussed below (Figure 4-19; NWMO11ab). The analysis focused on accessible shoreline exposures of the Devonian Lucas Formation (Figure 4-19a; NWMO11ab). These results are also compared with joint orientation information determined during detailed core logging (NWMO11k). The results confirm that the surface data are generally consistent with the subsurface data and, further, that both are broadly consistent with the regional dataset.



Note: Modified from Figure 16a of the 2D seismic survey report (WATTS09).

Figure 4-18: 2D Seismic Line #1 with Interpreted Faults



Notes: Base map in (a) indicates areas covered during detailed outcrop fracture mapping analysis at locations proximal to the Bruce nuclear site and Inverhuron Provincial Park (NWMO11ab). Rose diagrams of joint and vein data collected are shown in (b) – (d). (b) Joint orientation data from the Bruce Peninsula in the northwestern part of the regional study area (ARMSTRONG93). (c) and (d) are joint and vein orientation data, respectively, measured during the detailed fracture analysis (NWMO11ab). Modified from Figure 2.35 of the Geosynthesis (NWMO11c).

Figure 4-19: Compilation of Regional- and Site-Scale Joint and Vein Data

<u>Outcrop Data</u>

Exposed bedrock near the Bruce nuclear site is restricted to fine- to medium-grained, light grey limestone and dolostone of the Devonian Lucas Formation (Figure 4-20). The rock is observed as discontinuous, shallowly SW-dipping, pavements along the shoreline of Lake Huron immediately adjacent to the Bruce power plant and further to the south around Inverhuron Provincial Park (Figure 4-19a). Bedding attitude is locally deflected due to sediment compaction over the top of 1 to 2 m diameter stromatolite mounds. At a larger scale, aerial photograph interpretation of surface bedding traces indicates that bedding layers are locally deflected into 40 to 100 m diameter dome and basin features (NWMO11ab).

Only systematic joint sets were looked at during the fracture analysis. Their observable characteristics are identified below (NWMO11ab).

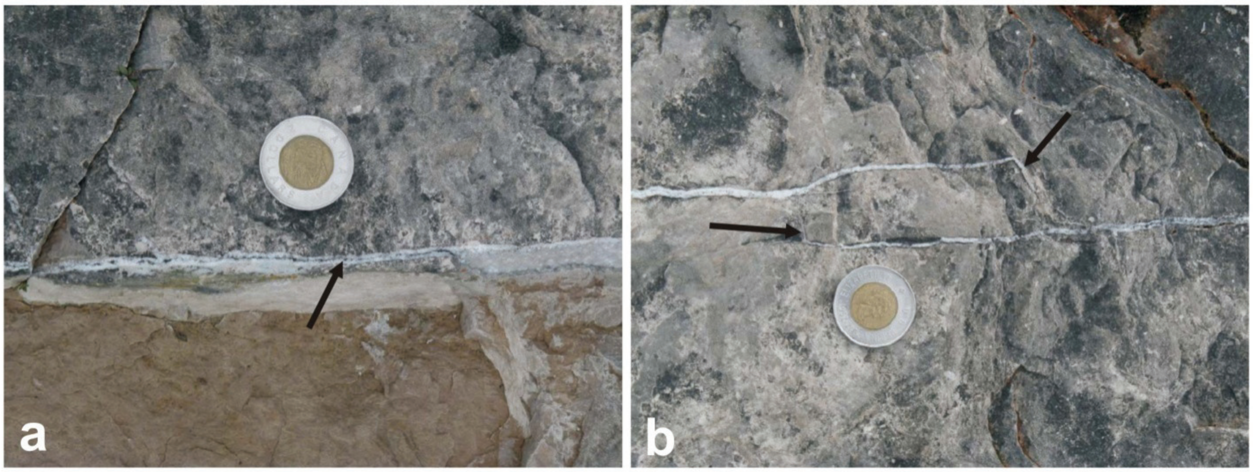
- Both joint and vein sets share common orientations with subtle variations. Two main sets are distinguished, one trending ENE and the other NNW (Figure 4-19c and Figure 4-19d). These two statistically dominant sets overshadow a very minor third set of SE-striking joints and veins.

- Most joints are closed and tight and those with measurable aperture have been widened by solution processes (karst) or creep. Joints only exhibit carbonate mineral infilling (Figure 4-20a and Figure 4-20b), with no iron oxide filling or coatings, indicating a lack of groundwater penetration along joint surfaces.

- Only 10 of the 610 measured joints and veins displayed horizontal offsets with both sinistral and dextral displacement, ranging from 2 mm to 150 mm, observed on both the ENE- and N-striking sets with no systematic distribution noted. No significant faults, or evidence of brittle or ductile faulting in the rocks, were observed in the study area.

In several places, fracture propagation and mineral precipitation are interpreted to have been synchronous based on the curved and branching morphology of observed calcite-filled veins suggestive of multiple cycles of hydraulic fracturing and mineral precipitation (Figure 4-20a and Figure 4-20b). Such features are indicative of fracture propagation under conditions of elevated pore fluid pressure. Given that both joints and veins share common orientations, it is likely that most fractures observed in the Lucas Formation formed under conditions of elevated pore fluid pressure experienced during either Acadian or Alleghenian orogenesis (NWMO11ab).

The two main outcrop-scale NNW- and ENE-trending joint and vein set orientations are broadly consistent with joint orientations measured elsewhere throughout the regional study area and in southern Ontario, including data from the Bruce Peninsula (compare Figure 4-8 and Figure 4-19b with Figure 4-19c and Figure 4-19d), and appear to be part of the regional fracture system in the Silurian and Devonian strata of the Bruce Peninsula, Manitoulin Island, and northern Michigan. In particular, the NNW-trending set is concentric with respect to the outline and structure contours of the Michigan Basin. A broad basin-centred subsidence event coincided with deposition of the middle Devonian Dundee Formation and Traverse Group strata in the Michigan Basin (HOWELL99). Radial tensile stresses generated during this event provide a plausible mechanism for developing the basin-scale concentric fracture set in general, and the NNW-trending fracture set in the study area in particular (NWMO11ab).



Notes: Calcite-filled veins in limestone (Lucas Formation) characterized during the outcrop fracture mapping study (NWMO11ab). In (a), the vein trends 119^0 and is filled with calcite. A thin dark discontinuous seam of wall rock occurs in the centre of the vein (indicated by arrow), indicating its crack-seal nature. In (b), overlapping calcite-filled veins with interacting (bridging) tips (indicated by arrows) suggest that the veins likely propagated as fluid-pressurized cracks (hydrofractures). Coin for scale in both photos. Modified from Figures 3.6 and 3.7 of the fracture mapping report (NWMO11ab).

Figure 4-20: Calcite-Filled Veins Exposed Along the Shoreline of Lake Huron Near the Bruce Nuclear Site

The geometrical relationships discussed above suggest a contemporaneous late Paleozoic age for formation of the NNW- and ENE-trending fracture sets. A neotectonic origin for the ENE-trending fractures (e.g., HOLST82, GROSS91) is difficult to reconcile with an interpreted late Paleozoic timing for formation of the NNW-trending fractures given that detailed fracture mapping suggests these two sets formed contemporaneously. Recent work re-analysing the paleo-stress field of the Appalachian Basin suggests that some of these ENE-trending joint sets distributed throughout the basin are actually late Paleozoic (Pennsylvanian-Permian) in age (ENGELDER06). They now simply share a common orientation with a prominent neotectonic joint set (HANCOCK89). Therefore there is no genetic significance to the similarity in orientation between the ENE-trending fracture population and the present in-situ maximum horizontal stress. The origin of the vein filling material and the timing of the main fracture forming event, for both the NNW and ENE fracture sets, is best interpreted as late Paleozoic in age (NWMO11ab).

<u>Core Logging Activities: Vertical Borehole Results</u>

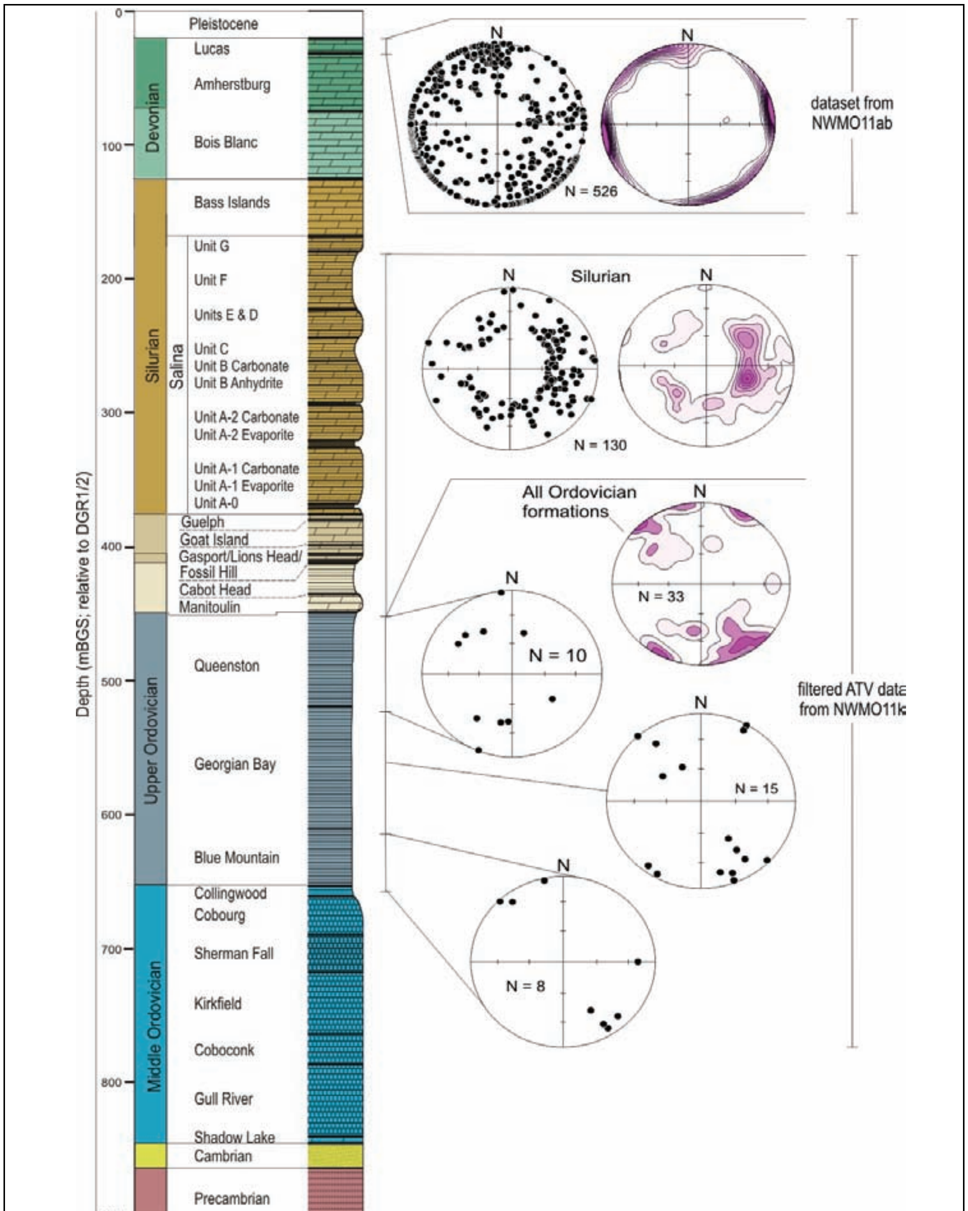
Details of the borehole layout at the Bruce nuclear site are given in Figure 3-1. Boreholes DGR-1, DGR-2, DGR-3, and DGR-4 were drilled to approximate depths of 462, 862, 869, and 857 mBGS, respectively, and are subvertical, never exceeding tilts of 1.5°, 1°, 4.5°, and 4°, respectively (NWMO11k). Core logging and Acoustic Televiewer (ATV) images represent the primary means of structural data collection. The former gives information primarily on occurrence and approximate dip of fractures while the latter can quantify both occurrence and orientation through the analysis of the elliptical traces of fractures on the borehole wall. Figure 4-21 shows a plot of ATV-derived natural fracture data in the subsurface separated by formation, as well as data compiled during the outcrop fracture mapping study (NWMO11ab). The ATV data have been filtered to only include features that dip >35° from horizontal (NWMO11k). The borehole data for the Ordovician are sparse with only 33 total measurements across all formations (Figure 4-21). This value highlights the lack of natural fractures in the subsurface beneath the Bruce nuclear site. Peak Ordovician fracture orientations trend ENE and ESE (Figure 4-21). A much larger dataset for the Silurian interval (130 measurements) exhibit a diffuse spread of data (Figure 4-21), possibly due to salt dissolution processes.

A much larger number of shallowly dipping to subhorizontal fractures were distinguished by the ATV survey, for example see Figure 3.67 and Figure 3.69 of the DGSM (NWMO11k). There is some uncertainty surrounding the identification of these occurrences as true fractures because many of the subhorizontal features evident on the ATV logs may actually be thin beds whose lithology is different from the host rock (e.g., thin siltstone and limestone beds within a host shale formation).

Core Logging Activities: Inclined Borehole Results

As noted above, vertical boreholes have an inherent sampling bias against steeply dipping structural features. Inclined boreholes DGR-5 and DGR-6 were drilled so that a statistically meaningful lateral section of rock could be sampled for quantification of the joint and vein distribution within the subsurface. The majority of steeply inclined joints within the Ordovician section occur in the Georgian Bay and Blue Mountain formations, with only three in the Collingwood Member and none in the Cobourg and Sherman Fall formations.

The inclined-drilling program was also designed to test for the existence of NNW-striking vertical faults proximal to the DGR, which were interpreted based on the results of the 2D seismic survey (WATTS09). DGR-5 was oriented such that it would potentially intersect a northward extension of one such fault structure. DGR-6 was oriented such that it would transect a similarly oriented structure at depth, if it exists. Continuous core retrieved from both inclined boreholes showed no indication of the existence of either one of these potential faults (NWMO11k). Based on the geological data available as part of DGR site characterization work, there are no indications of the presence of inclined or vertical faults in the area surrounding the proposed DGR defined by boreholes DGR-1 to DGR-6 (NWMO11k).



Notes: Data are plotted as poles to the plane of measurement on equal-area lower hemisphere projections. Surface dataset is compiled from the outcrop fracture mapping study (NWMO11ab). Subsurface dataset is from ATV logging of DGR boreholes (NWMO11k). From Figure 2.37 of the Geosynthesis (NWMO11c).

Figure 4-21: Natural Fracture Orientations from Surface and Subsurface Datasets

Other results that are consistent with a lack of significant evidence for faulting at the Bruce nuclear site include:

- The range in Middle and Upper Ordovician total stratigraphic thickness of between 396.1 and 398.3 m, as determined from logging of the DGR boreholes (Table 4-2), is remarkably similar to the total Ordovician thicknesses of 393.5 and 393.1 m for the proximal Kincardine #1 Union Gas and Texaco #6 boreholes, respectively (NWMO11k); and

- A neotectonic remote-sensing and field-based study that analysed Quaternary landforms for the presence of seismically induced soft-sediment deformation concluded that the Bruce nuclear site has not likely experienced any post-glacial tectonic activity (NWMO11v).

Ordovician Cap Rock Seal

An assessment of the cap rock integrity and seal potential of the DGR cap rock was undertaken based upon evaluation of the seal quality of cap rocks within petroleum deposits in the Appalachian and Michigan Basins (NWMO11y), for example, black shales of the Marcellus Formation. The purpose of the cap rock study was to explore whether the package of Upper Ordovican shale-dominated rocks at the Bruce nuclear site would provide a natural barrier to migration of fluids. The cap rock for the proposed DGR includes the Middle Ordovician organic shale-rich Collingwood Member and the overlying Upper Ordovician shale-dominated Blue Mountain, Georgian Bay and Queenston formations totalling >200 m of low-permeability shale-rich rocks overlying the proposed Bruce nuclear site. Main conclusions reached by the study which attest to the longevity in seal integrity of the Bruce nuclear site cap rocks are included below (NWMO11y).

- Limited hydrocarbon maturation at the Bruce nuclear site is a result of subsidence that reached a total burial depth of approximately 1.5 km and certainly no more than 2 km, creating temperatures that only marginally crossed the oil generation window (~70 ºC for the top of the Collingwood Member). This lack of thermal maturity precluded the development of gas-generated Natural Hydraulic Fractures (NHFs), and this relationship was confirmed by extensive coring. In contrast, gas-generating conditions within the Appalachian Basin led to extensive and pervasive NHF development.

- The compartmentalized distribution of hydrocarbons at the Bruce nuclear site as shown in Figure 4-16 suggests that these Upper Ordovician shales act as a barrier to hydrocarbon migration and therefore provide an adequate seal.

- The youngest strata in the regional study area affected by basement-seated faults are the Ordovician-aged Trenton Group limestones (ARMSTRONG10). The lack of any appreciable volume of HTD at the Bruce nuclear site (NWMO11k) argues against the likelihood of a proximal major Paleozoic fault system having been active in the vicinity in the ancient past that would have disrupted the seal integrity of the cap rocks. The indication that seismically imaged fault structures beneath the site do not breach the Upper Ordovician cap rock supports this assertion (WATTS09).

- The Appalachian Basin has gas traps below the Marcellus black shale that reach more than 70% of the overburden stress. The Marcellus black shale is also overpressured throughout the northern Appalachian Basin, leaving no doubt about its effectiveness as a regional seal. In a similar manner, the underpressured nature of the Ordovician shales beneath the Bruce nuclear site (see Section 4.4) indicates that this sedimentary package represents a long-lived and stratigraphically controlled cap rock seal.

Therefore, the shale-dominated cap rocks at the Bruce nuclear site represent a natural >200 m thick seal that has demonstrated long-term integrity over geological time and is well suited to continue acting as a primary barrier to contaminant transport in the subsurface (NWMO11y).

4.1.3 Geology Summary

The Paleozoic sedimentary rocks beneath the Bruce nuclear site are predictable, include multiple natural barriers to contaminant transport, and are located in a seismically quiet environment. A summary of the key lines of evidence which support this assertion is provided below.

- The occurrence of individual bedrock formations, facies assemblages, marker horizons, major mineralogy, and hydrocarbon and karst distributions are predictable and traceable at the site-scale (Section 4.1.2). Comparing the Paleozoic bedrock stratigraphy encountered in the DGR boreholes to that derived from an assessment of historic oil and gas well records demonstrates traceability at the local scale (e.g., Texaco #6 well) and indicates a consistency with the regional stratigraphic framework (ARMSTRONG06, NWMO11aa).

- The thickness and orientation of bedrock formations encountered beneath the Bruce nuclear site are highly consistent (NWMO11k). Within an area of approximately 1.5 km^2 enclosing the DGR footprint, information derived from the deep drilling and coring program confirms that Ordovician formation thickness variations are on the order of meters. Formation dips within the same chronostratigraphic sequence are uniformly 0.59° +/- 0.08° (≈10 m/km) to the southwest towards the Michigan Basin (Section 4.1.2.2).

- The results of the 2D seismic reflection survey (19.7 km of data collected), which provides evidence for the traceable nature of the bedrock stratigraphy, included an interpretation of possible subvertical basement faults or fault zone structures. Importantly, no potential fault structure was interpreted to have breached the Upper Ordovician shale cap rock (WATTS09). The inclined drilling of DGR-5 and DGR-6 targeted two of these interpreted structures, which are located proximal to the DGR footprint. Continuous core retrieved from both boreholes showed no evidence of faulting through the target interval (NWMO11k).

- Evidence supporting the occurrence of steeply oriented linear and elongate hydrothermally dolomitized reservoirs within the Ordovician carbonate rocks is absent with no proximal deep-seated fault system identified (NWMO11k).

- Detailed fracture mapping of the Lucas Formation that outcrops along the shore line adjacent to the Bruce nuclear site found no evidence of significant faults or shear zones. Observed fractures are predominantly joints and calcite-filled veins with only a small number showing brittle displacements (10 out of 610) associated with measurable horizontal offsets ranging from 2 to 150 mm. The joints and veins both exhibit systematic sets trending NNW and ENE. Their formation is interpreted to be synchronous and is attributed to late Paleozoic subsidence of the Michigan Basin (NWMO11ab).

- Mapped faults are not known to penetrate Paleozoic sedimentary rocks younger than Ordovician age within the regional study area (ARMSTRONG10). This is consistent with the results of the detailed fracture mapping study (NWMO11ab), and the 2D seismic survey. The present tectonic setting is stable and is expected to remain so for well beyond the 1 Ma design life of the repository.

- Present-day karst features are confined to the shallow groundwater zone and this zone is effectively isolated from the deeper groundwater system beneath the site. This interpretation is supported by the observed distribution of halite within the deep system (Figure 4-15).

- Site characterization activities found no evidence for any economical accumulation of hydrocarbon resources beneath the Bruce nuclear site (NWMO11k). The low degree of thermal maturity, which barely reached the oil window in terms of hydrocarbon generation, precluded the development of gas-generated NHF, which could have disrupted the Upper Ordovician cap rock seal. These results, coupled with the low average TOC (< 1 %) within the Upper Ordovician, argue against the likelihood of commercial quantities of shale gas occurring within the DGR footprint. The discrete and compartmentalized distribution of the minor hydrocarbon showings at the site attests to the longevity of the cap rock seal.

4.2 Geomechanics

4.2.1 Introduction

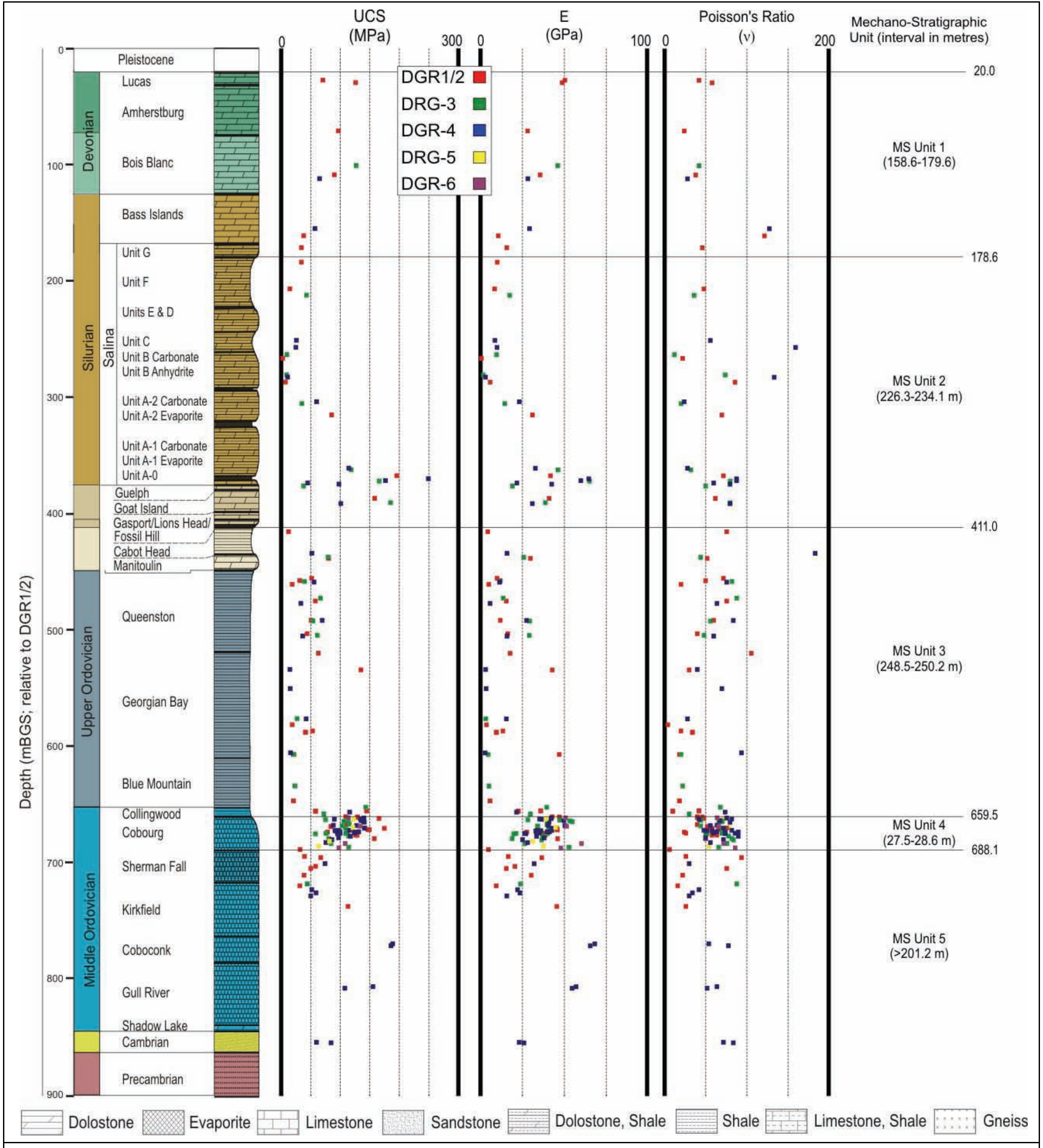
The purpose of this section is to present a summary of the geomechanical properties of the deep sedimentary formations at the Bruce nuclear site. This includes establishing the existing geomechanical knowledge as it relates to site material strength properties and ground stress distribution. The information is drawn from a recently compiled report on the regional geomechanical setting of southern Ontario (NWMO11n), site specific data collected during site characterization work (see Chapter 5 of NWMO11k), and the Geosynthesis (NWMO11c).

4.2.2 Geomechanical Properties: Rock Strength and Deformation

The multi-phase geomechanical analysis, undertaken as a part of the geoscientific site characterization work (NWMO11k), included testing of samples from boreholes DGR-1 through DGR-6 with the aim to provide a comprehensive suite of site specific geomechanical data of the rock material. Tests undertaken included, uniaxial compression, triaxial compression, cross anisotropic, free and semi-confined swelling, and long-term strength degradation tests (NWMO11k). The resultant dataset includes important rock material parameters such as peak intact rock strength, elastic modulus, and Poisson's ratio (Figure 4-22). A key outcome of these studies is the recognition of a systematic distribution of mechano-stratigraphic units in the subsurface beneath the Bruce nuclear site (Figure 4-22).

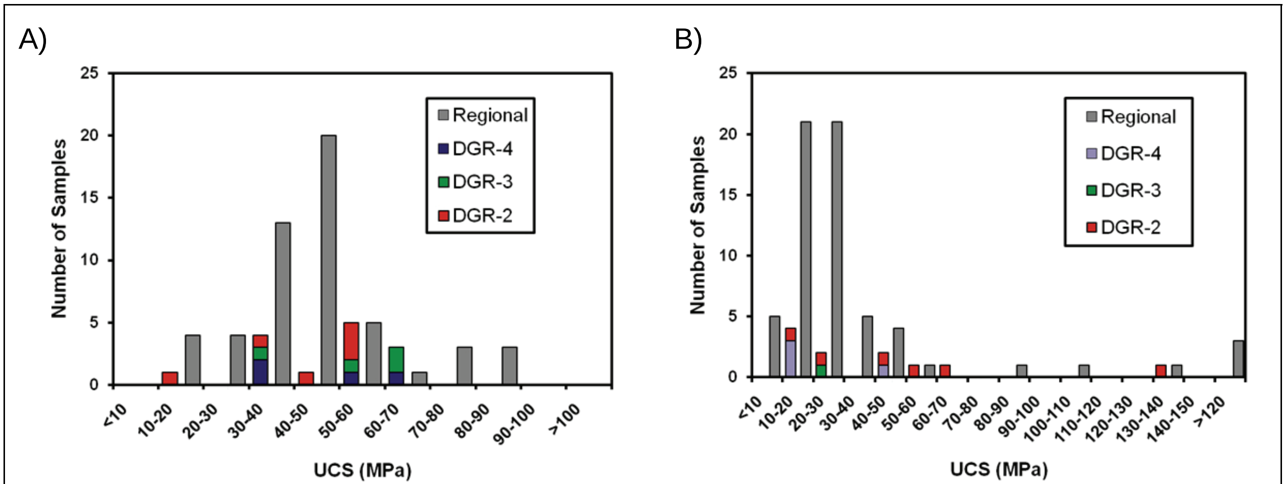
The following sections are mainly focused on the DGR host rock; the Cobourg Formation of middle Ordovician age (Trenton Group) and a portion of the caprock (Queenston and Georgian Bay formations) of upper Ordovician age. Only brief descriptions of the overlying rocks are included. As mentioned in Section 4.1.2.1, all mention of the Cobourg Formation below refers only to the lower argillaceous limestone member of this formation.

To determine the intact strength of the caprock, uniaxial compression testing was carried out on a total of 14 Queenston and 11 Georgian Bay samples from DGR-2 through DGR-4. From these tests, key parameters such as the Uniaxial Compressive Strength (UCS), Elastic Modulus (E), and Poisson's ratio (ν) were measured. Results plotted in Figure 4-22 and Figure 4-23 show that the shales have a moderate strength with estimated mean values of 48 MPa and 32 MPa for the Queenston and Georgian Bay formations, respectively. Regional UCS data of both rock formations, also plotted in Figure 4-23, show that both regional- and site-scale data sets are within the same range (NWMO11n).



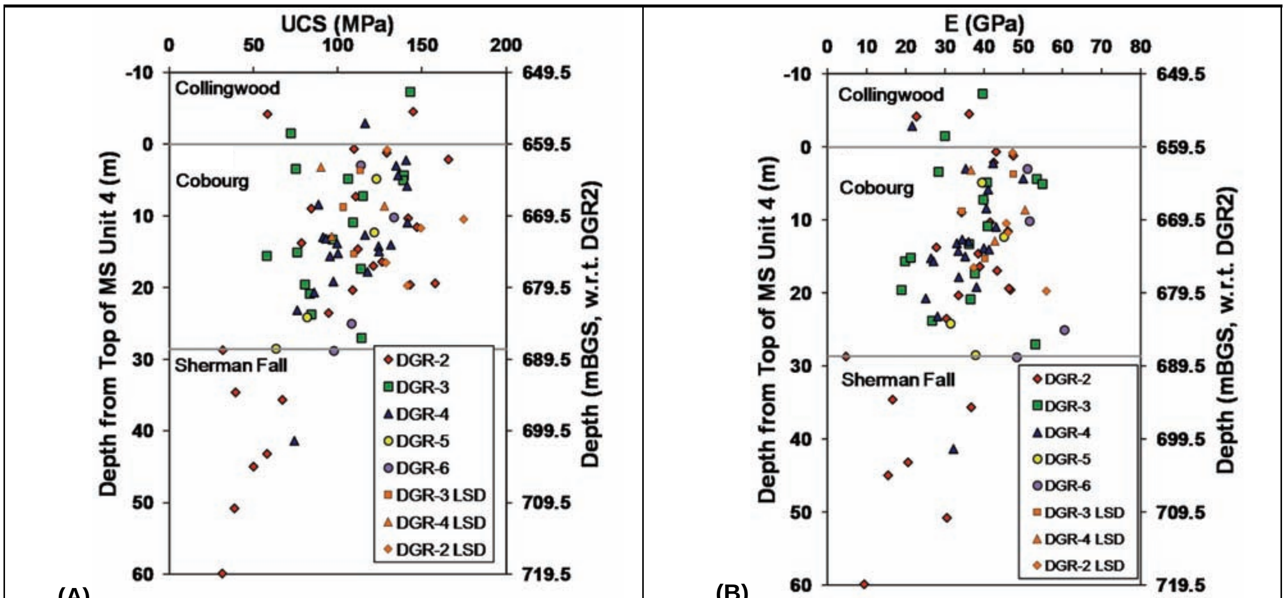
Notes: MS = mechano-stratigraphic. Figure is based on information in the DGSM (NWMO11k) and modified from Figures 3.10 and 3.21 of the Geosynthesis (NWMO11c).

Figure 4-22: Reference Stratigraphic Column Showing UCS Test Results and Mechano-Stratigraphic Units at the Bruce Nuclear Site



Notes: Includes regional (grey) and site-scale (colours; DGR-2, -3 and -4 boreholes) data for shales of the (A) Queenston Formation, and (B) Georgian Bay Formation. From Figure 3.14 of the Geosynthesis (NWMO11c).

Figure 4-23: UCS of Ordovician Shales



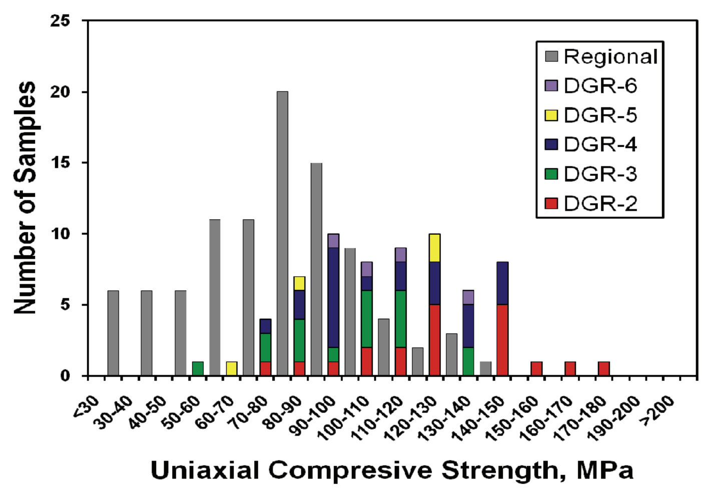
Notes: (A) is based on data from Figure 3.3a of the Geosynthesis (NWMO11c). (B) is based on data from Figure 3.3b of the Geosynthesis (NWMO11c). Long-term strength degradation (LSD) values are from laboratory testing as described in the DGSM (NWMO11k).

Figure 4-24: (A) UCS and (B) Elastic Modulus of the Cobourg Formation from DGR-2 to DGR-6

Peak UCS for the Cobourg Formation, as determined from the results of 67 samples, ranges from 58 to 175 MPa (Figure 4-22 and Figure 4-24a), with an arithmetic mean of 113 MPa and a standard deviation of 25 MPa. The corresponding elastic modulus (Figure 4-22 and Figure 4-24b) has a mean value of 39 GPa. The Cobourg Formation can be classified as a high strength rock with an average modulus ratio (LAM07). These results reflect positively on the stability of deep underground excavations at the DGR horizon. These results also compare favourably with other sedimentary formations considered internationally for long-term radioactive waste management programs, as discussed in Chapter 7 (Table 7.1 therein) of the Geosynthesis (NWMO11c).

A comparison of DGR versus regional UCS results for the Cobourg Formation reveals that the former have a considerably higher average peak strength value (Figure 4-25). This strength increase is likely attributed to greater sampling depths beneath the Bruce nuclear site, along with mineralogical variation (i.e., clay fraction), improved sample preservation methods, and/or the quality of the laboratory testing.

The UCS results from DGR-2 through DGR-6 show a consistent distribution and range within the formation when they are plotted versus depth (Figure 4-22). The variation in strength noted in the UCS test results is due to the variation in material properties within the formation, induced damage while drilling caused by unloading, and local platen interference and/or other boundary effects during laboratory testing.



Note: From Figure 3.4 of the Geosynthesis (NWMO11c).

Figure 4-25: Cobourg Formation UCS Measurements from DGR-2 to DGR-4 and Regional Data

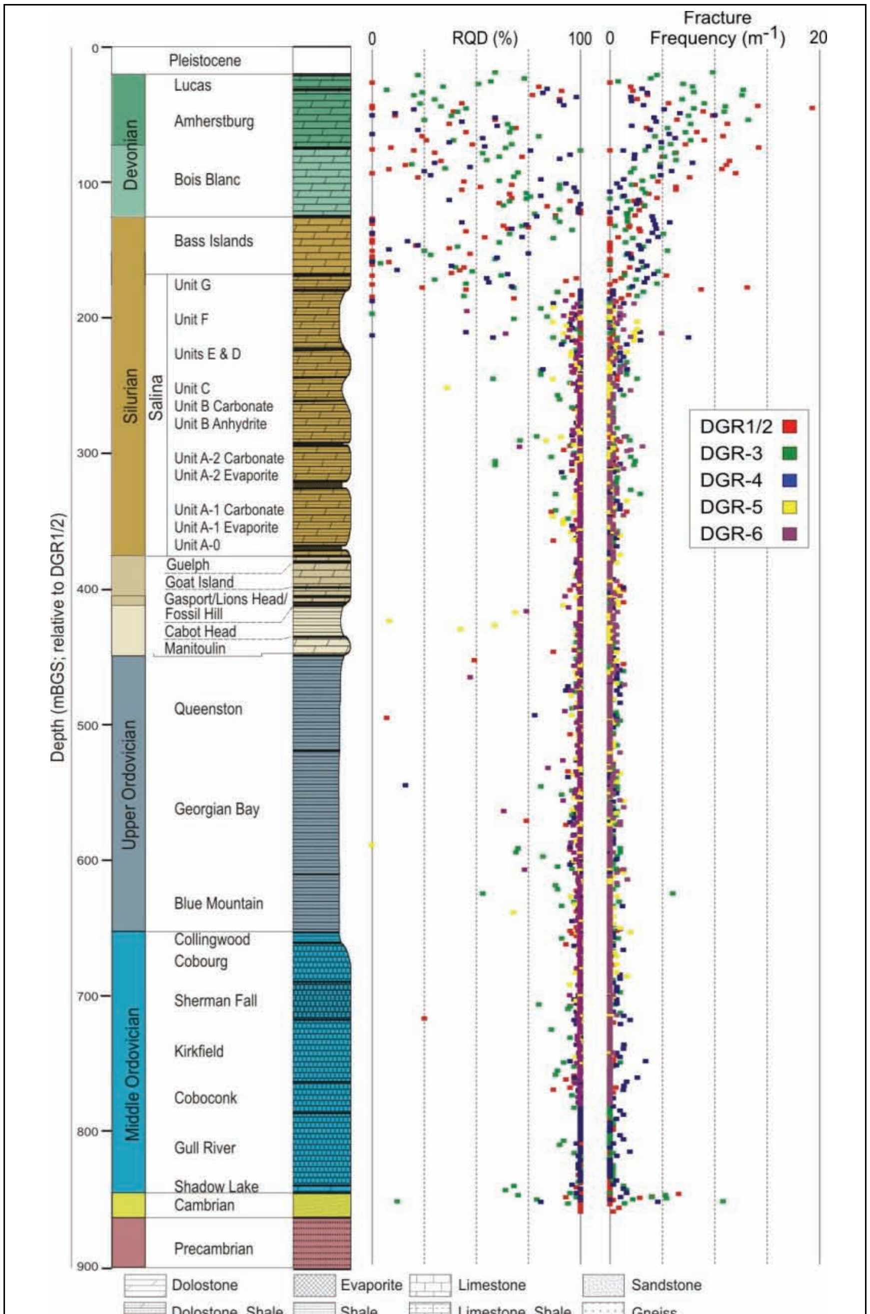
The discontinuity data from the DGR series of deep boreholes also provides an opportunity to further characterize the rock mass. Competent rock formations, illustrated by their high RQD values and low fracture frequencies, were encountered in formations below 200 m in boreholes DGR-1 through DGR-6 (Figure 4-26). The upper 200 m of rock consists mostly of dolostones which contain highly fractured and permeable zones with highly variable RQD values. Based on RQD, the Cobourg Formation is classified as an excellent quality rock, has a very low fracture frequency and few inclined to vertical joints (none were encountered in the DGR series of boreholes). Rock joint orientation measurements and spacing were obtained from the two inclined boreholes (DGR-5 and DGR-6) in Silurian and Ordovician rocks. Fractures at depth are tight and usually cemented with gypsum, anhydrite, halite and/or calcite.

Precedent construction experience with the excavation of underground openings in southern Ontario reveals that excavated openings in the Ordovician shale and Ordovician limestone are likely to be dry and stable (see Section 4.1 of NWMO11n).

4.2.3 In-Situ Stress Magnitude

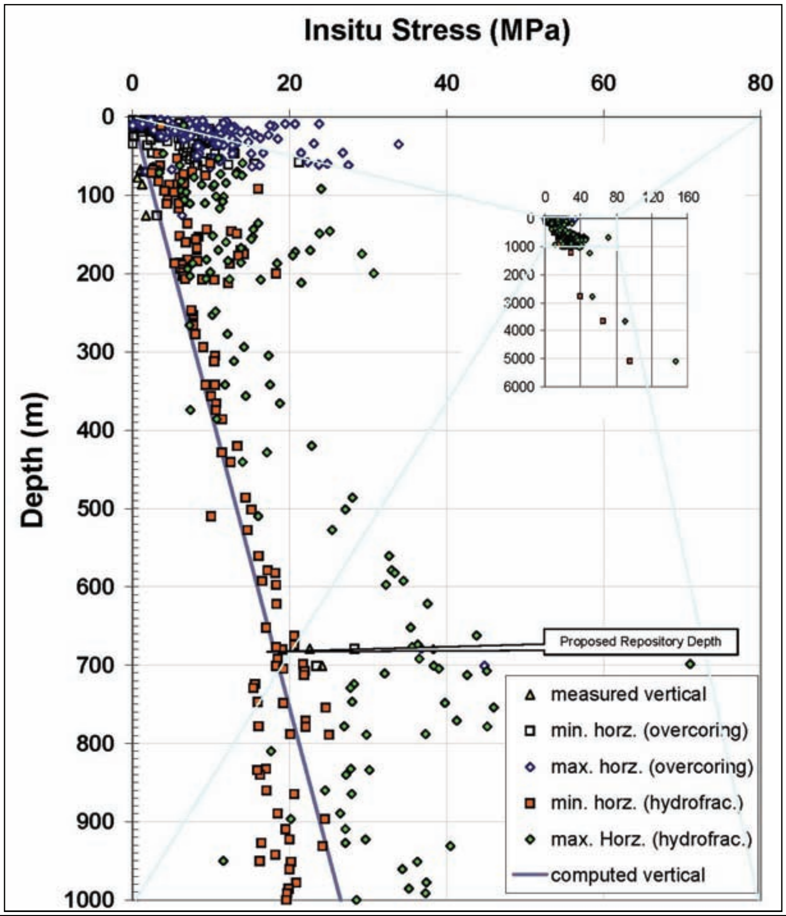
The regional in-situ stress data in Paleozoic sedimentary rock from over 20 sites in the Great Lakes region (NWMO11n) indicates the presence of relatively high horizontal compressive stresses within a thrust fault regime, where vertical stress (σ_V) < minimum horizontal stress (σ_h) < maximum horizontal stress (σ_H). Figure 4-27 shows a regional compilation of maximum and minimum horizontal stresses (σ_H & σ_h) plotted as a function of depth. The stress measurements for shallow bedrock (100 m or less) were made using the over-coring method, while virtually all of the deeper measurements were conducted using the hydrofracture technique. The only deep overcoring data available were collected at 670 mBGS from within the Norton Mine in Ohio. There is a large scatter in both hydraulic fracture and over-coring measurements, particularly in the shallow zone above 200 mBGS, and in the deeper zone below 700 mBGS, from many sites (Figure 4-27).

At the site-scale, borehole core and ATV data from DGR-1 to DGR-4 were analyzed to determine the physical response of these deep boreholes to the surrounding stress field. ATV inspection detected no evidence of borehole breakouts or drilling-induced tension fractures after the completion of the boreholes and during Westbay casing replacement exercises after approximately 24 months for DGR-2 and 6 months for DGR-3 (NWMO11k). Therefore, the primary aim of this analysis was to back-calculate an in-situ stress magnitude profile for the subsurface that would be consistent with the observed borehole wall stability. Assuming a 100% of UCS threshold rock strength the maximum allowable horizontal stress for each section of the borehole was estimated based on the observation of no failure along borehole walls (VALLEY10). The results are summarized in Figure 4-28. The 100% UCS threshold, which represents no failure, is shown on the figure by a green line.



Notes: Figure is based on data collected from boreholes DGR-1 to DGR-6 (NWMO11k) and modified from Figure 3.10 of the Geosynthesis (NWMO11c).

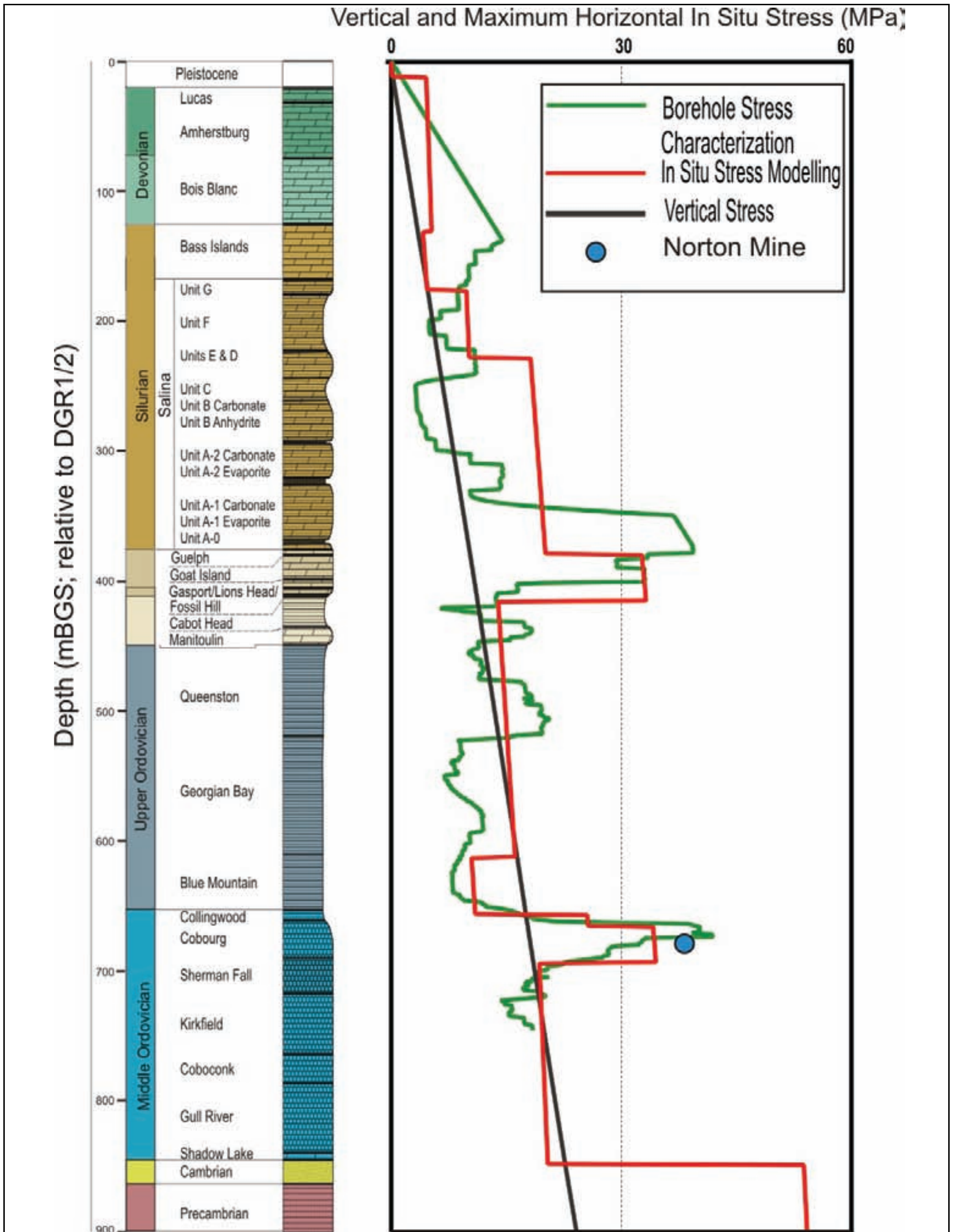
Figure 4-26: RQDs and Fracture Frequencies for the Paleozoic Succession at the Bruce Nuclear Site



Notes: Stress is plotted versus vertical depth. Includes both hydrofracturing and overcoring results. The inset figure includes a larger dataset extending to just beyond 5 km depth. From Figure 5.3 of the Regional Geomechanics report (NWMO11n).

Figure 4-27: Principal Stress Distribution in the Appalachian and Michigan Basins

A model of the DGR stratigraphy was also constructed using *FLAC3D* (red line on Figure 4-28) to further evaluate the vertical distribution of in-situ stress within the sedimentary succession in the subsurface below the Bruce nuclear site (NWMO11t). The model simulates the stiffness variability of individual rock formations oriented in the direction of the maximum horizontal principal stress. The model also included the lack of borehole breakout constraint and used 100% UCS as the borehole wall strength. The model was strained horizontally in both directions to simulate tectonic strains observed at the Norton mine, in Ohio, which has a similar depth horizon and stratigraphy. The results indicate that stiffness contrasts in adjacent rock units play a significant role governing formation-specific in-situ stress distributions.



Notes: Numerical modelling results (red line) plotted against vertical stress profile (black line) and the absence of borehole failure constraint based on borehole wall strength of 100% UCS (green line) (NWMO11n, NWMO11t, VALLEY10).

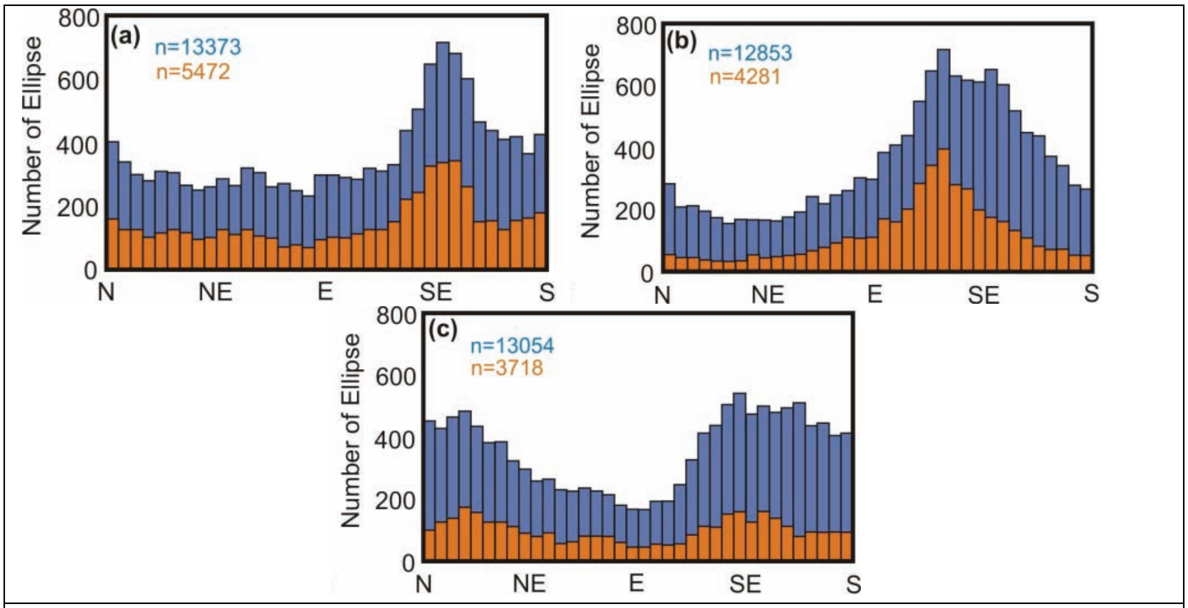
Figure 4-28: Comparison of Stress Constraints and Over-Coring Stress Measurements at the Bruce Nuclear Site

In summary, the in-situ stress ratios at the repository horizon (about 680 mBGS) with σ_V assumed equal to the approximate gravity load of superincumbent materials, are estimated to range from 1.5 to 2.0 for σ_H/σ_V and from 1 to 1.2 for σ_h/σ_V (NWMO11k).

4.2.4 Orientation

The principal sources for estimating regional in-situ stress orientations are the database compiled for the World Stress Map project (HEIDBACH08) and the regional in-situ stress database as described in the Regional Geomechanics – Southern Ontario report (NWMO11n). In brief, the regional principal horizontal in-situ stress is consistently oriented in a northeasterly to east-northeasterly direction throughout northeastern North America, including southwestern Ontario, and the Bruce nuclear site, in particular. These data are reliably constrained by numerous surface and borehole measurements including shallow (<100 m) overcoring measurements and deep (up to about 5 km) hydrofracturing measurements (NWMO11n).

ATV logs from DGR-1 through DGR-4 utilized ellipticity detection analyses to fit ellipses on borehole sections measured from the acoustic travel time logs over 10 cm intervals. From the analysis, the lengths of the ellipses' long and short axes, as well as their orientations, were determined (Figure 4-29).



Notes: Peaks on all figures indicate orientation of the minimum horizontal in-situ stress for all orientations (blue) and for axis ratios greater than 1.0025 (orange) (Redrawn from VALLEY10).

Figure 4-29: Ellipse Long Axis (Apparent Minor Horizontal Stress) Orientation Histograms for DGR-1 and DGR-2 (a), DGR-3 (b) and DGR-4 (c)

The results reveal the length difference between the ellipse axes is typically less than 0.5%. The orientations of the long axis of the ellipses are erratic for most of the borehole length in DGR-1, DGR-2 and DGR-4, except in the Cobourg, Sherman Fall and Kirkfield formations (660 – 760 mBGS) where the orientations are systematic in a SE (138º in DGR-1 and DGR-2 and 131º in DGR-4) direction. The same systematic SE (141°) borehole elongation in the Ordovician limestones was observed in borehole DGR-3. It appears that the systematic SE borehole elongation could be stress related i.e., the direction of the maximum horizontal stress is NE. This orientation is consistent with the regional trend (NWMO11n).

4.2.5 Geomechanics Summary

The following is a summary of the main findings of the geomechanical characterization.

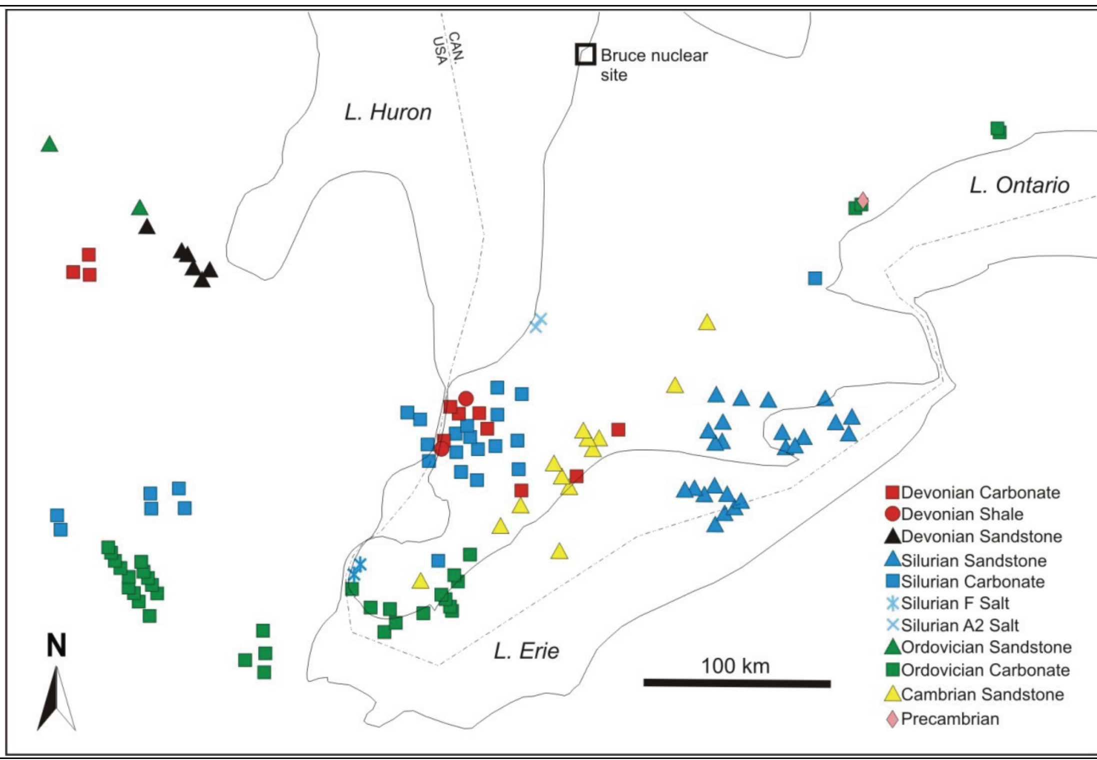
- The Cobourg Formation, which is the proposed DGR host rock at the Bruce nuclear site, is found to be very competent and massive with high RQD and UCS values.
- The values of geomechanical parameters for the Cobourg Formation determined from site specific testing agree favourably with the regional database assembled. The only exception being that UCS values for the site are significantly higher those for the region.
- Geomechanical testing of the Ordovician shale cap rock at the site indicates a moderate rock strength within the same range as the regional data set. The shale formations are generally tight with few sealed joints.
- Borehole observations indicate that the NE to ENE orientation of the maximum horizontal stress at the site appears to be similar to the stress orientation in the Michigan Basin and to the general trend of in-situ stresses in eastern North America.
- Stress analyses to evaluate estimated horizontal magnitudes were carried out, assuming that one principal stress is vertical. The absence of breakouts observed in the deep DGR boreholes permits the setting of an upper bound on the allowable maximum horizontal stress magnitude. At the repository horizon, the range of stress ratios is estimated to be: σ_H/σ_V from 1.5 to 2.0; σ_h/σ_V from 1.0 to 1.2.

4.3 Hydrogeochemistry

The information presented below is based on a comprehensive examination and integration of the regional hydrogeochemistry of southern Ontario (NWMO11q) and detailed site characterization activities specifically related to understanding the hydrogeochemical evolution at the Bruce nuclear site (see Chapter 4 of NWMO11k). The objective of this section is to present a summary of these works.

Section 4.3.1 provides a summary of the hydrogeochemical framework of the Michigan Basin at the regional-scale in terms of the age (i.e., residence time) and origin of the porewater and groundwater, the mechanisms controlling solute transport, and the processes responsible for the observed evolution in porewater and groundwater chemistry. The purpose of this integration is to develop an understanding of the hydrogeochemical evolution of the Bruce nuclear site, discussed in Section 4.3.2.

An important data source for the regional hydrogeochemical setting described below is a compilation of research undertaken at the University of Waterloo, hereafter referred to as the UW database. The UW database, gathered over a period of 25 years, includes information regarding characterization of formation fluids from within the Paleozoic sedimentary succession underlying southwestern Ontario. The UW database is included as an appendix in the Regional Hydrogeochemistry – Southern Ontario report (NWMO11q), and sampling locations for the database are shown in Figure 4-30.



Note: Modified from Figure 2.5 of the Regional Hydrogeochemistry report (NWMO11q).

Figure 4-30: Sampling Locations for the UW Database

4.3.1 Regional Hydrogeochemical Framework of the Michigan Basin

Saline fluids occur at all levels in the Michigan Basin, and although the associated sedimentary rocks were deposited in a marine environment, the salinity of the Michigan Basin fluids (TDS commonly > 200 g/L) is generally much higher than that of seawater (TDS ~ 35 g/L). Salinity is often classified based on the TDS load of the fluid such that (as in CARPENTER78): freshwater TDS <1.0 g/L; brackish water TDS between 1.0 and 10 g/L; saline water TDS between 10 and 100 g/L; and brine TDS >100 g/L.

At the regional-scale, the geochemistry of waters in the sedimentary sequence is characterized by a two-layer system (NWMO11q).

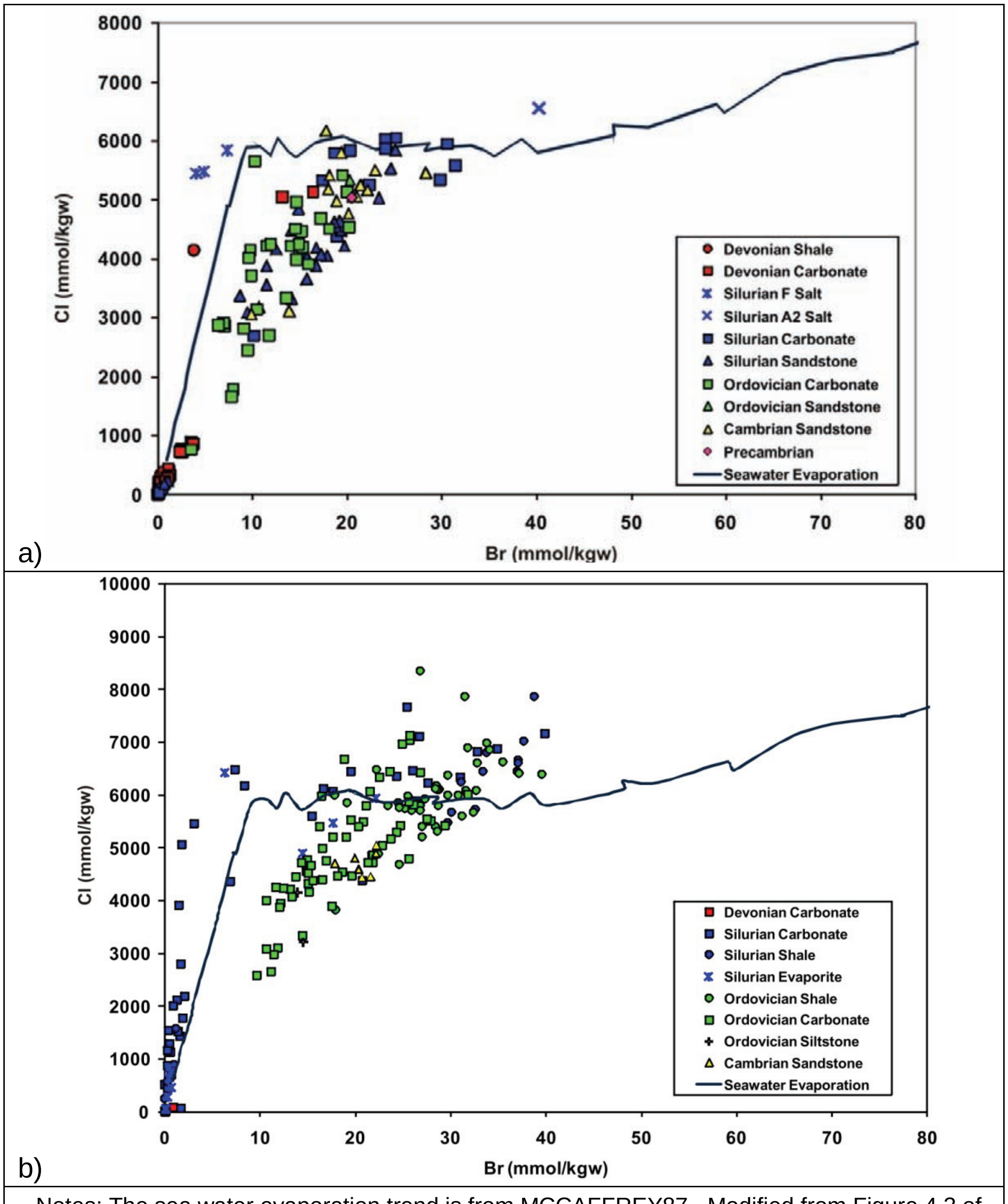
- A shallow groundwater system occurring at depths of up to approximately 170 mBGS and containing fresh through brackish waters. Waters in the shallow zone have $\delta^{18}O$ and $\delta^{2}H$ isotopic compositions suggesting that they are mixtures of dilute, recent, or cold-climate waters with more saline waters.

- An intermediate to deep system at depths greater than 200 mBGS. These waters are brines, as indicated by characteristically elevated TDS values (200 to 400 g/L), and these brines have stable isotopic signatures that are enriched in ^{18}O ($\delta^{18}O$ values of -6 to +3 ‰) and ^{2}H ($\delta^{2}H$ values of -55 to +20 ‰) relative to the Global Meteoric Water Line (GMWL). The information for this system is based predominantly on waters sampled from hydrocarbon reservoirs.

4.3.1.1 Origin and Evolution of Sedimentary Brines

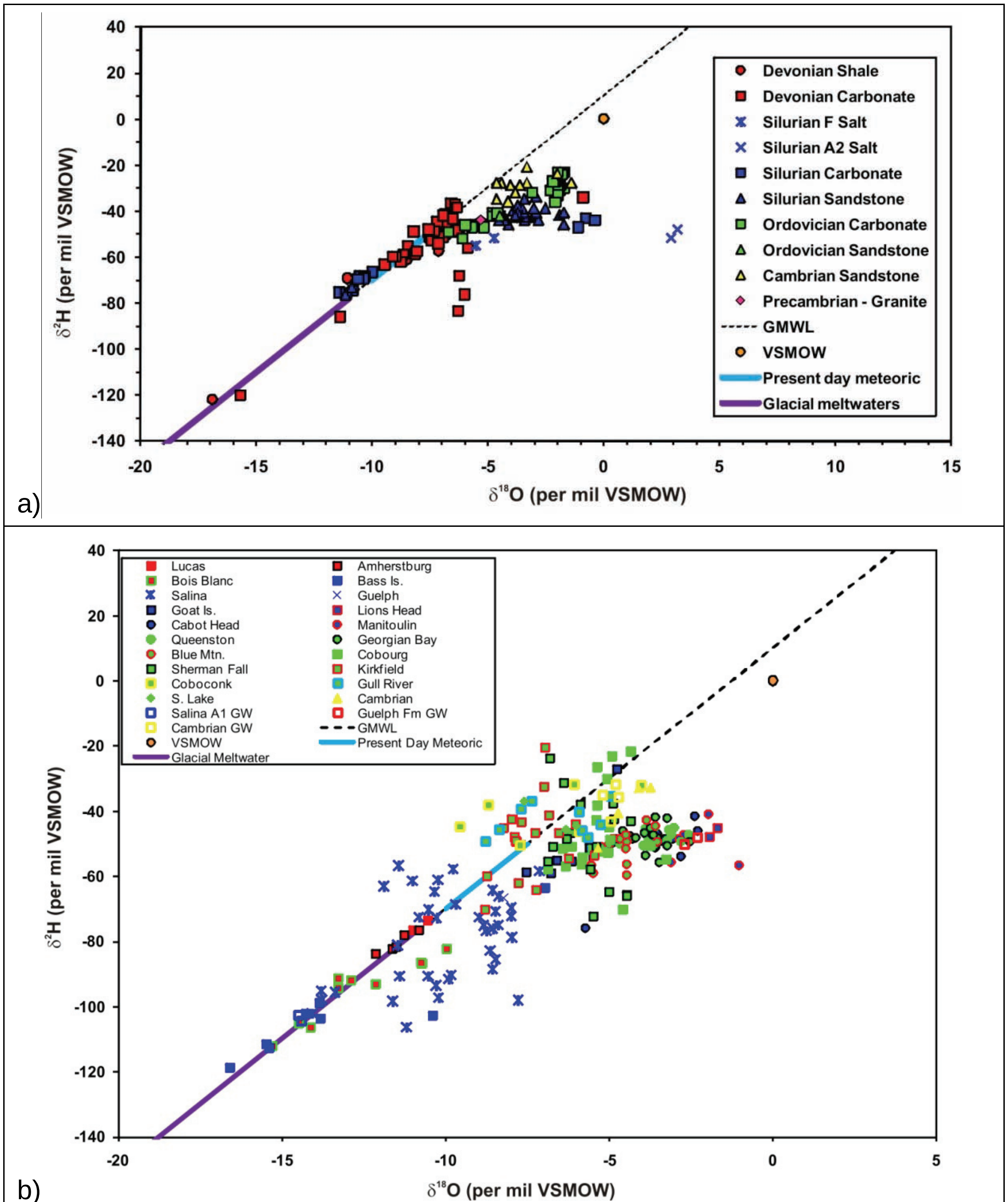
The brines in the Michigan Basin are considered to have originated by evaporation of ancient seawater (WILSON93a, WILSON93b, NWMO11q). For a full discussion of the origin of the sedimentary brines within the Michigan Basin, the reader is referred to Chapter 3 of the Regional Hydrogeochemistry – Southern Ontario report (NWMO11q). The regional and site-specific data for chloride (Cl) versus bromide (Br) and $\delta^{18}O$ versus $\delta^{2}H$ are presented in Figure 4-31(a and b) and Figure 4-32 (a and b), respectively. The trends observed at the regional and site-scale are very similar, suggesting both a common origin for the brines, and a common evolution.

Deviations from the sea water evaporation curve on a plot of Cl versus Br can aid in interpretation of processes that have influenced the evolution of the brine composition through time (MCCAFFREY87), such as mixing of fluids from different sources. The Cl-Br plot in Figure 4-31a from the UW database (NWMO11q) displays trends that indicate: i) dilution of brines by lower salinity water, and ii) dissolution of halite. Dilution is indicated for samples that plot below the sea water evaporation curve on a trend toward the origin, and dissolution of halite is indicated for samples that plot above the sea water evaporation trend.



Notes: The sea water evaporation trend is from MCCAFFREY87. Modified from Figure 4.3 of the Geosynthesis (NWMO11c).

Figure 4-31: Chloride Versus Bromide Concentrations for a) the UW Database, and b) Groundwater and Porewater Samples from the Bruce Nuclear Site



Notes: (a) Represents all fluids within the UW database. From Figure 6.5 of the Regional Hydrogeochemistry report (NWMO11q). (b) Represents groundwater and porewater samples collected at the Bruce nuclear site. Modified from Figure 4.59 of the DGSM (NWMO11k).

Figure 4-32: Hydrogen Versus Oxygen Isotopic Signatures

Infiltration of lower salinity water, such as meteoric water, glacial melt water, normal sea water, or water of hydrothermal origin, could contribute to the observed dilution trends. Figure 4-31b shows the Cl and Br data from groundwater and porewater collected during site characterization activities at the Bruce nuclear site. The trends in the data are very similar to the regional data, suggesting an evaporated seawater origin for the brine, with subsequent modification by processes such as dilution, halite dissolution, and water-rock interaction.

The $\delta^{18}O$ and $\delta^{2}H$ data presented in Figure 4-32a are consistent with the Cl-Br data presented in Figure 4-31a in that they indicate mixing has occurred in the shallow formations between saline brines and more dilute water(s). Most of the samples that display evidence of mixing with meteoric water are from Devonian and Silurian formations, which, in southern Ontario, occur at shallow depths and are commonly overlain by unconsolidated glacial overburden. The deep sedimentary formations of Ordovician and Cambrian age plot primarily to the right of, and below, the GMWL, indicative of long time periods of water-rock interaction. Similar trends are evident in the data from the Bruce nuclear site shown in Figure 4-32b.

When compared to the regional data, the shallow sedimentary formations (Devonian and Silurian) at the Bruce nuclear site may show more influence of mixing with glacial and/or meteoric water(s) (refer to Figure 4-32b) due to their shallower depth relative to samples taken from the same sedimentary formations nearer to the Chatham Sag in southern Ontario (refer to Figure 4-1 and Figure 4-30).

Fluid Migration and Solute Transport Mechanisms

The presence of hypersaline brines in sediments should result in a gravitationally stable system, and fluid flow would not be expected without a large pressure perturbation to the system. Fluids in sedimentary basins also do not flow without changes to hydraulic gradients (KYSER03). Possible driving forces for these changes, which can prompt groundwater flow and solute transport within the context of the geologic history of the Michigan Basin, include orogenesis (see Figure 4-7 and related discussion in Section 4.1.1.2), evaporation, and glaciation. The results of studies that have examined fluid migration and solute transport associated with orogenesis, evaporation, and glaciation are summarized below (see also NWMO11c).

- Fluid migration would likely have occurred within permeable sedimentary units, for example the Cambrian sandstones and dolomitized Ordovician carbonates, in response to hydraulic gradients and crustal motion related to Taconic, Acadian, and Appalachian orogenesis.

- Restricted marine conditions during the Silurian and Devonian periods led to periods of sea water evaporation, which would have created unstable high salinity brine layers in the upper stratigraphic levels of the basin, leading to the formation of hypersaline brines. Overturn by density-driven advection (CONIGLIO94, WINTER95) and diffusion are possible solute transport mechanism under such conditions. Although the Silurian sedimentary rocks are underlain by low-permeability Upper Ordovician shale, localized fracture and fault systems may have provided the opportunity for dense brine to migrate downward into the underlying Ordovician, Cambrian and Precambrian rocks (CONIGLIO94, DAVIES06). In the absence of localized fracturing, the concentration gradients between the underlying sedimentary porewaters and the hypersaline fluids would have resulted in downward diffusion of solutes.

- Barker and Pollock (BARKER84) noted that the natural gas chemistries in samples from the Michigan Basin were distinct from the natural gas chemistries within the Appalachian Basin, indicating that there has been no significant migration of gases between the basins. This interpretation is consistent with isotopic analyses of Ordovician brines (MCKENNA92, DOLLAR88, DOLLAR91), which indicate that groundwater from Ordovician formations within the Michigan Basin have a different evolution than fluids in the Appalachian Basin.

- Oil-field brines obtained near the eastern edge of the Michigan Basin in Ontario have strontium (Sr) isotopic compositions that are very similar to samples from deeper within the Michigan Basin suggesting intra-basin fluid migration over distances of hundreds of kilometres (MCNUTT87).

- Sherwood Lollar et al. (SHERWOOD94), using isotopic and compositional indicators, concluded that hydrocarbons to the southeast of the Algonquin Arch display elevated thermal maturities consistent with migration from the Appalachian Basin. Conversely, gas hydrocarbons from northwest of the Algonquin Arch do not display elevated maturities and are therefore not likely sourced from the Appalachian Basin, indicating the lack of detectable migration (mixing) between the basins (SHERWOOD94).

- Pb isotope ratios ($^{207}Pb/^{204}Pb$ and $^{208}Pb/^{204}Pb$) for galena northwest of the Algonquin Arch in the Michigan Basin indicate a crustal source that is distinct from the Pb in galena samples from the Appalachian Basin (FARQUHAR87) southeast of the arch, and lends support to the interpretation of intra-basin, but not inter-basin, fluid migration for the Michigan and Appalachian basins.

- Sedimentary basin fluid migration is also evidenced by the existence of HTD:

  - Hydrothermal-dolomite-hosted oil and gas reservoirs in the Black River and Trenton groups within southern Ontario and Michigan are presumed to have also formed as a result of brine migration during the Taconic Orogeny (DAVIES06). Middleton et al. (MIDDLETON93) measured homogenization temperatures ranging between 100 and 220 °C in primary fluid inclusions from the fracture-related dolomite in oil and gas fields in the Chatham Sag region of southern Ontario (see Figure 4-1 for location). These temperatures are substantially higher than those likely to be generated during peak burial of the sedimentary sequence suggesting the influence of these dolomitizing hydrothermal fluids (CONIGLIO94).

- On the basis of carbon and strontium isotope data, sea water-derived fluids are thought to be responsible for regional-scale dolomitization in the Middle Silurian Guelph Formation (CONIGLIO03). Primary fluid inclusions temperatures of between 65 and 130 °C indicate that the fluids were hydrothermal in nature (CONIGLIO03, after ZHENG99).

- Several authors suggest that fracture-related dolomitization and hydrocarbon migration in the Michigan Basin likely occurred during the Late Paleozoic to Early Mesozoic (PROUTY88, HURLEY90, BUDAI91).

- High fluid pressures at the base of glacial ice sheets are also potentially able to drive fluid migration. Although glacial events are recognized periodically throughout geologic history, there are no known events that would have affected the Michigan Basin between Upper Silurian and Pleistocene time (PRICE99). Fluid migration could also occur in response to pressure gradients formed by tilting of the basin during differential isostatic rebound following deglaciation.

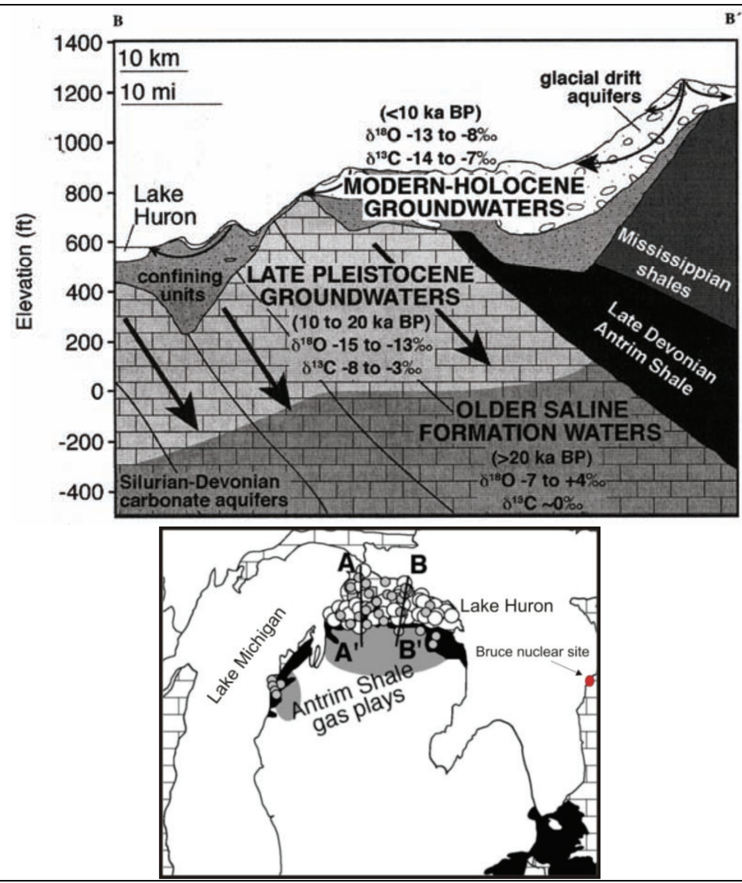
Isotopic Evidence for Pleistocene and Post-Pleistocene Infiltration Events

The widespread occurrence of ancient brines in the Michigan Basin demonstrates that, under conditions prevalent since the Paleozoic, it has not been possible for hydraulic heads generated in freshwater aquifers at the top boundary of the basin to displace the deep basin brines.

- Glacial melt water can be pressurized beneath continental ice sheets during interglacial periods to levels in excess of ambient heads, and has been driven to depths of several hundred metres in Paleozoic aquifers around the periphery of the Illinois and Michigan basins (see MCINTOSH05, MCINTOSH06; PERSON07 and references therein). The conceptual model developed by McIntosh and Walter (MCINTOSH06) for Pleistocene infiltration around the margins of the Michigan Basin is presented in Figure 4-33. Their research suggests that glacial melt water has penetrated to depths up to 200-300 m in Silurian-Devonian carbonate aquifers in northern Michigan on the northern margin of the Michigan Basin.

- Stable O isotope data provide the best evidence for infiltration and cross-formational mixing of glacial melt water, which displays strongly depleted $\delta^{18}O$ values (between -25 and -11 ‰), and this cold-climate water can be distinguished from: i) hypersaline basinal brines which have $\delta^{18}O$ values ranging between -6 and +5 ‰ (WILSON93a) and ii) modern recharge in southwestern Ontario which has $\delta^{18}O$ values typically ranging between -11 and -7.5 ‰.

- Although stable O and H isotopic data demonstrate that fresh glacial melt water has infiltrated around the periphery of the Michigan Basin, the composition of the water has been significantly altered by mixing with ancient hypersaline brines and by dissolution of evaporite minerals (refer to Figure 4-32a and b). Evidence for these changes in water chemistry is reviewed in detail by McIntosh and Walter (MCINTOSH05, MCINTOSH06).



Notes: Lower figure (modified from MCINTOSH06) indicating position of section line B-B' used in upper figure. Modified from Figure 4.2 of the Geosynthesis (NWMO11c).

Figure 4-33: Conceptual Model Showing Ancient Brine at Depth, Cold-Climate Water Infiltrated to Mid-Depths, and Modern Meteoric Water Near Surface

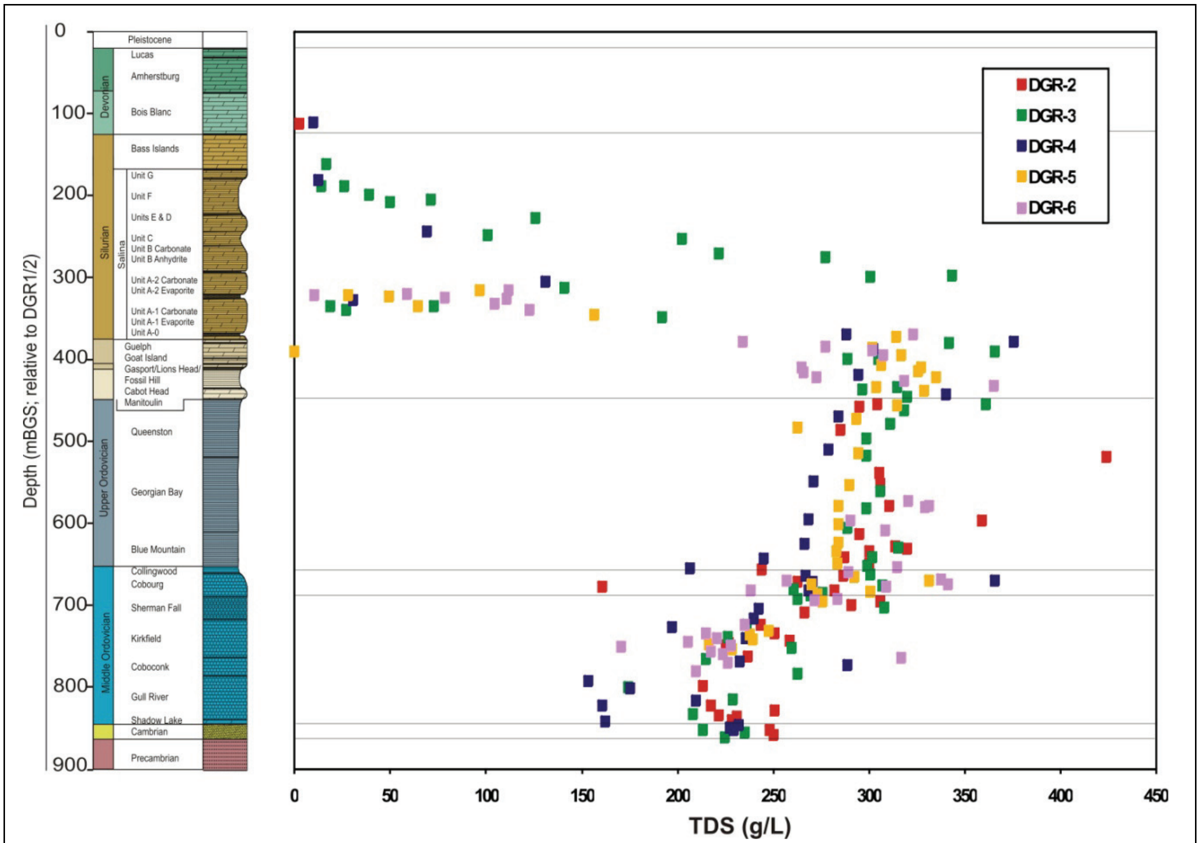
4.3.2 Hydrogeochemical Data from the Bruce Nuclear Site

In a similar manner as the regional discussion above, hydrogeochemical site characterization activities at the Bruce nuclear site have focused on the collection of data that will assist in identifying the residence time and origin of the porewaters and groundwaters underlying the Bruce nuclear site (e.g., NWMO11k). In particular, these results provide evidence regarding the extent of meteoric water and/or glacial melt water infiltration, allow for estimation of the redox conditions present, and provide constraints on the processes and timing of solute transport, in the key host (Cobourg Formation) and bounding Ordovician rocks for the proposed DGR. As in Section 4.1.2.1, all mention of the Cobourg Formation below refers only to the lower argillaceous limestone member of this formation. The results are considered below in terms of the distinguishable shallow, and intermediate to deep, groundwater systems. The Cambrian unit is also discussed separately. The discussion is based primarily on results from natural tracer, major ion, and gas characterization analyses undertaken as part of the site characterization activities (NWMO11k). Following this, the conceptual model and numerical modelling results for the hydrogeochemical evolution of the Bruce nuclear site are provided.

The hydrogeochemical characteristics of the porewaters and groundwaters, described below, are obtained by direct sampling in the case of groundwater (HEAGLE10, JACKSON10), and by use of leaching/extraction techniques for estimation of porewater composition in low-permeability rocks (CLARK10a, CLARK10b, KOROLEVA09). The six deep boreholes, DGR-1 through DGR-6, as well as the two existing shallow bedrock monitoring wells (US-3 and US-7) and an additional shallow monitoring well (US-8), were instrumented with MP38 multi-level casings manufactured by Westbay Instruments Inc., which allow groundwater samples to be obtained from packer-isolated intervals.

4.3.2.1 Groundwater and Porewater Characterization at the Bruce Nuclear Site

The distribution of TDS with depth beneath the Bruce nuclear site, presented in Figure 4-34, allows for the distinction of groundwater systems relevant to the following discussion. In a similar manner as the regional two-layer system, a shallow system of fresh to brackish water is defined for the overburden unit and the bedrock interval from the Lucas and Amherstburg formations to a depth of approximately 170 mBGS, which corresponds to the top of the Salina G Unit encountered at a reference depth of 169.3 mBGS in DGR-1/2 (Table 3.1 of NWMO11k). These TDS concentrations are relatively low compared to groundwater and porewater samples from the underlying intermediate to deep system. As Figure 4-34 indicates, the TDS values increase with depth from within the Salina F unit to the base of the Silurian (Guelph to Manitoulin formations). In the Ordovician rocks, TDS values are relatively high (most fluids have TDS > 200 g/L). TDS values are stable from the Queenston Formation to the Collingwood Member, and then decrease with depth, but typically maintain concentrations > 200 g/L in the carbonate-rich Cobourg to Gull River formations. At the base of the profile, within the Shadow Lake and Cambrian formations, TDS values increase slightly, but are still lower than the values measured within the Ordovician shales.



Note: Modified from Figure 4.54 of the DGSM (NWMO11k).

Figure 4-34: TDS Concentration versus Depth for DGR Boreholes

4.3.2.2 Shallow Groundwater System Characterization

The shallow groundwater system is characterized by two different water types. Within the overburden aquifer, the water is classified as Ca:Na-HCO_3 with low TDS. In the upper bedrock (above 170 mBGS), the dominant cations yield a Ca:Mg-HCO_3 water with TDS of approximately 0.5 to 5.0 g/L. The TDS of the groundwater samples from the US-series wells indicate a transition from fresh to brackish water (within the Lucas and Amherstburg formations), and brackish water throughout the Bois Blanc and the upper part of the Salina G Unit. Generally, solute concentrations in US-3 are slightly greater than US-7 and US-8, but the molal ratios are similar in each borehole, and concentrations increase with depth in each borehole.

Major Ions

Ferrous iron, or reduced iron (Fe^{2+}), concentrations in the US-series samples were between 0 and 1.3 mg/L. Where there was dissolved ferrous iron in the groundwater, the reduction-oxidation state may be classified as iron-reducing. This classification is supported by the core logs for DGR-1, DGR-3 and DGR-4, which note the presence of pyrite near the base of the Amherstburg Formation. Pyrite is indicative of ferrous iron in solution, resulting in precipitation of FeS_2. Pyrite is inconsistently observed through the Bois Blanc and Bass Islands formations in DGR-1, DGR-3 and DGR-4. Although pyrite was identified in the cores, sulphide was not detected in the groundwater samples.

Colorimetric and potentiometric measurement of Dissolved Oxygen (DO) showed concentrations were below 2 mg/L in most groundwaters sampled from US-series wells, except for one measurement of 6.3 mg/L in US-8 at a depth of 170.2 mBGS. These low oxygen levels indicate DO is limited in the shallow groundwater. Iron staining is observed in rocks of the Lucas, Amherstburg and Bois Blanc formations, however, and is likely due to ferric iron, or oxidized iron (Fe^{3+}), which is commonly associated with relatively oxidizing conditions. Isolated oxidized zones may occur in the upper flow system (Lucas, Amherstburg and Bois Blanc formations) based on the presence of iron staining within these rocks (NWMO11k).

The observed low ferrous iron concentrations and low DO contents (<2 mg/L) in the groundwater, combined with the presence of iron and pyrite in the cores, suggests oxygen is almost absent in the shallow groundwater, and that the redox conditions are in a transition from near-anaerobic to iron-reducing (NWMO11k).

Alkalinities measured in the field range between 100 and 330 mg/L as $CaCO_3$, with pH ranging between 6.8 and 8.5. The alkalinity in the samples is derived from HCO_3^-, which is the dominant anion in the groundwater due to carbonate dissolution. The major ion chemistry profile for the shallow groundwater system (US-8 data) is shown in Figure 4.44 of the DGSM (NWMO11k).

Oxygen and Hydrogen (^{18}O, ^{2}H, ^{3}H)

The stable water isotope data ($\delta^{18}O$ and $\delta^{2}H$) for shallow bedrock groundwaters collected from US-series wells, as well as drill waters, are plotted in Figure 4-35 and compared to the GMWL. Figure 4-35 shows the shallow bedrock groundwaters grouped by Middle to Lower Devonian dolostones (Lucas, Amherstburg and Bois Blanc formations) and Upper Silurian dolostones (Bass Islands and Salina G Unit). For comparison purposes, the groundwater samples collected from the Salina A1 Unit carbonate aquifer, the Guelph Formation, and the Cambrian sandstone in DGR boreholes are also shown. The following features can be observed in Figure 4-35.

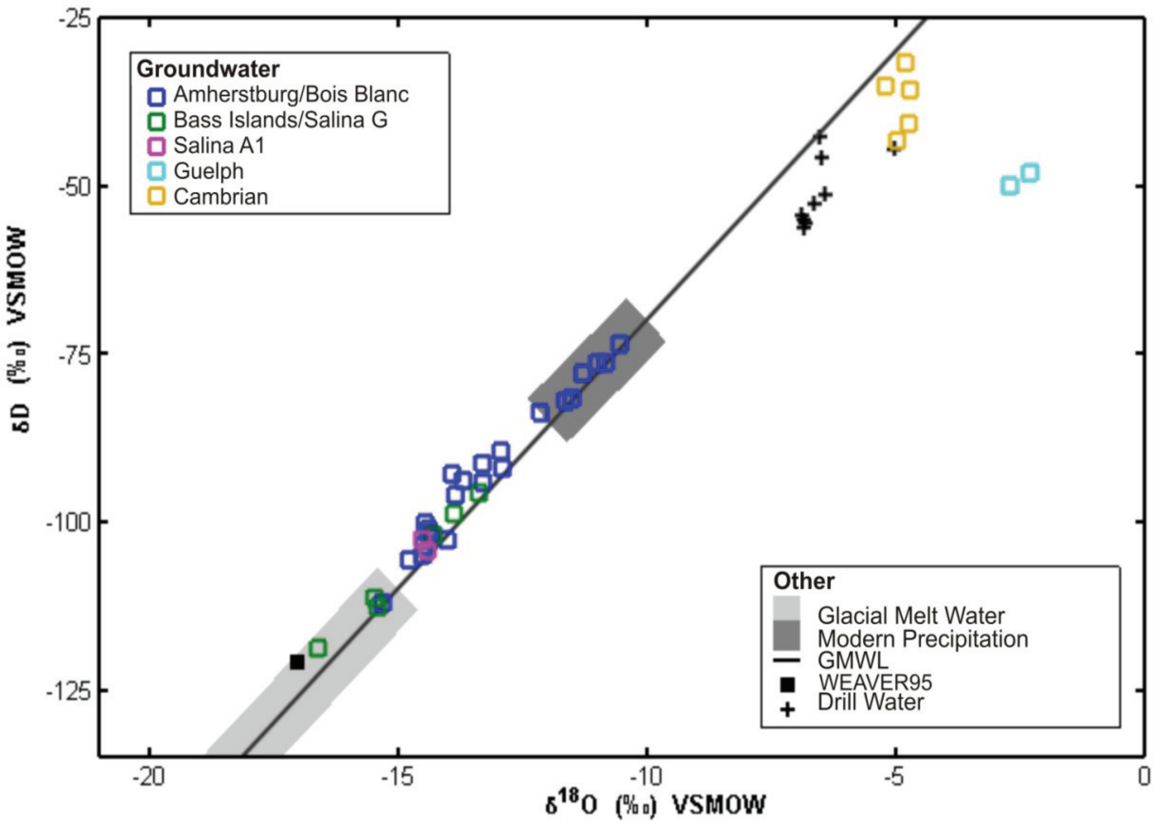
- Lake Huron water used for drilling has a characteristic evaporative enrichment signature. As well, the Cambrian groundwater is significantly enriched and plots close to the GMWL. Both of these waters plot remotely from the Devonian and Upper Silurian dolostone groundwaters, suggesting that the shallow bedrock groundwaters are not influenced by drill water or Cambrian sandstone water.

- The groundwater values in the Devonian and Silurian aquifers plot between modern precipitation (mean ~-12 ‰ for $\delta^{18}O$) and glacial meltwater (i.e., -20 to -15 ‰ for $\delta^{18}O$), indicating that these groundwaters are mixtures containing both glacial melt water and modern precipitation (NWMO11q).

Tritium units (TU) are a measure of the concentration of ^{3}H in a given sample. One tritium unit (1 TU) is equal to one ^{3}H atom per 10^{18} hydrogen atoms. Most groundwater samples from the US-series wells had < 35 TU (<4.13 Bq/L), and 14 out of 29 samples had tritium counts below the detection limit for direct counting analysis (<6 TU, or 0.708 Bq/L). Tritium in precipitation at the Bruce nuclear site is elevated and averaged 1700 TU (200.6 Bq/L) during 2005-2006 (BP08). Although the $\delta^{18}O$ and $\delta^{2}H$ ratios indicate the groundwater is of atmospheric origin (Figure 4-35), the low tritium counts suggest the groundwater does not contain recent atmospheric water that is affected by activities at the Bruce nuclear site.

Chloride

The trend toward low solute concentrations toward the surface, as indicated by low Cl and Br concentrations (refer to Figure 4-36), likely results from diffusive or advective mixing of surface-derived meteoric water with the shallow formational fluids. This interpretation is supported by $\delta^{18}O$ and $\delta^{2}H$ data for groundwater and porewater from the Bruce nuclear site (refer to Figure 4-35).



Notes: Also shown is the range of modern precipitation (FRITZ87) and the range and best estimate of glacial meltwater for Southern Ontario (ARAVENA95, WEAVER95). Modified from Figure 4.47 of the DGSM (NWMO11k).

Figure 4-35: Cross-Plot of δ^2H (δD) Versus δ^{18}O for Drill Waters and Groundwater Samples from US-3, US-7, US-8, and DGR Boreholes

4.3.2.3 Intermediate to Deep System Groundwater and Porewater Characterization

The intermediate to deep groundwater system is characterized by a transition from the brackish Ca-SO_4 water observed in the Silurian G Unit (TDS ~10 g/L), to an increasingly concentrated Na-Cl type (saline) brine from the Silurian C Unit down to the base of the Cambrian (244.6 to 860.7 mBGS. The high TDS value measured at the base of the Queenston Formation (~423 g/L) is not considered to be representative of the porewater TDS in this interval and is, instead, interpreted to be the result of mineral salt dissolution during the porewater extraction process. The underlying Precambrian fluid chemistries have not been characterized at the Bruce nuclear site, but investigations regarding the chemistries of shield brines have been the subject of extensive study across southern Ontario as discussed briefly in Section 4.3.3.

Natural Tracers

Analysis of natural environmental tracer profiles (such as chloride, bromide, and the stable isotopes of oxygen and hydrogen) in the porewaters of low-permeability sedimentary rocks can be a powerful approach for assessing the transport properties of potential host rock formations for nuclear waste management at time and spatial scales relevant to a DGR.

The Cl and Br profiles below the Bruce nuclear site are presented in Figure 4-36 and stable isotope profiles are presented in Figure 4-37.

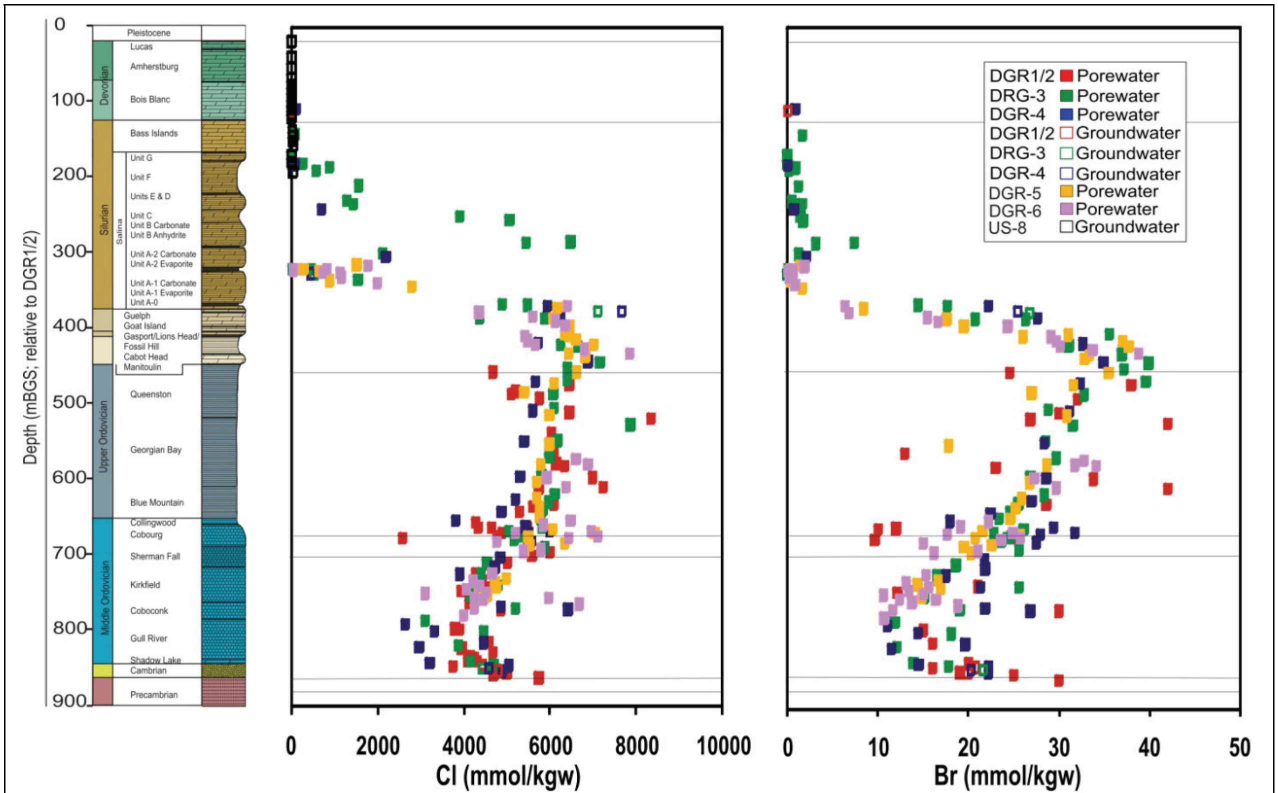
Trends in the data should be considered in terms of deviations from some initial baseline condition. For these tracers, that condition could be considered to be their respective concentrations in the ancient evaporated sea water from which the Michigan Basin brines are presumed to have been derived (WILSON93a, WILSON93b). The baseline $\delta^{18}O$ is best represented by a value of -2 ‰ for all of the sedimentary formations (GRAF65, DOLLAR88, WILSON93a, WILSON93b). An initial Cl concentration of 6 to 7 mol/kgw is proposed for the Silurian and Devonian fluids to represent evaporated sea water, and an initial Cl concentration of 0.6 mol/kgw is suggested for the Ordovician and Cambrian formation fluids as a representation of normal marine sea water. These baseline values are assigned to maintain consistency with the evolutionary history of the Michigan Basin.

The following features are observed in the natural tracer data.

- There is a decrease for all tracers from the Guelph Formation upward through the Silurian. The presence of high horizontal hydraulic conductivity (K_H) zones in the Silurian, and the corresponding abrupt variations in tracer profiles with depth through the Silurian sediments, suggest that dilution may have occurred by a combination of advective mixing and diffusion. See further discussion in Section 4.4.1 below and in Section 4.9 of the DGSM (NWMO11k).

- There is a less pronounced but persistent trend toward depleted $\delta^{18}O$ values, reduced Cl and Br concentrations, and enriched $\delta^{2}H$ values below the Ordovician shale.

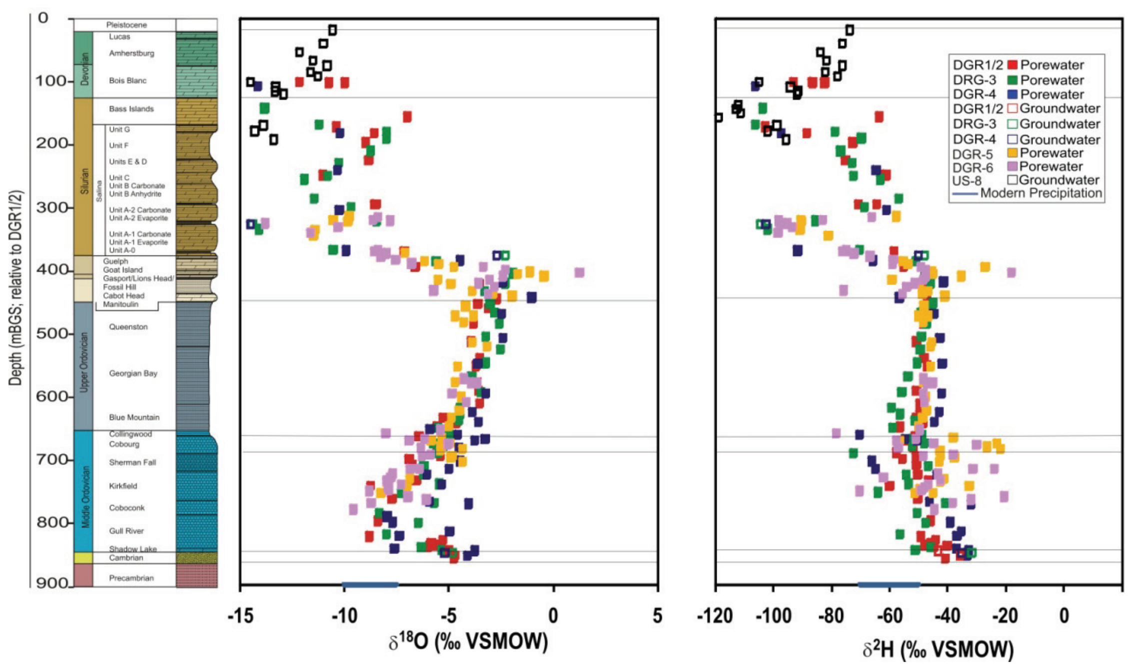
- The trends toward depleted $\delta^{18}O$ values, and reduced Cl and Br concentrations, below the Ordovician shale, are interrupted at the Cambrian where the tracer values become more enriched.



Notes: Chloride profile (left) is modified from Figure 4.53 of the DGSM (NWMO11k). Bromide profile (right) is modified from Figure 4.55 of the DGSM (NWMO11k).

Figure 4-36: Vertical Depth Profiles for Natural Tracers Cl and Br Determined in Porewater and Groundwater

Little is known about the timing of exposure of the Devonian rocks in southern Ontario to infiltration. If something close to the present-day erosion level was exposed during the Pleistocene, then the cyclic nature of glacial-interglacial periods in the past 1 to 2 Ma, as discussed in Section 4.5.1 below (see also inset of Figure 4-78), would have resulted in repeated infiltration events in the Devonian (and possibly Silurian) stratigraphy of southern Ontario, with subsequent diffusive equilibration of the formation waters in the low-permeability sediments with fresh water during interglacial periods. These processes may explain the trends toward depleted ^{18}O and ^{2}H (Figure 4-37) and decreased Cl and Br concentrations (Figure 4-36) that are observed above the Silurian Guelph Formation and discussed in Section 4.3.2.2. The most ^{18}O- and ^{2}H-depleted signatures (δ values of -14.5 and -110 ‰, respectively), which indicate the presence of glacial melt water, occur in the thin aquifer of the Salina A1 Unit carbonate encountered at a reference depth of 325.5 to 328.5 mBGS in DGR-1/2 (Figure 4-37). This occurrence is not interpreted as representing vertical infiltration to this depth at the site, but as representing flow within the Salina A-1 Unit from where it subcrops and is recharged east of the site.



Notes: Oxygen profile (left) is modified from Figure 4.61 of the DGSM (NWMO11k). Deuterium profile (right) is modified from Figure 4.62 of the DGSM (NWMO11k).

Figure 4-37: Vertical Depth Profiles for Natural Tracers ^{18}O and ^{2}H Determined in Porewater and Groundwater

With increasing depth, the general trend in the data in the Middle Ordovician is toward a gradual depletion in ^{18}O and decreasing salinity. Coincident with the depletion of ^{18}O, there is minor enrichment in ^{2}H (Figure 4-37). In contrast with the natural tracer profiles in the Silurian, the very low K_H values in the Ordovician limestones, as discussed in Section 4.4.1 below and in Section 4.9 of the DGSM (NWMO11k), and the smooth nature of the downward depletion trends, suggest that solute transport in this deeper system is dominated by diffusion. The time period required to form such trends in the profiles by diffusion is expected to be on the order of tens to hundreds of millions of years, and is discussed in more detail in Sections 4.3.3.1 and 4.3.3.2.

The groundwater and porewater profiles change within the Cambrian sandstone, where tracer concentrations shift back toward values representative of Cambrian groundwater sampled from southwestern Ontario oilfields (Figure 4-31a and Figure 4-32a). Possible mechanisms to explain the chemical signature in the Cambrian at the Bruce nuclear site are discussed in Section 4.3.4.

Water-Rock Interaction

Water-rock interaction must be considered as a possible explanation for the observed $\delta^{18}O$ and $\delta^{2}H$ profiles with depth. At elevated temperatures, reactions with calcite and illite-smectite clays could lead to an increase in $\delta^{18}O$ values (as is commonly observed in sedimentary basin brines), but such reactions cannot easily explain the decrease in $\delta^{18}O$ to values as low as -8.78 ‰ in the Middle Ordovician carbonates. The dolomite content in the Middle Ordovician limestone increases versus depth (Figure 3.5 of NWMO11k), coincident with the decrease in $\delta^{18}O$ values versus depth. If it is assumed that the porewater in the system is static, a very long porewater residence time is available and it may be possible that the observed $\delta^{18}O$ profiles have evolved in response to isotopic equilibration with dolomite. Using $\delta^{18}O$ values for Middle Ordovician dolomite from Coniglio and Williams-Jones (CONIGLIO92) and dolomite-water fractionation factors from Vasconcelos et al. (VASCONCELOS05) and Chacko and Deines (CHACKO08), the isotopic composition of pore water in equilibrium with dolomite can be calculated over a reasonable temperature range (25 to 45 °C).

Results of these calculations indicate that equilibration with dolomite could result in porewater $\delta^{18}O$ values from -13.1 to -2.7 ‰. These results suggest that isotopic equilibration with dolomite might explain the observed decrease in $\delta^{18}O$ values with depth.

Although water-rock interaction might provide an explanation for the $\delta^{18}O$ profile, it is not apparent that water-rock interactions could explain the observed ^{2}H enrichment versus depth in the Middle Ordovician. It is well known that ^{2}H partitions preferentially to the fluid during mineral hydration reactions (e.g., feldspar to clay transformations) (CLARK97) and this fractionation may have operated throughout the Ordovician units as detrital feldspars were altered to clay minerals. However, mass-balance requirements suggest that any resulting ^{2}H enrichment of the porewater should be proportional to the ratio of sheet-silicate content to porosity. Regarding illite and chlorite content (Figure 3.7 of NWMO11k), there is no significant increase versus depth in the Middle Ordovician as would be expected if mineral hydration reactions were responsible for the observed ^{2}H enrichment in the porewater.

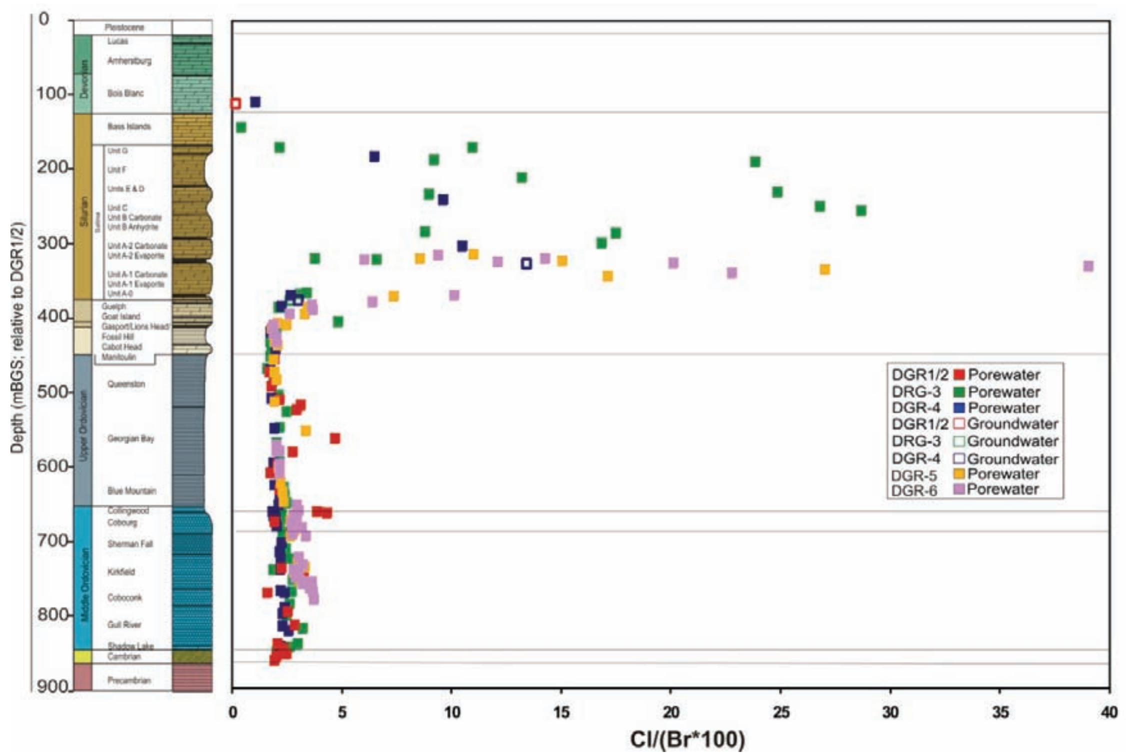
Fluid Mixing

In contrast to water-rock interaction, the Middle Ordovician trends for all tracer profiles could result from one or more mixing events with water at depth that is relatively depleted in ^{18}O, has lower Cl and Br concentrations, and is enriched in ^{2}H. This could not be the brine that is currently contained in the Cambrian sandstone because it has a higher salinity and more enriched isotopic composition than the porewater in the Middle Ordovician carbonates (Figure 4-37 and Figure 4-36). However, the relatively high permeability in the Cambrian sandstone could have allowed changes in the groundwater composition at some point in the geologic past, provided the appropriate driving mechanism(s) for fluid migration were present. The question arises as to whether groundwater in the Cambrian aquifer, or groundwater in the underlying shield, could have provided a suitable end member to generate these mixing trends. The current state of knowledge regarding groundwater in the Precambrian shield and in the Cambrian is discussed in Sections 4.3.3 and 4.3.4, respectively.

The relatively constant Cl/Br ratios in the Ordovician and Cambrian rocks suggest that halite dissolution does not have a significant influence on the Cl concentration in the porewater (Figure 4-38). The elevated Cl/Br ratios in the Salina Formation suggest that these porewaters have been influenced by halite dissolution. The occurrence of halite within the Ordovician units, as shown in Figure 4-15, suggests that hypersaline brine was present at depth within the Middle Ordovician at some time in the geologic past.

Gas Characterization

Methane (CH_4), carbon dioxide (CO_2) and helium (He) were extracted from samples of groundwater and core (CLARK10a, CLARK10b). The isotopic compositions $\delta^{13}C$ (CH_4 and CO_2), $\delta^{2}H$ (CH_4) and $^{3}He/^{4}He$ ratios were determined for the gases. The approach of normalizing the total mass of extracted gas (CH_4 and CO_2) to the porewater content was adopted. This approach does not provide an accurate measure of dissolved gas content in cases where gas occurs in other forms, such as in a separate gas phase, dissolved in liquid hydrocarbons, or sorbed to solid forms of organic carbon; however, as measured (mass of gas per mass of rock normalized to water content), the concentrations can be compared to the solubility limits for the gases in brine. Values in excess of the solubility limits provide evidence for the presence of either a separate gas phase or gas in association with solid organic carbon or liquid hydrocarbons.



Note: Modified from Figure 4.58 of the DGSM (NWMO11k).

Figure 4-38: Cl/Br Ratios versus Depth for DGR Boreholes

<u>Methane and Carbon Dioxide</u>

The CH_4 and CO_2 data are reported in units of mmol/kgw but, as discussed above, they should not be considered to be exactly equivalent to porewater aqueous concentrations. The concentrations of CH_4 and CO_2, and the respective stable isotopic data, are presented in Figure 4-39 and Figure 4-40. There are a number of features observed consistently in the CH_4 and CO_2 data from the DGR drill cores.

- Low CH_4 concentrations are observed near the surface and down to a depth of approximately 300 mBGS, which corresponds to the top of the Upper Silurian Salina A2 Unit.

- Elevated CH_4 concentrations occur in proximity to the hydrocarbon-containing Guelph Formation at 375 to 410 mBGS (see also OBERMAJER00). The overlying Salina A1 and A2 units may represent a low-permeability barrier to gas transport upward from the Guelph Formation.

- The CH_4 concentration increases gradually downward through the Ordovician Queenston Formation shale and then remains at a near constant value through the Georgian Bay Formation shale.

- There is a pronounced increase in the CH_4 concentration in the interval represented by the Blue Mountain Formation shale and the Collingwood Member at 617 to 660 mBGS.

- The CH_4 concentration in the Middle Ordovician limestones and the underlying Cambrian sandstone is low relative to the overlying Blue Mountain shale and the Collingwood calcareous shale.

- The CO_2 data (Figure 4-40) display a step-wise increase, with the lowest concentrations occurring from surface downward to the Guelph Formation, intermediate concentrations from the top of the Guelph Formation down to the bottom of the Blue Mountain Formation shale, and highest concentrations in the Middle Ordovician carbonates.

The stable isotope data provide important insight into the origin of the CH_4.

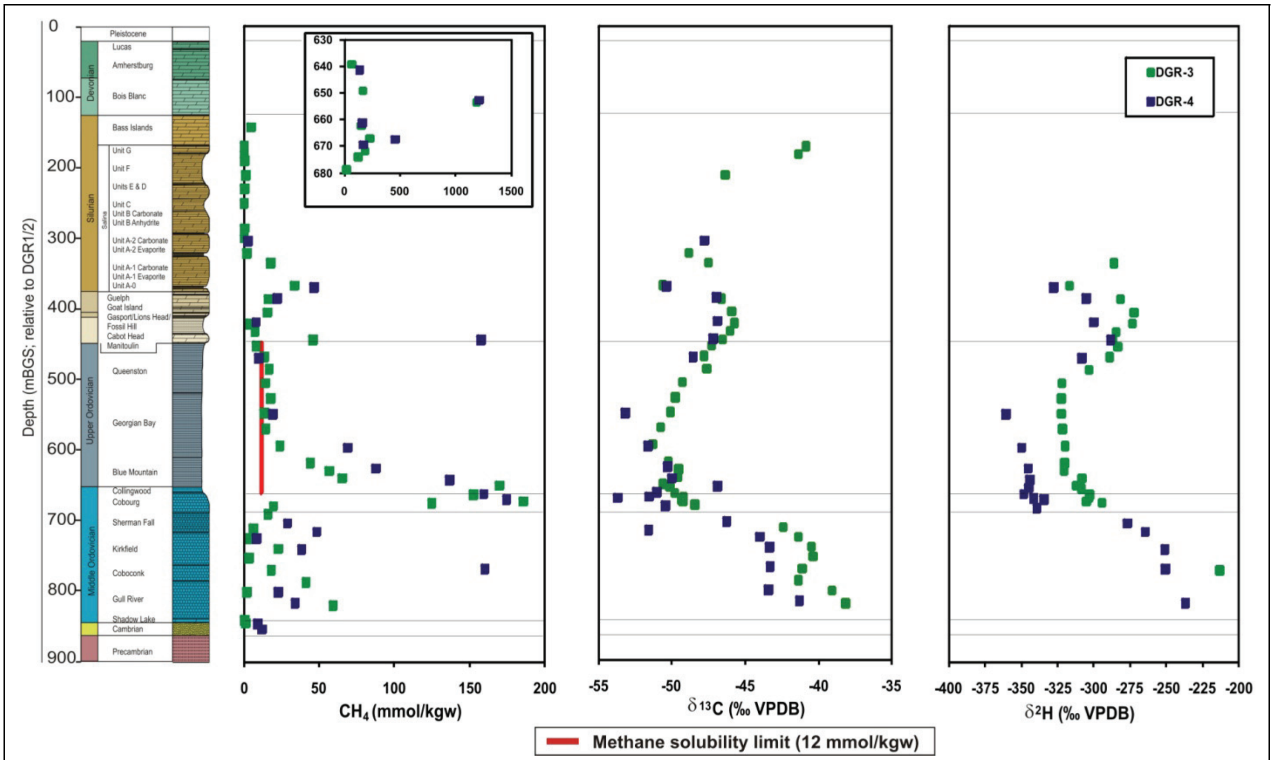
- The $\delta^{13}C$ and $\delta^{2}H$ data for CH_4 display a clear separation between the Upper Ordovician shales and the Middle Ordovician carbonates (Figure 4-39).

- The stable isotope data from CH_4 have been plotted on the variation diagram from Whiticar (WHITICAR99) and they define two fields: one field represents CH_4 of biogenic origin in the Upper Ordovician shales, and a second field represents CH_4 of thermogenic origin in the Middle Ordovician carbonates (Figure 4-41).

The generation of thermogenic gas requires temperatures in excess of ~70 °C (HUNT96), and such a condition has probably not prevailed since maximum burial in the Carboniferous (see Figure 4-7 and discussion in Section 4.1.1.2). It is therefore likely that the thermogenic gas is very old. The age of the biogenic CH_4 contained in the Ordovician rocks is unknown, and two possibilities are listed below.

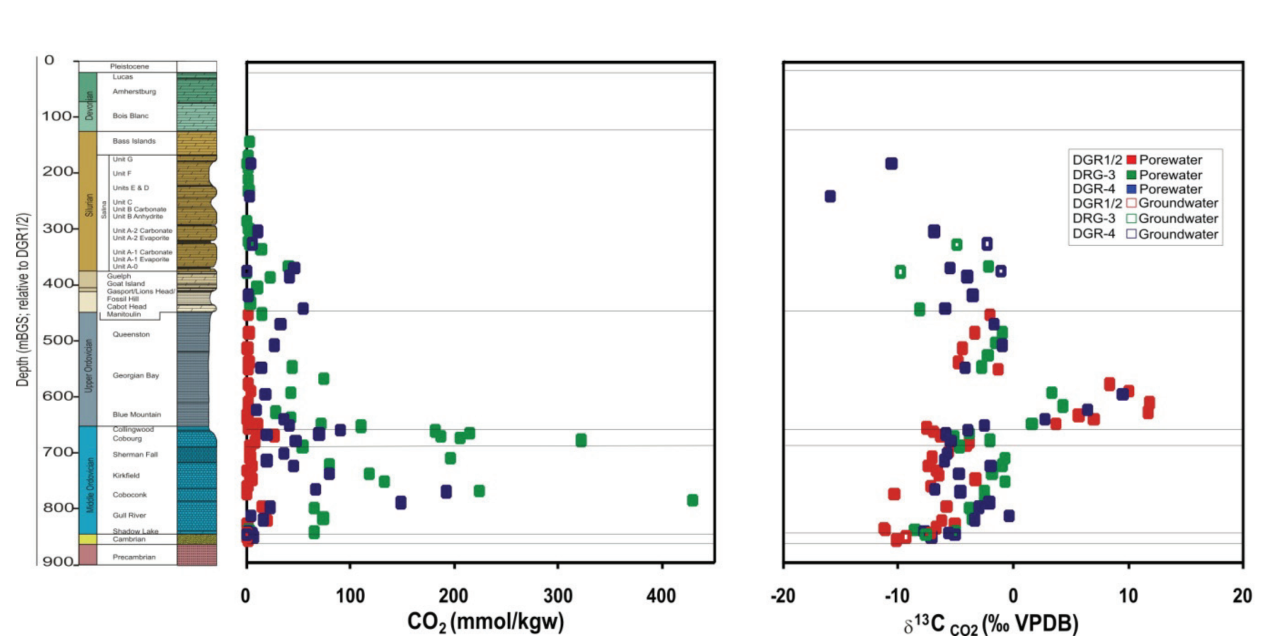
- If the biogenic gas is young, or perhaps even accumulating via methanogenesis at the present time, then there should be viable and active methanogens in the Blue Mountain shale. The presence of active methanogens is highly unlikely due to the high salinities and low water activities (0.6 to 0.7) measured in the Ordovician sediments. A preliminary microbiological investigation did not find evidence of viable bacteria within the Ordovician limestones (STROESS08), suggesting that microbes, if present within the sediments at depth, are most likely in a dormant state.

- The alternative interpretation is that the biogenic gas is relatively old and immobile. This is possible if the aqueous CH_4 concentrations are at saturation in the porewater and sections of the profiles with elevated CH_4 content can be explained either by the presence of a discrete gas phase, or by the partitioning of CH_4 into solid organic carbon or liquid hydrocarbons. The CH_4 concentrations exceed presumed solubility limits in the Collingwood Member, the Blue Mountain Formation shale, and, in most samples obtained from the Georgian Bay Formation shale and the lower portion of the Queenston Formation shale (Figure 4-39), suggesting that CH_4 may occur in a separate gas phase or in associated with organic carbon or liquid hydrocarbons in these zones.



Notes: Upper dataset (DGR-3) and lower dataset (DGR-4). Modified from Figure 4.67 of the DGSM (NWMO11k).

Figure 4-39: Concentration Distributions for CH_4 and $\delta^{13}C$ and $\delta^{2}H$ in CH_4



Note: Modified from Figure 4.69 of the DGSM (NWMO11k).

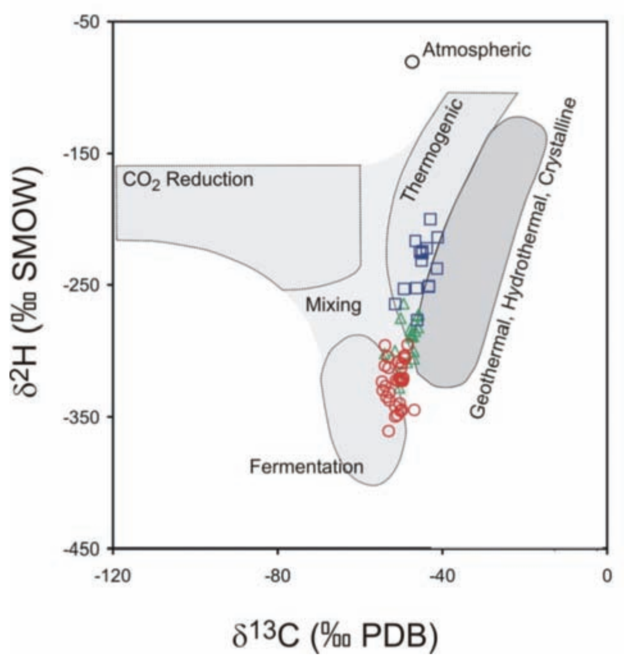
Figure 4-40: Concentration Distribution for CO_2 Versus Depth (Left), and Corresponding Distributions of $\delta^{13}C$ in CO_2 (Right)

In addition, there appears to be a lack of solute migration in response to the existence of isotopic gradients. There are at least two possible explanations for the apparent retardation of diffusive transport and the full discussion can be found in Section 4.4.3.1 of NWMO11c.

- Sorption and dissolution/exsolution reactions between CH_4 and solid organic carbon, or liquid hydrocarbons, respectively, cause a decrease in apparent diffusion coefficients.

- Infill or occlusion of porosity in the Cobourg Formation by precipitation of secondary minerals would also act to inhibit solute transport.

The observed separation of biogenic gas above, from thermogenic gas below, provides evidence that there has been little or no cross-formational mixing by advection while the gas has been resident in the system. It appears that neither the biogenic nor the thermogenic gas is mobile, at least in the vertical direction, and this immobility may reflect slow accumulation over a very long period of time. Given that high salinities and low water activities appear to inhibit microbial activity within these sediments, it may be that the biogenic gas is of Paleozoic age.

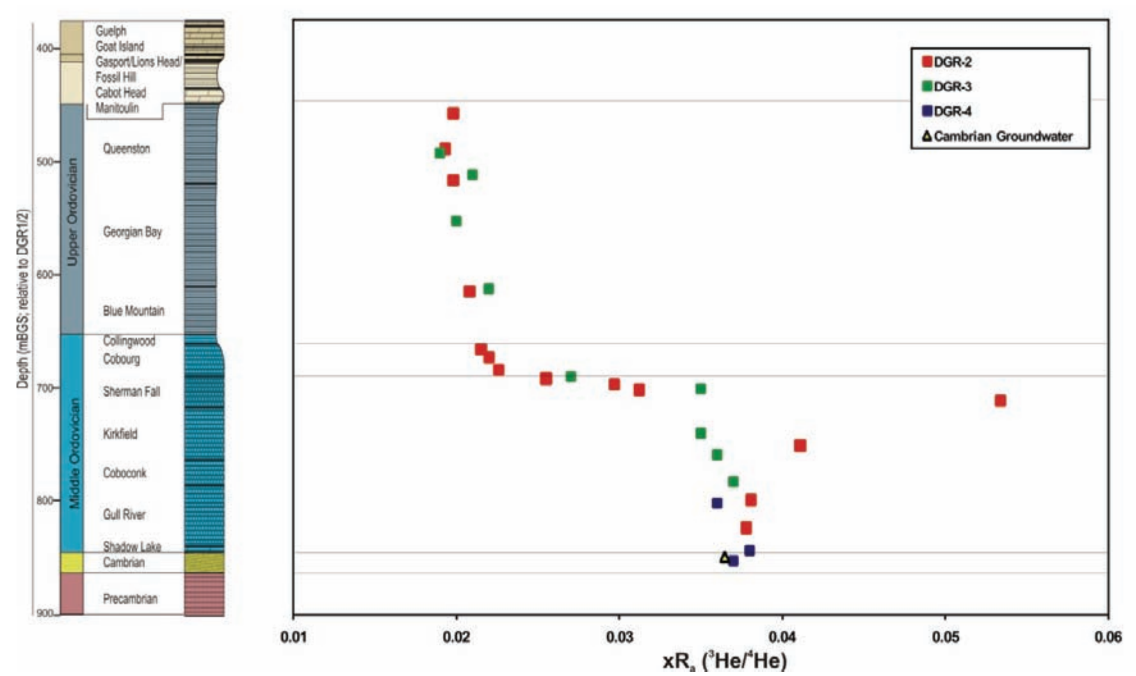


Notes: Green triangles represent data from the Queenston Formation and above, red circles from the Cobourg, Blue Mountain and Georgian Bay formations, and the blue squares from below the Cobourg Formation. Data is from the DGSM (NWMO11k) and the background plot is modified from Whiticar (WHITICAR99).

Figure 4-41: Discrimination Diagram Indicating Fields for CH_4 of Biogenic (CO_2 Reduction and Fermentation) and Thermogenic Origin

Helium

Profiles of $^3He/^4He$ for DGR-2, DGR-3 and DGR-4 are presented in Figure 4-42. The data are presented as the isotope ratio in the sample (R_s) normalized to the isotope ratio in air (R_a) such that $xR_a = R_s/R_a$. The data are remarkably consistent among the three drill cores, and they define two distinct regions of differing isotope ratio separated at the base of the Cobourg Formation, with xR_a of approximately 0.02 within and above the Cobourg Formation, and xR_a of approximately 0.035 below. Consistent with observations from the CH_4 data, the clear separation between regions of differing He isotope composition indicates that there has been very little cross-formational mixing of helium between the Middle Ordovician limestones and the Upper Ordovician shales, and suggests that there is a barrier to solute migration within the Cobourg Formation.



Note: Modified from Figure 4.75 of the DGSM (NWMO11k).

Figure 4-42: Vertical Profiles of Helium Isotopic Ratios ($^3He/^4He$) from DGR-2, DGR-3 and DGR-4

Redox Conditions in the Ordovician Shale and Carbonate

Redox conditions can be defined in terms of the principal redox couples that reflect the oxidation state at a given depth (e.g., Fe^{3+}/Fe^{2+}; SO_4^{2-}/S^{2-}; CO_2/CH_4). It is commonly possible to determine the dominant redox couple by analysis of dissolved gases, stable carbon isotope ratios, and the distribution of redox-sensitive minerals. Mineralogical and geochemical evidence (SCHANDL09, SKOWRON09) indicates that sulphide minerals (predominantly pyrite) and organic carbon are common throughout the stratigraphic sequence, particularly below the Silurian. The presence of these materials suggests that redox conditions range from sulphate reducing to methanogenic.

The presence of CH_4 suggests that the redox conditions are strongly reducing throughout most of the Ordovician. The redox conditions are in the range of iron- or sulphate reduction to methanogenesis, with E_h values estimated at -150 mV for the whole of the Ordovician sedimentary sequence (NWMO11k).

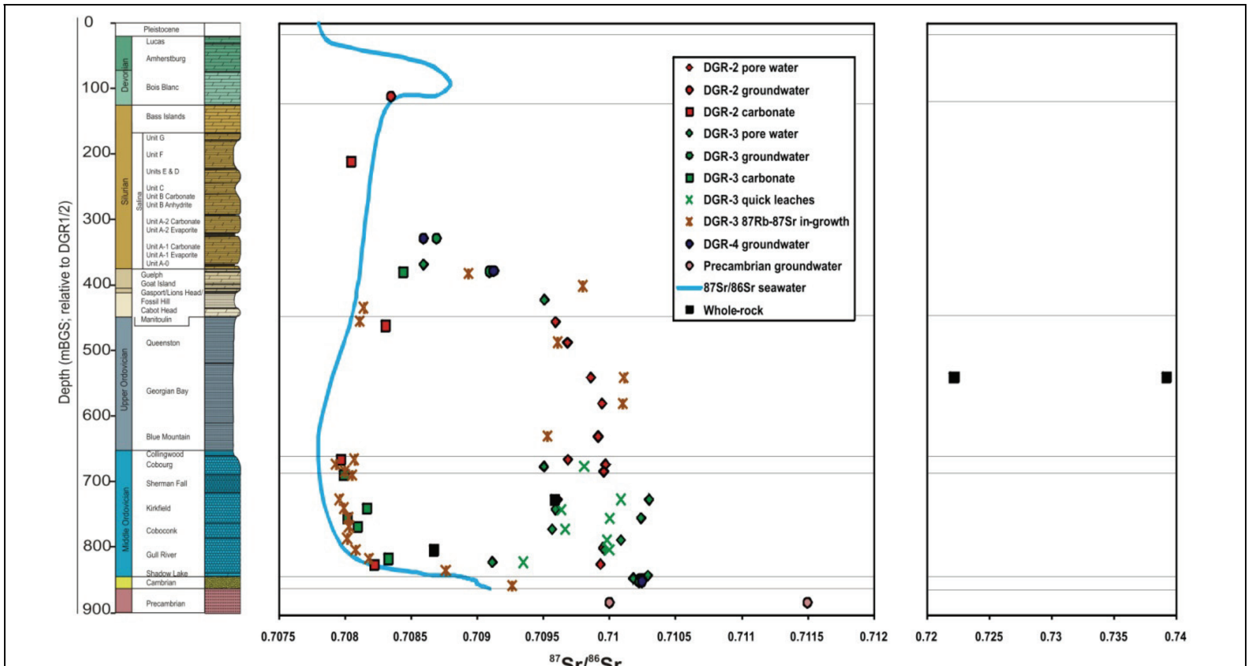
Strontium Isotopes

The $^{87}Sr/^{86}Sr$ ratios in the porewater and the host rocks were determined by Clark et al. (CLARK10a, CLARK10b). Consistent with the results for strontium (Sr) isotopic analysis of oilfield groundwater from the Michigan Basin reported by McNutt et al. (MCNUTT87), the $^{87}Sr/^{86}Sr$ ratios from Cambrian groundwaters and from the Ordovician and Silurian porewaters at the Bruce nuclear site are more radiogenic than the Paleozoic seawater curve (Figure 4-43). With the exception of the Ordovician shale units, the $^{87}Sr/^{86}Sr$ signatures of the porewater are more radiogenic than those of the host rocks. There are three possible explanations for the ^{87}Sr enrichment in the porewater. These include:

- Ingrowth of ^{87}Sr from ^{87}Rubidium (Rb) decay since the Ordovician;
- Leaching of ^{87}Sr from old shield-derived siliciclastic material in the shales and the argillaceous component of the limestones; and
- Transport of Sr upward from an ^{87}Sr-enriched brine source in the underlying Precambrian shield.

The observed ^{87}Sr enrichment in the Ordovician must have resulted from some combination of the three processes described above, but the respective contributions cannot be resolved quantitatively. In any case, the presence of radiogenic Sr throughout the Ordovician indicates extremely long time periods for water-rock interaction and/or diffusive transport of radiogenic Sr upward from the shield.

Above the Guelph Formation aquifer, the $^{87}Sr/^{86}Sr$ ratios for Silurian porewater and groundwater at the Bruce nuclear site approach the values of the enclosing host rock and the seawater curve. The convergence demonstrates the dominance of the Silurian sea water $^{87}Sr/^{86}Sr$ signature in the evaporite minerals (anhydrite) and non-argillaceous limestones of the Salina units. A significant decrease in Sr concentrations in the Upper Silurian and Devonian formations (Bois Blanc, A1 carbonate) is also observed (Figure 4-44), further demonstrating that the shallow groundwaters have been diluted, most likely due to the influx of glacial melt water and/or meteoric water in these relatively high permeability zones.



Notes: Seawater $^{87}Sr/^{86}Sr$ curve is from VEIZER05. Modified from Figure 4.65 of the DGSM (NWMO11k).

Figure 4-43: Depth Profiles for $^{87}Sr/^{86}Sr$ in Groundwater, Porewater and Host Rocks at DGR-2, DGR-3 and DGR-4

4.3.2.4 Solute Transport Mechnisms: Evidence for Diffusion

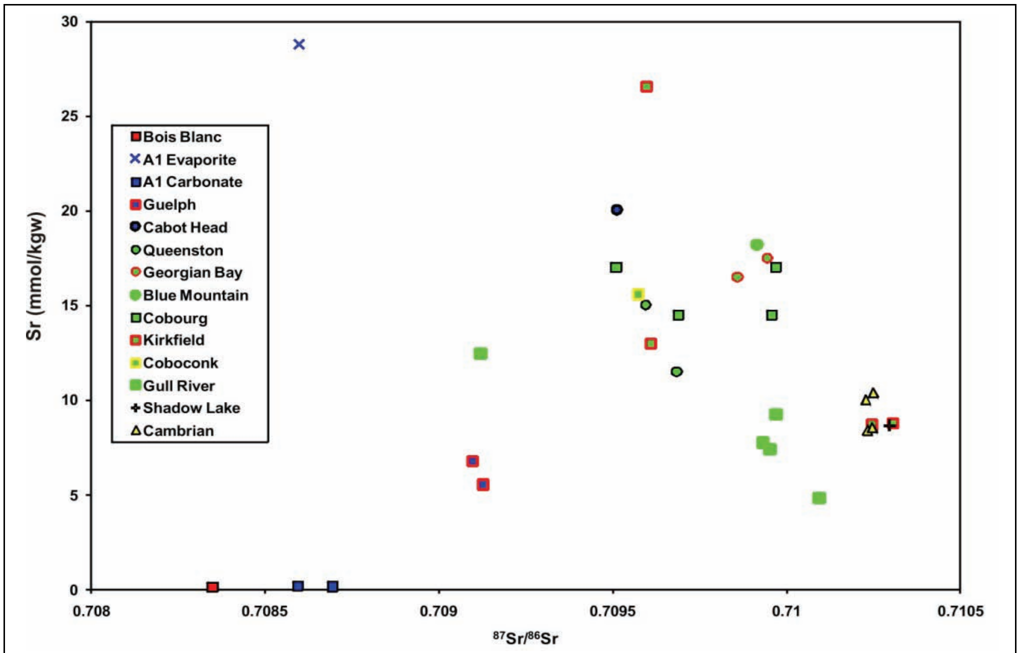
Laboratory-scale diffusion measurements were undertaken to determine effective diffusion coefficients (D_e) for rock samples from the Silurian and Ordovician sections of the stratigraphy (Figure 4-45). The D_e measurements were conducted with sodium iodide (NaI) and tritiated water (HTO) tracers, using radiography (NAI only) and through-diffusion (NaI and HTO) methods (AL10a, AL10b). The through-diffusion technique is well established and data acquired with this method have been published by numerous authors (e.g., VANLOON03, VANLOON07). The radiography technique (TIDWELL00) was modified for application to samples from the DGR project and was benchmarked against results from the through-diffusion method (CAVÉ09). The results are summarized below and are discussed in detail in Section 5.3.5 of NWMO11c.

- With the exception of just a few samples from the Upper Silurian, the D_e values measured from DGR drill cores are all less than 10^{-12} m^2/s (Figure 4-45).

- The highest values occur in the Upper Silurian Salina B, C, E and F units, with values greater than 10^{-11} m^2/s in the silty shale of the Salina B.

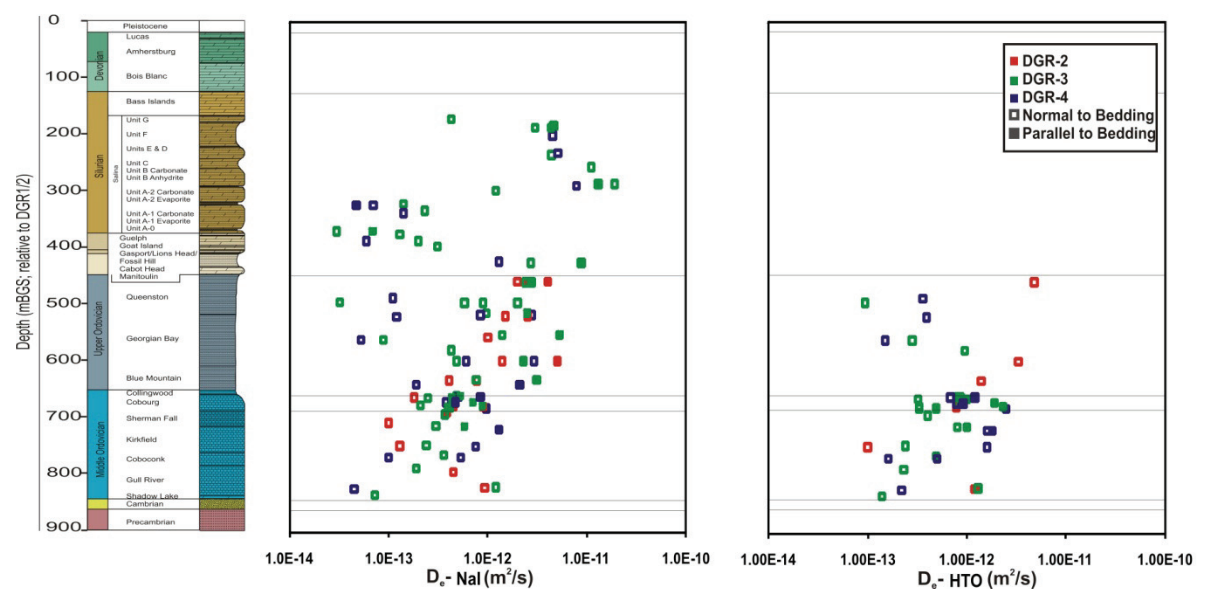
- The lowest values, on the order of 10^{-14} m^2/s, are obtained in the gypsum-anhydrite layers of the Salina A0-A2 units, in the carbonate "hardbeds" within the Georgian Bay Formation, and in several limestone samples from the Gull River Formation.

- The majority of the data are in the range $10^{-13} < D_e < 10^{-11}$ m^2/s, with Lower Silurian and Upper Ordovician shale samples representing the higher end of this range because of their relatively high porosity (7 to 9 %). The lower porosity of the Middle Ordovician limestones (< 2%) yields lower D_e values which cluster in the range $10^{-13} < D_e < 10^{-12}$ m^2/s, with only a few samples displaying values greater than 10^{-12} m^2/s.



Note: Modified from Figure 4.16 of the Geosynthesis (NWMO11c).

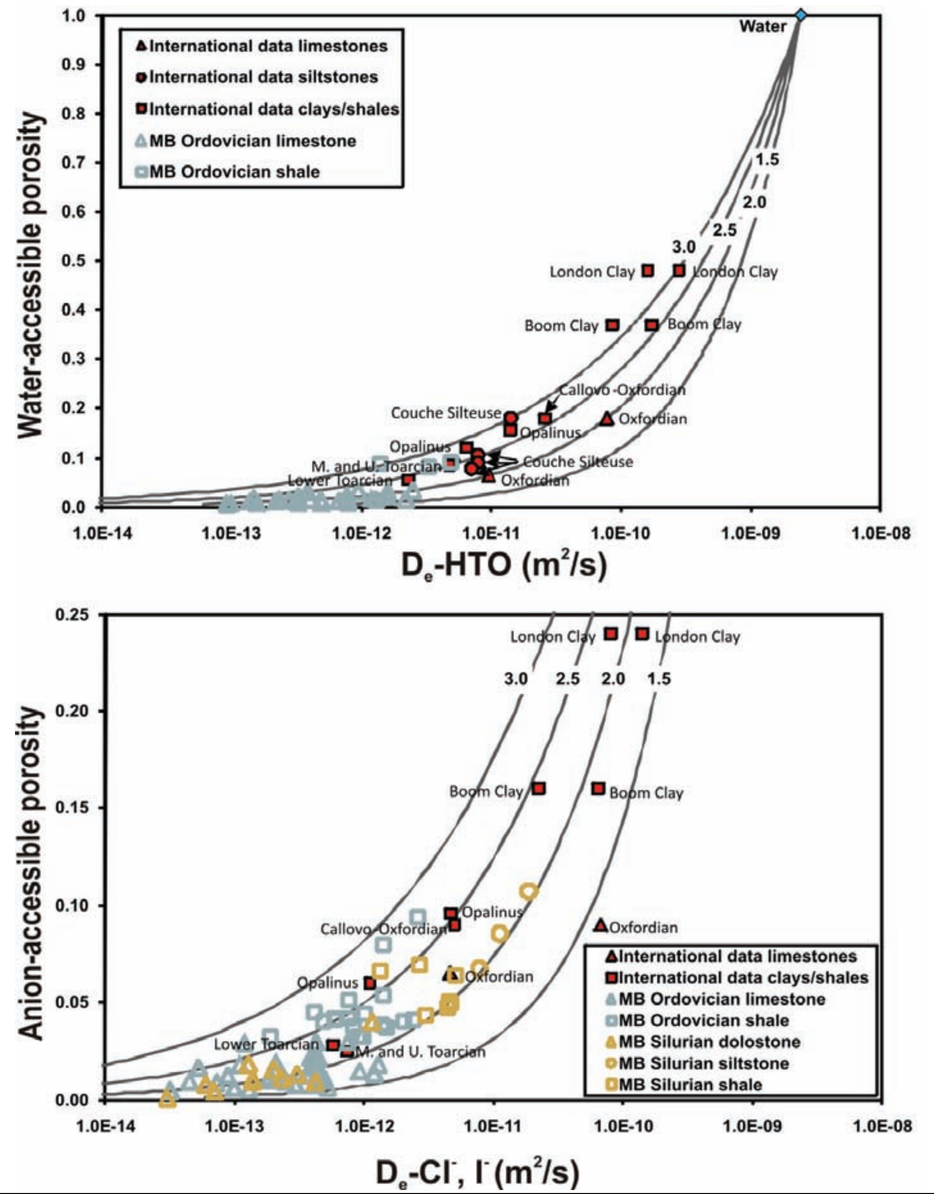
Figure 4-44: $^{87}Sr/^{86}Sr$ versus Sr Concentration for DGR Groundwaters and Porewaters



Notes: The D_e values were determined by radiography using NaI tracer and/or through-diffusion using NaI or HTO tracer. Modified from Figure 4.38 of the DGSM (NWMO11k).

Figure 4-45: Effective Diffusion Coefficients (D_e) Versus Depth

In order to obtain an international perspective, the porosity and diffusion coefficient data from the DGR cores are compared to corresponding data obtained from argillaceous rocks by researchers involved with radioactive waste programs in other countries (Figure 4-46). The Claytrac Project is sponsored by the NEA of the Organization for Economic Co-operation and Development (OECD) and the project included an effort to compile the results of diffusion studies conducted at nine different European sites (OECD09). In the data compilation, the D_e values for HTO tracer range from 2.3×10^{-12} to 2.8×10^{-10} m^2/s, and for anionic tracers from 5.7×10^{-13} to 1.6×10^{-10} m^2/s. The values collected from DGR samples are generally lower than the D_e values obtained from the European site characterization programs (Figure 4-46).



Notes: MB – Michigan Basin. Solid lines represent the exponential term, m, in the Archie's Law Relationship between diffusivity and porosity. Plot includes an international data set (OECD09) and DGR data from the DGSM (NWMO11k). See text for further discussion. Modified from Figure 6.6 of the DGSM (NWMO11k).

Figure 4-46: Comparison of D_e Values from DGR Drill Cores with Data from International Programs Involving Argillaceous Sedimentary Rocks

4.3.3 Illustrative Modelling of the Bruce Nuclear Site Geochemistry

In this section, a conceptual model is presented to describe the hydrogeochemical evolution at the Bruce nuclear site. This model is consistent with the regional-scale information presented in Section 4.3.1 and provides insight into the natural tracer profiles for the site described above in Section 4.3.2, including:

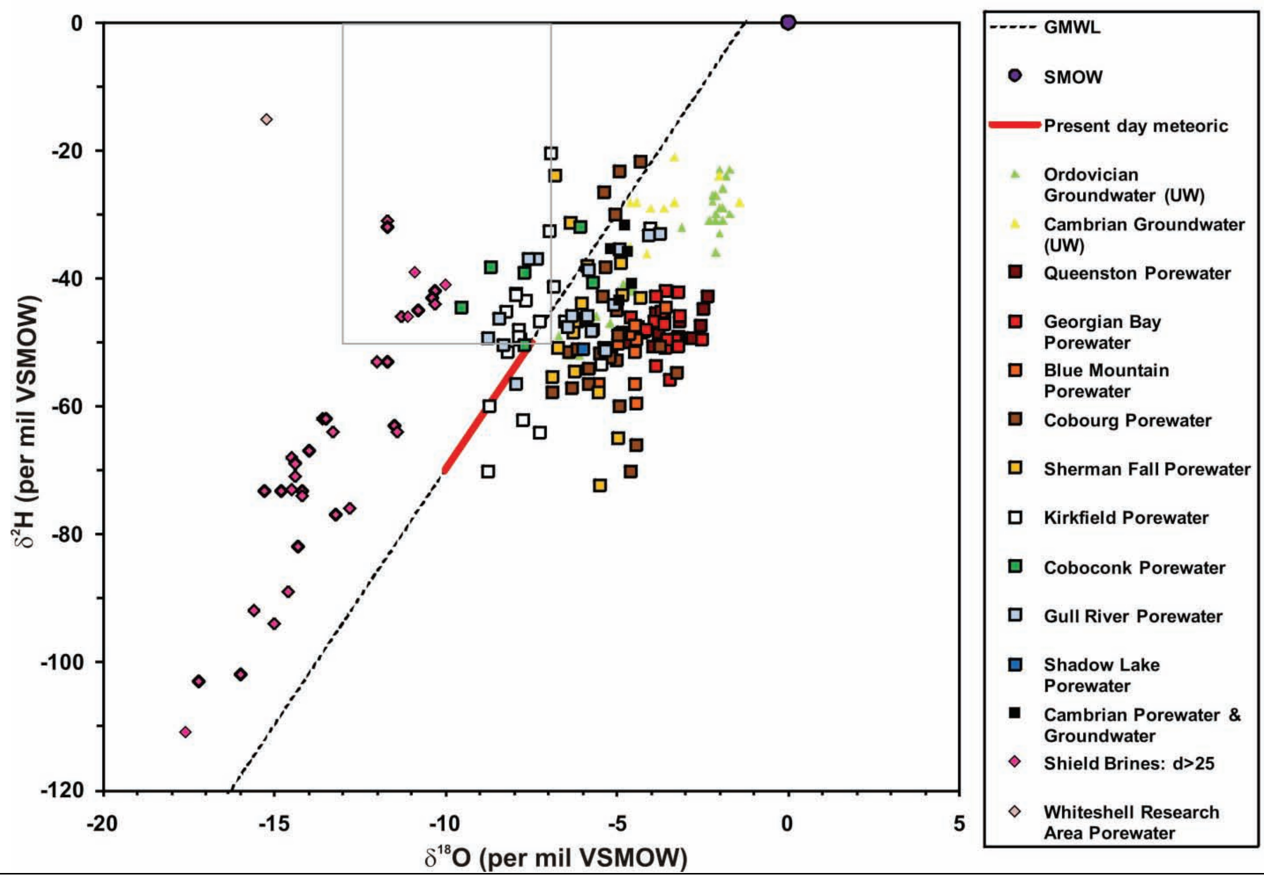
- The large decrease in concentration for all tracers from the top of the Guelph Formation upward through the Silurian; and
- A less pronounced, but persistent, trend toward depleted $\delta^{18}O$ values, decreased Cl and Br concentrations, and enriched $\delta^{2}H$ values occurs in the Middle Ordovician limestone.

The conceptual model has been adopted because of its ability to describe the observed geochemical profile trends for almost all of the data collected at the Bruce nuclear site. The conceptual model is tested with numerical modelling, described in Section 4.3.3.2. One feature of the natural tracer profiles that the model cannot simulate is the current Cambrian fluid chemistry, suggesting that its fluid evolution may be more complex. Possible explanations for the current chemical profile of the Cambrian are presented in Section 4.3.4.

In order to model the fluid evolution, the composition of potential end members for mixing must be established. Because the composition of groundwater in the Precambrian shield below the Michigan Basin and below the Bruce nuclear site is not known, a potential end member composition for the Precambrian was assumed. ^{2}H-enrichment, coupled with ^{18}O-depletion relative to the GMWL, are consistent characteristics of old groundwater in a shield setting.

Various authors have proposed isotopic compositions for a hypothetical shield groundwater end member based on mixing trends observed at various locations across Canada where the shield is shallow or exposed (Sudbury, Yellowknife and Manitoba; FRITZ82, FRAPE84, FRAPE87, PEARSON87, BOTTOMLEY99, DOUGLAS00, BOTTOMLEY03, BOTTOMLEY04, BOTTOMLEY05, GREENE08). The typical compositions range from $\delta^{2}H$ = -50 to 20‰ and $\delta^{18}O$ = -13 to -7‰ (FRITZ82, FRAPE84, PEARSON87, BOTTOMLEY99). Given that the porewater and groundwater in the shield underlying the Michigan Basin is likely to be at least as old as, and perhaps several hundred million years older than, shield groundwater studied in exposed regions of the Canadian Shield, it is expected that the isotopic composition of shield brines underlying the basin would be characterized by strong ^{2}H enrichment and depleted $\delta^{18}O$ values and this assumption is the basis for the Precambrian fluid composition utilized in the hydrogeochemical modelling. Further discussion on this point can be found in Section 4.4.6 and in Table 4.3 of the Geosynthesis (NWMO11c).

The proposed shield-brine end member responsible for the observed mixing trends (shown in Figure 4-47 along with data from the UW database, the Bruce nuclear site, and various shield locations across Canada) plots to the left of the GMWL, and the ^2H enrichment that is required to cause this shift occurs as a result of water-rock interactions over long periods of geologic time.



Notes: The grey box indicates proposed range in composition for a shield end member. Also shows groundwater brine samples from Ordovician carbonates and Cambrian sandstone from the UW database. For shield brines,'d' represents deuterium excess, where d = $\delta^2H - 8 \times \delta^{18}O$. Modified from Figure 4.19 of the Geosynthesis (NWMO11c).

Figure 4-47: $\delta^{18}O$ versus δ^2H for Ordovician and Cambrian Porewater from DGR-2, DGR-3 and DGR-4

4.3.3.1 Conceptual Model

<u>The Ordovician Tracer Profiles: Diffusion from Above</u>

Diffusion downward from a Silurian source could provide an explanation for the salinity profile because the original porewater in the Ordovician would be expected to be close to normal seawater, and the high-salinity porewater in the overlying Silurian evaporites would create a strong downward gradient for diffusive transport. In support of this hypothesis, numerical modelling of diffusive transport downward from the Silurian suggests that the observed natural tracer profiles in the Ordovician could be generated over a period of approximately 300 Ma (see discussion in Section 4.3.3.2 below).

The presence of halite in the Middle Ordovician carbonates (HERWEGH08) can be explained by asserting that localized halite occurrences formed by concentration mechanisms, such as hydration reactions (DREVER79) or hyperfiltration (BREDEHOEFT63, KHARAKA73).

The "diffusion from above" conceptual model is summarized below.

- Deposition of the Cambro-Ordovician sequence under normal marine conditions, followed by deposition of the Silurian and Devonian, created a condition with high-TDS porewater overlying porewater of normal marine composition. This established a natural concentration gradient that promoted a downward mass flux of salts by diffusion.

- A very long period (~300 Ma) of diffusive transport followed, during which the high-salinity profile propagated downward into the Upper and Middle Ordovician by diffusion. During the same period, water-rock reactions in the underlying shield and Cambrian sediments caused the deep groundwater isotopic characteristics to evolve toward a shield signature with enriched $\delta^{2}H$ and depleted $\delta^{18}O$ values.

The very long period of diffusion-dominated transport and water-rock reaction required to justify the interpretations presented in the diffusion from above conceptual model is supported by multiple lines of hydrogeochemical evidence.

- The enriched $\delta^{18}O$ signatures of most of the Ordovician fluids relative to the GMWL are indicative of long time periods for water-rock interaction (i.e., long residence times).

- Separation between biogenic CH_4 in the Upper Ordovician shales and thermogenic CH_4 in the Middle Ordovician carbonates (Section 4.3.2.3), and between He with different $^{3}He/^{4}He$ ratios in the Upper Ordovician shales and the Middle Ordovician carbonates (Section 4.3.2.3), suggests that advective mixing has not occurred and diffusive transport is extremely slow.

- The presence of radiogenic Sr in the Upper Ordovician shale and the Middle Ordovician carbonate porewaters suggests that the radiogenic Sr must have been derived either from in-growth from ^{87}Rb decay, leaching from the siliciclastic sediments, or diffusion upward from a ^{87}Sr-enriched end member in the shield (Section 4.3.2.3). All of these possibilities require extremely long time periods.

Devonian and Silurian Tracer Profiles: Glacial Melt Water Infiltration

In addition to the diffusion from above model, a glacial melt water infiltration scenario is also proposed to explain the natural tracer profiles observed for the Devonian and Silurian porewaters and groundwaters at the Bruce nuclear site. The observed decrease in salinity and the depleted $\delta^{18}O$ values that are apparent from the top of the Guelph Formation to ground surface suggest that a combination of glacial melt water and recent meteoric water have contributed to the shallow fluid chemistries. Based on the geologic history of the site, these signatures are best explained by episodic infiltration of meteoric and/or glacial melt water during the Pleistocene.

4.3.3.2 Numerical Modelling Results

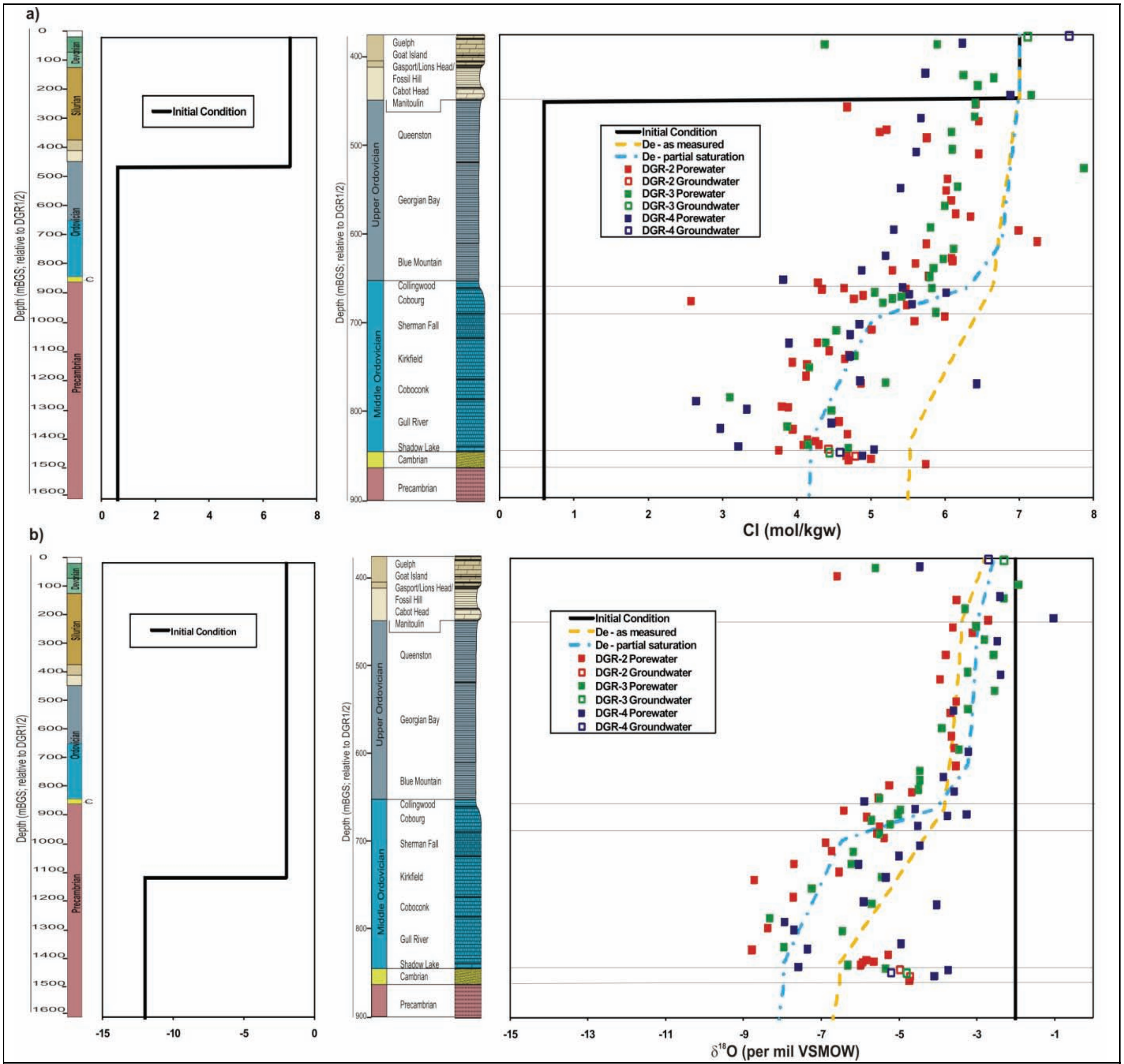
Diffusion From Above Conceptual Model – Tracer Profiles in the Ordovician

The numerical model presented in Figure 4-48 is not intended to be unique, but rather is intended to provide a test, through reasoned illustrative modelling, of various elements of the conceptual model described. Details on the model justification and the modelling parameters can be found in Section 4.5.2 of NWMO11c. The key results that can be drawn from the modelling are indicated below.

- The principal controls on the shape of the simulated profiles are the boundary conditions, the contrast in D_e between the Upper and Middle Ordovician, and the effect of partial saturation in lowering the D_e values at the boundary between the Upper and Middle Ordovician. See Section 4.5.2.3 of the Geosynthesis (NWMO11c) for further discussion.

- The diffusion from above conceptual model is able to explain the observed natural tracer profiles of the Ordovician fluids. The numerical simulations are able to reproduce the measured Cl and $\delta^{18}O$ profiles, and the data are particularly well matched under the partial saturation case, indicating that partially saturated conditions (or conditions that result in a decrease in D_e; e.g., secondary mineral precipitation) may exist within the Ordovician shales and carbonates.

- The profiles are best matched for both Cl and $\delta^{18}O$ under partially saturated conditions for a time period of 300 Ma, assuming diffusive transport only. The simulated profiles are consistent with the site-specific data, supporting the hypothesis that solute transport in the Ordovician sediments is diffusion dominated.

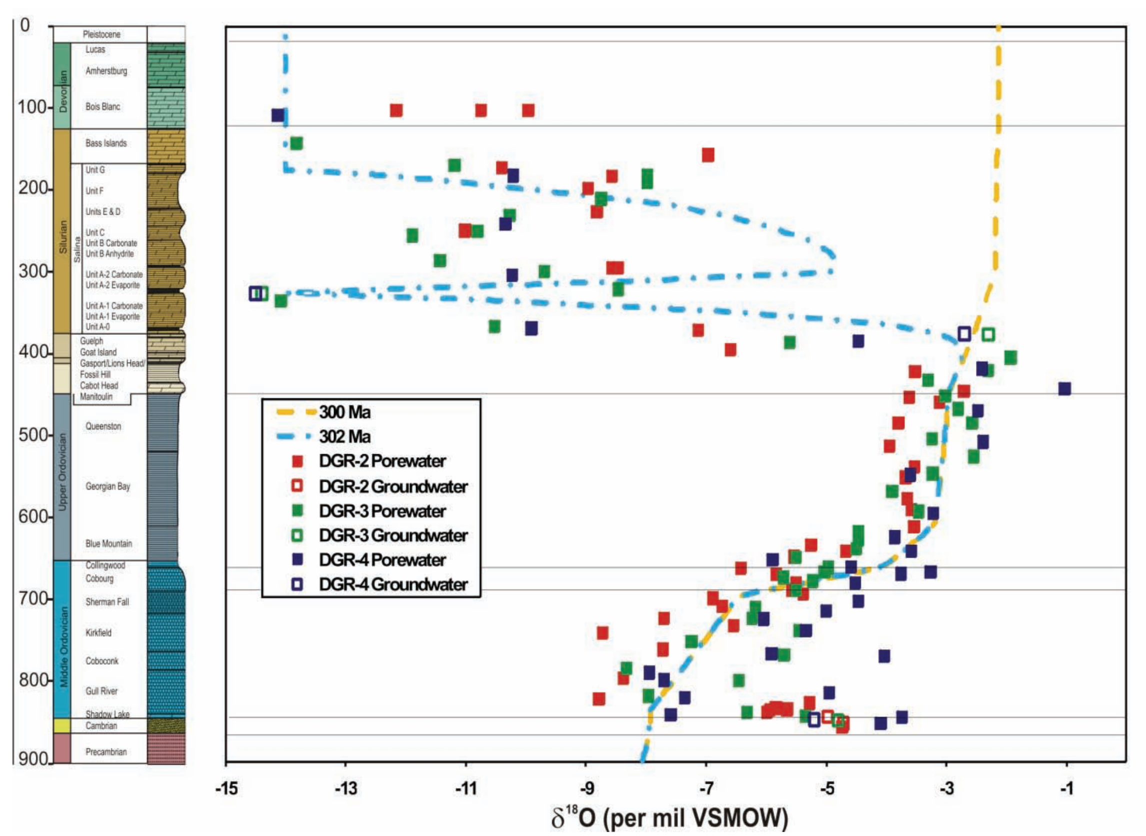


Notes: (a) Salinity (Cl) tracer profile develops as a result of salt diffusion downward from the Silurian. (b) $\delta^{18}O$ profile results from diffusive mixing with shield brine at the base of the profile. X-axis for left side plot in (a) and (b) is same as right side ($\delta^{18}O$ per mil VSMOW). Modified from Figure 4.21 of the Geosynthesis (NWMO11c).

Figure 4-48: Results of the "Diffusion from Above" Modelling Scenario

Glacial Infiltration – Tracer Profiles in the Silurian and Devonian

There is considerable uncertainty in attempting to translate the conceptual model into a numerical model to describe advective and diffusive mixing between basin water and infiltrating glacial and/or meteoric water. The most important issues include: 1) when did these units “open up” to glacial and meteoric water infiltration; 2) did they open up sequentially, or all at once; and 3) what was the volume and duration of glacial melt water infiltration? The model results are presented in Figure 4-49.



Note: Modified from Figure 4.22 of the Geosynthesis (NWMO11c).

Figure 4-49: Results of $\delta^{18}O$ Diffusion Simulation (dashed lines) Compared to Measured Porewater $\delta^{18}O$ Data

The model results in Figure 4-49 describe a general depletion in $\delta^{18}O$ values upward through the Silurian and Devonian that is generally consistent with the site data. Therefore, in support of the conceptual model, it is suggested that there is a glacial melt water component in many of the shallow system (Devonian and Silurian) fluids and in the Salina A1 Unit carbonate aquifer. A relatively poor fit in the upper units of the Salina Formation, however, suggests that the hydrogeochemical history of these rocks is more complex than has been represented in the model.

4.3.4 Cambrian Fluid Chemistry

The Cambrian chemistry displays a distinct rebound in the natural tracer profiles relative to the overlying Ordovician carbonates. The rebound in the profiles, as shown in Figure 4-37 and Figure 4-36, is abrupt compared to the gradual decline in concentrations and isotope ratios observed with depth through the Ordovician carbonates. The composition of the Cambrian groundwater below the Bruce nuclear site is very similar to Cambrian groundwater samples from elsewhere in southern Ontario (refer to Section 4.5.4 in NWMO11c). The similarity between the present-day brine in the Cambrian below the Bruce nuclear site and Cambrian and deep Ordovician brines elsewhere in the Appalachian and Michigan basins, respectively, suggests that the Cambrian fluid underlying the Bruce nuclear site originated at depth within the Michigan Basin.

The hydraulic conductivity of the Cambrian aquifer is approximately six orders of magnitude higher than that of the overlying Middle Ordovician limestones as discussed in Section 4.4.1 below and in Section 4.9 of the DGSM (NWMO11k). The groundwater in the Cambrian sandstone would be more susceptible than porewater in the Ordovician carbonates to advection-driven changes in composition through geologic time.

Under the influence of diffusion, it is expected that such an abrupt concentration gradient would be attenuated over time. Conventional hydrogeologic rationale would suggest that this feature of the profiles could represent a geologically recent movement of groundwater in the permeable Cambrian formation, thereby disrupting the mixing relationship that had developed previously between basin and shield end members. Assuming that the Cambrian fluid composition represents a recent change, the mechanism responsible for the re-supply of basin water is not known. Based on the evolutionary history of the Michigan Basin, the possible driver(s) for fluid migration from basin centre in the recent geologic past are rather limited. These drivers include 1) fluid migration in response to the anomalous pressures deep in the Michigan Basin (BAHR94) and/or 2) fluid migration in response to differential uplift of the basin due to repeated isostatic adjustments related to glaciation and deglaciation. Irrespective of the mechanism(s) responsible for the current Cambrian fluid chemistry beneath the Bruce nuclear site, the fundamental hypothesis that solute migration with the Ordovician sediments is diffusion dominated is well supported by the hydrogeochemical data presented above and the hydrogeological modelling data which will be discussed in Section 4.4.

4.3.5 Hydrogeochemistry Summary

The following points may be made in support of the hydrogeochemical suitability of the Bruce nuclear site for the proposed DGR.

- The current understanding regarding the origin of brines from the Michigan Basin indicates that they were formed by evaporation of sea water and subsequently modified by dilution, halite dissolution, and water-rock interaction processes. The regional data (Cl-Br, ^{18}O-^{2}H) and the data from the Bruce nuclear site are very similar, indicating that the brines at both the regional scale and the site scale are of similar origin and evolution.

- The widespread occurrence of ancient brines in the basin demonstrates that, under most conditions prevalent since the Paleozoic, it has not been possible for hydraulic heads generated in freshwater aquifers to drive infiltration events capable of displacing the brines. Glacial melt water infiltration has been identified to maximum depths of 200-300 mBGS along the northern margins of the Michigan Basin. Consistent with regional observations, glacial melt water infiltration is identified to a maximum depth of 328.5 mBGS at the Bruce nuclear site within the Salina A1 Unit carbonate.

- At the Bruce nuclear site, concentrated brines occur at all depths below the top of the Silurian Guelph Formation.

- ^{18}O enrichment with respect to the GMWL in the majority of the Ordovician porewaters suggests long periods of water rock interaction (i.e., long residence times in the sedimentary system).

- Separation between biogenic CH_4 in the Upper Ordovician shales and thermogenic CH_4 in the Middle Ordovician carbonates, as well as the separation between He with different $^{3}He/^{4}He$ ratios in the Upper Ordovician shales and the Middle Ordovician carbonates, suggests that diffusion is extremely slow and that there is a barrier to vertical solute migration within the Cobourg Formation.

- Radiogenic $^{87}Sr/^{86}Sr$ ratios in the Middle and Upper Ordovician porewater are interpreted to result from a combination of water-rock interaction, in-situ ^{87}Rb decay, and diffusion of ^{87}Sr upward from an enriched end member in the shield. All of these mechanisms indicate a very long residence time, on the order of tens to hundreds of millions of years.

- The redox conditions in the Ordovician and Cambrian formations are strongly reducing, in the range of iron- and/or sulphate reduction and methanogenesis.

- Illustrative modelling suggests that the time frames required for the development of the salinity and $\delta^{18}O$ profiles within the Ordovician sediments are on the order of 300 Ma; the results are consistent with the assertion that solute transport in the Ordovician is diffusion dominated.

4.4 Hydrogeology

The objective of hydrogeological modelling is to assist in developing the safety case for the proposed DGR at the Bruce nuclear site. This assistance is provided by characterizing and analyzing the groundwater system in the deep geologic formations by creating robust numerical groundwater models (e.g., NWMO11p). In order to develop an understanding of groundwater migration and mass transport in the deep geological units, it is especially pertinent to ensure that the basis for the numerical models is developed from sound geologic interpretations and conceptual models (i.e., NWMO11c, NWMO11k). This will contribute to a more accurate distribution of unit properties such as permeability for a given numerical model and an appropriate realization of the domain geometry. The distribution of permeability is of importance due to the requirements of sufficient thickness, lateral continuity, and predictability of the geologic units contributing to the performance of the proposed repository.

The analyses of the modelling study were designed to gain insight on regional-scale and site-scale groundwater system hydrodynamics and evolution relevant to understanding groundwater pathways and solute migration from the location of the proposed DGR in the Cobourg Formation (Figure 4-9). A primary focus of the numerical modelling study is the investigation of the hypothesis, as discussed in Chapter 3, that solute transport in the Ordovician sediments is diffusion dominant.

This section summarizes the hydrogeological modelling performed for the DGR project at the Bruce nuclear site. It is a summary of the work that is described in detail in the Hydrogeologic Modelling report (NWMO11p).

4.4.1 Conceptual Model

From a hydrogeological viewpoint, the Michigan Basin can be conceptualized as a closed system, closed in the sense that groundwater flows neither in nor out from outside the basin. Recharge occurs where formations crop out (or subcrop) and discharge occurs into lakes and streams at topographical low points. Thus, gravitational driving forces are strongly controlled by the topographic relief of the basin.

The salinity (or density) distribution within the stratigraphic column exerts a strong influence on flow. All concentrations increase significantly below the shallow groundwater system. Without significant driving forces, dense brines at depth cannot be displaced by fresh waters entering the system at the surface, so the deep brines in the Michigan Basin are likely to be effectively stagnant (i.e., mass transport is diffusion dominant), as discussed in Section 4.3.2.4.

The regional-scale modelling integrated aspects of the Geosynthesis studies in one framework through the development and analysis of a regional and site-scale geosphere conceptual model. The conceptual model for the DGR site was defined by both the field and laboratory investigations of the site characterization study (NWMO11k) and by the work of the other studies of the Geosynthesis program (NWMO11c). The Regional Geology report (NWMO11m) together with the data from the DGR boreholes defines the geologic framework of the conceptual model. Hydraulic parameters for the model hydrostratigraphic units were defined using data from the DGR site boreholes and from lab analyses of cores. Borehole data included hydraulic conductivities from straddle-packer hydraulic tests and pressure measurements from the Westbay MP38 and MP55 multi-level groundwater monitoring systems.

The regional-scale and site-scale pore water chemistry was defined by both the Geosynthesis (NWMO11c) and the Regional Hydrogeochemistry report (NWMO11q), as well as data from the boreholes of the site characterization program. Rock cores and opportunistic water samples were used to define the spatial distribution of the TDS concentration and fluid density. Core analyses yielded estimates of porosity, Elastic modulus, Poisson's ratio, water saturations and gas saturations. Layer dependent specific storage coefficients and one-dimensional loading efficiencies were calculated using appropriate field and laboratory data. The Long-Term Climate Change study defined the glacial loading and the evolution of the formation properties for paleohydrogeologic analyses (NWMO11r). The numerical model of the Bruce nuclear site requires the development of constitutive models that relate the fluid density to the fluid TDS concentration. The linking of the field program to the development of the parameters of the numerical models adds to the confidence and robustness of conclusions developed from the modelling.

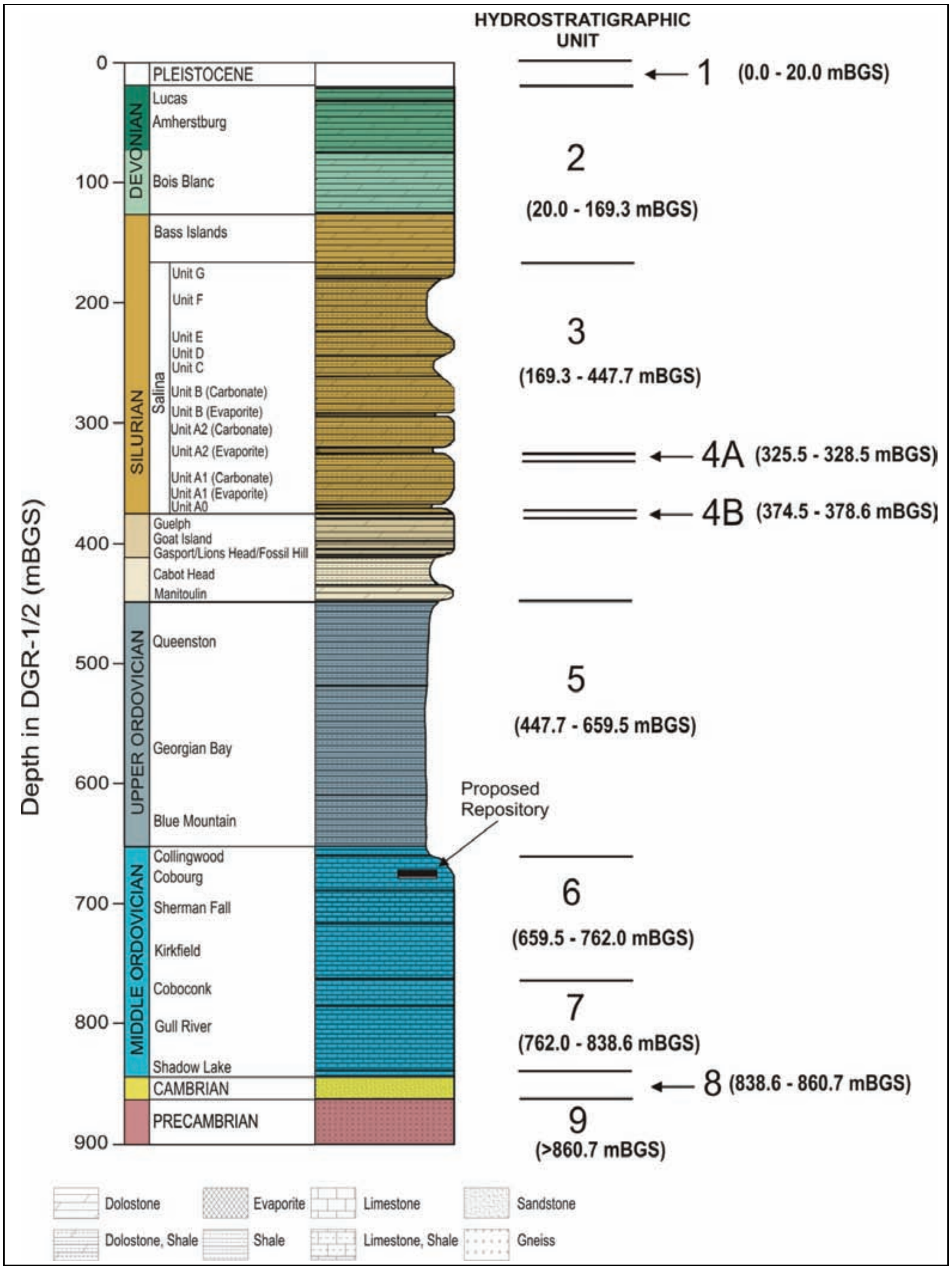
The stratigraphic units observed at the Bruce nuclear site are grouped into three groundwater regimes associated with different high-permeability units that behave independently of one another because they are separated by low-permeability strata (NWMO11k). These three regimes are described below.

- Shallow Regime: The shallow hydrogeological regime includes surficial Pleistocene deposits, Devonian strata, and the Silurian Bass Islands Formation. It extends to the top of the Salina G Unit which is encountered at a reference depth of 169.3 mBGS in DGR-1/2 (Figure 4-50). Groundwater within the permeable bedrock regime flows from recharge areas toward Lake Huron, where it discharges. Groundwaters and porewaters are transitional from fresh Ca:Mg-HCO_3 water (TDS ~0.5 g/L) near the top of the bedrock to brackish Ca-SO_4 water (TDS ~5.0 g/L) at the bottom of the shallow regime (Figure 4-34). Representative horizontal hydraulic conductivities (K_H) range from $8x10^{-8}$ to $1x10^{-4}$ m/s (Table 4-4).

Solute migration within this permeable shallow groundwater regime is driven principally by advection.

- Intermediate Regime: The intermediate hydrogeological regime extends from the top of the Salina G Unit to the top of the Queenston, and occurs between reference depths of 169.3 to 447.6 mBGS in DGR-1/2 (Figure 4-50). This is a predominantly low-permeability regime (K_H = 5x10^{-14} to 3x10^{-10} m/s) with groundwater flow likely restricted to the two permeable aquifer zones (K_H = 5x10^{-9} to 2x10^{-8} m/s) present at DGR-1 reference depths of 325.5 to 328.5 mBGS in the top of the Salina A1 Unit and at 374.5 to 378.6 mBGS in the Guelph Formation. These aquifers appear to be recharged east of the Bruce nuclear site where they outcrop (or subcrop) along the Niagara Escarpment, and discharge into Lake Huron in different locations tens of kilometres from the Bruce nuclear site. Groundwaters and porewaters in this intermediate regime are transitional from saline Ca-SO_4 water (TDS ~10 g/L) near the top to Na-Cl brine (TDS ~370 g/L) in the Guelph Formation (Figure 4-34).
- Deep Regime: The deep hydrogeological regime extends from the top of the Queenston to the top of the Precambrian. It occurs at reference depths of 447.6 to 860.7 mBGS in DGR-2 (Figure 4-50). This deep regime consists of the Upper Ordovician shales, the Trenton and Black River group limestones, and the Cambrian sandstone. The rocks of the Upper Ordovician and Trenton Group are of exceptionally low horizontal hydraulic conductivity (K_H = 4x10^{-15} to 1x10^{-13} m/s), and are significantly underpressured. Porewaters in these units are Na-Cl brine with TDS of 220 to 300 g/L that decrease in concentration with depth (Figure 4-34). These hydrogeological properties indicate a regime with no advection of brine, and a regime in which gas flow would also be diffusion controlled. The deeper Black River Group and Cambrian are overpressured and exhibit increased horizontal permeability relative to the overlying units. The formation horizontal hydraulic conductivities decrease upwards from the Cambrian sandstone (K_H =3x10^{-6} m/s) through the Shadow Lake Formation (K_H = 1x10^{-9} m/s) to the Gull River and Coboconk (K_H = 2x10^{-12} and 2x10^{-11} m/s). Groundwaters and porewaters in this group are Na:Ca-Cl to Na-Cl brine with TDS of about 200 to 235 g/L (Figure 4-34).

A further subdivision of these three bedrock groundwater regimes beneath the Bruce nuclear site was proposed in the DGSM (NWMO11k). As depicted in Figure 4-50, nine hydrostratigraphic units, including the Precambrian were identified. A detailed description and justification for this subdivision is provided in the DGSM (NWMO11k).



Note: From Figure 4.106 of the DGSM (NWMO11k).

Figure 4-50: Reference Stratigraphic Column Showing Hydrostratigraphic Units at the Bruce Nuclear Site

Table 4-4: Base-Case Hydrogeological Parameter Values for Regional-Scale and Site-Scale Modelling

Period	Formation	K_H [m/s]	K_V [m/s]	K_H:K_V	θ	ρ [kg/m^3]	TDS [g/L]	S_s [m^{-1}]	ζ	τ
Quaternary	Drift	1.00E-08	5.00E-09	2:1	0.2	1000	0	9.90E-05	0.99	4.00E-01
Devonian	Kettle Point	3.00E-09	3.00E-10	10:1	0.1	1006	9	1.50E-06	0.8	1.20E-01
	Hamilton Group	2.20E-11	2.20E-12	10:1	0.1	1008	12	1.50E-06	0.8	1.20E-01
	Dundee	8.40E-08	8.40E-09	10:1	0.1	1005	8	1.50E-06	0.8	1.20E-01
	Detroit River Group	5.90E-07	2.00E-08	30:1	0.077	1001	1.4	1.40E-06	0.84	9.40E-02
	Bois Blanc	1.00E-07	1.00E-08	10:1	0.077	1002	3.2	1.40E-06	0.84	9.40E-02
Silurian	Bass Islands	5.00E-05	1.70E-06	30:1	0.056	1004	6	2.00E-06	0.92	2.80E-01
	Unit G	1.00E-11	1.00E-12	10:1	0.172	1010	14.8	1.10E-06	0.55	3.00E-03
	Unit F	5.00E-14	5.00E-15	10:1	0.1	1040	59.6	9.50E-07	0.68	4.90E-02
	Unit F Salt	5.00E-14	5.00E-15	10:1	0.1	1040	59.6	9.50E-07	0.68	4.90E-02
	Unit E	2.00E-13	2.00E-14	10:1	0.1	1083	124	6.50E-07	0.51	5.70E-02
	Unit D	2.00E-13	2.00E-14	10:1	0.089	1133	200	6.40E-07	0.53	6.40E-02
	Units B and C	4.00E-13	4.00E-14	10:1	0.165	1198	296.7	9.50E-07	0.38	8.40E-02
	Unit B Anhydrite	3.00E-13	3.00E-14	10:1	0.089	1214	321	6.90E-07	0.53	1.00E-03
	Unit A-2 Carbonate	3.00E-10	3.00E-11	10:1	0.12	1091	136	7.20E-07	0.46	1.20E-02
	Unit A-2 Evaporite	3.00E-13	3.00E-14	10:1	0.089	1030	45.6	5.80E-07	0.53	1.00E-03
	Unit A-1 Carbonate	1.40E-08	9.70E-13	14912:1	0.023	1120	180.2	4.10E-07	0.82	1.20E-02
	Unit A-1 Evaporite	3.00E-13	3.00E-14	10:1	0.02	1229	343.7	4.50E-07	0.83	1.80E-03
	Niagaran Group	3.60E-09	2.50E-13	14431:1	0.026	1206	308.4	2.70E-07	0.66	1.20E-02
	Reynales / Fossil Hill	5.00E-12	5.00E-13	10:1	0.031	1200	300	2.90E-07	0.62	6.20E-01
	Cabot Head	9.00E-14	9.00E-15	10:1	0.116	1204	306	1.10E-06	0.6	3.20E-02
	Manitoulin	9.00E-14	9.00E-15	10:1	0.028	1233	350	7.50E-07	0.86	6.40E-03
Ordovician	Queenston	2.00E-14	2.00E-15	10:1	0.073	1207	310	9.00E-07	0.71	1.60E-02
	Georgian Bay / Blue Mtn.	3.50E-14	3.30E-15	11:1	0.07	1200	299.4	1.20E-06	0.79	8.80E-03
	Cobourg	2.00E-14	2.00E-15	10:1	0.015	1181	272	2.60E-07	0.8	3.00E-02
	Sherman Fall	1.00E-14	1.00E-15	10:1	0.016	1180	270	4.90E-07	0.88	1.70E-02

Period	Formation	K_H [m/s]	K_V [m/s]	K_H:K_V	θ	ρ [kg/m^3]	TDS [g/L]	S_s [m^{-1}]	ζ	τ
	Kirkfield	8.00E-15	8.00E-16	10:1	0.021	1156	234	4.90E-07	0.85	2.40E-02
	Coboconk	4.00E-12	4.00E-15	1000:1	0.009	1170	255	4.60E-07	0.93	3.60E-02
	Gull River	7.00E-13	7.00E-16	1000:1	0.022	1135	203	4.90E-07	0.85	1.40E-02
	Shadow Lake	1.00E-09	1.00E-12	1000:1	0.097	1133	200	7.40E-07	0.56	7.60E-02
Cambrian	Cambrian	3.00E-06	3.00E-06	1:1	0.071	1157	235	3.70E-07	0.34	1.30E-01
Precambrian	Upper Precambrian	1.00E-10	1.00E-10	1:1	0.038	1200	300	2.60E-07	0.49	9.50E-03
	Precambrian	1.00E-12	1.00E-12	1:1	0.005	1200	300	1.50E-07	0.88	7.20E-02

Notes: Modified from Table 4.3 of the Hydrogeologic Modelling report (NWMO11p). The following parameters (not previously defined) are described in Section 4.4.4 in the context of their use in the hydrogeologic models:

K_V – vertical hydraulic conductivity
Θ – porosity
ρ – fluid density
S_s – specific storage
ζ – one-dimensional loading efficiency
τ – tortuosity

For the regional-scale model, a groundwater divide (no-flow boundary) may be assumed to exist below the center of Lake Huron. The Cambrian is absent over the Algonquin Arch to the southeast of the DGR site. The OGSR data (e.g., OGSR04) and the 3DGF model (NWMO11aa) discussed in Section 4.1.1.1 indicate that the Cambrian is also absent northeast of the DGR site, as shown in the Figure 4-3 cross-section. To the south, units such as the Cambrian are discontinuous as a result of compartments and traps, with these being more prevalent in the Niagara Megablock region defined in Figure 4-4 (SANFORD85, CARTER96, ARMSTRONG06).

Physical hydrogeological attributes of the conceptual model, which are explored and illustrated through systematic numerical simulations at basin, regional and site specific scales, include the list below.

- Only the shallow system receives recharge from present-day precipitation at the Bruce nuclear site and surrounding region.
- The shallow system is isolated from the intermediate system by the low permeabilities of the Salina Formation.
- The intermediate system aquifers may be recharged where they crop out (or subcrop) near the Niagara Escarpment east of the Bruce nuclear site, however, the very high density of the water in aquifers such as the Guelph impedes the flow of the recharge water to the location of the proposed DGR at the Bruce nuclear site.
- Vertical advection through the system at the Bruce nuclear site is effectively non-existent because of the presence of hundreds of metres of lateral continuous, near-horizontally layered, low-permeability sediments.
- Diffusion is the dominant transport mechanism in all the low permeability units such as the Ordovician sediments and also the dominant vertical transport mechanism within the intermediate and deep groundwater regimes.
- Hydraulic gradients are upwards from the permeable Cambrian, which is over-pressured relative to density-compensated hydrostatic conditions, through the Black River Group to the Trenton Group.
- The Upper Ordovician shales and Trenton Group limestones are significantly underpressured and, at least at the present, act as a hydraulic sink for flow from both below and above.

Data from the DGR field program, for example the dataset from borehole DGR-4 shown in Figure 4.102 of the DGSM (NWMO11k), support the assertion that these Ordovician strata are underpressured; fluid saturations indicate the possible presence of a discontinuous immiscible gas phase as discussed in Section 4.3.3 of the DGSM (NWMO11k). Qualitative indications of the presence of gas come from several different sources (hydraulic testing, core logging, laboratory testing). Gas saturations were calculated for seven Silurian formations or units and nine Ordovician formations, as discussed in Section 4.3.3 of the DGSM (NWMO11k). Multiple conceptual models invoked to explore the underpressures observed in the Ordovician strata, are discussed in Section 4.4.4.1.

4.4.2 Modelling Strategy

It is standard practice in radioactive waste programs around the world to perform an analysis of the Features, Events, and Processes (FEPs) that affect the suitability and safety of a potential repository site. A catalog of FEPs specifically for argillaceous formations proposed as host rocks for repositories has been developed (MAZUREK03). Numerical modelling, whether as part of site-characterization, Geosynthesis, performance assessment or safety assessment, provides an important tool in the evaluation of the FEPs that may be relevant to the long-term safety of a repository. With regard to the hydrogeological setting and performance of the proposed DGR at the Bruce nuclear site, the geology (in a broad sense including both hydrogeology and hydrogeochemistry) provides the primary features to be evaluated. Events of concern include glaciation, and the primary processes of interest are the transport processes of advection, mechanical dispersion and diffusion, two-phase flow, glacial loading and unloading, and recharge induced by glaciation. The numerical models that are the basis for the investigation of FEPs honour the data from the DGR site characterization progam with spatial and temporal upscaling being minimized. The most important FEP considered in the hydrogeologic study is solute transport in the Ordovician sediments.

The hydrogeological modelling strategy adopted for the proposed DGR at the Bruce nuclear site was to explore the FEPs relevant to the performance of the geologic barrier hosting and isolating the DGR. The strategy was not one of trying to create a single calibrated model that could reasonably reproduce all the observed characteristics of the system, but rather to understand what FEPs were truly relevant and place bounds on the performance of different elements of the overall system. This strategy was developed because it is not feasible to fully characterize the strata of the Michigan Basin beyond the site-scale over an area of thousands of square kilometers, nor is such a characterization necessary to demonstrate the safety of the proposed DGR. Thus, the modelling strategy entailed the identification of FEPs that might be relevant to DGR performance, and then performing the modelling necessary to determine if they were in fact relevant, and if so, what their ranges of possible behaviours implied with respect to DGR performance.

The features of the hydrogeologic environment that were considered necessary to include in modelling are listed below.

- The geologic framework (stratigraphy, unit thicknesses, lateral extent and geometrical relationships).
- Hydrogeological and hydromechanical properties of the strata (hydraulic conductivity, specific storage, porosity and one-dimensional loading efficiencies).
- Hydraulic head distribution.
- Solute distributions, including environmental isotopes.
- Relative fluid saturations.
- Diffusion properties of the strata.
- Hypothetical undetected faults.

The only events that were identified as potentially affecting the performance of the DGR were glaciations and/or the presence of an undetected vertical transmissive fault. Both of these scenarios and their influence on mass transport mechanisms were examined through numerical modelling.

The processes that were considered to be potentially operative and relevant to DGR safety include:

- Advection;
- Mechanical dispersion;
- Diffusion;
- Two-phase flow;
- Physical (matrix diffusion) and chemical (sorption) retardation processes;
- Glacial loading and unloading; and
- Recharge induced by glaciation.

Physical and chemical retardation processes were purposely omitted from the modelling performed because they act only to increase the safety of the DGR. If the repository is otherwise safe, these processes simply increase the margin of safety. Retardation processes are included in the safety assessment analysis for the DGR.

From a modelling perspective, consideration of these FEPs led to the following broad modelling strategy.

- Model what the system would look like at equilibrium (base-case), using parameter values that honour the site characterization program as described in the DGSM (NWMO11k), geologically reasonable boundary conditions, and assuming full water (or brine) saturation. Compare equilibrium solution to current observations (e.g., head, solute distributions), and estimate performance measures for the equilibrium system.

- Model alternatives to the base-case, varying boundary and/or initial conditions, parameter values, loading conditions, etc., and incorporating alternative processes such as two-phase flow. Compare alternative solutions to current observations (e.g., head, solute distributions), and estimate performance measures for the alternative systems.

- Model at different scales, or using different codes, as appropriate to the issue/process to be addressed.

- Identify aspects of the performance of the system that are robust (invariant through all alternative models) and those that are sensitive to the modelling assumptions/parameters.

- Identify factors, if any, which may lead to concerns about the ability of a DGR in the Cobourg Formation to safely contain and isolate the L&ILW.

The parameter perturbation and scenario analyses of the hydrogeological modelling study involves a large number of simulations using four different numerical models. This comprehensive design provides confidence in the study conclusions through the development of multiple lines of model evidence linked to field observation. The final step of the numerical modelling study is to determine what has been learned about the system and its performance in relation to the fundamental hypotheses of site suitability introduced in Chapter 3.

4.4.3 Computational and Numerical Models

This study uses four different numerical models and two different computational models to evaluate groundwater flow and solute transport. These models consider:

- Regional-scale saturated density-dependent flow for a domain with an area of approximately 18,000 km^2 centred on the DGR (see Section 4.4.4.1 and 4.4.4.2);

- Site-scale saturated density-dependent flow for a domain with an area of approximately 400 km^2 centred on the DGR (see Section 4.4.4.3);

- Density-dependent flow for an approximately 677 km east-west cross section of the Michigan Basin (see Section 4.4.4.4); and

- One-dimensional two-phase gas and water flow analyses of a stratigraphic column at the DGR (see Section 4.4.4.5).

The regional-scale, site-scale, and cross-section modelling was accomplished using FRAC3DVS-OPG (FRACtured 3D Variably Saturated-OPG) as described in the Hydrogeologic Modelling report (NWMO11p). To investigate the hypothesis that the underpressures in the Ordovician sediments may indicate the presence of a gas phase, the two-phase air and water model TOUGH2-MP was used (PRUESS99).

4.4.4 System Performance Measures

Common measures of the performance of a groundwater system include equivalent freshwater head or environmental head and the derived pore water velocity, the solute concentration for a conservative tracer, average water particle paths and travel time, the Péclet number of molecular diffusion (BEAR88, HUYSMANS05), and Mean Lifetime Expectancy (MLE) (NORMANI07). MLE represents the average time it would take conservative, non-sorbing, non-decaying particles to travel from a given point to potential outflow points in the environment under the influence of advection, mechanical dispersion, and diffusion.

The Péclet number defining the ratio between the rate of solute transport by advection and the rate of solute transport by molecular diffusion (BEAR88, HUYSMANS05) is:

$$Pe = \frac{VL}{D_e} \qquad (4.1)$$

in which V is the pore water velocity, L is a characteristic length, and D_e is the effective diffusion coefficient calculated as the product of the tortuosity (τ) of the porous medium [-] and the molecular diffusion coefficient (D_m) [L^2/T], where T is time. A Péclet number <0.4 is indicative of solute transport that is dominated by molecular diffusion (BEAR88). The scale length is that of the mean grain or pore size or any other characteristic medium length (BEAR88). A value of L = 1 m was used in this study to provide conservatively high estimates of the Péclet number.

4.4.4.1 Regional-Scale Model

The purpose of the regional-scale model was to examine aspects of the hydrogeology of the Bruce nuclear site in three dimensions at a scale large enough to include natural hydrogeologic boundary conditions for the surface. The simulations are designed to illustrate that processes affecting groundwater flow and transport can be evaluated without assumed boundary conditions for the shallow domain exerting an undue influence on predicted outcomes. The primary focus of the regional-scale model is the assessment of solute transport in the Ordovician sediments. Specifically, whether or not transport is diffusion dominant in the present state, and whether or not it will remain diffusion dominant during glacial episodes.

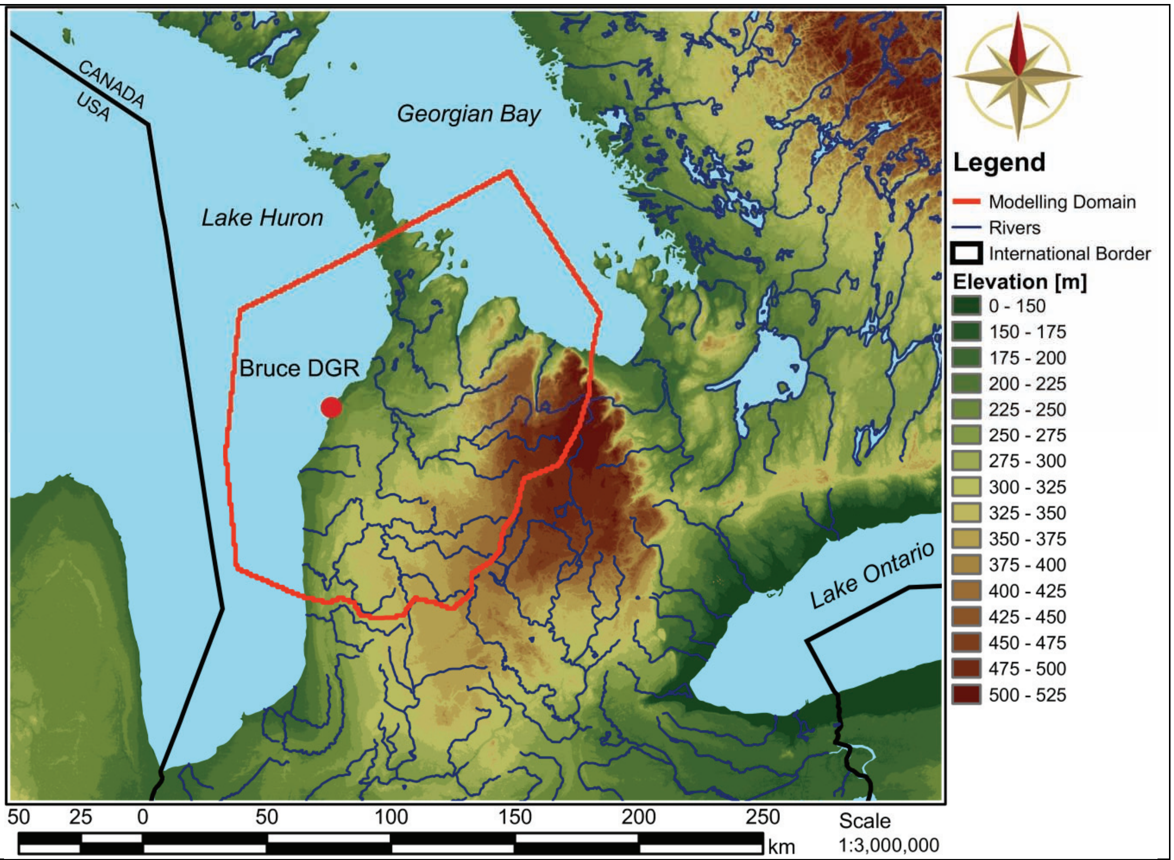
<u>Model Domain and Spatial Discretization</u>

The spatial scale required to assess solute transport in the Ordovician shale and limestone is on the order of kilometres or less. For paleoclimate analyses and to fully characterize flow in the more permeable units such as the Guelph dolostone and the Cambrian sandstone, a considerably larger spatial domain is required. Ideally, the spatial domain should include the outcrop and subcrop for the permeable units, such as the Guelph, that are potential pathways for solute migrating from the Ordovician at the location of the proposed DGR. The regional-scale spatial domain meets this criterion for all of the permeable units above the Cambrian. While it does not strictly meet this criterion for the discontinuous Cambrian, potential pathways in the Cambrian can be investigated through scenario analyses.

The regional-scale modelling domain boundary shown in Figure 4-51 was chosen from within the 3DGF domain shown in Figure 4-2 and described in detail in the 3DGF model report (NWMO11aa). The southeastern portion of the boundary follows the regional surface water divides surrounding the Bruce nuclear site. Based on the assumption that the water table is a subdued reflection of surface topography, the topographic divides are a reasonable choice for the upper flow regime and for the higher permeability Guelph (represented in the numerical model as a unit of the Niagaran) within the intermediate flow regime, which outcrops or subcrops in the model domain (NWMO11p).

The modelling domain includes the local topographic high in southern Ontario, and extends to the deepest portions of both Lake Huron and Georgian Bay. The conceptual model hypothesizes that at a point in all units/formations beneath Lake Huron, either a divide for groundwater flow occurs or horizontal flow is negligible (NWMO11p). The eastern boundary of the domain is west of the Algonquin Arch.

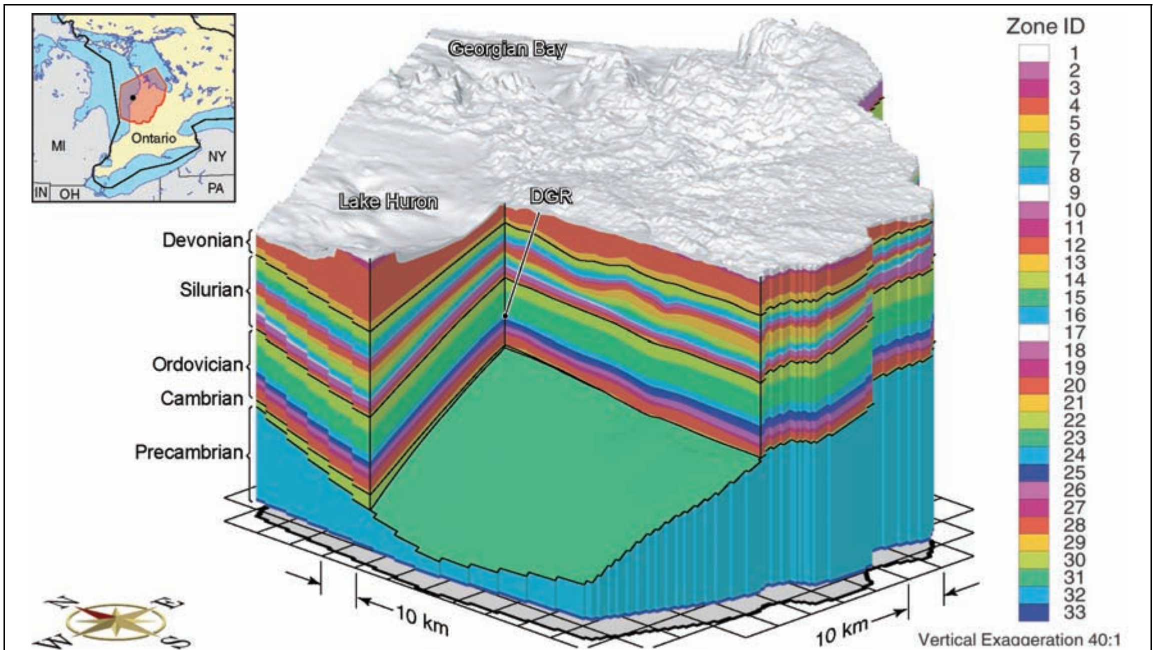
The potential energy gradients that occur at depth in the Michigan Basin will be reduced due to the presence of dense saline groundwater found within the formations of the deeper regimes. Where these formations outcrop at recharge areas, there will be a potential for fresh water to infiltrate the geologic units and displace higher density water until there is a balance between the elevation gradient and the density gradient. At this equilibrium point, the energy gradient will approach zero. With the dense brine, there will be associated higher viscosities, which will act to further impede flow. The combination of negligible horizontal energy gradients with dense brine and low permeabilities in the deep groundwater regime leads to a system that is dominated by diffusion (NWMO11p).



Note: From Figure 1.1 of the Hydrogeologic Modelling report (NWMO11p).

Figure 4-51: Location of Proposed DGR Site, Regional-Scale Modelling Domain, Land Surface Elevations and River Courses

The base-case data set for the regional-scale model consists of 39 model layers, with each of the 31 top layers corresponding to a unit in the stratigraphic section provided by the geologic framework model. The bottom 8 layers are associated with the Precambrian (7 layers) and the upper Precambrian (1 layer). A block-cut view of the assigned geologic layer zone identifiers is shown in Figure 4-52; the sub-layers for the Precambrian are not identified in the figure. Note that the vertical exaggeration is 40:1 in this figure, and others, describing the regional-scale spatial domain. Each zone identifier is associated with a specific geologic layer or geologic grouping. The layers and their constituent geologic units are listed in Table 4-5, along with the measured layer thicknesses at DGR-1/2. Due to lack of differentiation of some units in the regional database (NWMO11aa, NWMO11n), it was necessary that the Guelph Formation, the Goat Island and Gasport members of the Lockport Formation, and the Lions Head Member of the Amabel Formation be combined into a single layer referred to as the Niagaran Group for the modelling discussed in this report (e.g., NWMO11p), as indicated on the stratigraphic column for the Bruce nuclear site shown in Figure 4-5. The Georgian Bay Formation, Blue Mountain Formation, and Collingwood Member are also combined into a single model layer (NWMO11p).



Note: From Figure 2.22 of the Hydrogeologic Modelling report (NWMO11p).

Figure 4-52: Block-Cut View of FRAC3DVS-OPG Zone Identifiers for the Layers in the Regional-Scale Model

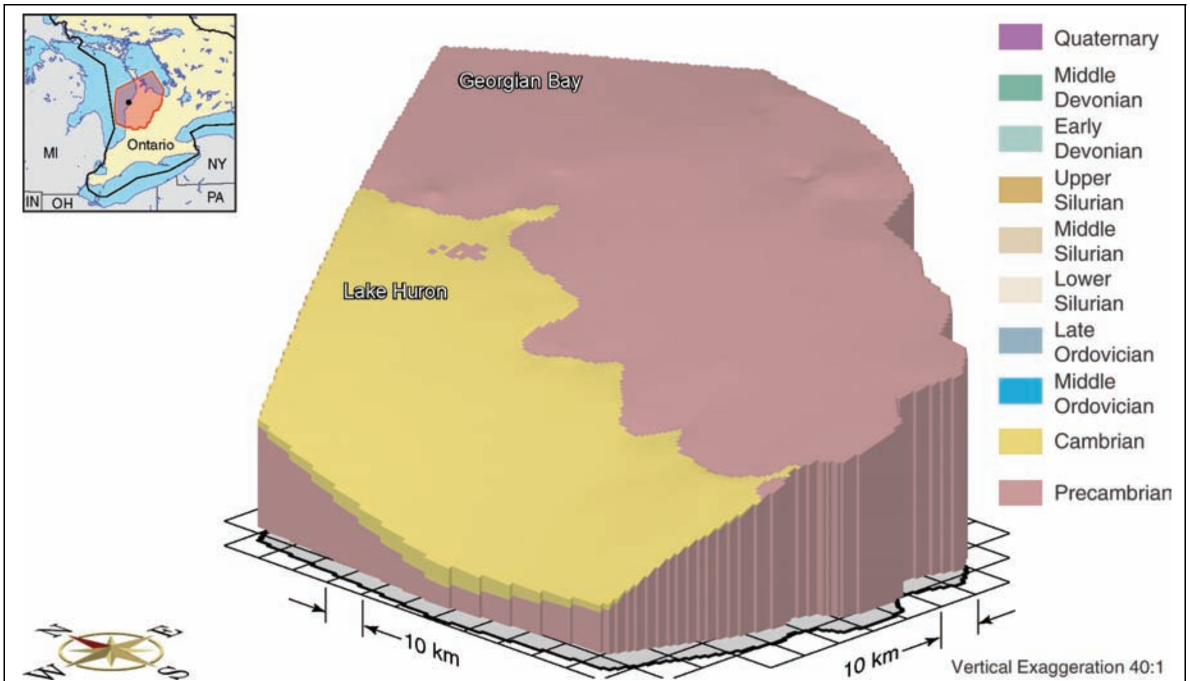
Table 4-5: FRAC3DVS-OPG Model Layers and Corresponding Geologic Units

Period	Stratigraphic Unit	Model Layer	Model Layer Number	Layer Thickness at DGR-1/2 (m)
Quaternary	Drift	Drift	1	--
Devonian	Kettle Point	Kettle Point	2	--
	Hamilton Group	Hamilton Group	3	--
	Dundee	Dundee	4	--
	Lucas	Detroit River Group	5	55.0
	Amherstburg (top 20 m)			
	Amherstburg (lower 25 m)			
	Bois Blanc	Bois Blanc	6	49.0
Silurian	Bass Islands (upper 20 m)	Bass Islands	7	45.3
	Bass Islands (lower 25 m)			

Period	Stratigraphic Unit	Model Layer	Model Layer Number	Layer Thickness at DGR-1/2 (m)
	Salina G	Salina G	8	9.3
	Salina F	Salina F	9	44.4
	Salina E	Salina E	10	20.0
	Salina D	Salina D	11	1.6
	Salina C	Salina C and B	12	46.6
	Salina B carbonate			
	Salina B evaporite	Salina B evaporite	13	1.9
	Salina A2 carbonate	Salina A2 carbonate	14	26.6
	Salina A2 evaporite	Salina A2 evaporite	15	5.8
	Salina A1 upper carbonate	Salina A1 carbonate	16	41.5
	Salina A1 carbonate			
	Salina A1 evaporite	Salina A1 evaporite and A0	17	7.5
	Salina A0			
	Guelph	Niagaran	18	34.3
	Goat Island			
	Gasport			
	Lions Head			
	Fossil Hill	Fossil Hill	19	2.3
	Cabot Head	Cabot Head	20	23.8
	Manitoulin	Manitoulin	21	12.9
Ordovician	Queenston	Queenston	22	70.3
	Georgian Bay	Georgian Bay/Blue Mountain	23	133.6
	Blue Mountain			
	Collingwood			
	Cobourg	Cobourg	24	28.6
	Sherman Fall	Sherman Fall	25	28.0
	Kirkfield	Kirkfield	26	45.9
	Coboconk	Coboconk	27	23.0
	Gull River	Gull River	28	53.6
	Shadow Lake	Shadow Lake	29	5.2
Cambrian	Cambrian	Cambrian	30	16.9
Precambrian	Upper Precambrian	Upper Precambrian	31	--
	Precambrian	Precambrian	32-39	--

Note: Based on information in Tables 2.1 and 4.1 of the Hydrogeologic Modelling report (NWMO11p).

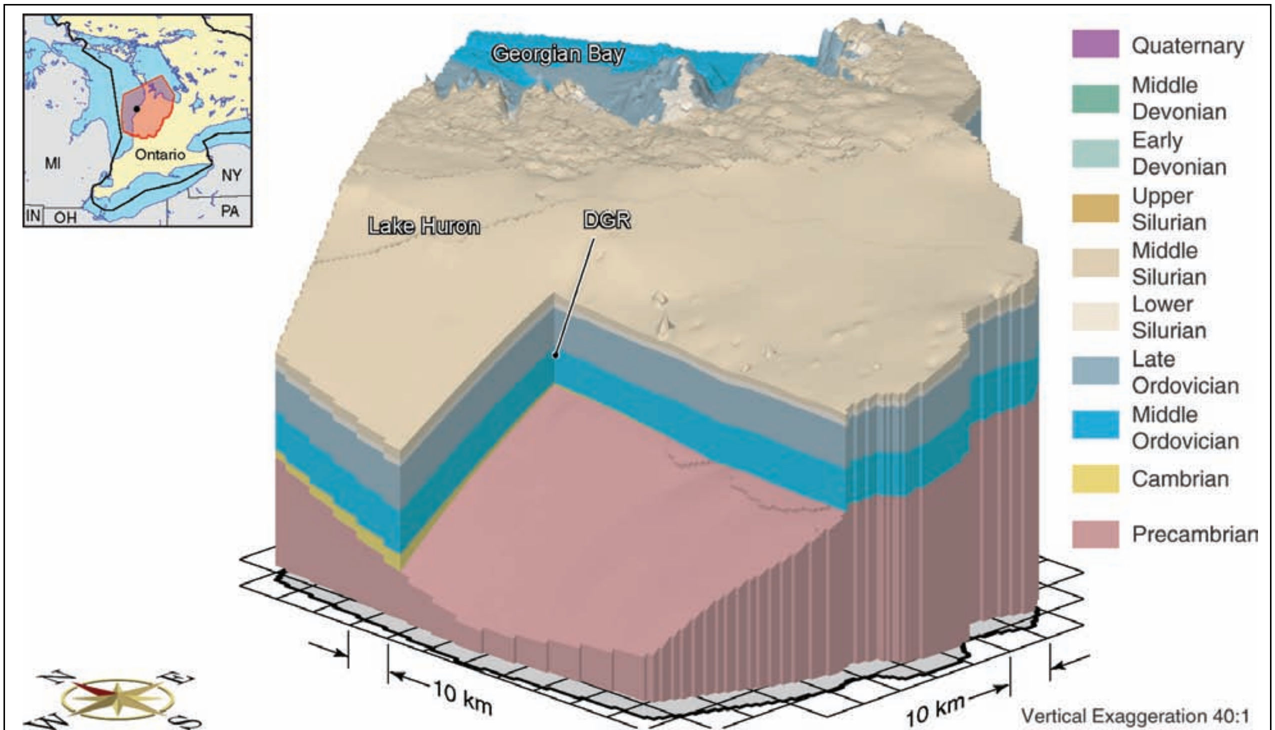
The geologic reconstruction also makes use of the outcrop limits or extent of the various geologic units. The Cambrian Formation pinches out against the Precambrian on the western flank of the Algonquin Arch (CARTER96). A 3D view of the Cambrian Formation as represented in the modelling grid is shown in Figure 4-53. An important attribute of this permeable unit is that it is present only over the more westerly part of the domain.



Note: From Figure 2.26 of the Hydrogeologic Modelling report (NWMO11p).

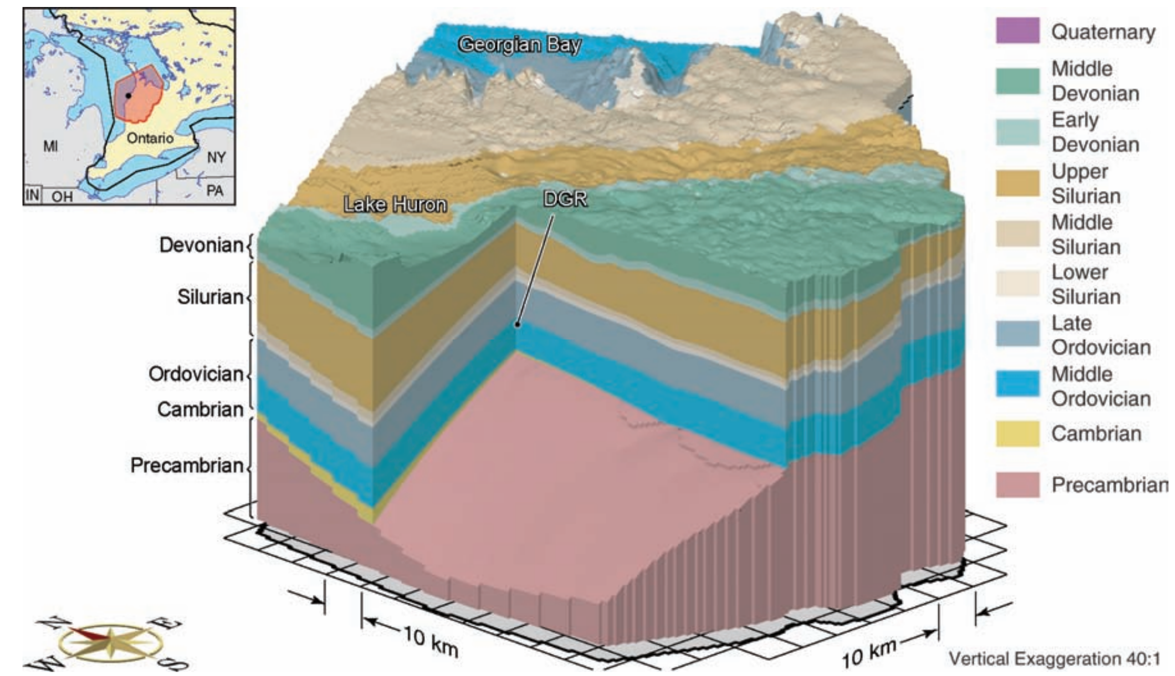
Figure 4-53: Block-Cut View Showing Spatial Extent of the Cambrian (Yellow), Underlain by the Precambrian Basement (Pink), for the Regional Modelling Domain

A view of the Middle Silurian geologic units (top of the Niagaran Group) is shown in Figure 4-54; the portion of the surface appearing rougher represents outcrops or subcrops, and has been defined using Ontario Geolgoic Survey Digital Bedrock topography and overburden thickness mapping. The zone with a smooth surface corresponds to the portion of the Niagaran that is overlain by the Upper Silurian. Pinnacle reef structures are visible as protuberances in the Middle Silurian surface to the right of the DGR location on Figure 4-54. A view of the subcrop of all geologic units below the Quaternary drift deposits is shown in Figure 4-55.



Note: From Figure 2.27 of the Hydrogeologic Modelling report (NWMO11p).

Figure 4-54: Block-Cut View Showing Spatial Extent of the Middle Silurian (Top of the Niagaran Group) for the Regional Modelling Domain



Note: From Figure 2.28 of the Hydrogeologic Modelling report (NWMO11p).

Figure 4-55: Block-Cut View Showing Subcrop of the Bedrock Units Beneath Quaternary Drift Deposits for the Regional Modelling Domain

Model Parameters

The hydrogeologic parameters defined in this section are based on the DGR borehole investigations (NWMO11k) and are applied to the regional-scale and site-scale numerical models. The relationship between the site-scale lithology and the lithology applied to the numerical models is shown in Table 4-5. Model layer thicknesses vary over the modelling domain; the thicknesses of the layers as measured at the DGR-1/2 location are also given in Table 4-5.

The base-case parameter values used for the regional-scale and site-scale groundwater modelling are given in the Hydrogeologic Modelling report (NWMO11p). Horizontal hydraulic conductivity (K_H) values were derived from the field studies (Section 4.4.1) and described in Section 4.9 of the DGSM (NWMO11k). Vertical hydraulic conductivity (K_V) values were estimated for most units by assuming an anisotropy ratio (K_H/K_V) of 10:1. Higher anisotropy ratios were assumed for units in which both high-K and low-K layers were aggregated, particularly the Salina Unit A-1 carbonate, the Niagaran Group, and the Black River Group, because K_V is dominated by the lowest K in a succession of strata. Little to no anisotropy was assumed for the high-K drift and Cambrian aquifers, and for the Precambrian.

The specific storage (S_s) and one-dimensional loading efficiency (ζ) were calculated based on preliminary data on the Elastic modulus (E), Poisson's ratio (ν), mineral grain modulus (K_s) for the rock formations, the coefficient of vertical compressibility (β') for the drift, porosity (θ), and the fluid density (ρ) as described in the DGSM (NWMO11k).

The fluid density values used were determined from the TDS concentrations as detailed in the Hydrogeologic Modelling report (NWMO11p). The tortuosity (τ) varied by layer and was calculated from the iodide effective diffusion coefficient (D_e) and the porosity (θ) described in Section 4.4 of the DGSM (NWMO11k), the free solution diffusion coefficient for iodide of 1.66×10^{-9} m^2/s (CRC83), and assuming that only 50% of the porosity was accessible to iodide diffusion. In the case of the Niagaran Group and other combined-formation model layers, their parameters were calculated using parameter-appropriate averaging of the site formation parameters (NWMO11p).

Because of grid Péclet number constraints, a longitudinal dispersivity of 500 m was used in the regional-scale model, along with a transverse to longitudinal dispersivity ratio of 0.1, and a vertical to horizontal dispersivity ratio of 0.01 (NWMO11p).

The Precambrian underlies the sedimentary deposits of the Michigan Basin. Due to a paucity of site-specific data for the Precambrian, both the hydraulic conductivity and TDS concentrations below the Cambrian or Shadow Lake formations are based on characteristics derived from studies of the Canadian Shield. Relationships between both horizontal and vertical permeability and the depth below ground surface of the Precambrian were applied to the Precambrian depth data in the 3DGF to provide Precambrian permeability data to be used in the modelling (e.g., NORMANI09). Within the area of the site-scale model, the Precambrian hydraulic conductivity ranged between approximately 1×10^{-12} and 1×10^{-10} m/s.

The salinity of groundwater generally increases with increasing depth in plutonic rock on the Canadian Shield. The highly saline pore fluids can have TDS concentrations up to 300 g/L (BOTTOMLEY02; FRAPE87). The Hydrogeologic Modelling report (NWMO11p) developed an initial TDS distribution for the Precambrian rock required for the pseudo steady-state model based on Figure 2b in Frape and Fritz (FRAPE87). The Hydrogeologic Modelling report also developed a general expression relating TDS concentration to density for use in modelling groundwater flow in the Michigan Basin on a variety of scales (NWMO11p).

A summary of the base-case regional-scale parameters is presented in Table 4-4. Different parameter values were used to investigate specific scenarios. The values used and their justification are discussed in the sections detailing the individual scenarios.

Flow Boundary Conditions

Various surface boundary conditions were applied to the regional modelling domain. A Dirichlet (fixed-head) hydraulic boundary condition was applied to the top nodes of the domain to set the water table 3 mBGS, regardless of streams or other inland water bodies such as lakes or wetlands, but not less than the elevation of Georgian Bay or Lake Huron which were set to a mean water elevation of 176 m. The scale of the model and the size of grid blocks precluded the inclusion of any hydrologic features, other than characterizing the water table as a subdued reflection of surface topography. For the regional-scale grid, the elevation of the water table was estimated at grid block nodes.

Both the sides and bottom of the modelling domain were specified as a zero-flux boundary condition. Zero-flux lateral boundary conditions are appropriate for the shallow groundwater system and the Salina A1 Unit upper carbonate aquifer because both their recharge areas and their presumed discharge areas in Lake Huron are included in the model domain. The high-permeability Niagaran Group and Cambrian Formation, however, might be thought to have the potential to allow influx and efflux across the model boundary. The use of the no-flux boundary condition for the Niagaran beneath Lake Huron is consistent with the hypothesis that at a point in units/formations beneath Lake Huron, either a divide for groundwater flow occurs or horizontal flow is negligible. As described previously, the Cambrian is known to pinch out east of the Bruce nuclear site toward the Algonquin Arch. Any potential pathways that may exist in the Cambrian to the west and northwest can be investigated through scenario analyses.

Initial Conditions and Solution of Density-Dependent Flow

Salinity plays an important role with regard to fluid flow at the proposed DGR. The higher density of the deeper fluids inhibits active flow at depth (PARK09). The methodology used to develop a solution for regional-scale density-dependent flow is described in detail in the Hydrogeologic Modelling report (NWMO11p).

A final freshwater head distribution for the base-case analysis was obtained through a multi-stage process equilibrating initial distributions of freshwater head and TDS (NWMO11p). After reaching pseudo-equilibrium at 1 Ma, the model produces salinity profiles that are compatible with the geological framework, boundary conditions and hence the groundwater domain. In the northeastern part of the model domain, brine will be flushed from shallow layers because of a combination of the absence of a source term for brine and the effect of meteoric recharge near Georgian Bay where the Ordovician formations outcrop. This is contrasted to the deeper Ordovician shale and limestone units in the western portion of the domain which, because of the absence of a velocity to transport the brine from the system, will maintain a high salinity concentration. The proposed DGR repository is located within this area. At such a location, stagnation of the groundwater is expected due to both the low permeability of the Ordovician units and the effect that density will have on reducing energy gradients.

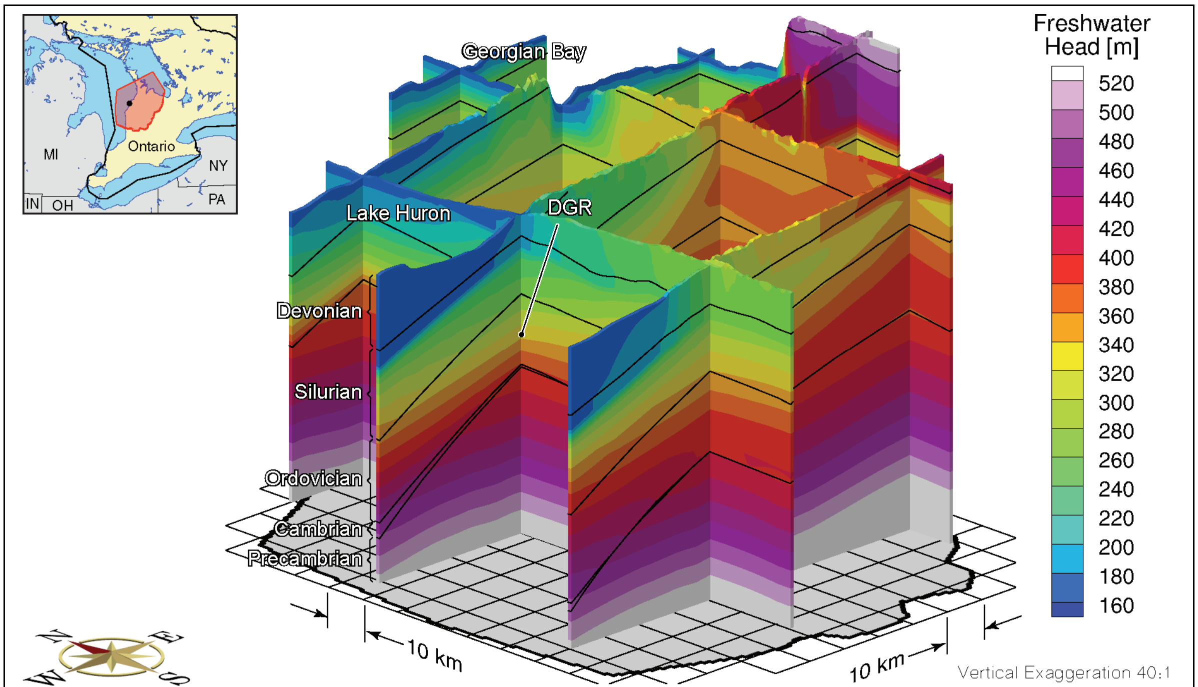
Base-Case Simulations

The base-case regional-scale model attempted to replicate the observed present-day conditions using the geological framework model, hydraulic parameters, transport parameters, porewater solute concentrations, and boundary conditions based on observations, analyses, and interpretations of this state. Importantly, the model also assumes that the system is completely water (or brine) saturated. The initial conditions of TDS concentration and equivalent freshwater heads assumed for the model evolved to a pseudo-equilibrium solution for this state. The objective of the analysis, in part, was to reveal system behaviour, identify observed attributes that may be the signature of a different state, and assess dominant transport mechanisms with particular emphasis on the Ordovician sediments.

Given the boundary conditions applied to the base-case model, the surface water level for Lake Huron of 176 mASL represents the minimum head possible in the model; the observed fluid underpressures in the Ordovician and Lower Silurian units beneath the site are clearly a consequence of a different state than that described by the base-case conceptual model. The pressures may be the result of rock dilation, from either glacial unloading or significant removal of mass through erosion that was at a rate that is greater than that of water influx to these low-permeability units from the over and underlying units with higher pressure; the pressure distribution is still evolving. Alternatively, the low pore fluid pressures may indicate the presence of a trapped non-wetting gas phase, the impact of osmosis, or the result of crustal flexure. In any case, the base-case model, as formulated, cannot be expected to produce Ordovician underpressures.

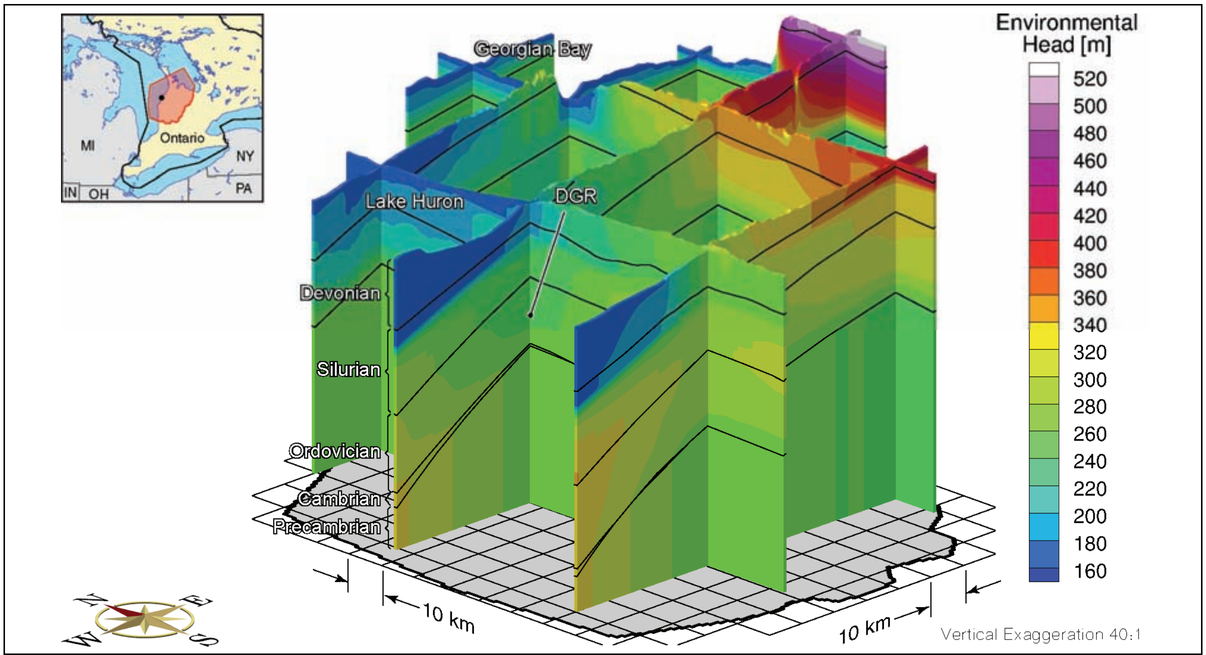
Modelling of the pressure profile at the DGR boreholes can be approached from two perspectives: an assessment of the cause of the underpressures of the Ordovician and Lower Silurian and the overpressures of the Cambrian, or an assessment of the evolution of the pressures from their current state. The former analysis would require either realizations of the previous state of the regional-scale system or the simulation of immiscible, two-phase flow of gas and water. Realizations of the previous state of the system during the most recent episodes of glaciation are described in Section 4.4.4.2. An analysis of two-phase water and gas flow using the model TOUGH2-MP is developed in Section 4.4.4.5 for a one-dimensional column. An assessment of the future evolution of the pressures cannot be undertaken at the regional-scale due to a lack of reliable data on the pressures at other locations in the domain; however, an analysis at the site-scale is developed in Section 4.4.4.3.

The shallow groundwater regime above the Salina is dominated by flow that mimics topography. Beneath the shallow groundwater zone, the heads are not controlled to the same extent by the local elevation of the surface. The main control for the horizontal component of the density-dependent energy gradient at depth is the elevation difference between Lake Huron and the topographic high at the Niagara Escarpment. The head signature will be transmitted from the outcrop area and will be dissipated, depending on the energy gradient, across the domain (Figure 4-56). At a given location, the vertical component of the energy gradient is controlled by the difference in the environmental heads between the more permeable units that are separated by low-permeability units (Figure 4-57). For the regional domain, the higher permeability Cambrian (where present) and Niagaran Group formations are separated by the low-permeability units of the Ordovician and Lower Silurian. The Niagaran is confined in the southwestern part of the domain by the overlying low-permeability units of the Salina. Flow in the Niagaran where it is unconfined is controlled by surface topography.



Note: From Figure 4.11 of the Hydrogeologic Modelling report (NWMO11p).

Figure 4-56: Fence View of Freshwater Heads that have Equilibrated at 1 Ma to the Temporally Varying TDS Distribution

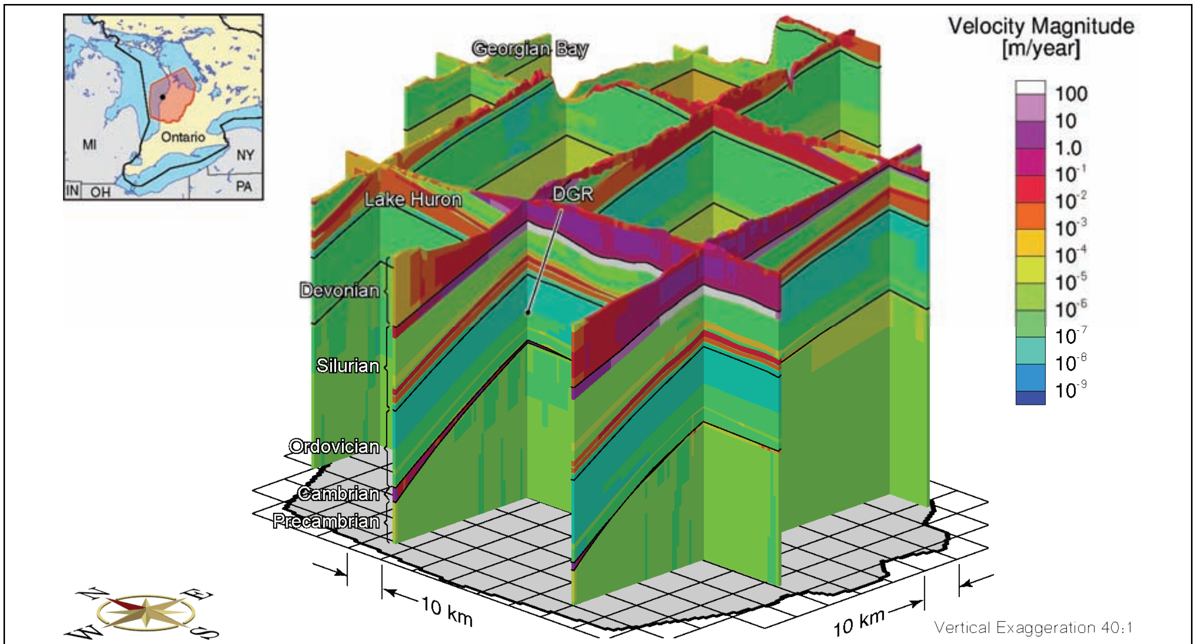


Note: From Figure 4.15 of the Hydrogeologic Modelling report (NWMO11p).

Figure 4-57: Fence View of the Base-Case Environmental Heads that have Equilibrated at 1 Ma to the Temporally Varying TDS Distribution

The over-pressured environmental heads observed in the permeable Cambrian at the DGR-4 borehole, as shown in Figure 4.102 of the DGSM (NWMO11k), are under predicted in the pseudo-steady-state analysis with the base-case parameters, initial conditions and boundary conditions. Several causes can be postulated for why the observed Cambrian pressures are higher than those modelled. Because the Cambrian pinches out east of the Bruce nuclear site, it does not outcrop or subcrop within the modelling domain and, therefore, is not connected to any recharge area in the model. Within the model, the Cambrian head is generated by the fluid density distribution and depth of the Cambrian. In actuality, the Cambrian may derive its head from a higher elevation recharge area outside the model domain and/or from connection to the centre of the Michigan Basin where it is several kilometres deep with a significant column of higher density saline fluids above. Either of these possibilities would require continuity of the Cambrian's permeability over much of the basin.

In addition to the elevation component of the gravitational gradient imposed by the topographic high at the Niagara Escarpment, the density of the brine in the deep groundwater zone will have an impact on the energy gradients. The salinity profile for the base-case at a pseudo-equilibrium time of 1 Ma consists of relatively fresh groundwater for the shallow groundwater zone and an area with much higher TDS concentrations for the intermediate and deep groundwater zone (below the Salina, where present). The shallow groundwater zone will remain devoid of salinity because the continual inflow of meteoric water through recharge to the zone will dilute any salinity that diffuses upward through the Silurian or Ordovician. The brine concentrations in the low-permeability Ordovician units at the Niagara Escarpment, where the Silurian is absent, will also experience some flushing as well; however, the higher density groundwater found in the deeper zone, that has a higher energy than water with low TDS, will prevent any significant penetration of freshwater. The TDS transition zone occurs across the Salina (e.g., Figure 4-34).



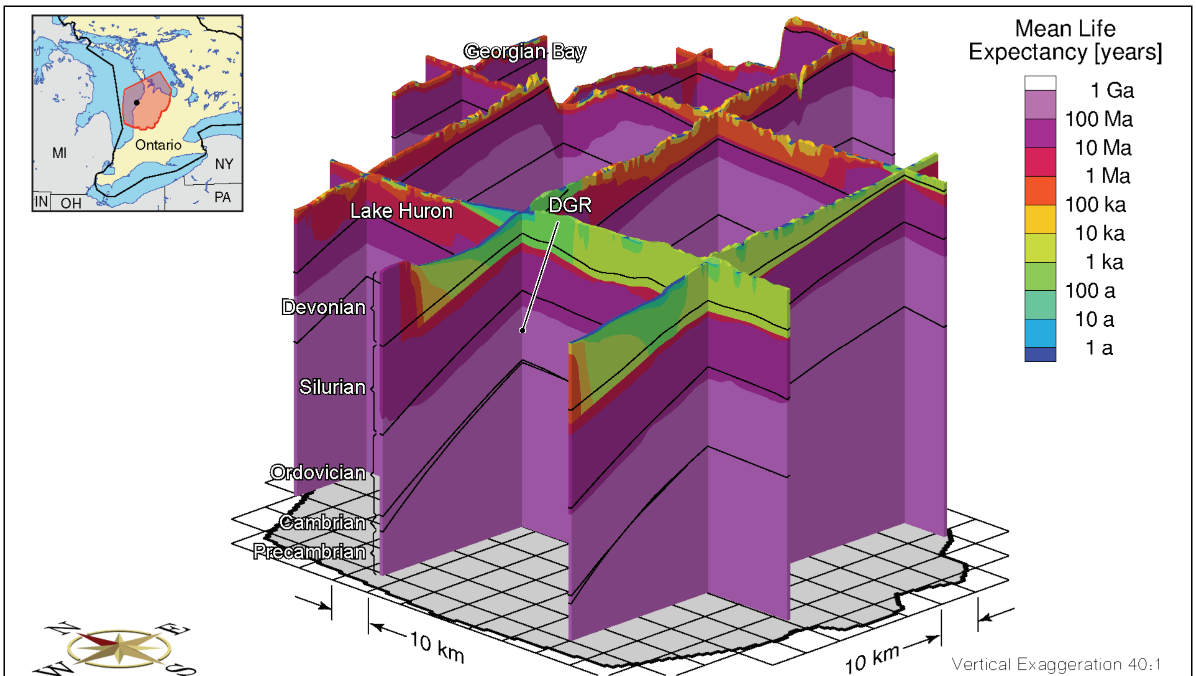
Notes: Diffusion is the dominant mechanism wherever velocity is <10^{-4} m/a. From Figure 4.17 of the Hydrogeologic Modelling report (NWMO11p).

Figure 4-58: Fence View of Base-Case Pore Water Velocity Magnitude

The base-case pore water velocity magnitudes are presented in Figure 4-58. The highest velocities occur in the more permeable shallow groundwater zone. The lower velocities beneath Lake Huron and Georgian Bay are the result of the absence of a horizontal gradient. The reduction of the velocities in the Salina Group is shown as the greenish band below the upper reddish-purple band at the DGR location, while the higher velocities of the Niagaran in the Silurian appear as the first orange/red band above the indicated DGR position.

Above the Niagaran, higher velocities are also evident in the Salina A1 Unit carbonate. Within the Ordovician in the vicinity of the proposed DGR, the groundwater pore velocities are less than $1x10^{-6}$ m/a; the pore water velocity estimated for the Cobourg Formation is $2x10^{-7}$ m/a. The estimated Péclet number for the Cobourg Formation for the base-case analysis is $2x10^{-4}$, indicating that solute transport in the Ordovician will be diffusion dominated.

The performance measure selected for the evaluation of the groundwater system is the MLE (Figure 4-59). The general trend for the MLE is similar to that found in the head and velocity distributions. The shallow groundwater zone has significantly shorter MLEs compared to the deep groundwater zones. The areas of recharge versus discharge can be noted in the figure as the recharge areas have a high MLE while the discharge areas have low MLEs. The groundwater area surrounding the proposed DGR is calculated to have an MLE of 164 Ma for the base-case regional-scale conceptual model.



Note: From Figure 4.21 of the Hydrogeologic Modelling report (NWMO11p).

Figure 4-59: Fence View of Base-Case MLE

Alternative Simulations

To evaluate the potential effects of alternative conceptualizations of various features of the base-case model, different boundary conditions, and different parameter values, a variety of alternative simulations were performed with the regional-scale model. Two sets of simulations involved variations in the hydraulic conductivity of the Precambrian and of the Cambrian. Another set examined the effects of varying the surface boundary conditions. One simulation evaluated the effect of changing the lateral boundary condition on the model. An extensive suite of simulations focused on the effects of past glacial cycles is described in Section 4.4.4.2. The alternative simulations were designed to reveal the attributes of the groundwater system that are important in the development of a safety case for a deep geologic repository and to investigate the sensitivity of the numerical solution to selected parameters. The performance measures for the analysis includes MLE and the Péclet number for the Cobourg Formation.

Table 4-6 provides a matrix of the alternative simulations performed, showing how different conditions and assumptions were combined (NWMO11p). The scenario names in Table 4-6 correspond to the prefix of the file names for the computer runs. The "f" designates the FRAC3DVS-OPG computational model, the "r" designates the regional-scale model, the middle descriptor of "base" designates that the analysis is a perturbation of the base-case regional-scale model, while the third and fourth descriptors designate the scenario. The third descriptor "paleo" indicates that the analysis is one of the paleoclimate scenarios described in Section 4.4.4.2.

Table 4-7 lists the MLE estimates at the location of the proposed DGR for each of the non-paleo scenarios modelled.

Table 4-8 presents a summary of the Péclet numbers for the Cobourg Formation calculated using Equation 4.1 with a characteristic length of 1 m for each of the scenarios modelled. The Péclet numbers are all less than 10^{-3}, clearly supporting the hypothesis that solute transport in the Ordovician sediments is diffusion dominant. Vertical pore velocities would have to be three orders of magnitude greater than the modelled scenarios indicate before advection would constitute a significant transport mechanism. No plausible parameter variation could cause such an increase in velocity.

Table 4-6: Matrix of Regional-Scale Simulations Performed

Factor Varied	Condition	fr-base	fr-base-up	fr-base-hkp	fr-base-hbc	fr-base-rech	fr-base-paleo	fr-base-paleo-biot	fr-base-paleo-gas	fr-base-paleo-head80	fr-base-paleo-head30	fr-base-paleo-zero-head	fr-base-paleo-le-zero	fr-base-paleo-nn9921	fr-base-paleo-openbnd
Precambrian Conductivity	Uniform		x												
	Vary with depth	x		x	x	x	x	x	x	x	x	x	x	x	x
Weathered Precambrian Conductivity	At least 1 x 10^{-10} m/s	x	x		x	x	x	x	x	x	x	x	x	x	
	At least 1 x 10^{-8} m/s			x											x
Lateral Boundary Conditions	Neumann Zero Flux	x	x	x		x	x	x	x	x	x	x	x	x	
	Dirichlet heads for A1 Carb., Niagaran, and Cambrian														x
	High K open to surface				x										
Surface Boundary Conditions	Dirichlet	x	x	x	x										
	Neumann					x									
Paleo Surface Boundary Conditions	Dirichlet 100% ice thickness						x	x	x				x	x	x
	Dirichlet 80% ice thickness									x					
	Dirichlet 30% ice thickness										x				
	Dirichlet 0% ice thickness											x			
Paleo Simulation	nn9930						x	x	x	x	x	x	x		x
	nn9921													x	
Hydromechanical Coupling	Biot coeff. = 1.0						x		x	x	x	x	x	x	x
	Biot coeff. = 0.5							x							
Presence of Gas Phase	No gas phase						x	x		x	x	x	x	x	x
	Partial gas phase								x						
Loading Efficiency	Actual						x	x	x	x	x	x		x	x
	Zero												x		
Paleo Cycles	1 – 120 ka						x	x	x	x	x	x	x	x	x
	2 – 240 ka						x								

Note: Modified from Table 4.10 of the Hydrogeologic Modelling report (NWMO11p).

Table 4-7: MLE at the Location of the Proposed DGR for Alternative Modelling Scenarios

Simulation	Cobourg MLE (Ma)
fr-base (no density)	155
fr-base	164
fr-base-hkp	164
fr-base-up	161
fr-base-rech	172
fr-base-hbc	44
Note: Modified from Table 4.12 of the Hydrogeologic Modelling report (NWMO11p).	

Table 4-8: Péclet Numbers for the Cobourg Formation from Regional-Scale Analyses

Simulation	Péclet Number [ℓ = 1 m]
fr-base (no density)	2.17E-04
fr-base	2.91E-04
fr-base-hkp	3.13E-04
fr-base-up	3.04E-04
fr-base-rech	2.72E-04
fr-base-hbc	7.59E-04
Note: Modified from Table 4.11 of the Hydrogeologic Modelling report (NWMO11p).	

Conclusions from Regional-Scale Modelling

Modelling of a variety of different scenarios has shown that:

- The head conditions in the Niagaran and Cambrian that drive advective flow through the Ordovician have no significant effect on transport through the Ordovician because that transport is so strongly dominated by diffusion;
- Opening the lateral boundaries to the surface reduced the MLE from a range of 148 to 172 Ma to only 44 Ma, providing a robust demonstration that a DGR in the Cobourg Formation can effectively isolate radionuclides. None of the changes

modelled altered the condition of an upward gradient from the Cambrian to the Niagaran, except for scenario fr-base-hbc; and

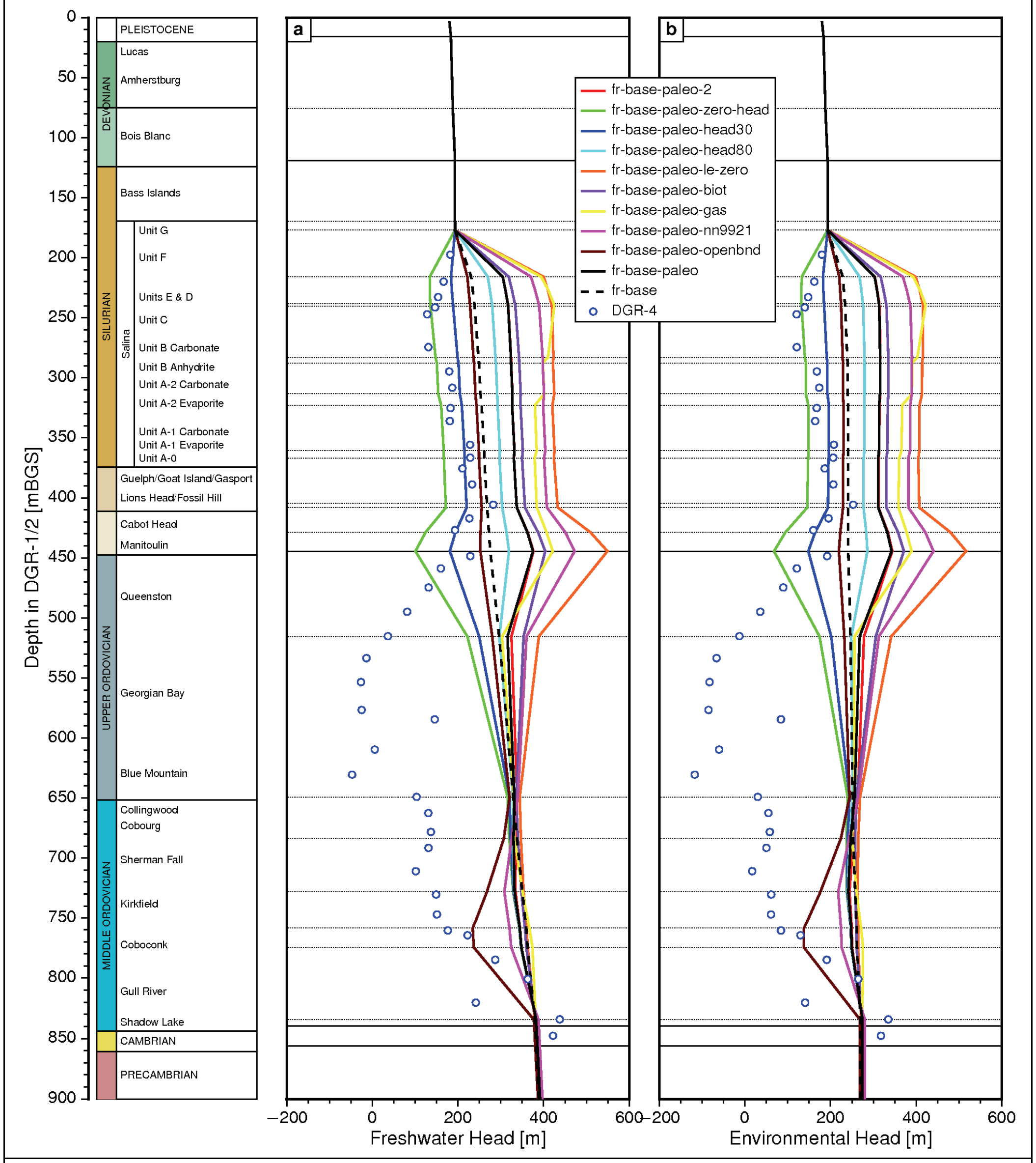
- The Péclet numbers are all less than 10^{-3}, clearly supporting the hypothesis that solute transport in the Ordovician sediments is diffusion dominant.

4.4.4.2 Regional-Scale Paleoclimate Modelling

The objective of the paleoclimate modelling work was to investigate the impact of glaciation and deglaciation on the dependent variables and performance measures of the regional-scale model. Of importance is the impact on solute transport in the Ordovician sediments and the pressures in the various units at the Bruce nuclear site. Regional-scale paleoclimate modelling described in the Hydrogeologic Modelling Report (NWMO11p) was performed by using information produced by the University of Toronto Glacial Systems Model (UofT GSM), detailed in the Long-Term Climate Change report (NWMO11r). Ice thicknesses, permafrost depths, and lake depths from a 120,000 year (120 ka) long glacial simulation were applied as boundary conditions to the regional-scale model to evaluate the groundwater system response to glaciation (Figure 4-60). In addition to a base-case scenario which used the UofT GSM paleoclimate model nn9930 (NWMO11r), alternative scenarios were modelled in which:

- Hydraulic pressures beneath the temperate ice sheet were varied from 80% to 30% of the ice sheet thickness;
- The loading efficiency of the rock units was reduced to zero;
- The Biot coefficient of the rock units was reduced to 0.5;
- A free gas phase was present in the pores of most rock units;
- Two 120 ka paleoclimate cycles occurred in succession;
- Different ice-sheet loading and unloading histories were simulated based on predicted UofT GSM outcomes (NWMO11r); and
- High-permeability units were given open lateral boundaries.

The base-case paleoclimate model (scenario fr-base-paleo) shows slight Silurian and Queenston overpressures at the end of the paleoclimate cycle, and no underpressures (Figure 4-60). Including a second consecutive paleoclimate cycle (fr-base-paleo-2) had little effect on these results. Decreasing the hydraulic boundary condition at surface (fr-base-paleo-head80 and fr-base-paleo-head30) causes Silurian and Upper Ordovician pressures to decrease, with slight underpressures appearing for a surface hydraulic boundary condition set to 30% ice thickness. A free-draining boundary condition at the base of the glacier (fr-base-paleo-zero-head) causes more underpressure in the Silurian and Upper Ordovician (Figure 4-60).



Notes: Freshwater and environmental heads for DGR-4 are shown as measured on August 24, 2009. Modified from Figure 5.25 of the Hydrogeologic Modelling report (NWMO11p).

Figure 4-60: Plot of Freshwater Head and Environmental Head Results from Paleoclimate Simulations

The base-case used loading efficiencies defined by the properties of the model layers (refer to Table 4-4). Decreasing the loading efficiency, whether directly (fr-base-paleo-le-zero), by introducing gas (fr-base-paleo-gas), or by lowering the Biot coefficient (fr-base-paleo-biot), causes overpressures in the Silurian and Upper Ordovician to increase from the base-case results. Placing open boundaries on the high-permeability units (fr-base-paleo-openbnd) holds Silurian heads largely unchanged and produces Middle Ordovician underpressures, with essentially hydrostatic conditions everywhere else.

The alternate paleoclimate model nn9921 (fr-base-paleo-nn9921) produced more overpressures than the base-case nn9930 model in the Silurian and Upper Ordovician; the overpressures were intermediate between those from the zero loading efficiency case and those from the lowered Biot coefficient case. Slight underpressures developed from the Sherman Fall Formation to the Gull River Formation (Figure 4-60).

Most of the paleoclimate scenarios affected heads only in the Silurian and Upper Ordovician. None of the paleoclimate scenarios produced Upper and Middle Ordovician underpressures like those observed at DGR-4, nor could any plausible parameter variations. Increasing hydraulic diffusivity, whether by increasing hydraulic conductivity or decreasing specific storage, would allow the system to respond more rapidly to glacially induced perturbations and return to equilibrium conditions more rapidly.

<u>Conclusions from Regional-Scale Paleoclimate Modelling</u>

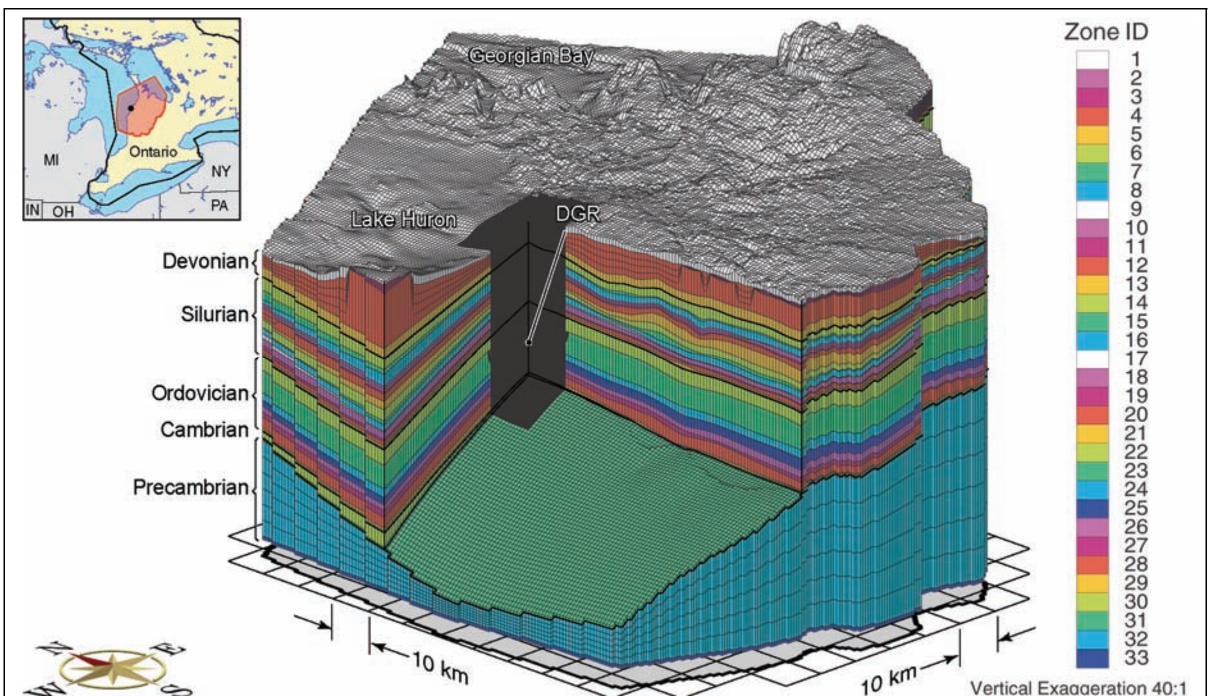
In summary, none of the paleoclimate scenarios modelled were able to produce a head profile similar to that observed in the DGR boreholes. The Ordovician underpressures that are observed do not appear to be the result of glacial loading and unloading (NWMO11p). None of the alternative scenarios showed recharge water penetrating below the middle of the Salina, or a different distribution of TDS in the system from the base-case scenario. Diffusion remained the dominant transport mechanism in the Ordovician in all scenarios.

4.4.4.3 Site-Scale Model

The objective of the site-scale hydrogeologic modelling was to investigate the evolution of the tracer plume originating from the proposed DGR location (the base-case model), the measured pressure profile in the DGR boreholes, and the impact of hypothetical undetected transmissive fracture zones connecting the Cambrian to the Niagaran (NWMO11p).

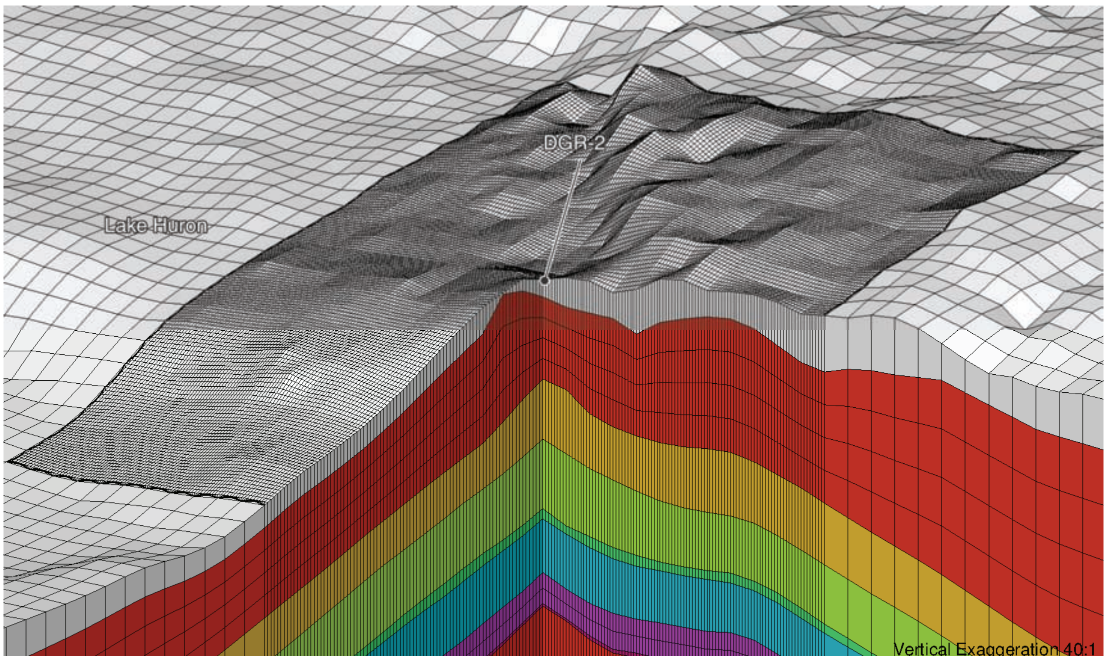
Model Geometry

The site-scale spatial domain relative to that of the regional-scale domain is depicted in Figure 4-61. The domain has a spatial extent of 19.1 km in the west-to-east direction and 18.9 km in the south-to-north direction centred on borehole DGR-2. The site-scale domain was discretized by using 6 columns (west-to-east sub-gridding) for each regional-scale column and 8 rows (south-to-north sub-gridding) for each regional-scale row. The resulting site-scale domain has 150 columns and 168 rows with each grid block being 127 m in the west-to-east direction and 112.6 m in the south-to-north direction. The areal discretization is shown in Figure 4-62. Sub-gridding was also used to refine the discretization of the Cobourg Formation with three layers being used in the site-scale model to represent the single regional-scale layer (Figure 4-63). The overlying Collingwood/Blue Mountain/Georgian Bay, Queenston and Niagaran layers were subdivided into 8, 4 and 3 layers respectively. The underlying Gull River, Kirkfield and Sherman Fall formations were further subdivided into 4, 2 and 3 layers in the site-scale model, respectively.



Note: From Figure 4.28 of the Hydrogeologic Modelling report (NWMO11p).

Figure 4-61: Regional-Scale Discretization Showing Location of Site-Scale Spatial Domain



Note: From Figure 4.29 of the Hydrogeologic Modelling report (NWMO11p).

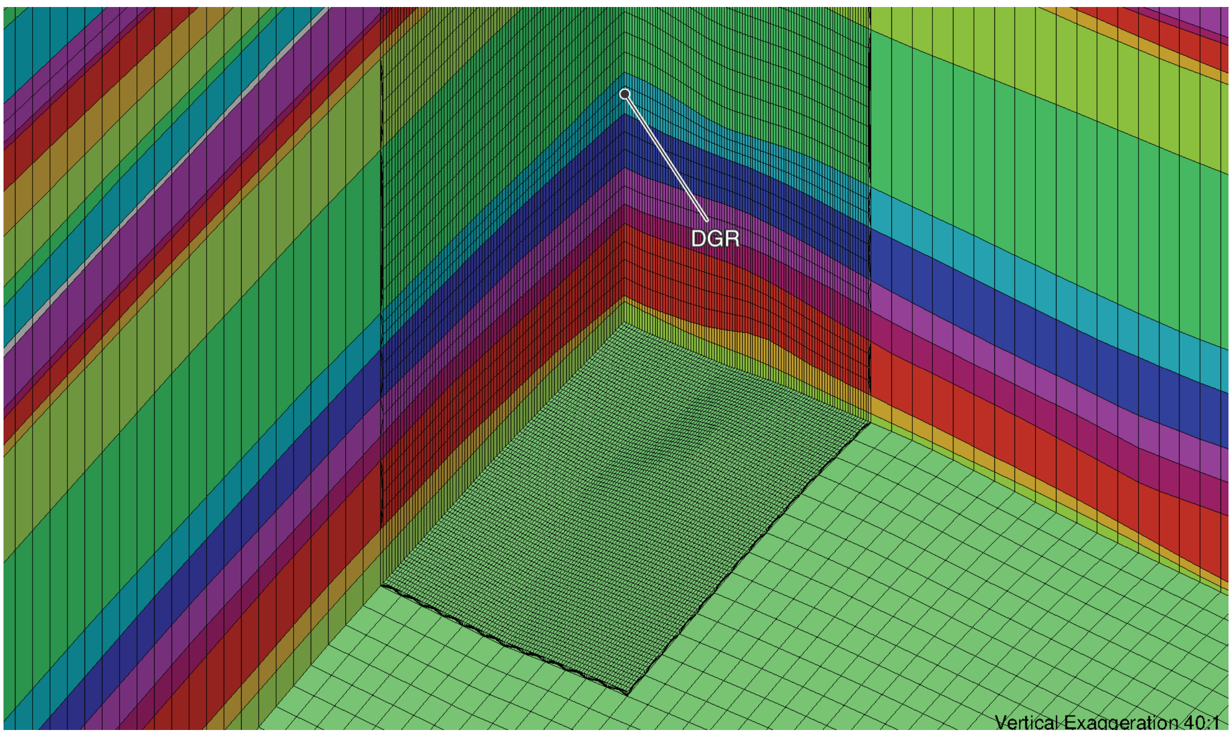
Figure 4-62: Regional-Scale Discretization Showing Site-Scale Discretized Spatial Domain

Model Parameters

The hydraulic properties used for the site-scale analyses are the values developed in the site investigation (NWMO11k). Each model layer was assigned unique but homogeneous properties. Using a grid Péclet number constraint, the longitudinal dispersivity coefficient was selected as approximately one half of the maximum length of the side of a site-scale grid block. As a consequence, the contribution to solute migration of mechanical dispersion may be overestimated. The boundary conditions for the embedment approach are those imposed on the regional-scale domain; the solution methodology is the same as that followed in the regional-scale analyses.

Base-Case Simulations

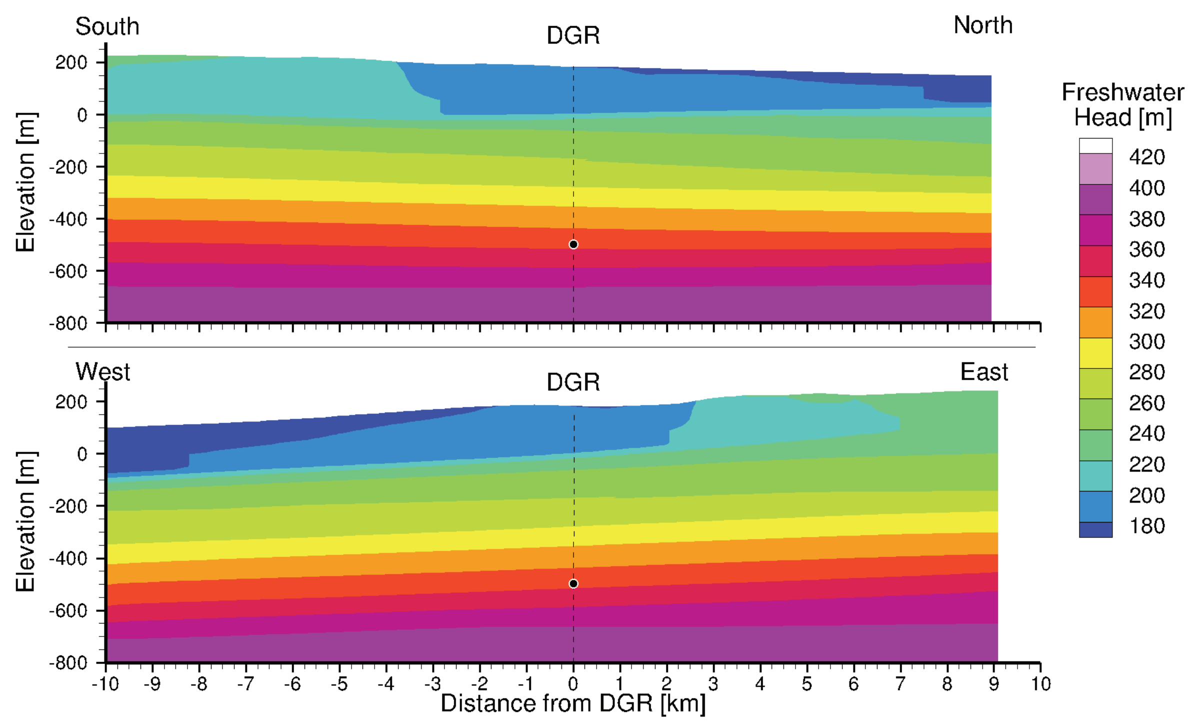
The purpose of the base-case site-scale simulations was to evaluate the transport of a conservative solute from the DGR to the edge of the modelling domain under equilibrium, fully water-saturated conditions (NWMO11p). The base-case site-scale solution for freshwater heads at a pseudo-equilibrium time of 1 Ma is presented in N-S and E-W cross-sections through the DGR location in Figure 4-64. The base-case site-scale pseudo-equilibrium solution for environmental heads is presented in cross-section view in Figure 4-65. Environmental heads can be used to estimate vertical gradients while the freshwater heads can be used to estimate horizontal gradients.



Note: From Figure 4.30 of the Hydrogeologic Modelling report (NWMO11p).

Figure 4-63: Regional-Scale Discretization Showing Vertical Details of Site-Scale Discretized Spatial Domain

The migration of a conservative tracer released to the Cobourg Formation at the proposed DGR site was investigated for the saturated base-case site-scale case. The source term for the conservative tracer was defined using prescribed concentrations of unity for the eight nodes of a grid block at the horizontal direction centre of the site-scale grid in the middle layer of the three layers used to discretize the formation. The analysis assumes that there is no decay of the source and that the solute neither decays nor adsorbs as it migrates, both highly conservative assumptions. The transport parameters used for the analysis are given in the Hydrogeologic Modelling report (NWMO11p).

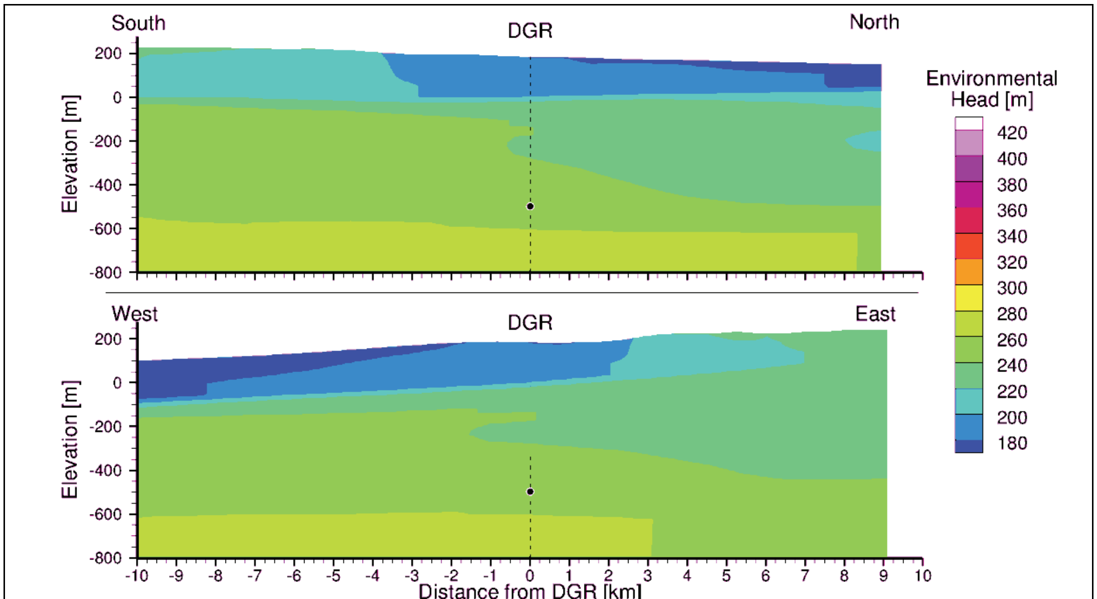


Notes: Cross-sections are parallel to cut faces on block diagram shown in Figure 4-61. From Figure 4.32 of the Hydrogeologic Modelling report (NWMO11p).

Figure 4-64: Cross-Sections of Freshwater Heads for the Base-Case Site-Scale Analysis with Equilibrated Regional-Scale Heads as the Initial Condition

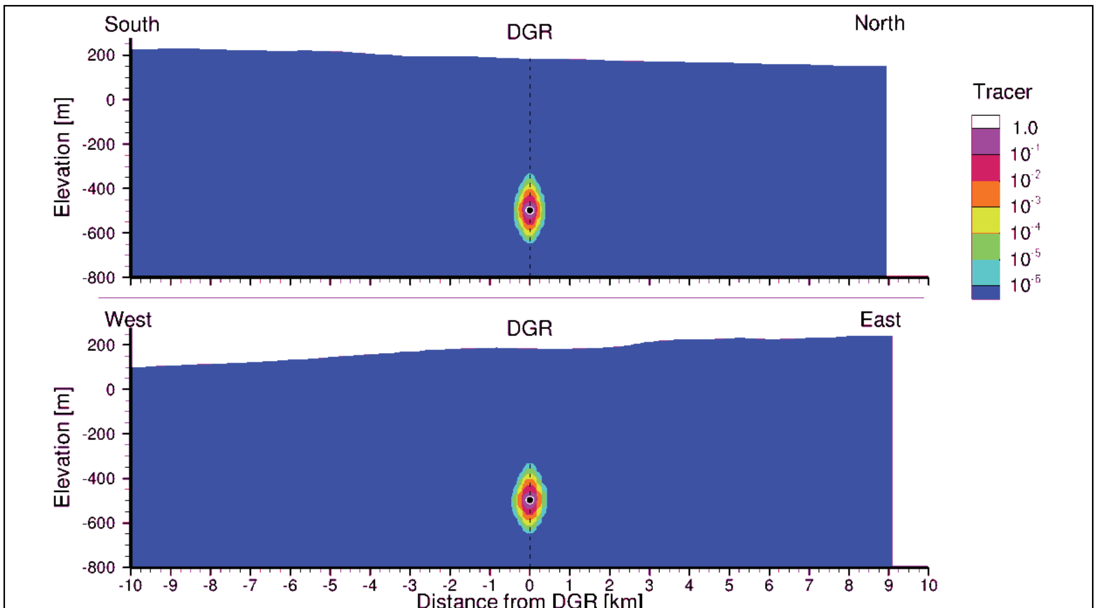
Cross-section views of the tracer distribution at 100 ka and 1 Ma are shown in Figure 4-66 and Figure 4-67, respectively. Tracer at a relative concentration exceeding 10^{-6} reaches the Niagaran and Cambrian by 1 Ma, but not by 100 ka.

One variation on the base-case was modelled, in which a weathered zone was incorporated in the upper Precambrian. Inclusion of the weathered zone resulted in no obvious differences in the spatial distribution of the tracer at 100 ka and 1 Ma, as the permeable Cambrian unit tends to diminish the impact of the weathered zone on the migration of the tracer plume.



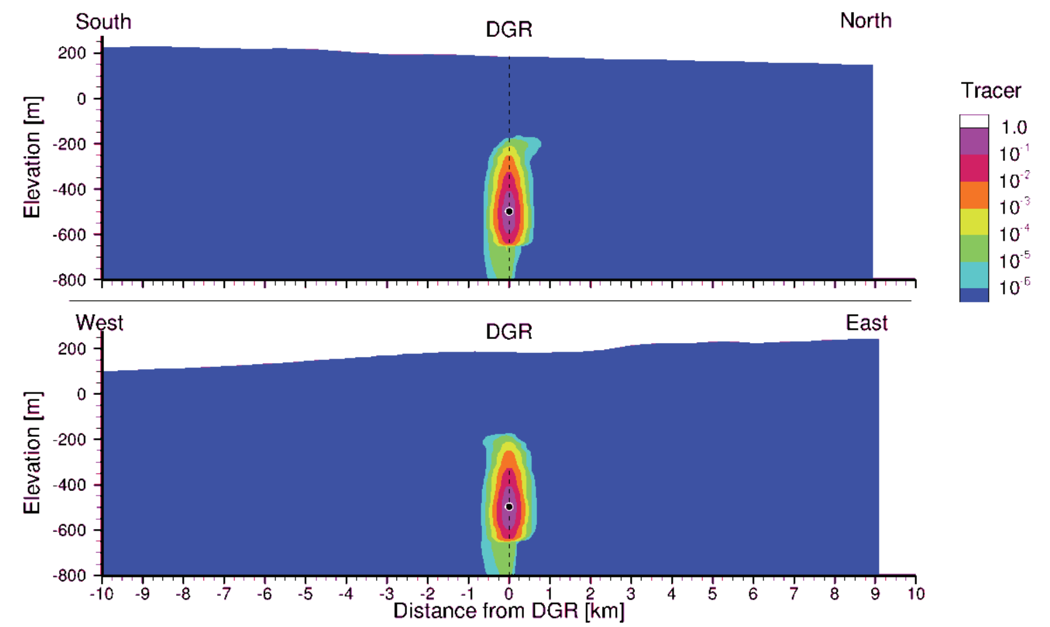
Note: From Figure 4.34 of the Hydrogeologic Modelling report (NWMO11p).

Figure 4-65: Cross-Sections of Environmental Heads for the Base-Case Site-Scale Analysis with Equilibrated Regional-Scale Heads as the Initial Condition



Note: From Figure 4.35 of the Hydrogeologic Modelling report (NWMO11p).

Figure 4-66: Cross-Section View of the Spatial Distribution of a Tracer at 100 ka with Equilibrated Regional-Scale Heads as the Initial Condition



Note: From Figure 4.36 of the Hydrogeologic Modelling report (NWMO11p).

Figure 4-67: Cross-Section View of the Spatial Distribution of a Tracer at 1 Ma with Equilibrated Regional-Scale Heads as the Initial Condition

Alternative Simulations

In addition to the base-case analysis of solute migration from the DGR, the site-scale model was also used to investigate two questions related to:

- The effect of different hydraulic anisotropies in the Black River Group on Ordovician underpressures; and
- The effect of a fracture zone connecting the Cambrian and Niagaran on pressure profiles.

The suite of base-case and alternative simulations performed using the site-scale model is illustrated in Table 4-9.

The environmental head distribution versus depth for the DGR-4 borehole is plotted in Figure 4.102 of the DGSM (NWMO11k) based on the pressure measurements in DGR-4 on June 6, 2008, August 24, 2009 and November 15, 2009. Relative to the ground surface elevation at DGR-4 of 181.6 mASL, the profile indicates that the Cambrian is overpressured while units in the upper Ordovician are significantly underpressured, thus reflecting a water deficit relative to the amount of water that would be in the pores for pressures that are hydrostatic relative to the elevation of the ground surface. The evolution of these pressures as they equilibrate to the present-day boundary conditions was investigated using the site-scale model assuming fully saturated (single phase) conditions. Instead of using the equilibrated regional-scale heads as the initial condition, as was done for the base-case modelling, this modelling used the August 24, 2009 measured environmental head profile at the DGR-4 borehole as the initial condition.

Table 4-9: Parameters and Initial Conditions for Site-Scale Analyses

Parameters						Underpressure in the Ordovician			
		fs-base	fs-base-hkp	fs-1km	fs-5km	fs-base-under-pressure	fs-10kv-under-pressure	fs-100kv-under-pressure	fs-1km-under-pressure
Initial Heads	Steady State	•	•	•	•				
	Underpressure					•	•	•	•
Weathered Zone in the Precambrian	20 m		•						
	0 m	•		•	•	•	•	•	•
Fracture Zone	1 km			•					•
	5 km				•				
Anisotropy in the Black River Group	10:1						•		
	100:1							•	
	1000:1	•	•	•	•	•			•
Note: Modified from Table 4.14 of the Hydrogeologic Modelling report (NWMO11p).									

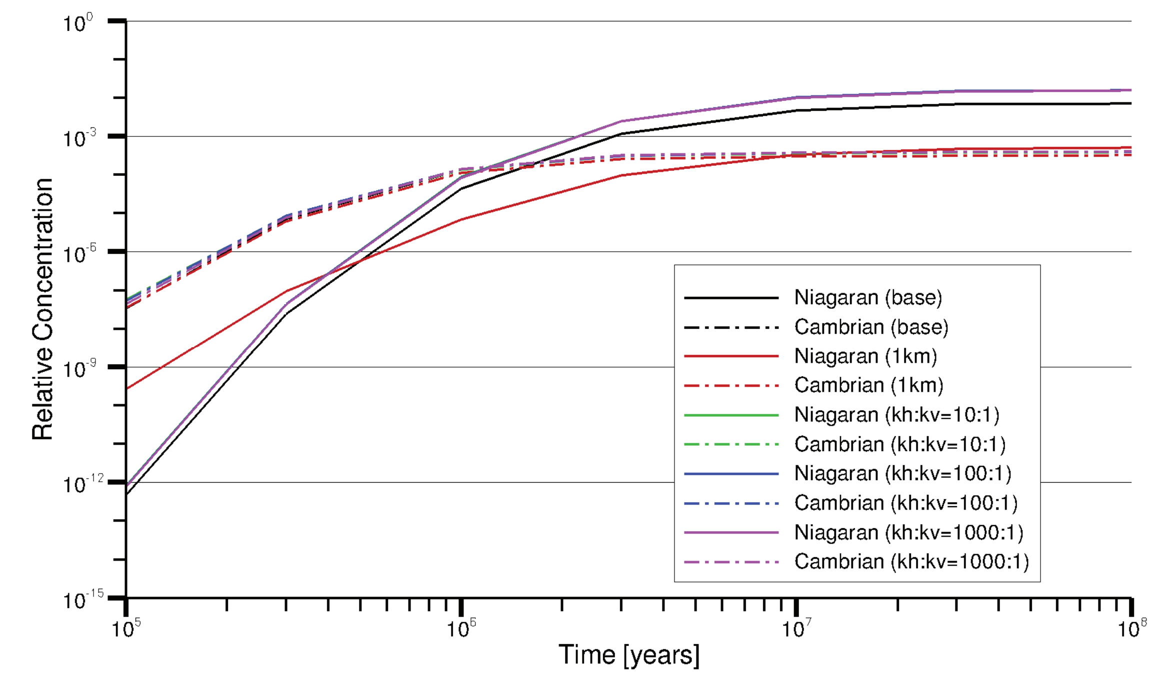
The environmental head profile in Figure 4.102 of the DGSM (NWMO11k) indicates an upward gradient from the Cambrian to the Ordovician and a downward gradient from the Niagaran to the Ordovician. To simulate the evolution of the measured pressure gradient using the site-scale model, the initial heads for each site-scale layer were calculated from the pseudo-equilibrium heads from the sub-gridded regional-scale model by subtracting the difference between the pseudo-equilibrium and measured heads at the DGR-4 borehole for a given layer. The procedure ensured that the horizontal gradients in each model layer of the adjusted model were the same as those calculated for the base-case site-scale model.

Instead of the Dirichlet (prescribed head) boundary condition that was used for the base-case site-scale analysis, the lower Silurian, Ordovician and Cambrian units were assigned a zero-flux Neumann boundary condition. Pressure support for the Niagaran was provided by the Dirichlet boundary condition retained on that unit, while pressure support for the Cambrian was provided by using a Dirichlet boundary condition for all layers of the Precambrian with the freshwater head level being determined by the measured head for the Cambrian in the DGR-4 borehole. It is noted that the Cambrian sandstone is not continuous across the site-scale model domain. The horizontal gradient across the Precambrian was maintained to be that of the base-case site-scale analysis.

One issue investigated was the effect on the evolution of the Trenton Group and Upper Ordovician underpressures of the vertical hydraulic conductivities for the units of the Black River Group, the Shadow Lake, Gull River and Coboconk formations. In addition to the base-case vertical-to-horizontal hydraulic conductivity anisotropy ratio of 0.001 for the Black River Group, ratios of 0.1 and 0.01 were used to assess the sensitivity of the head profile to the anisotropy ratio. Horizontal hydraulic conductivities for the Ordovician units were held constant while vertical hydraulic conductivities were varied to obtain the desired anisotropy.

For all three cases, the results show that a downward gradient from the Niagaran to the Ordovician persisted for over 300 ka. The pressure and related water deficit in the Ordovician was met by approximately 1 Ma. Steady-state pressures were reached by 3 Ma with an upward gradient developing from the Cambrian to the surface. With an anisotropy ratio of 0.1, the overpressurization of the Cambrian propagates quickly through the Black River Group such that the hydrostatic state with minimal vertical hydraulic gradient through these units was reached by 10 ka. For all three cases, the water deficit in the Ordovician was met by very slow influx from the Cambrian and/or the Niagaran Group.

The tracer-breakthrough curves at the Niagaran and the Cambrian for the three alternative anisotropy cases for the Black River Group are plotted in Figure 4-68. Not all curves are visible because they largely overlap. The similarity of the breakthrough curves despite the different head conditions and anisotropies confirms the conclusion that solute transport in the Ordovician is dominated by diffusion and that the impact on solute transport of pore velocity in the deep Ordovician limestones is negligible.



Notes: Initial condition for all but the base-case results is from the measured DGR-4 environmental head profile shown in Figure 4-60. Base-case (Black lines) results used equilibrated steady-state heads as the initial condition. Red lines are for the fracture zone case. Green, blue and purple lines indicate varying vertical-to-horizontal hydraulic conductivity anisotropy ratios. See text for further discussion. From Figure 4.37 of the Hydrogeologic Modelling report (NWMO11p).

Figure 4-68: Tracer-Breakthrough Curves at the Niagaran Group and Cambrian for Various Site-Scale Simulations

The possible presence of faults at the Bruce nuclear site was investigated using 2D seismic reflection surveys as described in Section 4.1.2.3. As discussed, angled boreholes DGR-5 and DGR-6 were drilled/cored through two inferred potential fault structures, but found no evidence of their presence. Consequently, no faults are believed to be present in the vicinity of the proposed DGR. Nevertheless, the impact on both the pressure evolution in the Ordovician limestone and shale and the migration of a tracer from the Cobourg Formation was investigated for cases in which a hypothetical undetected fault connecting the Cambrian sandstone and the Niagaran Group is located at an arbitrary distance from the tracer source grid block. The fault was conceptualized as a vertical discrete fracture zone oriented in the north-south direction at a distance west of the tracer grid block. This provides a conservative analysis, as the impact of a fracture zone east of the site would be lessened by the possible absence of the Cambrian. An equivalent porous medium approach was used to characterize the 2-km-long fracture zone, which was assigned a hydraulic conductivity of 3.0 x10^{-6} m/s and a width of 1 m. The configurations investigated include a hypothetical discrete fracture zone 1 km west of the tracer source zone grid block and a fracture zone 5 km west. Simulations were performed using both the equilibrium-state pressure distribution and the observed environmental head profile at DGR-4 as initial conditions.

The tracer-breakthrough curves at the Niagaran and the Cambrian for the case with a fracture zone at 1 km and the DGR-4 environmental head profile are plotted in Figure 4-68. The effect of the fracture zone is to increase the very low (10^{-12} to 10^{-7}) relative concentrations in the Niagaran before approximately 40 ka, but reduce the later peak concentration to less than 10^{-3}, while tracer breakthrough to the Cambrian is almost unchanged from the other cases considered. A fracture 5 km west of the proposed DGR site was found to be too far from the DGR to have a significant impact on the evolution of tracer plumes in the Cambrian and Niagaran. As for all other cases, solute transport in the Ordovician is dominated by diffusion.

Conclusions from Site-Scale Modelling

The site-scale model was used to investigate the evolution of a conservative tracer plume originating from the proposed DGR site assuming fully saturated conditions. The model results are described below.

- The choice of initial head conditions or anisotropy was found to make no difference in the transport of the tracer—transport in the Ordovician was dominated by diffusion, not advection.

- The effects on tracer transport of permeable faults connecting the Cambrian and Ordovician were also evaluated for both initial pressure conditions. A fault at 5 km from the tracer source had no effect whatsoever on tracer transport, while a fault at 1 km led to tracer migrating from the Cambrian to the Niagaran. For all simulations using the DGR-4 environmental head profile to define initial conditions, a downward gradient from the Niagaran to the Ordovician persisted for over 300 ka years. The pressure and related water deficit in the Ordovician was met by approximately 1 Ma. Steady-state pressures were reached by 3 Ma with an upward gradient developing from the Cambrian to the surface.

- Under water saturated conditions, the head profile through the Ordovician is irrelevant to DGR performance-transport is diffusion dominated. If the assumption of full water saturation is invalid, that is, if a discontinuous gas phase occupies some portion of the Ordovician pore space, diffusion of solutes through the Ordovician will be even slower than shown by the site-scale model because of the phase-dependence of diffusion coefficients.

4.4.4.4 Michigan Basin Cross-Section Model

A cross-section of the Michigan Basin was modelled to investigate the over-pressures measured in the Cambrian beneath the Bruce nuclear site. The objective was to assess whether the geometry of the basin and the salinity distribution could contribute to overpressures in the high-permeability Cambrian at the Bruce nuclear site, sandwiched as it is between the low-permeability Precambrian and low-permeability Ordovician units. No attempt was made to calibrate the model to conditions in the Cambrian or in any other unit; the intent was simply to illustrate how Cambrian overpressures might arise.

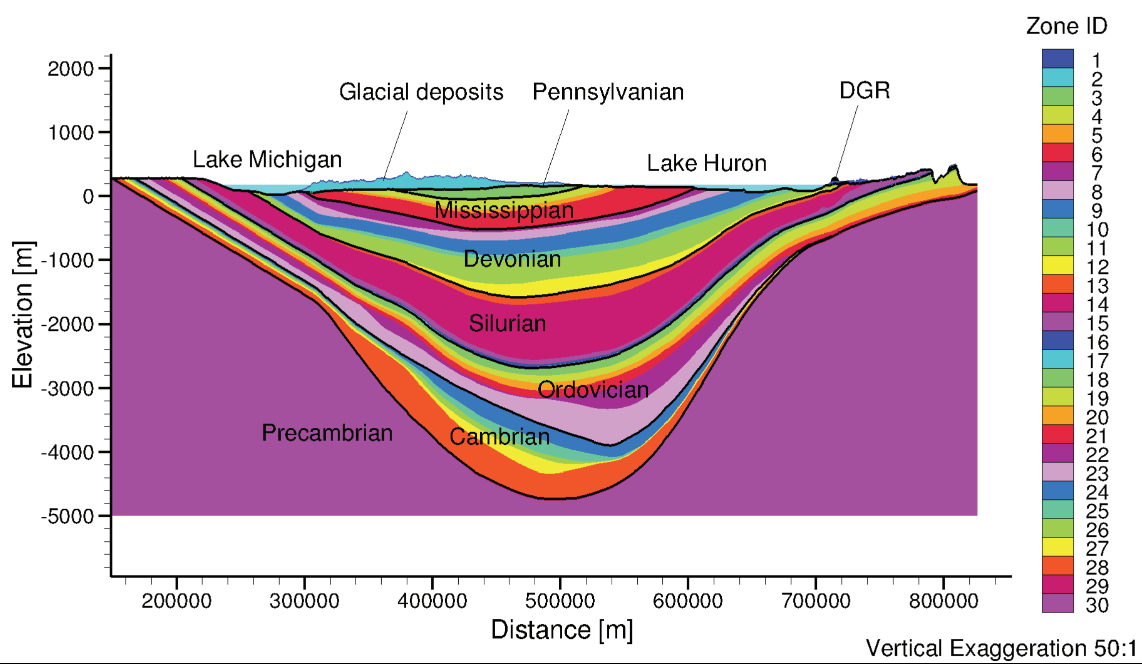
Model Domain and Mesh Generation

The Michigan Basin cross-section modelling domain extends laterally from southwestern Ontario to Wisconsin across Lake Huron, the State of Michigan, and Lake Michigan, a distance of approximately 677 km (Figure 4-69; also see Figure 2.6 in NWMO11m). The vertical elevations range from approximately -5,000 m depth at the lowest point in the Precambrian to 509 m at the highest point on the Niagara Escarpment. The Cambrian sandstone outcrops in Wisconsin and is absent at the Algonquin Arch. The Cambrian also outcrops in the upper peninsula of Michigan and north of Sault Ste. Marie, Ontario.

The domain under zero mASL, where a density-dependent flow simulation was necessary due to the high salinity in the Michigan Basin groundwater system, was finely discretized into a planar hexahedral mesh with 1,355 columns, 600 rows, and 1 block in thickness to create a vertical two-dimensional mesh. These hexahedral elements have sides of 500 m in the horizontal direction by 10 m in the vertical direction by 1 m in thickness. The non-orthogonal mesh above sea level has 100 evenly distributed layers with 1,355 nodes each. The elevation of the nodes for each layer were determined from the geological framework model. Given the fact that the continuity of each geologic unit was strictly maintained, 30 stratigraphic units for the Michigan Basin cross-section were mapped to the mesh (Figure 4-69).

Flow Boundary and Initial Conditions

The eastern boundary of the domain is the water divide for the surface water system, and conceptualized as a Neumann no-flow boundary condition. The western boundary roughly corresponds to the surface water divide between Lake Michigan and the Mississippi River in Wisconsin, and can also be conceptualized as a Neumann no-flow boundary condition. The bottom of the Michigan Basin cross section is in Precambrian granitic gneiss with very sparse fractures (NWMO11k). Therefore, a Neumann no-flow boundary condition was assumed for the bottom of the model. The elevations of the nodes at the top of the model domain are defined by either the digital elevation model or the lake bathymetry. For surface nodes including those occupied by Lake Huron and Lake Michigan, the assigned prescribed head was set as the elevation minus 3 m, but not less than the 176 m Lake Huron and Lake Michigan water elevation. The imposed surface boundary condition permits recharge and discharge to occur as determined by the surface topography and the hydraulic conductivity of the top model layer. The assigned head represents a water table occurring at an assumed depth of 3 mBGS.



Note: From Figure 6.3 of the Hydrogeologic Modelling report (NWMO11p).

Figure 4-69: Stratigraphic Zones for the Michigan Basin Cross-Section Model

Hydraulic and Transport Parameters

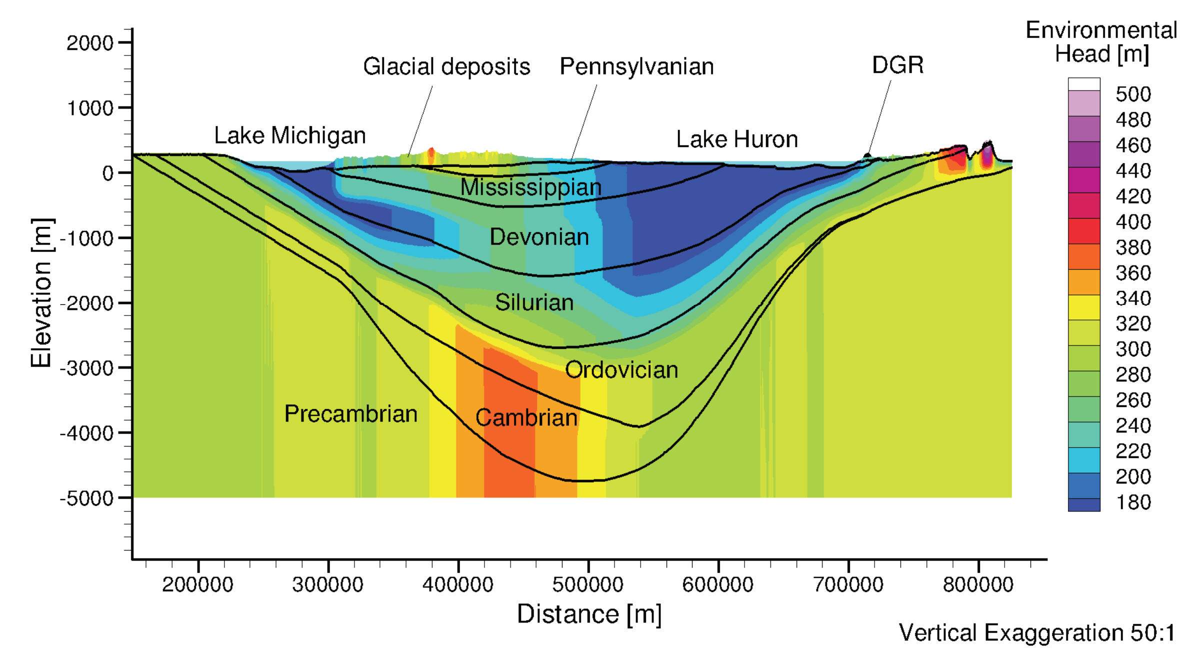
The base-case data set for the conceptual model consists of 30 model layers (Figure 4-69), with each hydrostratigraphic layer corresponding to a unit in a stratigraphic section with associated hydraulic conductivities, anisotropy ratios, porosities, and specific storage values (NWMO11p). For those geologic units existing at the Bruce nuclear site, the hydraulic parameter values were inherited from the regional-scale model. The variation of hydraulic conductivity in the Precambrian with depth was calculated using the relationship of Normani (NORMANI09). Some other Michigan Basin geologic units, such as the Saginaw, Marshall, Ancell Group, and Prairie du Chien, pinch out to the west of the proposed DGR and are therefore absent in the 3DGF model (NWMO11aa). Their values were either derived from the literature or estimated by appropriate assumptions (NWMO11p).

TDS

Salinity plays an important role with regard to a density-dependent groundwater system. For the base-case scenario, the initial prescribed TDS distribution was developed from data presented in published literature (FRAPE87, HANOR79 and LAMPE09) which is discussed in the Hydrogeologic Modelling report (NWMO11p). An equilibrium head solution was reached by allowing an initial freshwater head distribution to equilibrate to the defined TDS distribution for 10 Ma (Figure 4-70); no change was observed in the equivalent freshwater head distribution after this time.

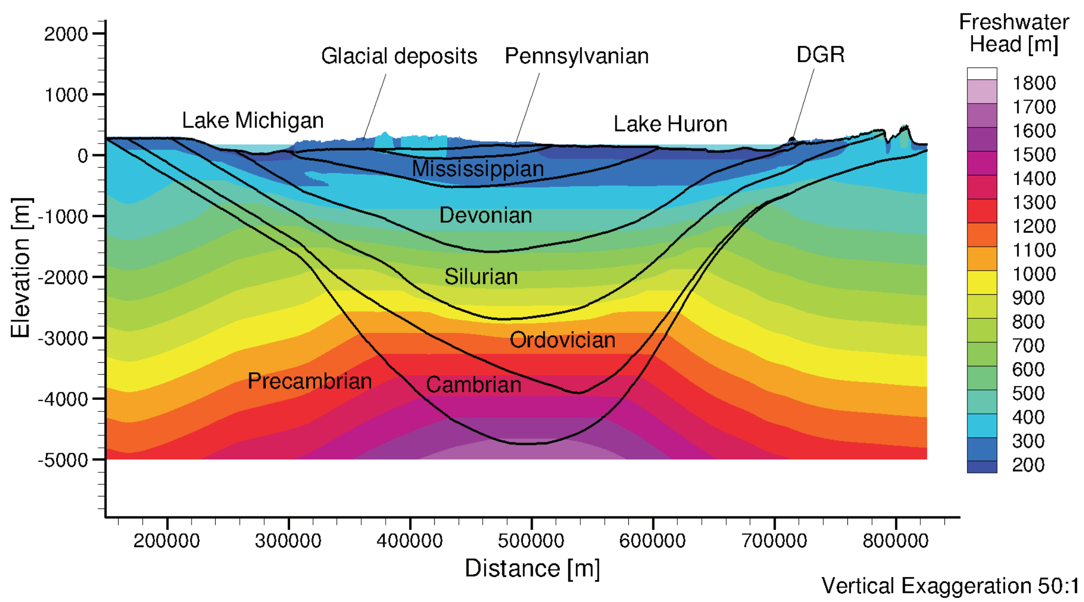
Results from Michigan Basin Cross-Section Model

The environmental head distribution presented in Figure 4-70 shows that the gradient that controls vertical flow is effectively non-existent in the Cambrian and high-permeability Lower Ordovician units in the central portion of the basin, and is upward through the lower permeability Ordovician units. Figure 4-71 shows that the freshwater head gradient controlling horizontal flow is also effectively nonexistent in the Cambrian.



Note: From Figure 6.8 of the Hydrogeologic Modelling report (NWMO11p).

Figure 4-70: Equilibrium Environmental Heads for Defined TDS Distribution



Note: From Figure 6.7 of the Hydrogeologic Modelling report (NWMO11p).

Figure 4-71: Equilibrium Freshwater Heads for Defined TDS Distribution

The density-dependent simulation resulted in a calculated equivalent freshwater head in the Cambrian at the location of the DGR of 472.6 m and a calculated environmental head of 305.3 m. The August 24, 2009 measured freshwater head and estimated environmental head in the Cambrian at the DGR-4 borehole are 422.1 m and 317.6 m respectively. An upward environmental head gradient is predicted in the analysis. Relative to the ground surface at 181.6 mASL, the measured overpressures in the Cambrian are reconstructed by the Michigan Basin cross-section model.

Conclusions from the Michigan Basin Cross-Section Analyses

The Michigan Basin cross-section analyses indicate that the overpressures in the Cambrian can be attributed to topography, the spatial distribution of fluid density and the geometry of the various stratigraphic layers in the Michigan Basin.

4.4.4.5 1D Two-Phase Model

A one-dimensional two-phase air-water analysis was performed using TOUGH2-MP (PRUESS99) to determine whether or not the presence of a free gas phase could lead to a non-hydrostatic pressure profile between the Guelph and Cambrian formations. This study was motivated by the pressure profiles defined by straddle-packer testing and Westbay monitoring, as shown, for example, in Figure 4.102 of the DGSM

(NWMO11k). The primary feature of interest shown by these profiles is underpressures in the Upper Ordovician and Trenton Group strata. A secondary feature of interest is apparent discontinuities in the pressure profiles, as isolated intervals of normal pressure or overpressure are sandwiched between underpressured intervals.

The scope of the two-phase air-water analysis was limited to demonstrating the effects that the presence of a gas phase in the Ordovician sediments (and a conductive feature) could have on water-phase pressures. The two-phase modelling included an attempt to determine whether or not localized overpressures could be related to the presence of gas-containing features having different two-phase flow properties from the surrounding rock. A detailed sensitivity analysis was not performed. For all simulations, the measured Westbay pressure and estimated head profiles from August 2009 for the DGR-4 borehole, shown in Figure 4.102 of the DGSM (NWMO11k), provided a qualitative comparison to the model results—no attempt was made to calibrate the model to the DGR-4 data by parameter adjustment.

Two-phase properties and characteristic curves were used in the simulations (NWMO11p). The base-case scenario for the TOUGH2-MP modelling assumed that a uniform gas saturation was present in all strata from the Gasport to the Coboconk (inclusive) as an initial condition. The evolution of the saturation profile and associated gas and water pressure profiles was modelled for 4 Ma while gas saturations in the Cambrian and Guelph were held at zero. An alternative scenario was modelled in which the initial gas saturation in all units was zero, gas was introduced uniformly for 200 ka from the Coboconk to the Queenston inclusive, and the system evolved for another 800 ka. A variant on both scenarios included a thin zone at a depth of 585 mBGS in the Georgian Bay Formation having a different capillary pressure versus saturation curve than the rest of the formation.

For the base-case simulations, the initial gas saturation for the units between the Coboconk and the Gasport was set to 0.17, resulting in an initial water saturation of 0.83. The Gull River and Shadow Lake were assumed to have an initial gas saturation of zero. The initial saturations were used to determine the capillary pressure within a formation. The initial water pressure was specified to account for hydrostatic conditions in the Guelph Formation, and hydrostatic conditions with 120 m overpressure in the Gull River and Shadow Lake formations. Initial water pressures were set to zero between the Guelph Formation and the Gull River Formation. The initial gas pressure was calculated from the water pressure minus the capillary pressure (NWMO11p).

For the alternative scenario, the initial gas saturation was set to zero for all units, resulting in an initial water saturation of 1.0. The initial water pressure was defined by a linear profile between the pressure measured in the Cambrian sandstone in DGR-4 and the pressure measured in the Guelph at DGR-4. The boundary conditions, properties and parameters for the analyses were identical to those used for the base-case simulations.

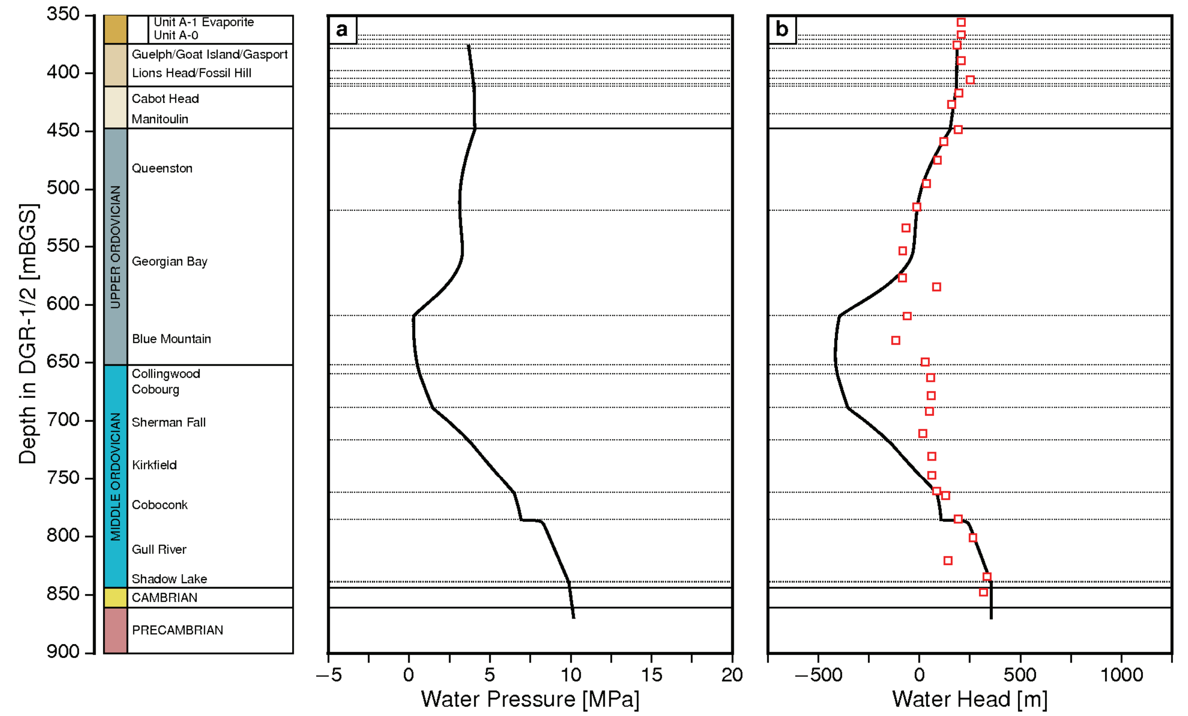
To determine whether or not localized overpressures could be related to conductive layers having different two-phase flow properties from the surrounding rock, separate TOUGH2-MP simulations were performed for both the base-case and alternative scenarios that included a thin zone (termed a "fracture" by NWMO11k and NWMO11p) with different two-phase properties in the Georgian Bay Formation at a depth of 585 m, represented using a single block with a height of 0.5 m. The initial saturations and initial water pressures were identical to those assumed for the cases lacking this feature.

Base-Case Simulations

For the base-case scenario in Figure 4-72 the water pressure and water head at 400 ka are shown. The pressure profiles are still evolving at this stage in the simulation. In Figure 4-72a, the water head is negative within the middle formations and remains overpressured in the Gull River and Shadow Lake formations. Figure 4-73 shows that the water pressure and water head had increased by 1.25 Ma. Other model results showed that this increase was related to dissipation of the gas phase. The dissipation of the gas phase occurs as a result of both gas transport as a separate phase from the domain as well as partitioning of the water vapour and air phases from the gas to the water phase and then diffusion in the solution phase to the bounding layers (Guelph and Cambrian).

Migration of the gas phase is sensitive to the relative permeability versus saturation curves for both the water phase and the gas phase, while diffusion of the air in the water phase is sensitive to the diffusion model used in the analysis. With the diffusion model used, the gas phase has completely dissipated by 3 to 4 Ma; alternative diffusion models might allow either more rapid dissipation or slower dissipation. Regardless, the results for the water head at 1.25 Ma as shown in Figure 4-73 indicate that underpressures in the Ordovician sediments could be related to the presence of a gas phase.

Including a fracture in the base-case simulations at 585 mBGS altered the pressure profile. The effects of the fracture on pressure at 300 ka are seen in Figure 4-74, with the discontinuity created by the fracture clearly evident. The water pressure in the fracture feature could be adjusted by choosing a different capillary pressure versus saturation curve for the fracture, but no attempt was made to adjust either the capillary pressure versus saturation curves or the relative permeability versus saturation curves in order to yield a better comparison between the modelled results and the measured pressures in the DGR-4 borehole. Gas pressures, but not saturations, are continuous throughout the formation. The fracture feature exhibits a high gas saturation and high water pressure relative to the adjacent Georgian Bay Formation. The profiles changed only slightly as the modelling period was extended from 300 to 500 ka (NWMO11p).



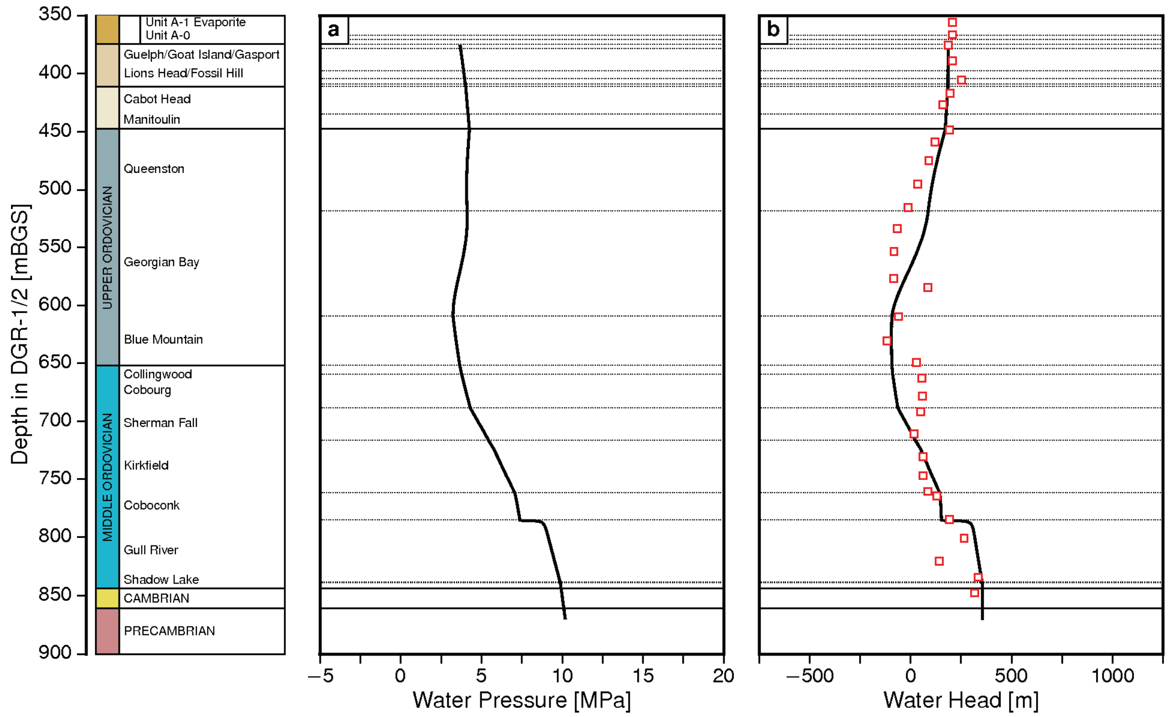
Notes: (a) Water pressure, (b) Freshwater head with posted August 24, 2009 measurements (□) from DGR-4. From Figure 6.24 (top two panels) of the Hydrogeologic Modelling report (NWMO11p).

Figure 4-72: Two-Phase Flow Analysis at 400 ka for the Base-Case Scenario

<u>Alternative Scenario</u>

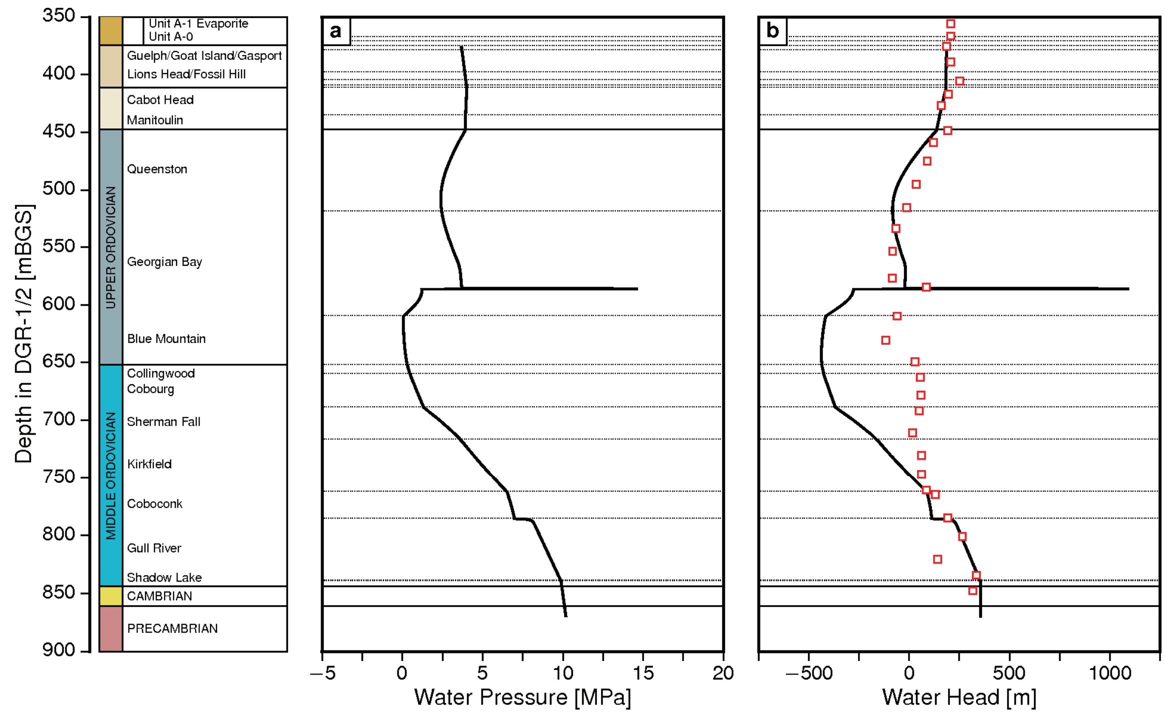
For the alternative scenario, air was introduced uniformly for 200 ka from the Coboconk to the Queenston inclusive to provide a temporary gas source. The total amount of air introduced per unit length of rock was assumed to be 98% of the air that would be contained in a volume of rock with a water saturation of 95% and a gas saturation of 5% (NWMO11p). Alternate gas generation rates were not investigated in this study.

After time zero, air generation greater than the amount that can be accommodated by the pore compressibility results in increased water pressures in the Ordovician. The water that is being displaced by the gas phase migrates from the domain under the resulting efflux water gradients. The air in the gas phase partitions into the water phase and migrates from the domain through diffusion in the solution phase. The air also migrates from the domain as a separate phase. After gas generation ceases, the dissipation of both the displaced water phase and the air phase results in a decrease in the water pressure, eventually resulting in underpressures. The water pressure is sensitive to the pore compressibility while the pore volume is sensitive to the high air entry gas pressure for the capillary pressure versus saturation curves for the Ordovician rock. The underpressures that develop at 1.0 Ma are shown in Figure 4-75. The water pressures compare favourably with the measured pressures in the DGR-4 borehole. Continued diffusion of air in the solution phase results in the gradual dissipation of the air phase and a return of the water pressures to a hydrostatic state. The rate of return is sensitive to the diffusion coefficient. Thus, generation of a gas phase can result in the development of underpressures in the water phase that may persist for hundreds of thousands of years, if not longer periods of time.



Notes: (a) Water pressure, (b) Freshwater head with posted August 24, 2009 measurements (□) from DGR-4. From Figure 6.26 (top two panels) of the Hydrogeologic Modelling report (NWMO11p).

Figure 4-73: Two-Phase Flow Analysis at 1.25 Ma for Base-Case Scenario



Notes: (a) Water pressure, (b) Freshwater head with posted August 24, 2009 measurements (□) from DGR-4. From Figure 6.28 (top two panels) of the Hydrogeologic Modelling report (NWMO11p).

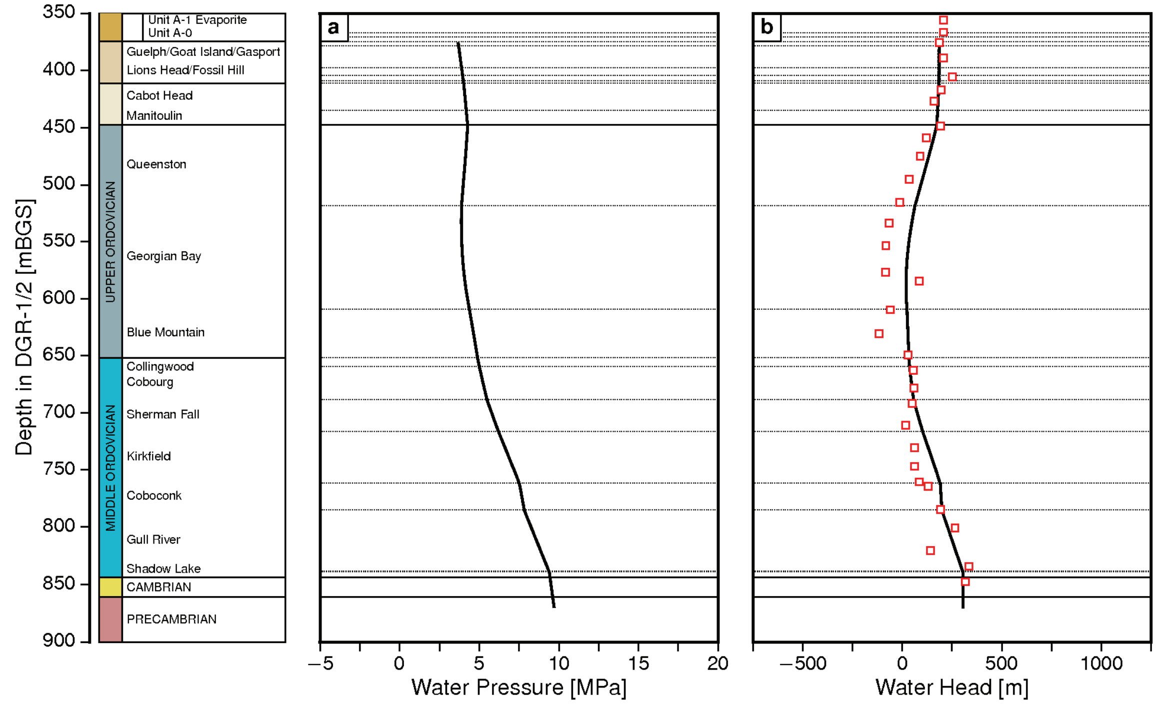
Figure 4-74: Two-Phase Flow Analysis at 300 ka with a Fracture Zone at 585 mBGS

Including a fracture at 585 mBGS in the model with air generation described above results in the pressure distributions shown in Figure 4-76 at 1.0 Ma. Compared to the pressure distributions without the fracture (Figure 4-75), the primary difference is an offset toward higher water pressures at the elevation of the fracture. Notably, this offset is maintained below the fracture.

Conclusions from 1D Two-Phase Modelling

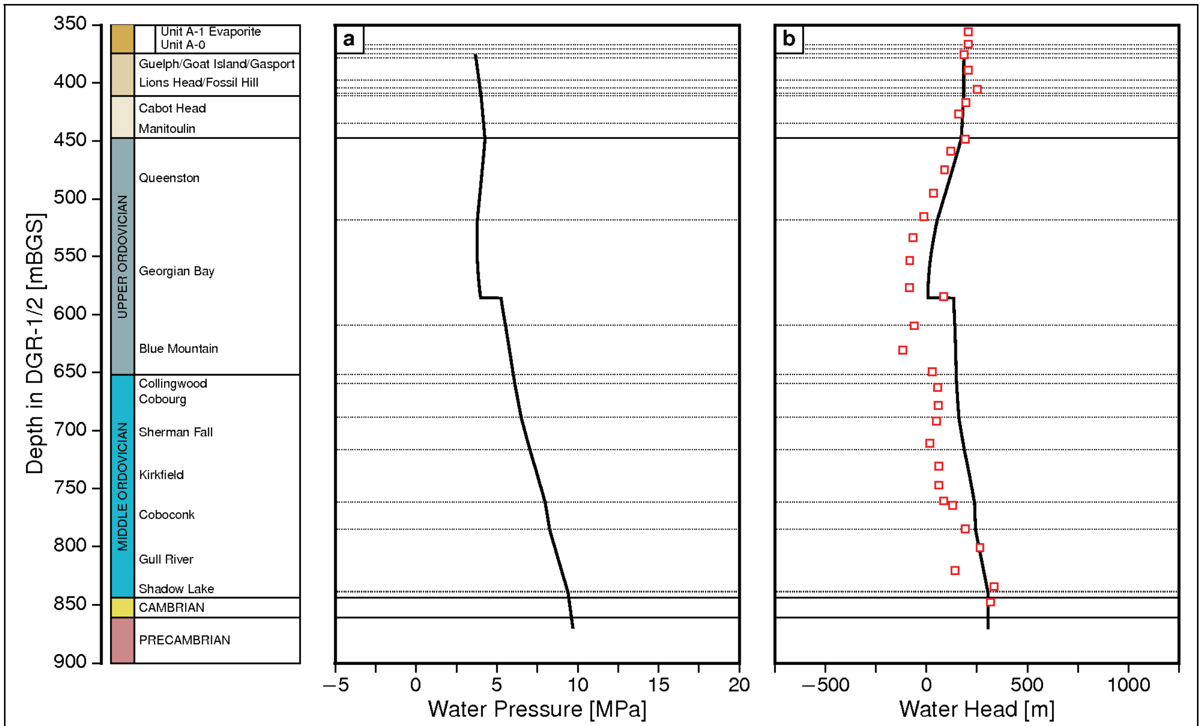
The 1D two-phase modelling shows that water-phase underpressures, such as those observed in the Ordovician rock in the DGR boreholes, can be caused by the presence of a gas phase. The modelling shows that water pressure is sensitive to the rock-dependent capillary pressure versus saturation relationships. The results demonstrate that gas saturations vary significantly throughout the rock column. The modelling also shows that significant discontinuities in the phase saturations can occur at the boundaries between formations having different two-phase properties, as well as at heterogeneities in the rock mass such as fractures. Fractures may also have a much higher water pressure than the surrounding rock.

From a solute-transport perspective, higher gas-phase saturation and lower water-phase saturation in a fracture as compared to the adjacent rock will result in a reduction of the water-phase diffusion in the fracture through its dependence on the water-phase saturation. This implies that water-phase diffusion can be significantly reduced as a result of the presence of zones in the rock with higher gas saturation.



Notes: (a) Water pressure, (b) Freshwater head with posted August 24, 2009 measurements (□) from DGR-4. From Figure 6.33 (top two panels) of the Hydrogeologic Modelling report (NWMO11p).

Figure 4-75: Two-Phase Flow Analysis at 1 Ma with Air Generation



Notes: (a) Water pressure, (b) Freshwater head with posted August 24, 2009 measurements (□) from DGR-4. From Figure 6.35 (top two panels) of the Hydrogeologic Modelling report (NWMO11p).

Figure 4-76: Two-Phase Flow Analysis at 1 Ma with a Fracture Zone at 585 mBGS and Air Generation

4.4.4.6 Conclusions from Hydrogeological Assessment

Regional-Scale Base-Case Model

- The regional-scale base-case model represents an equilibrium state condition toward which the present-day system may be evolving. In all cases, the MLE for solutes originating at the proposed DGR location is significantly greater than 10 Ma, providing a robust demonstration that a DGR in the Cobourg Formation can effectively isolate radionuclides for any period of concern (NWMO11p).

Site-Scale Model

- The choice of initial head conditions or anisotropy makes no difference in the transport of a conservative tracer through the Ordovician—that transport is dominated by diffusion, not advection.

- The presence of nearby permeable faults would not alter these conclusions. If the assumption of full water saturation is invalid, that is, if a gas phase occupies some portion of the Ordovician pore space, diffusion of solutes through the Ordovician will be even slower than shown by the site-scale model because of the phase-dependence of diffusion coefficients (NWMO11p).

- For all simulations using the DGR-4 environmental head profile to define initial conditions, the site-scale model showed that a downward gradient from the Niagaran to the Ordovician persisted for over 300 ka. The pressure and related water deficit in the Ordovician was met by approximately 1 Ma. Steady-state pressures were reached by 3 Ma with an upward gradient developing from the Cambrian to the surface, consistent with the results of the regional-scale base-case model (NWMO11p).

- The paleoclimate scenarios showed that glaciation would affect heads only in the Silurian and Upper Ordovician. None of the paleoclimate scenarios produced Upper and Middle Ordovician underpressures like those observed at DGR-4, nor could any reasonable parameter variations. Thus, the Ordovician underpressures that are observed do not appear to be the result of glacial loading and unloading.

- None of the alternative scenarios showed recharge water penetrating below the upper Salina, or a different distribution of TDS in the system from the base-case scenario. Diffusion remained the dominant transport mechanism in the Ordovician in all scenarios (NWMO11p).

Michigan Basin Cross-Section Model

- Cambrian overpressures result from the spatial distribution of fluid density and the geometry of the various stratigraphic layers in the Michigan Basin (NWMO11p).

1D Two-Phase Model

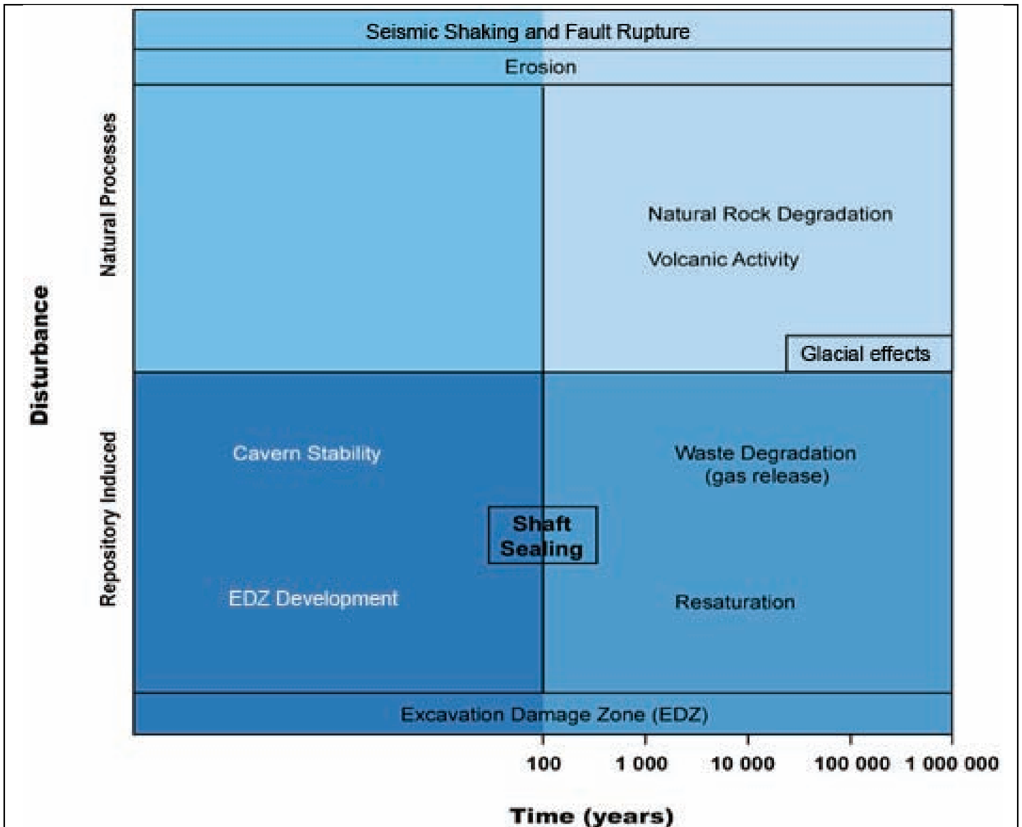
- Water-phase underpressures, such as those observed in the Ordovician rock in the DGR boreholes, can be caused by the presence of a gas phase. Water pressure is sensitive to the rock-dependent capillary pressure versus saturation relationships, and gas saturations should not be expected to be continuous but may vary significantly throughout the rock column. The most significant effect of a separate gas phase is to reduce the rate of diffusion through the Ordovician, further contributing to the safety of the DGR (NWMO11p).

Some of the key findings of the work, analyses and interpretations of the hydrogeologic modelling study are summarized in the following points.

- The deep groundwater system is isolated; it is unaffected by surface perturbations.
- The permeability of the Ordovician sediments is extremely low. This is a necessary requirement for the existence of the abnormal pressures and high gradients observed in the DGR boreholes.
- The sedimentary sequence at the DGR site provides multiple barriers in both the deep and intermediate zones; for simulations that honour the site data, solute transport in the Ordovician layers is diffusion dominant as is transport in the low permeability lower and middle Silurian sediments.
- The calculated fluid velocities in the Ordovician layers are extremely low and primarily vertical.
- There is no evidence to support the existence of permeable connected pathways, proximal to the proposed DGR site, through the sedimentary sequence of the deep groundwater zone; the presence of permeable pathways is inconsistent with the abnormal pressures measured in the DGR boreholes.
- A solute released from the horizon of the proposed DGR in the Cobourg Formation would migrate by diffusion through the Ordovician sediments to the overlying Niagaran Group and/or to the thin underlying Cambrian layer.
- Based on density-dependent saturated analyses, it will take more than 3 Ma for the observed underpressure in the Ordovician limestone and shale at the DGR site to equilibrate to the over-pressures observed in the underlying Cambrian sandstone and the overlying Niagaran Group.
- The abnormal pressures observed in the DGR boreholes could not be explained by paleoclimate analyses that use appropriate parameters, boundary conditions and glaciation/deglaciation scenarios.
- The underpressure in the Ordovician limestone and shale can be explained by the presence of a non-wetting immiscible gas phase in the rock and two-phase air and water analyses.
- The measured discontinuities in Ordovician water pressures can be explained by the presence in the rock of a non-wetting gas phase and layers with different two-phase flow properties.

4.5 Future Evolution of the Bruce Nuclear Site

The evolution of the Bruce nuclear site over the next 1 Ma depends on which natural and repository-induced processes will factor into its evolution (Figure 4-77).



Note: From Figure 6.1 of the Geosynthesis (NWMO11c).

Figure 4-77: Factors Influencing the Future Evolution of the DGR

Natural geological processes, which need to be considered in this capacity, were they to affect the Bruce nuclear site are shown in Figure 4-77.

- Glaciation, including studies of glacial erosion, glacial loading and permafrost development and associated changes in groundwater recharge.

- Other natural geologic processes such as seismicity, fault rupture/reactivation and volcanism.

- The concentrated occurrence of natural resources also involves natural processes. If these occurrences exhibit economical viability the potential for future human intrusion exists.

The second part of this section looks at repository-induced processes, as well as the effects that the natural processes listed in the first two bullets may or may not have on the proposed repository. This discussion is organized around the:

- Shaft seal analysis; and

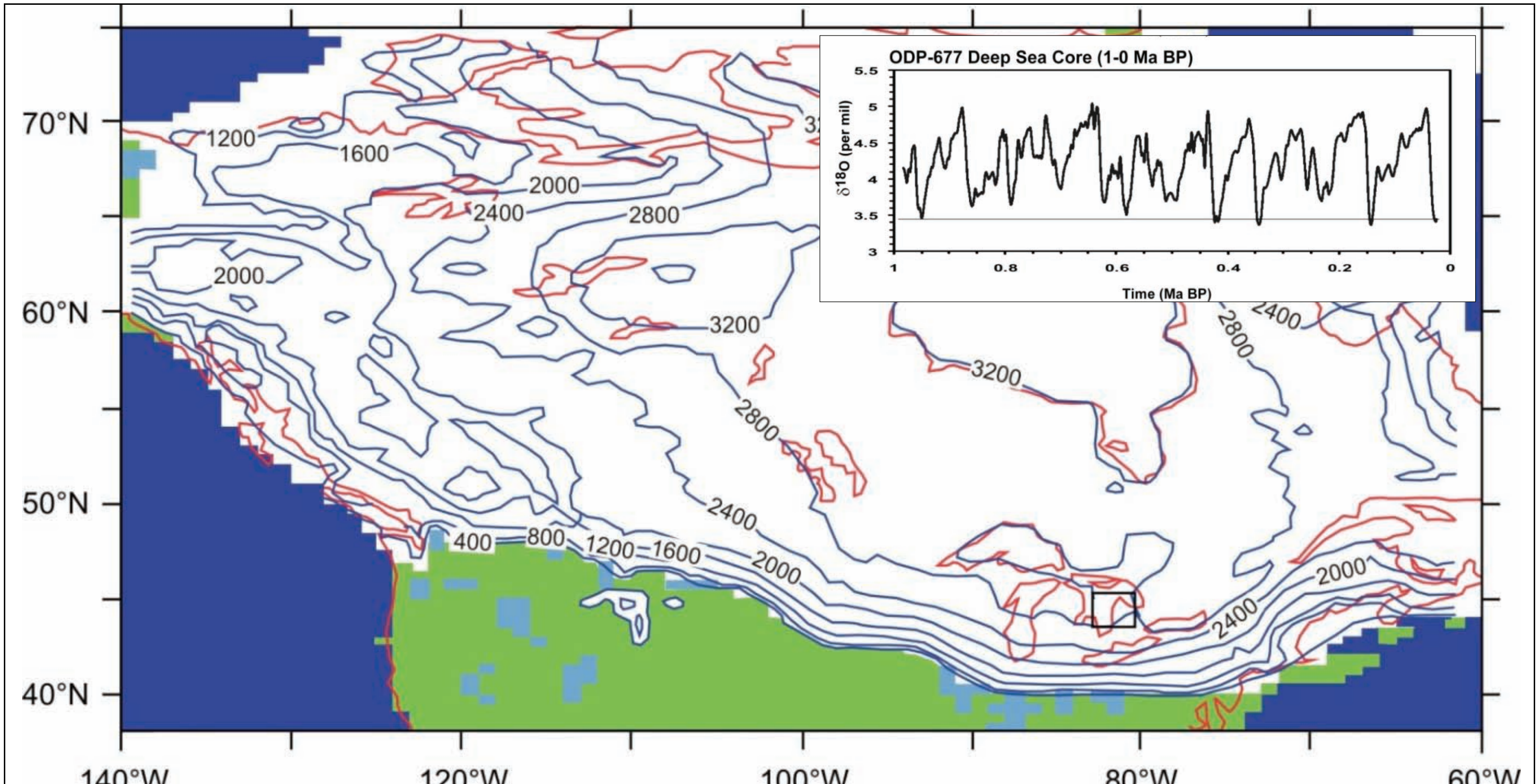
- Long-term cavern and pillar stability analysis.

4.5.1 Glaciation

The Quaternary Period represents the last 2.588 million years of geologic history. During the last approximately 1 Ma of this period the North American continent has endured nine cycles of glaciation occurring approximately every 100 ka (NWMO11r). Glacial cycles occur due to small changes in effective solar insolation caused by the changing geometry of Earth's orbit around the Sun over time. Evidence of these cyclical glacial episodes are observed by the fluctuation in oxygen isotope concentration in deep sea core samples as shown in the inset of Figure 4-78 (see also Figure 2.1a of NWMO11r). In each of the nine cycles, the glaciation phase has lasted approximately 90 ka and the deglaciation phase approximately 10 ka. If a reglaciation of the Canadian land mass should occur again in the future, such an event is most likely to begin approximately 60 ka from present or later, depending on the CO_2 levels at that time (NWMO11r).

The Long-Term Climate Change report provides a detailed account of the glaciation process and a series of constrained numerical simulations using the UofT GSM to predict future glacial conditions as they may affect the Bruce nuclear site (NWMO11r).

Quaternary sediments and the altered landscape and physiography in the regional study area remain as physical evidence of past glaciations (CHAPMAN84; see also Figure 2.16 of NWMO11c). The Quaternary overburden at the Bruce nuclear site has been discussed briefly in Section 4.1.2.1 (see also NWMO11k).



Notes: Last glacial maximum was ~25 kaBP (NWMO11r). Contour interval in metres. From model nn9921 (NWMO11r) and reproduced here from Figure 2.17 of the Geosynthesis (NWMO11c). Inset shows glacial advance-retreat cycles for the past 1 Ma as indicated by the saw-tooth patterned oxygen isotope curve (Figure 2.1a of NWMO11r).

Figure 4-78: Laurentide Ice Sheet Thickness at Last Glacial Maximum

During the most recent glacial episode, which began approximately 120 kaBP, the Late Pleistocene Laurentide Ice Sheet (LIS) developed in the Arctic and advanced over most of Canada into the United States (NWMO11r). At Last Glacial Maximum (LGM), approximately 25 kaBP, the LIS surpassed 2,800 m in thickness over the most glaciated regions of the continent (Figure 4-78). Within the Great Lakes region, as the ice sheet retreated 14 kaBP, glacial melt waters from the retreating ice filled erosional depressions that evolved into the modern day Great Lakes Basin.

Climate change is a natural phenomenon that has been shown to occur over geologic time (NWMO11r). More recently, climate change has been shown to be forced by an increase in carbon dioxide (CO_2) levels in the atmosphere from combustion of fossil fuels. For the next century, the earth will be subjected to warming of its mean surface temperature as a consequence of increasing greenhouse gas concentrations. Steps taken today to reduce greenhouse gas emissions will not see meaningful results for decades to come. If the CO_2 levels are not reduced, then the onset of the next glacial cycle could be delayed.

The relevant characteristics of the glaciation process are discussed below in terms of glacial erosion, glacial loading and permafrost formation and groundwater recharge.

4.5.1.1 Glacial Erosion

A glacial erosion assessment report for the Bruce nuclear site (NWMO11u) looked at several independent types of geological evidence in order to assess the magnitude of total erosion which would likely occur over one glacial cycle, including:

- Historical and recent regional estimates of Quaternary erosion associated with the LIS;
- Empirical studies of glacial erosion on bedrock and sediment substrates in diverse settings and at different scales;
- Physical examples of glacial erosional processes in the regional study area;
- Extreme cases of erosion by ice, catastrophic glacial outburst floods and subglacial meltwater;
- Theoretical considerations of glacial erosion and their application in a model of erosion by the LIS; and
- The occurrence of sediment cover over the bedrock.

Relevant results from the UofT GSM (NWMO11r). The analysis estimates that total erosion for one glacial cycle ranges from ~200 m, the largest and most extreme amount, to a few meters, and perhaps no erosion and net deposition. In view of the absence of topographic features or other known factors that would tend to localize erosion by ice or water over the Bruce nuclear site, and the absence of evidence of preferential past erosion over the site, a realistic but still quite conservative site-specific estimate of total erosion extrapolated to the 1 Ma timeframe is 100 m (NWMO11u). Thus, glacial erosion is not expected to affect the DGR at a present-day depth of 680 mBGS.

4.5.1.2 Glacial Loading

Climate modelling of the LIS describes ice thicknesses of up to 2.8 km in southern Ontario during the LGM (Figure 4-78) (NWMO11r). This mass of ice significantly increased the normal stress as it moved across the proposed Bruce nuclear site. Such movement resulted in loading and unloading cycles on the underlying rock with every major ice-sheet advance and retreat.

The maximum crustal depressions from the equilibrium level occurred at LGM 25 kaBP and reached values in excess of 600 m (NWMO11r). After the ice retreated, the earth's surface has rebounded due to a process known as isostasy that is still occurring today. In the Great Lakes area, the continental isostasy contour represents zero with increasing uplift to the north of about 1.5 mm/a and subsidence to the south at about the same rate, thus indicating that the continent is tilting slightly upward in the north (NWMO11r).

Glacial loading causes the vertical and horizontal stresses at depth to increase. The horizontal stress increases due to both Poisson's effect and plate bending. During the LGM, it is calculated that the vertical stress increased by approximately 30 MPa and the horizontal stress by 2 MPa. Glacially induced shear stresses, which typically occur close to the surface along the glacial margins, were not considered in the analyses because previous work showed that these shear stresses are relatively minor compared to the vertical and horizontal normal stresses (LUND09).

4.5.1.3 Permafrost Formation (Changes in Groundwater Recharge)

Permafrost formation during a cycle of ice-sheet advance and retreat is a determinant of the extent to which water generated by the melting of a continental-scale ice sheet may infiltrate the subsurface. When permafrost exists, it inhibits flow to depth, whereas if the glacier is temperate such that permafrost decays beneath the insulating ice-sheer then enhanced recharge can occur. Modelling has shown that permafrost at the Bruce nuclear site seldom reached more than 60m depth (NWMO11r).

Glacial meltwater beneath continental ice sheets can be pressurized to achieve freshwater hydraulic heads far in excess of ambient heads during interglacial periods. These conditions have been effective in causing recharge of glacial meltwater to depths of several hundred metres in Paleozoic aquifers around the periphery of the Illinois and Michigan basins (MCINTOSH05, MCINTOSH06, PERSON07). Groundwater chemistry indicates that glacial meltwater may have entered the Devonian and Silurian formations above the Salina F Unit shale at the Bruce nuclear site. The low permeability of the deeper Silurian units provides a barrier to further infiltration of glacial meltwater. Modelling results presented in Section 5.4 of the Geosynthesis (NWMO11c) suggest that neither the presence nor the lack of permafrost below a glacier overlying the study area would impact or alter the intermediate to deep hydrogeologic system. Thus, the DGR will not be affected by any changes in groundwater recharge that might occur due to permafrost formation.

4.5.2 Geologic Disturbances

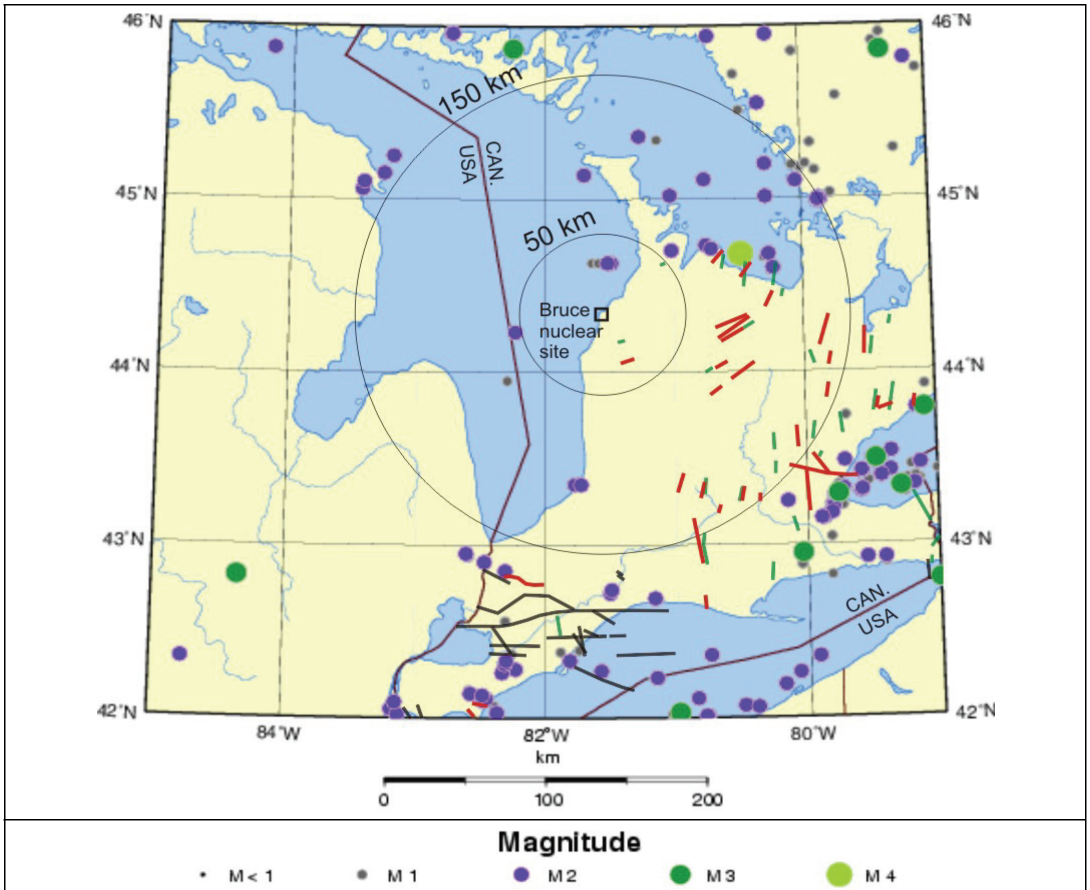
Natural geologic evolution of landmasses takes place over many millions or even billions of years. The last major orogeny to occur with effects in southern Ontario was the Alleghenian Orogeny some 250 Ma ago (see Section 4.1.1.2 and Figure 4-7). Since that time, southern Ontario has been tectonically stable with only mass wasting and glacial processes taking place. The following are three natural geologic processes that may affect the DGR over the next million years: seismicity, fault rupture/reactivation, and volcanism.

4.5.2.1 Seismicity

The regional study area is within the tectonically stable interior of the North American continent in a region characterized by low rates of seismicity. Figure 4-79 shows all known earthquakes in the region between 1985 and 2010, overlain with the mapped faults in southern Ontario as shown in Figure 4-4. Most recorded events have a magnitude of less than M3 (Nuttli Magnitude, which is the primary local magnitude scale used for reporting in the region), with rare occurrences of larger events within a 150 km radius from the Bruce nuclear site. Twenty-six events have been detected in this region since 1952 with a maximum magnitude of 4.3 (M4.3) and a focal depth of about 11 km measured 99 km northeast of the Bruce nuclear site (15 km north of Meaford, Ontario) (HAYEK10). The historical record is considered to be relatively complete for events of about M > 3.5. It has become more complete for lower magnitude events over the last 10 years owing to the increased station density in the region (HAYEK08).

To improve the detection of the local pattern of low-level seismicity, three highly sensitive borehole seismometer stations were installed within an approximate 40 km radius of the Bruce nuclear site during the summer of 2007, allowing the threshold for detection to be further lowered to M1.0. An objective of this new array is to capture microseismic events in the immediate area for the delineation of seismogenic features deep in the bedrock. The data collected since installation suggests that the regional study area experiences sparse seismic activity and there are no major seismogenic features or active faults of concern (Figure 4-79).

The above interpretation is supported by a recently completed remote-sensing and field-based assessment of neotectonic features and landforms within 50 km of the Bruce nuclear site (NWMO11v). The study looked at Quaternary land forms and soil exposures for evidence of potential neotectonic features such as soft sediment disturbance, paleoliquefaction and other features such as offset beaches. The study found no evidence for neotectonic activity post-dating the most recent glacial cycle within the area of investigation (NWMO11v).



Notes: Fault data is compiled in ARMSTRONG10. All seismic events plotted in local magnitude (M=Nuttli Magnitude). The circles around the Bruce nuclear site represent 50 km and 150 km radii. Refer to legend of Figure 4-4 for information on fault ages. Earthquake event data was generated from the database at www.EarthquakesCanada.ca. Modified from Figure 2.14 of the Geosynthesis (NWMO11c).

Figure 4-79: Seismicity in the Bruce Region From 1985 to 2010 Overlain with Mapped Faults in Southern Ontario

Currently, Canadian Hazards Information Services (CHIS) of the Geological Survey of Canada monitors and reports on seismic activity in the immediate region of the Bruce nuclear site on an annual basis (HAYEK08, HAYEK09, HAYEK10). CHIS (HAYEK08) reviewed historical seismicity for the Bruce area and noted that only three earthquakes have historically been detected within 50 km of the Bruce nuclear site prior to 2007. Three events occurred in Lake Huron about 20 km northwest of Southampton, with M1.7 to M2.1, and are the only seismic events recorded within 50 km of the Bruce nuclear site. The current and historical monitoring data confirm that the Bruce nuclear site is located in a seismically quiet area.

Seismic Hazard Assessment

As discussed in the previous section the regional study area in general and the Bruce nuclear site in particular experience infrequent seismic activity, with no apparent concentrations that might delineate regional seismogenic features or active faults. Twenty-six events have been detected within 150 km of the Bruce nuclear site since 1952 with a maximum magnitude of 4.3 measured (11 km focal depth) 99 km from the Bruce nuclear site at a location north of Meaford near Owen Sound in 2005 (HAYEK10).

In general, earthquakes in stable interior regions, such as the Bruce region, occur at depths of 5 to 20 km, on faults formed hundreds of millions of years ago during previous active tectonic episodes. In a review of 76 events in eastern Ontario and western Quebec with known focal depth, most (82%) were less than 10 km depth, with an average depth of 7 km (MA06).

A Probabilistic Seismic Hazard Assessment (PSHA) was performed for the Bruce region to estimate bedrock ground motions that are expected for probabilities of 10^{-3} to 10^{-6} per annum (NWMO11w). The peak ground accelerations obtained from the PSHA are summarized in Table 4-10. Table 4-10 also presents the results of a $4x10^{-4}$ per annum probability event determined from this study and that defined in National Building Code of Canada (NBCC05).

Table 4-10: Summary of Seismic Hazard Assessment Results

Event (Prob. of exceed. p.a.)	Peak Ground Acceleration (% g)
1/1000	1.7
1/2500*	4.4
1/100,000	18.7
1/1,000,000	60.1

Notes: p.a. = per annum; % g = percent of gravitational acceleration. *From NBCC05. Data from Table 6.1 of the Seismic Hazard Assessment (NWMO11w).

Ground shaking hazard is one of the greatest threats to surface facilities, and forms the basis for seismic design. For underground facilities, it is generally known that earthquake damage to underground workings is rare. A strong dependence of damage to peak ground acceleration and peak ground velocity was demonstrated using case studies that examined the extent of tunnel damage during earthquakes (POWER98, BACKBLOM02). Seismic analysis of a DGR cavern using ground motions of 10^{-5} and 10^{-6} annual probability events reveals that seismic shaking would not induce damage to the host rock other than dislodging already fractured rock mass around the opening.

4.5.2.2 Fault Rupture and Reactivation

Fault rupture or reactivation is a concern as it could compromise the isolation potential of the repository for the migration of radioactive waste degradation products. Findings from existing seismic information, a neotectonic investigation of Quaternary sediments, structural surface bedrock mapping, micro-seismic monitoring, a 2D seismic survey and associated angled drilling, regional geologic data and the possible effects of glaciation were reviewed and assessed to provide an understanding of the likelihood of fault rupture and reactivation at the Bruce nuclear site. In addition to the summary points listed in Section 4.1.3, the evidence below argues against any significant effect on DGR performance due to fault rupture or reactivation.

- Based on existing seismic information, the likelihood of fault rupture is extremely low, as it would require a moderate-to-large event to occur right at the repository site, with rupture to shallow depths. Furthermore, since the repository is sited in an area where no faults have been observed, it would require earthquake faulting to propagate into previously unfaulted rock. In addition, most earthquakes are deep, occur on pre-existing basement faults, and are very rare in the site area. There are no known seismic events in the region with a focal depth in the Paleozoic sequence.

- A micro-seismic monitoring network was installed and commissioned in August 2007. Thus far, the results show a lack of low level seismicity (>M1.0) within the vicinity of the Bruce nuclear site, implying the absence of seismogenic structures or faults within or in close proximity to the proposed DGR.

- An investigation undertaken to characterize deformation features within Quaternary land forms and soil exposures surrounding the Bruce nuclear site concluded that none of the features observed (e.g., soft sediment disturbance and paleoliquefaction) resulted from post-glacial neotectonic activity (NWMO11v).

- Renewed glacial ice-sheet cycles of advance and retreat, over the DGR, may result in periods of enhanced seismic activity. Based on the lack of evidence for surface faulting (NWMO11ab), neotectonic deformation (NWMO11v) and cross formational groundwater mixing at the site (NWMO11k), any seismic event that may have been induced by such glacial activity in the past must have occurred either deep in the Precambrian basement or was too small to disrupt the intact rock mass proximal to the Bruce nuclear site. The impact of subsequent glaciations, therefore, would most likely not be significant enough to result in fault rupture and propagation into the Paleozoic rock sequence.

4.5.2.3 Volcanism

The only recognized evidence of volcanic activity at the Bruce nuclear site is ancient and in the form of one 8 to 10 cm thick bentonite seam interpreted to represent altered volcanic ash. This bentonite is observed at the same stratigraphic horizon, approximately 7 m below the top of the Coboconk Formation, in penetrating deep DGR boreholes. Based on a regional correlation, this particular bentonite is one of several distinct ash layers deposited throughout the Appalachian and Michigan basins during episodic volcanic activity associated with the onset of the Taconic Orogeny on the southeastern margin of Laurentia approximately 454 MaBP (e.g., HUFF92, KOLATA98).

The majority of recognized Mesozoic magmatic activity is localized around pre-existing faults, which are presently at a considerable distance away (> 150 km) from the regional study area, and the Bruce nuclear site, in particular. This includes kimberlites and other mafic intrusions within the Canadian Shield (HEAMAN00) and the ca. 130-110 MaBP Monteregian Hills alkaline intrusions near Montreal, Quebec (MCHONE84), which are related to passage of an interpreted hotspot through this region (e.g., CROUGH81). Other recognized activity includes a suite of 173 MaBP Middle Jurassic ultramafic dykes which intrude Middle Ordovician strata in the Picton Quarry, Ontario (BARNETT84). A lack of active orogenic activity in southern Ontario under the currently stable tectonic regime suggests strongly that volcanic activity is not expected to influence the regional study area.

4.5.3 Natural Resources

Natural resources found within the regional study area include oil and gas, base metal (MVT) mineralization, bedrock aggregate, salt and groundwater. Only oil and gas will be discussed for the reasons listed below.

- Sphalerite concretions within Silurian dolomite on the Bruce Peninsula have attracted some base metal exploration interest for potential MVT deposits (e.g., SANGSTER71). Evidence of historical exploration (e.g., shafts, trenches) exists on the peninsula; however, no commercial MVT deposits have been found within southern Ontario (Section 10.2.1 of NWMO11m).

- Although a number of areas in the regional study area have been identified by the Ontario Geological Survey and Ministry of Natural Resources as containing significant resources of sand and gravel (OGSR04), it is concluded that none have been identified within 20 km of the Bruce nuclear site (Section 10.2 of NWMO11m).

- The Salina salt does not represent a commercial resource because it has been dissolved and removed beneath the Bruce nuclear site through natural processes in the Paleozoic.

- Groundwater resources in the vicinity of the Bruce nuclear site are obtained from shallow overburden or bedrock wells extending to depths of ca. 100 m into the permeable Devonian carbonates. At increasing depth, groundwater becomes brackish and then saline (non-potable) as discussed in Section 4.3.1, and yields decrease. These factors would prevent or discourage deep drilling for water resources.

The following summary of the distribution of oil and gas resources is based on the detailed description in Section 2.2.8 of the Geosynthesis (NWMO11c).

4.5.3.1 Oil and Gas

Commercial quantities of oil and gas have been discovered in over 300 separate pools or reservoirs within the Paleozoic succession in southwestern Ontario (SANFORD93, ROSE70, HAMBLIN08), as shown in Figure 2.20 of the Geosynthesis (NWMO11c). Of more than 21,000 documented wells drilled in Ontario, only 27 petroleum exploration wells have been drilled within a 40 km radius of the proposed DGR and there is no commercially active hydrocarbon extraction at present in this area (OGSR04). Current exploration interest is focused on targets in the southwestern tip of Ontario in Middle Ordovician carbonates and Upper Cambrian sandstones at depths of 800 to 1000 m (GOLDER05), and the majority of this is concentrated within the geographic triangle between London, Sarnia, and Chatham-Kent (NWMO11m). From an evaluation of existing literature (NWMO11m), and based on the site characterization undertaken at the site (NWMO11k), the probability of future identification of potential economic oil and/or gas resources at, or adjacent to, the Bruce nuclear site is low. This conclusion is based on several factors.

- Although porous Cambrian sediments have been identified in core within the regional study area, no commercial oil or gas accumulations were encountered during site characterization activities (NWMO11k).

- Site characterization activities found no structural, lithological, chemical or hydrological evidence to suggest that the Bruce nuclear site is proximal to an ancient Ordovician HTD system (NWMO11k).

- An average TOC content of the Upper Ordovician shales of less than 1.0% (Figure 3.14 in NWMO11k), the recognition of low thermal maturity throughout the regional study area which indicates that these sedimentary rocks only reached the lower threshold of the oil window (LEGALL81, OBERMAJER96, NWMO11y), and the absence of natural gas shows during drilling of the DGR boreholes (NWMO11k), argues against the likelihood of commercial accumulations of either thermogenic or biogenic shale gas beneath the Bruce nuclear site (NWMO11y).

- None of the Silurian reefs nearest to the DGR encountered commercially viable resources. In addition, the Bruce nuclear site is located within an inter-reef lithology (NWMO11m). Minor oil showings in the Silurian Guelph Formation from the DGR core are associated with non-commercial hydrocarbon accumulations (NWMO11k).

- The Devonian Hamilton Group provides the cap rock for Devonian hydrocarbon plays, however it is absent at the site. Similarly, the Upper Devonian Kettle Point Formation shale, which might represent good candidate biogenic shale gas plays in southwestern Ontario (e.g., HAMBLIN06), has been eroded away across the entire regional study area.

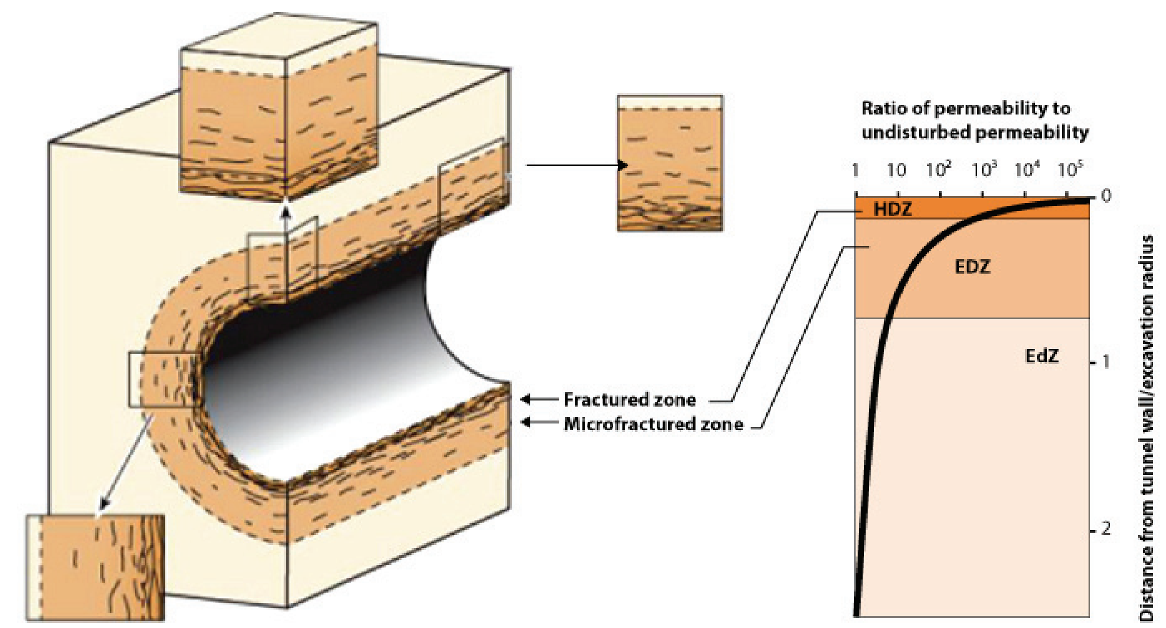
- Lateral traceability between the Bruce nuclear site boreholes and other proximal dry wells (e.g., Union Gas #1 and Texaco #6), demonstrates that locally around the Bruce nuclear site (~7 km radius), no pockets of oil or gas hydrocarbon are likely to exist.

4.5.4 Repository Induced Disturbances: Long-Term DGR Performance and Integrity

For long-term stability the quality of the rock mass containing the DGR must remain adequate such that the functionality of the DGR is not compromised. Numerical analyses with conservative assumptions have been used to assess the potential changes to the rock mass surrounding the repository and its access shafts (NWMO11t). An EDZ exists around underground openings where rock properties and conditions have been altered during excavation activities (NWMO11x). The EDZ was identified in the Safety Assessment as the primary pathway for the migration of radionuclides from the repository. In this section, the long-term evolution around the access shaft seal system is examined by means of geomechanical numerical modelling simulations. The geomechanical simulations capture the dominant mode of behaviour and understanding on the evolution of the EDZ with respect to specific sealing elements, in-situ stress environments, rock conditions, and pore pressure response in the rock mass due to the presence of water and gas during long-term repository evolution (NWMO11x).

4.5.4.1 EDZ

This section summarizes experience from geoscientific studies of the EDZ and its surrounding disturbed zone in sedimentary rocks (Figure 4-80). The bulk of these studies haves been performed as part of research programmes for radioactive waste disposal in argillaceous rocks. In particular, the section draws on in-situ experimental studies performed in Underground Research Laboratories.



Note: From Figure 6.7 of the Geosynthesis (NWMO11c).

Figure 4-80: Schematic Illustration Defining EdZ, EDZ, and HDZ for Unjointed Rock

The excavation of any underground opening creates a zone of disturbed rock around it. Within this disturbed zone there may exist a zone of damaged rock. In the past various definitions of the disturbed and damaged zones have been used. In this report, the EDZ around excavation openings is divided, as shown in Figure 4-80, into three categories (NWMO11x).

- The Highly Damaged Zone (HDZ) is a zone where macro-scale fracturing or spalling may occur. The effective permeability of this zone is dominated by the interconnected fracture system and may be several orders of magnitude greater than that of the undisturbed rock mass.

- The EDZ is a zone with hydromechanical and geochemical modifications inducing significant changes in flow and transport properties. These changes can, for example, include one or more orders of magnitude increase in (effective) flow permeability.

- The Excavation Disturbed Zone (EdZ) with possible hydromechanical and geochemical modifications but without material changes in flow and transport properties.

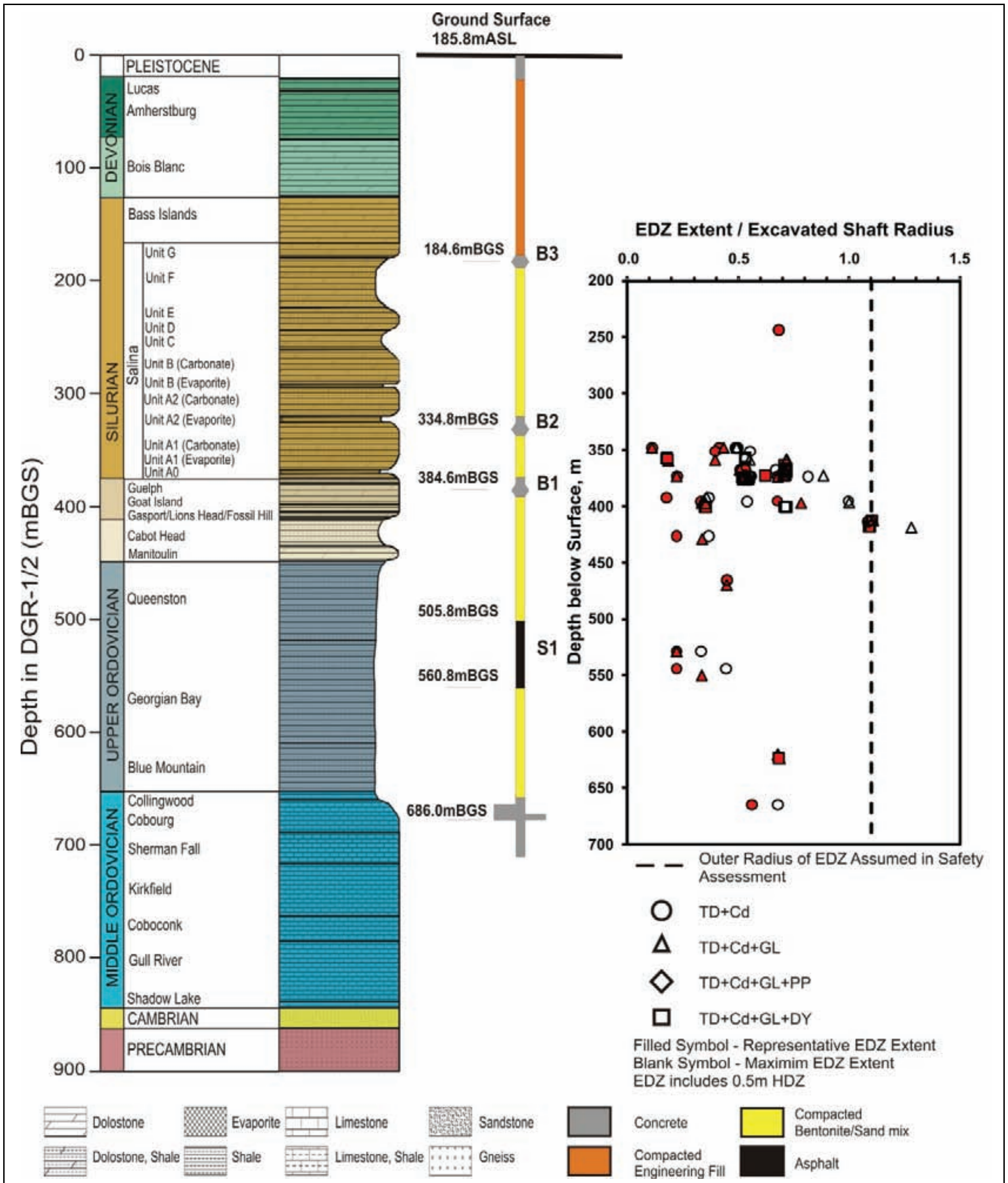
The alterations of the rock mass around excavation openings include induced fracturing, stress relaxation, desaturation, pore pressure disequilibrium, and chemical interaction near the excavated face. An overview of EDZ characteristics in terms of the process and mechanisms governing its occurrence, distribution, properties, and evolution is described in the Geosynthesis (NWMO11c).

4.5.4.2 Shaft Seal Analysis

Shaft seals are required to ensure the long-term integrity of the DGR. The purpose of this backfill/seal system is to inhibit gas/fluid migration along the shaft. The planned shaft seal system of the DGR consists of a series of sections with engineered backfill/seal material comprised of compacted engineered fill, four compacted bentonite/sand backfill columns, concrete bulkheads (B1 to B3), and an asphalt column (S1). Figure 4-81 shows the shaft seal system with the corresponding subsurface lithology beneath the site. Details of the shaft seal system are presented in Chapter 13 (see Figure 13-1).

The three concrete bulkheads are planned at horizons in the upper 4 m of the Salina A1 Unit, the Guelph Formation, and in the upper Salina F Unit, below the shallow groundwater system.

Shaft seal analyses were carried out on the concrete bulkhead (B1) and the asphalt column (S1) with a focus on the evolution of the damaged zone around the shafts. The knowledge gained from earlier studies on four trial seal configurations with different surrounding host rocks was also utilized to support the results of the current analysis (NWMO11t). The long-term shaft seal analysis covers the rock mass response in varied rock formations, specific seal behaviour (i.e., asphalt, concrete bulkhead), in-situ stress environment, and pore pressure response around excavated openings over a period of 1 Ma (NWMO11t).



Notes: Indicates maximum local and formation specific representative EDZ extents relative to excavated shaft radius. Loading abbreviations: Time Dependent Strength Degradation (TD), Concrete Degradation (Cd), Glacial Loading (GL), Pore Pressure (PP), Seismic Loading (DY). Modified from Figure 6.22 of the Geosynthesis (NWMO11c).

Figure 4-81: Distribution of EDZ Extent Along Shaft

A total of 48 numerical simulations were performed. The key scenarios of repository evolution listed below will be described in the next five sub-sections (NWMO11t):

Case	Description
1	Time-dependent strength degradation (base-case)
2	Strength degradation with additional effects of gas pressure build-up
3	Strength degradation with additional effects of glacial loading
4	Strength degradation with additional effects of seismic ground shaking
5	Combinations of all of the above loading scenarios

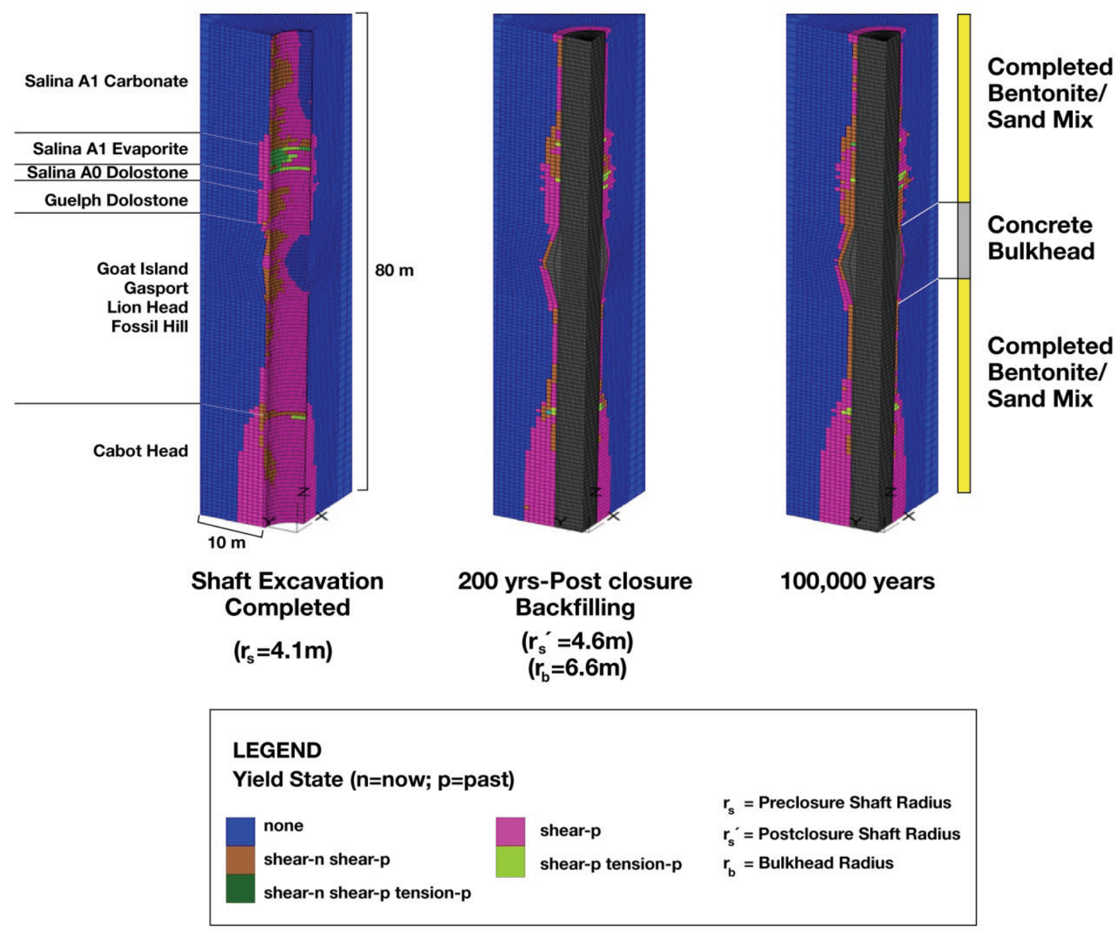
Rock mass properties used are based on the laboratory test results conducted on rock cores retrieved from boreholes DGR-1 to 6 from the relevant seal horizons. The rock support, content, rock mass behaviour, and in-situ stress condition were conservatively assumed for these analyses. Also, the initial shaft excavation, postclosure over-excavation and shaft backfill/seal placement were modelled based on the excavation and backfilling sequence described in Chapters 9 and 13 of this report.

Results of Shaft Seal Analysis

For selected seals in the analysis, several other long-term loading conditions were considered in parallel with the time-dependent strength degradation: glacial loading, pore pressure evolution, and seismic ground motions. Typically, each specific loading condition was simulated in the model that would result in worst-case increases in the extent of damage. These additional loading conditions are described in greater detail in the following section.

Case 1 Time Dependent Strength Degradation

Time-dependent strength degradation is a measure of how the rock will perform over a period of time under existing stress conditions after an opening has been excavated. The models generally showed that most of the EDZ developed soon after the completion of the initial shaft excavation phase (Figure 4-82). For most of the seals analyzed, the time-dependent strength degradation resulted in less than 20% to 50% increase in the extent of damage depending on rock material. The additional time-dependent loading conditions had limited effect on evolution of the damaged zone around the shaft/seals because of the stabilizing influence of the confining effect provided by the backfill-seal materials. The swelling pressure due to geological units and bentonite backfill is conservatively not considered in the modelling as they are anticipated to provide additional confinement to the rock. Specific observations from the shaft analysis are summarized below.



Notes: Colours other than blue represent yield and presence of EDZ. Modified from Figure 7.11 of the Geomechanical Stability Analysis report (NWMO11t).

Figure 4-82: Yield State - Concrete Bulkhead B1: Time-Dependent Strength Degradation

The relaxation of the rock mass around an opening, which mostly occurs during and shortly after excavation, and time-dependent strength degradation effects directly account for the majority of the EDZ formation and may persist on a diminutive scale for a period of 1 Ma, which was represented as a loss in cohesion in the model. A series of analyses were carried out to determine the normalized extent of the EDZ (scaling with excavated radius). The results for the base-case (strength degradation only) are summarized in Figure 4-83. Results of the 3D analyses (including excavation sequence and layer geometry) were compared with 2D results from finite-element analysis. Many of the rock masses in the sedimentary sequence could experience one or the other behaviour depending on the thickness of the rock unit being analyzed and so these two analyses bracket the expected response.

The EDZ is defined by the presence of yield in the model and represents a maximum estimate. For the highly fractured zone around the excavation, the HDZ (a subset of EDZ) estimates are based on increased rates of strain and negative net volumetric strain, and as such represent conservative estimates of fracture damage (NWMO11x). Figure 4-83 show the estimated HDZ and EDZ for the time dependent strength degradation base-case. Only the Cabot Head shale shows excessive extent of the EDZ. The EDZ in all formations analyzed indicates values less than those assumed in Safety Assessment. Along the shaft, the geometries of the EDZ and HDZ are expected to vary due to changing lithology and mechanical properties, intersection with geological related discontinuities, and changes in excavation methods (Figure 4-81).

The low permeability of the rock and shaft backfill makes the EDZ along the shaft behave as a serial system. During closure, the HDZ will be removed by over-excavation prior to the shaft being backfilled. Within this serial system, migration of a contaminant front can only occur through one local zone of EDZ at a time. As such, this local zone will not significantly enhance the transmissivity along the full shaft damage annulus.

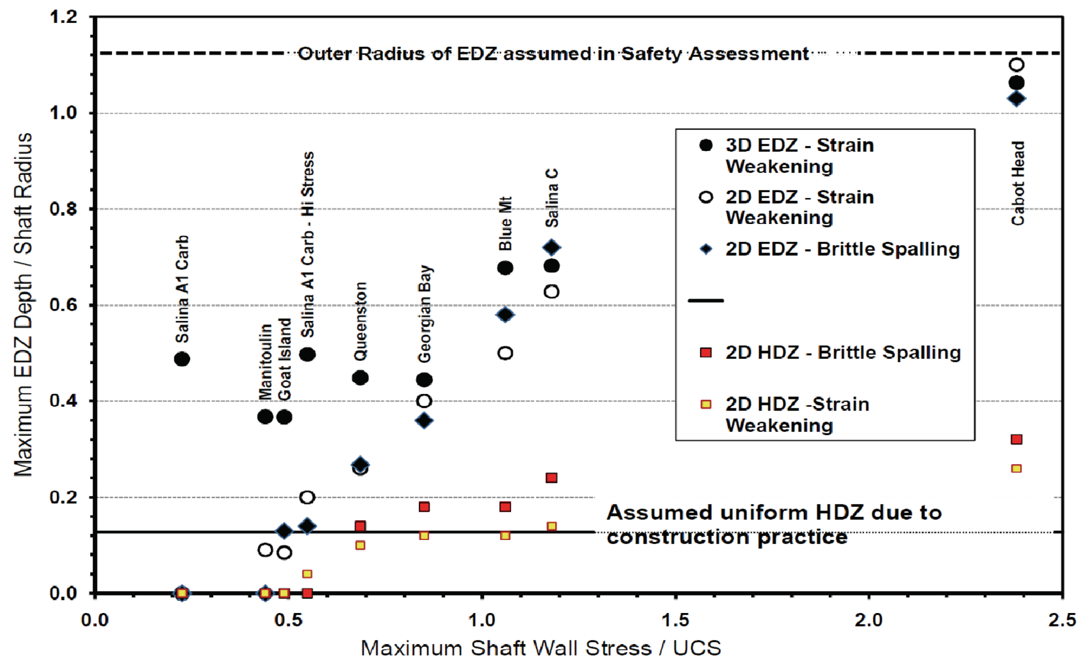
Case 2 Effects of Gas Pressure Build-up

Gas-generation processes within the repository were modelled and the pressure and flow response simulated for a 1 Ma timeframe. Seal B1 represents seal element at great depth subjected to the greatest water and gas pressures. Over the long-term, the pore pressure gradually increased to a steady-state value of approximately 3.9 MPa, as the shaft materials responded to pressure changes transmitted from the repository, and equilibrated to pressures in the surrounding intact rock.

The pore pressure had little short-term (initial excavation) effect on the extent of yielding, while the long-term pore pressure evolution resulted in some increased yielding around the shaft seal/backfill. The long-term pore pressure evolution combined with strength degradation and glacial loading could increase the extent of predicted damage locally by at most 1.4 m for seal S1.

Case 3 Effects of Glacial Loads

During the next advance of continental-scale glaciation, predicted to occur at least 60 ka in the future (see Section 4.5.1), it is anticipated that each seal will be subjected to glacial loading with a maximum vertical pressure of about 30 MPa (approximate ice thickness of 3 km; NWMO11r). An assumed horizontal stress increase of 2 MPa due to bending of the strata was also imposed in the simulation in addition to the in-situ stress profile described in Section 4.2.3.



Notes: Assumptions made for the safety assessment are shown as horizontal lines. Maximum EDZ and HDZ estimates in different rock units are generally consistent with the assumptions made. Exception is the weaker Cabot Head Formation. Modified from Figure 6.17 of the Geosynthesis (NWMO11c).

Figure 4-83: EDZ and HDZ Estimates for Base-Case

The addition of glacial loading combined with strength degradation had only minimal effect on the extent of damaged rock due to the confinement provided by the backfilling in the shaft. The effect of a single glacial event on the shaft EDZ is relatively small to almost negligible (Figure 4-84). Multiple events, thus, were not analyzed despite the evidence that multiple glacial events are expected during the next 1 Ma.

Case 4 Effects of Seismic Ground Shaking

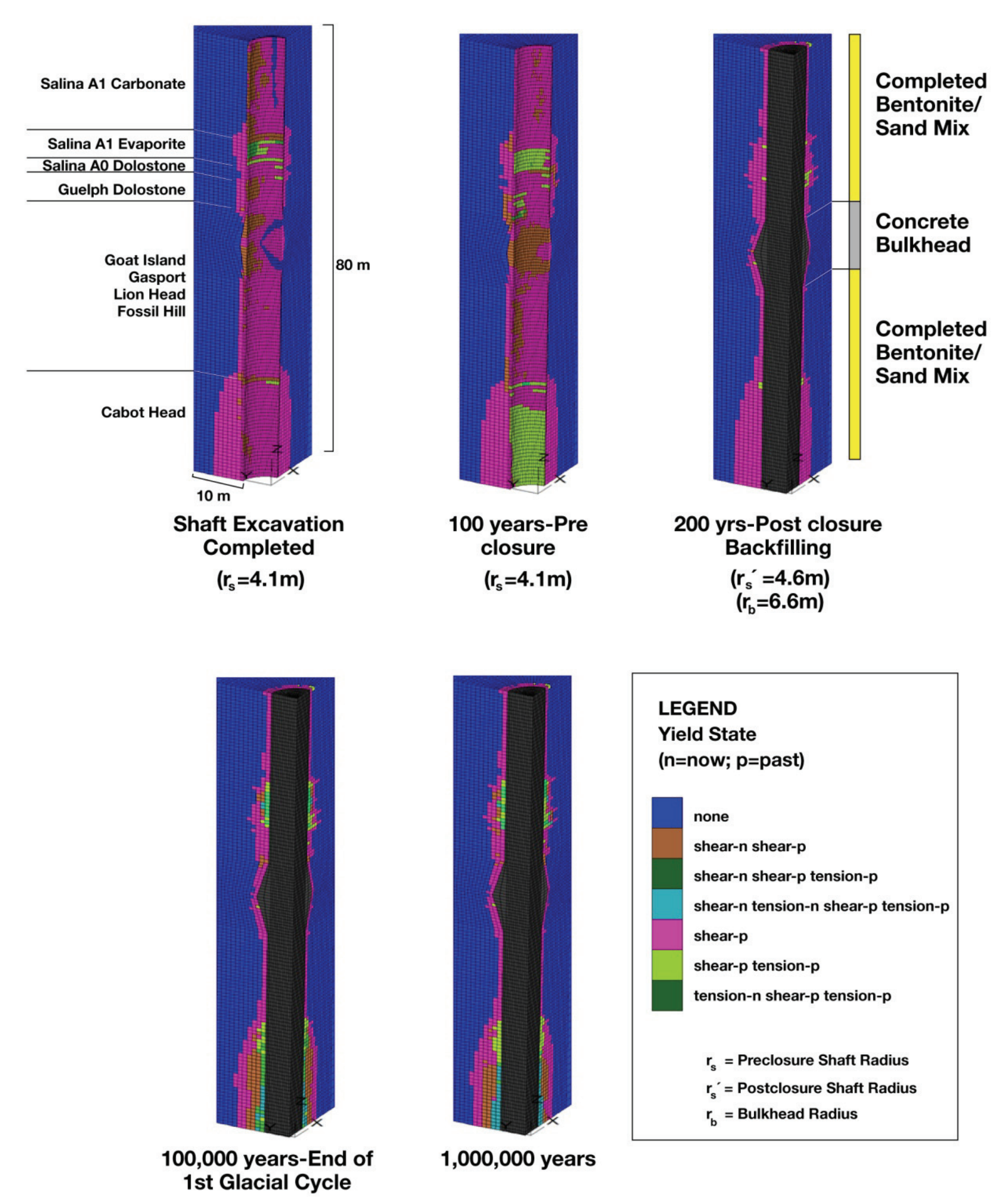
The effect of seismic ground shaking was evaluated by incorporating ground motions developed as part of the PSHA directly into the simulation (NWMO11w). Although time histories for a number of horizons were generated from the PSHA based on the P- and S-wave velocity profiles, only Seal B1 (Figure 4-81) was analyzed to provide insight on the seal behaviour under seismic conditions.

A dynamic analysis was run with time-dependent strength degradation and glacial loading until 67,200 years, the point at which the maximum glacial cycle had been reached. At this state, the model was subjected to the ground motions of three events with 10^{-6} per annum probability of exceedance. As shown in Figure 4-85, seismic loading had no effect on the extent of failure for seal B1. Because dynamic loading proved negligible, no additional seismic analyses were carried out for other seal elements.

Discussion of EDZ Evolution

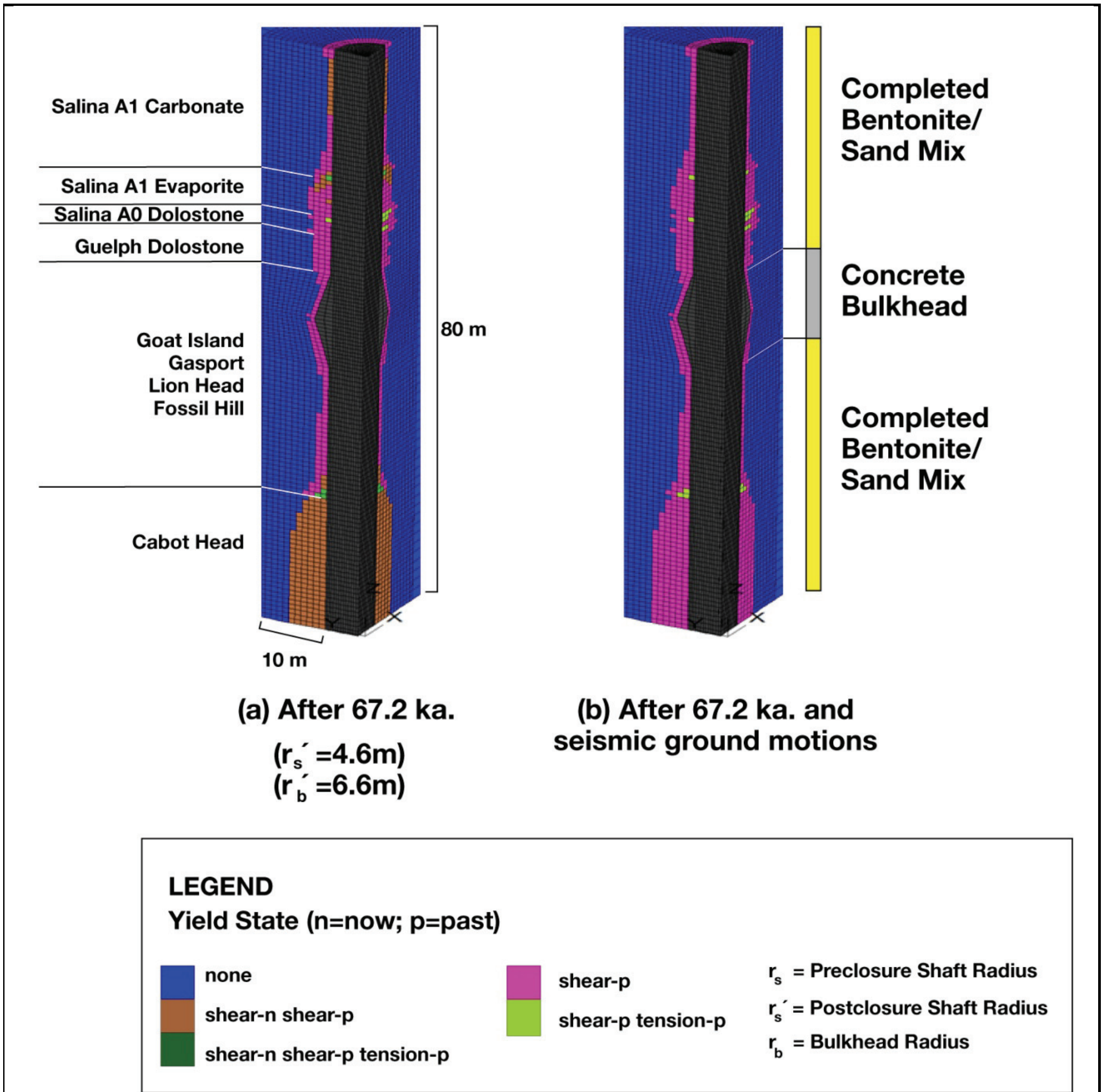
Due to the vertical geometry of the shaft, glacial loading has only a minor effect on differential ground stresses in the horizontal plane. Consequently, the effect of EDZ increase during glaciations is minor for the shaft. Similarly, pore pressure and seismic loading will not significantly increase the predicted EDZ around the shaft. The extent of the EDZ, including the HDZ, is typical 0.7 times or less of the shaft radius. Based on the assumption of controlled drill and blast excavation, the estimated thickness of the HDZ (from the shaft wall to the outer limit of the HDZ) is approximately 0.5 m, or 0.11 times the radius of the access shaft. Because of the anticipated high fracture interconnectivity, thus high permeability, within this zone, the HDZ will be over-excavated at DGR closure. Figure 4-81 shows the distribution of the EDZ with depth, showing representative EDZ extent and maximum EDZ extents in formations under study. Because of the low rock strength, the Cabot Head shale reveals a much more significant EDZ (1.28 times the shaft radius) than the remaining sedimentary sequence. The extent of damaged zones, EDZ and HDZ, behind the shaft wall is conservatively assumed to be 1.1 times the shaft radius uniformly along the entire shaft in the DGR Safety Assessment (reference).

Considering the 80 m distance between the two shafts, the interaction of rock mass due to the excavation of the main and vent shafts is negligible as the distance between two excavations is three diameters or greater.



Note: Modified from Figure 7.9 of the Geomechanical Stability Analysis report (NWMO11t).

Figure 4-84: Yield State – Concrete Bulkhead B1: Time-Dependent Strength Degradation + Glacial Load + Pore Pressure



Notes: After 67,200 years and before (a) and after (b) three seismic events of 10^{-6} annual exceedance frequency were applied. Modified from Figure 6.21 of the Geosynthesis (NWMO11c).

Figure 4-85: Yielding Zones Around the Shaft (Concrete Bulkhead B1)

4.5.4.3 Long-Term Cavern and Pillar Stability

A comprehensive suite of modelling analyses have been performed to test the repository design and the Cobourg Formation against the challenges imposed by stress, material strength degradation, fracture generation, seismic loading, pore pressure effects including gas pressure build up, and multiple glacial cycles (NWMO11t).

Each panel of the repository consists of caverns and pillars located in the high-strength Cobourg Formation. The overlying Collingwood Member is a high-strength shale and limestone unit with mechanical properties that are only slightly lower than those of the Cobourg Formation. The interval from 685 to 695 mBGS, comprising the lower 3 m of the Cobourg Formation and the upper 7 m of the Sherman Fall Formation, has lower strength than the overlying and underlying rocks, and is here named the "weak Sherman Fall". In the analyses discussed below, two floor horizons at 683 mBGS and at 679 mBGS, 2 and 6 m above the top of the weak Sherman Fall unit, were examined, with 6 m providing the optimum balance between floor and roof cover within the Cobourg Formation (NWMO11t).

Rock mass properties used in the modelling are from the laboratory test results conducted on rock cores retrieved from DGR-2 to DGR-4. The long-term strength of the Cobourg Formation is based on a lower bound consistent with the Crack Initiation (CI) threshold for the argillaceous limestone. A large number of tests were carried out to establish this critical limit. The resultant representative value (CI = 45 MPa) has been used in these analyses. Previous experience has shown that the mean CI represents the lower bound conservative assumption for long-term strength.

In all analyses, the vertical stress at the repository level was 18 MPa, while the horizontal stress was 36 MPa (i.e., stress anisotropy = 2 as a conservative case although the caverns will be oriented in the direction of the major principal stress).

The performance of the repository with time was broken up into a number of time intervals:

1. End of operation (100 years);

2. After long-term strength degradation (pre-glacial) (50 to 60 ka);

3. After one glacial period (~100 ka); and

4. After many glacial periods, multiple seismic events (to 1 Ma).

Analyses for cavern stability over the first 100 ka were based on a reasonable extrapolation of measured material parameters and consideration of long-term material properties. Stability of the cavern over the first three time intervals resulted in a limited degree of overbreak with no influence extending to the overlying shales or the underlying limestones.

Due to the increasing uncertainties inherent in predicting material behaviour and changes in the geological environment in the long-term, the stability of the repository beyond a few hundred thousand years must be defined as a self-arresting deterioration of each individual cavern and pillar. The long-term stability thereafter relies on material bulking (observed in mining and in natural caves). This is verified by complex simulations involving the unravelling and build-up of rock blocks within the cavern opening. Such material will expand in volume after failure from the roof and walls and will eventually fill up the cavern space and stabilize the rock mass by preventing further collapse. The choking of the excavation due to bulking is consistent with observations of collapses in large natural caverns where the groundwater flow has been diverted and the material is not dissolved but allowed to bulk. Typical bulked material in a collapsed limestone cavern is shown in Figure 4-86. This cavern is essentially stabilized by the bulking of material from the roof and walls.

Stability in this extreme case is defined as a self-stabilization of the collapsing repository horizon with the terminal settlements within the damage tolerances of the overlying strata, preserving the integrity of the natural barrier system. Consistent with observation model results indicate that the Cobourg Formation will provide a competent roof for the excavated caverns (NWMO11t).

Results of the following modelling analyses are discussed below:

- Time-dependent strength degradation pillar-scale analyses;
- Multiple glaciation analyses;
- Hydraulic fracture analyses;
- Seismic ground shaking analyses; and
- Stength parameter sensitivity analyses.

Time-Dependent Strength Degradation Pillar-Scale Analysis

Simulations of cavern and pillar evolution were performed using more realistic lower bound strength conditions. Figure 4-87 shows the results of modelling two scenarios using the pillar-centred model (pillars and caverns are assumed to be infinitely repeated in both directions). In both cases, the long-term lower bound strength of CI = 45 MPa was used for the Cobourg Formation. The floor invert level is set at 679 mBGS and 1.4 m of frictional material was included in the cavern to represent degraded waste. In the upper case shown, dry conditions were maintained while two glacial loading cycles were imposed. In the lower case shown, formation pore pressures and repository gas pressures were included while one glacial loading cycle was modelled. Using a lower bound long-term strength, the dry condition (no pore or gas pressure) is the conservative case, as it shows more fracture and bedding separation into the roof (larger EDZ and HDZ) at 100 ka than the case including pore and gas pressures.

Notes: Collapse has resulted in only minimal drainage and water flow. The collapse shown on the left has evolved over 100 ka. Modified from Figure 6.28 of the Geosynthesis (NWMO11c).

Figure 4-86: Examples of Bulking After Limestone Cavern Roof Collapse

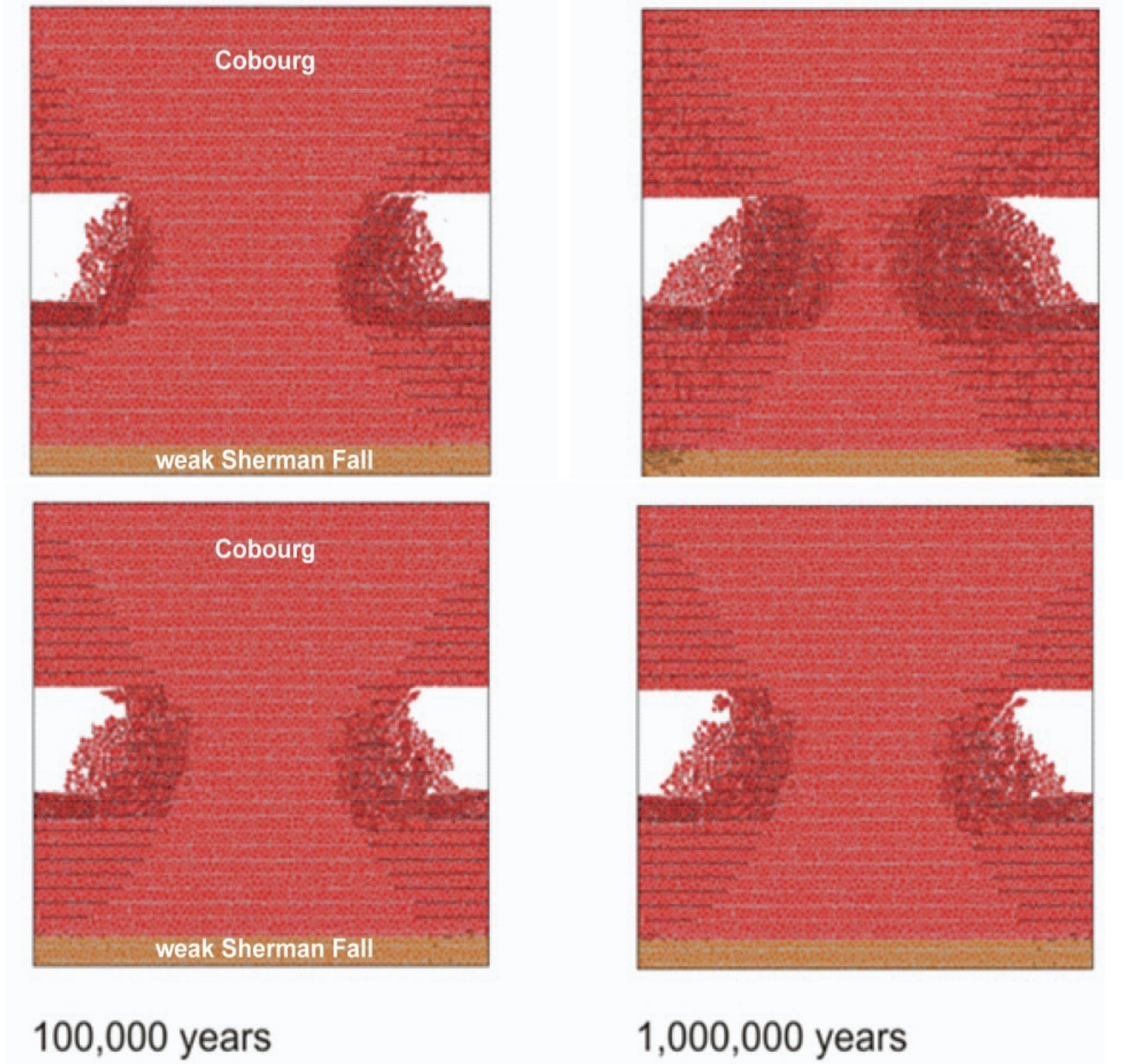
Multiple Glaciations

Time frames extending beyond 100 ka are not within the conventional scope of engineering geomechanics. It is not possible to predict the exact cycle or extent of glaciations in this time frame. Therefore, maximum glacial events (3 km of ice) were applied in the model at 60, 100, 200 and 300 ka, peaking 7 ka years later in each case. Each glacial event, predictably, induced additional damage to the caverns. This damaged material fails into the cavern with an increase in volume (typically 20-30%). Eventually, the failed and expanded material from the roof, floor, and walls chokes the cavern and provides a natural backfill. In old mining stopes and in natural caverns, this material will remain and prevent further collapse (in the absence of flowing water). Using the model parameters defined previously, the results for the 2nd, 3rd, and 4th glacial periods are shown in Figure 4-88. By the fourth glacial cycle, this discontinuum model shows that the cavern is effectively choked and further collapse is prevented indefinitely.

Potential for Hydraulic Fracturing

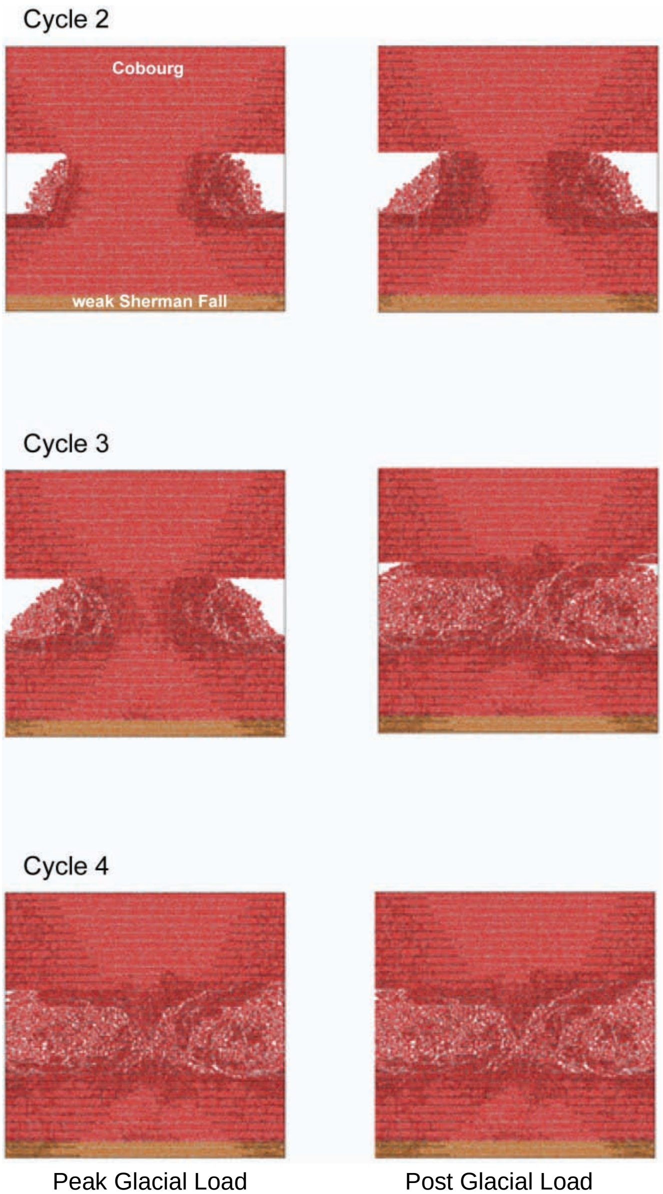
The effect of pore and gas pressure evolution on the structural performance of the repository was investigated using numerical modelling. The rock formations at the DGR are saturated under in-situ conditions. Corrosion of the waste inside the caverns will result in the generation of gases. During the postclosure phase of the repository, gases generated as a result of waste degradation will cause pressure changes inside the cavern as well as in the surrounding damaged zones due to diffusion of the gases into the available porosity. Because of the low permeability and high gas entry pressure of the intact Cobourg Formation, a significant amount of the gas will remain inside the caverns, resulting in a gradual build-up of gas pressure which has the potential to open any fracture planes that are normal to the minimum principal stress.

At the repository depth, the vertical stress of 18 MPa is the minimum principal stress and, therefore, the potential for hydraulically induced fracturing is greatest along the subhorizontal bedding planes (which are planes of weakness), if the gas pressure should exceed the vertical stress of approximately 18 MPa. For the analysis, two simplified cases of gas pressure histories were developed representing a base-case condition with gas pressure plateau at about 7 MPa at about 100 ka and an extreme case scenario with 15 MPa peak gas pressures approaching lithostatic pressure (NWMO11t). Figure 4-89 shows these gas pressure profiles comparing with profiles of various gas modelling cases for the normal evolution scenario in the repository with time (NWMO11aj, NWMO11t). The results of the analyses for the normal pressure evolution scenario in base-case with updated permeabilities that include the effects of gas pressures in the repository do not indicate any localized fracture development typical of hydraulic fracturing. No horizontal fractures were observed in the rock mass around the cavern.



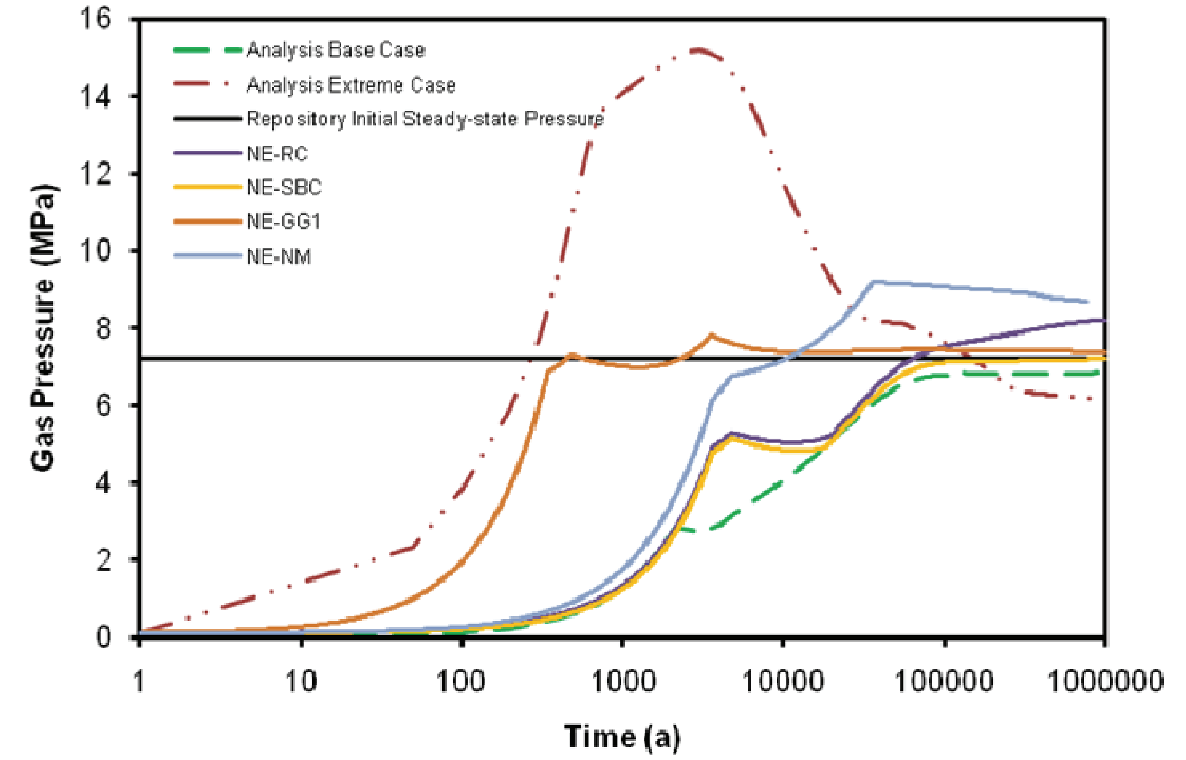
Notes: Top row loading conditions: time-dependent strength degradation; dry conditions (no pore or gas pressure); 1st glacial cycle started at 60 ka and reached its peak at 67.2 ka; 2nd cycle started at 100 ka and peaked at 107 ka; no seismic loading. Bottom row loading conditions: same as above but with pore and gas pressure; 1 glacial cycle started at 60 ka and reached its peak at 67.2 ka with ice removal by 80 ka. 6-m floor with 1.4 m degraded waste. Rubble is allowed to accumulate. DGR invert at 679 mBGS. Modified from Figure 6.26 of the Geosynthesis (NWMO11c).

Figure 4-87: Evolution of Cavern Outline and Pillar Damage with Lower Bound Strength



Notes: Time-dependent strength degradation and dry conditions; 1st glacial cycle started at 60 ka and peaked at 67.2 ka; 2nd cycle started at 100 ka and peaked at 107 ka; 3rd cycle started at 200 ka and peaked at 207 ka; 4th cycle started at 300 ka and peaked at 307 ka; no seismic load. DGR invert at 679 mBGS. Modified from Figure 6.27 of the Geosynthesis (NWMO11c).

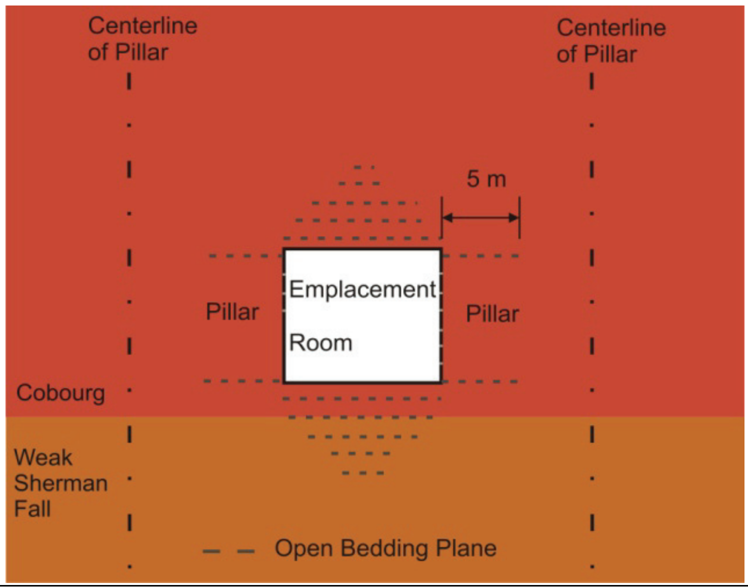
Figure 4-88: Evolution of Cavern Outline and Pillar Damage, Representative Case for Four Glacial Cycles



Notes: NE-RC = DGR Reference Case, NE-SBC = Steady-state Cambrian Overpressure Case, NE-GG1 = Increase Gas Generation Case, and NE-NM = Methanogenic Reaction Case (NWMO11aj). Modified from Figure 4.4 of the Geomechanical Stability Analysis report (NWMO11t).

Figure 4-89: Repository Gas Pressure Histories Used in Geomechanical Stability Analyses

To demonstrate the margin with respect to no hydraulic fracturing, the extreme case with a high gas generation rate with the maximum gas pressure of about 15.2 MPa was also investigated. The potential for fracturing of bedding planes due to gas pressure was investigated assuming that there is no time-dependent rock strength degradation, which is the case that promotes localized deformation along the bedding planes instead of distributed damage and fracturing of the rock matrix. The results of this calculation at 100 ka, shown in Figure 4-90, indicate that the 5-m-long shear (not tensile) fractures localize along bedding planes in the floor and crown of the cavern.



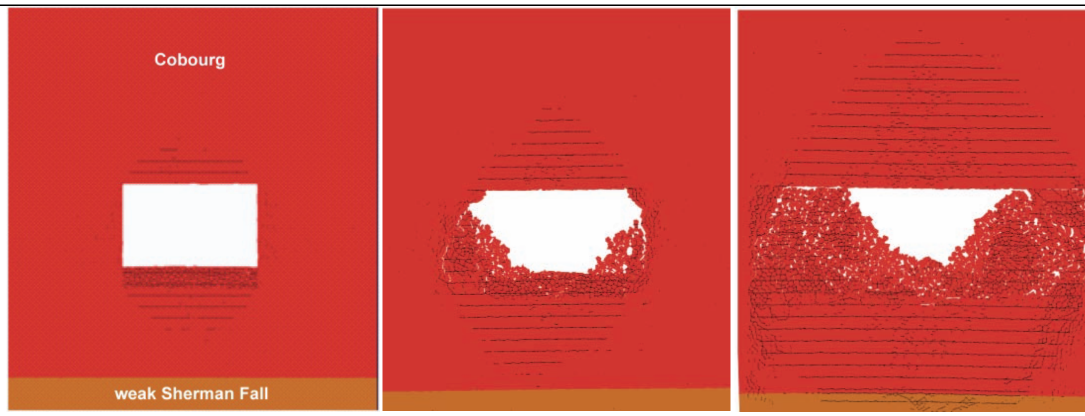
Notes: DGR invert is at 683 mBGS. Modified from Figure 5.12 of the Geomechanical Stability Analysis report (NWMO11t).

Figure 4-90: Opening of Bedding Planes (black lines) Around a Cavern at 100 ka Due to Extreme Gas Pressure History (15.2 MPa)

Seismic Analyses

Dynamic analyses were carried out to investigate the effect of seismic ground shaking on stability of the emplacement caverns for 6 ground motion time periods, 3 each at 10^{5} and 10^{-6} probabilities of annual exceedance (NWMO11t). The time histories at the repository level were derived from the transfer function defining the ground motion at depth to that at ground surface and were used in the cavern stability analyses. The dynamic analyses considered loading due to in-situ stresses, time-dependent strength degradation and glacial loading. Because the gas and pore pressure do not have significant effect on the cavern stability, they were not included in the load combinations with dynamic analysis. Three different times of the occurrence of each seismic event are analyzed: 1) before the first glacial cycle, 2) at the peak of the first glacial cycle, and 3) at the peak of the second glacial cycle. A total of 18 dynamic simulations were completed.

The results of the dynamic analyses for an M7.4 event at 200 km are shown in Figure 4-91, for cases of the seismic events occurring before the glacial loading, at the peak of the first glacial cycle and at the peak of the second glacial cycle, respectively. The seismic shaking of the considered magnitudes does not cause any additional damage or fracturing of the rock mass. That is particularly the case for the events occurring early, before glacial events, when the rock mass is relatively unfractured. The seismic shaking does promote unravelling of already fractured and loose rock mass. Unravelling can result in additional fracturing of the rock mass due to the reduction in confinement, but not as a result of seismically induced stress change or inertial forces. Consequently, the effect of seismic shaking appears to have more effect as the area of the damaged rock mass increases when the rock mass is subjected to more glacial events.



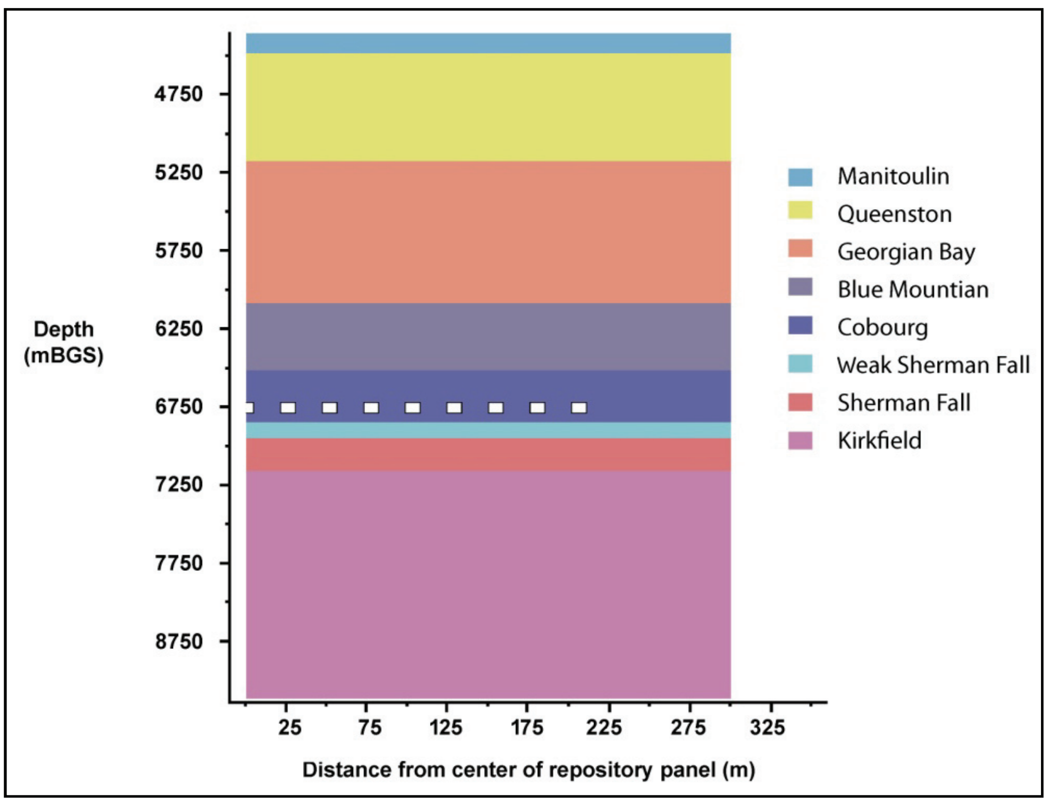
Notes: Shaking by an M7.4 earthquake at 200 km distance at 60 ka (left), 67.2 ka and one glaciation (middle), and 107 ka and 2 glaciations (right). Shaking occurs at glacial maximum. DGR invert is at 679 mBGS. From Figure 6.31 of the Geosynthesis (NWMO11c).

Figure 4-91: Cavern Models Subjected to Shaking

<u>Panel-Scale Analysis</u>

A panel-scale parametric analysis was performed to illustrate the sensitivity of the strength parameters and to quantify the level of conservatism in the pillar-scale models using CI stress as the lower bound strength. The model geometry indicating different geological units is shown in Figure 4-92. The analysis is two-dimensional in the plane of the cross-section along Panel 2. The model uses symmetry to include only one half of the panel length.

In the model, time-dependent strength of the Cobourg Formation was degraded and assumed to be equal to the long-term strength from the beginning of the simulations irrespective of the stress state. The Cobourg Formation long-term strength values in the pillar-scale analyses, was set as 45 MPa (40% of the laboratory UCS) which corresponds to the CI stress measured in the uniaxial laboratory compressive tests. While Damjanac and Fairhurst (DAMJANAC10) suggested that the CI stress as a lower bound for the long-term strength of rock, there is no physical evidence to suggest that actual rock strength reaches this lower bound in-situ. Rock masses have sustained deviatoric stresses for tens of millions to billions of years, yet their measured strength values do not correspond to CI stress. Hence, there is significant uncertainty in proposing a realistic long-term strength for rock other than the CI limit of 40% UCS in the case of Cobourg Formation. To investigate the sensitivity of the predictions of cavern and pillar degradation to the assumption of the long-term strength, the parametric analysis was carried out for six values of the Cobourg Formation long-term strength: 45 MPa (40% UCS); 54 MPa (49% UCS); 63 MPa (57% UCS); 72 MPa (65% UCS); 81 MPa (73% UCS); and 90 MPa (81% UCS).



Note: From Figure 6.1 of the Geomechanical Stability Analysis report (NWMO11t).

Figure 4-92: Geometry of the Model for Panel-Scale Parametric Analysis

The analysis considered estimated in-situ stresses, the effect of time-dependent strength degradation as discussed in the previous paragraph, and multiple glacial load cycles. The effect of gas and pore pressure was not included in this analysis because the pillar-scale analysis showed that this effect was not significant. Ten glacial cycles were simulated quasi-statically in each case.

Figure 4-93 shows the evolution of plasticity around caverns in the middle of the repository after 2 (left) and 8 (right) glacial cycles for the 6 values of long-term strength investigated. Each simulation shows regions of plastic deformation, where rock yields because stresses reach the yield strength. The continuum panel-scale models predict that the extent of the fractured rock mass around excavation openings increases with each glacial cycle. The increment of the damaged volume per glacial cycle increases as the modelled long-term strength decreases.

Considering the results of all the model runs, it was concluded that if the long-term strength of the Cobourg Formation is 40% of the UCS, the pillars are predicted to fail after 3 to 7 glacial cycles (NWMO11t). However, if the long-term strength is 49% UCS or greater, the continuum panel-scale model predicts that pillar core will remain elastic for at least 10 glacial cycles (NWMO11t). Considering uncertainties in the magnitudes of the peak glacial load (always assumed to be 30 MPa, i.e., the maximum load within the last 120 ka), frequency of recurrence of glacial events, and uncertainty in the long-term rock strength, the pillars and the panels are expected to be stable for at least 100 ka. However, sensitivity analyses also show that if the long-term Cobourg strength is equal to or greater than 72 MPa (or 65% UCS), the extent of the damaged (or plastically deformed) region does not increase with glacial cycles after the first glaciation. This implies that the pillars and the emplacement caverns will remain stable throughout modelled timeframe of 1 Ma. If the Cobourg long-term strength is equal to or greater than 81 MPa, the zone of damaged rock in the pillars remains confined to 1 m from the pillar wall (NWMO11t).

3D analyses were also conducted on the full panel configuration with yielded and intact central barrier pillars (NWMO11t). For the case with a 40 m wide central pillar, a small amount of plastic yielding is apparent in the Blue Mountain Formation over the barrier pillars. The barrier pillar itself does not show any sign of yielding. Kinematically, this indicates that the yielding in the Blue Mountain Formation shale cannot propagate upward and will be in the form of minor local degradation instead of extensive shear (with large strain). For the cases with yielded central pillar (single width pillar assumed to fail along with the other inter-cavern pillars), there is a slight increase in the degree of perimeter yielding in the Blue Mountain Formation shale. None of the cases indicates any damage or undue influence on the shale cap rock of the Georgian Bay or Queenston formations.

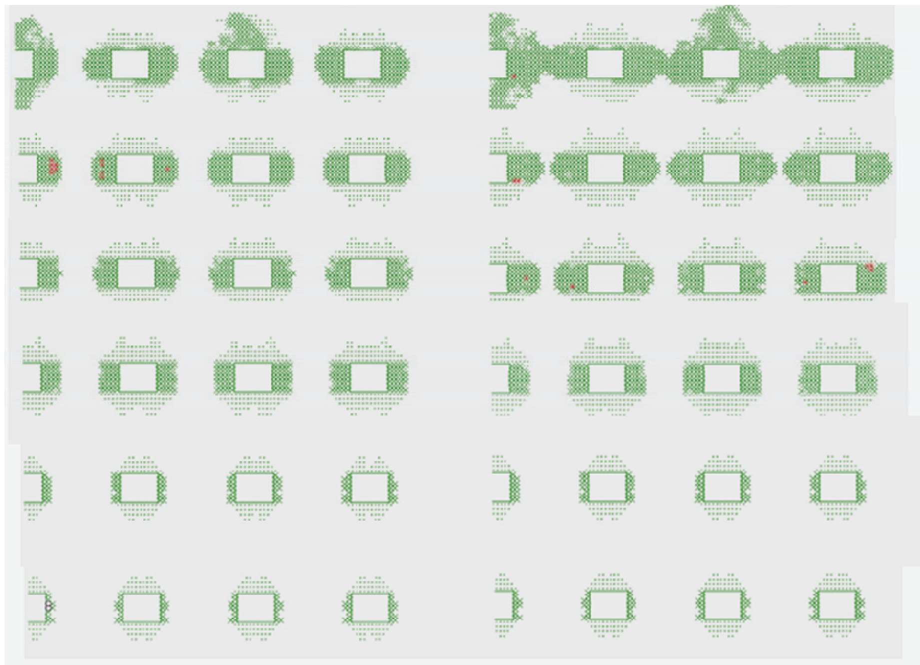
Long-term Strength = CI	Yield After 2 Glacial Cycles	Yield After 8 Glacial Cycles
CI = 45 MPa (40% UCS) CI = 54 MPa (49% UCS) CI = 63 MPa (57% UCS) CI = 72 MPa (65% UCS) CI = 81 MPa (73% UCS) CI = 90 MPa (81% UCS)		
Notes: Top is the base-case for this analysis. Bottom represents response using CI=90 MPa or no strength degradation. Yield indicators (dots around rectangular openings) represent EDZ development. Compiled from figures 6.2 to 6.7 of the Geomechanical Stability Analysis report (NWMO11t).		

Figure 4-93: Evolution of Plasticity Around Caverns After 2 and 8 Glacial Cycles for Different Long-Term Strengths

4.5.5 Future Evolution Summary

Based on the analysis of potential future natural processes and events that could occur at or near to the Bruce nuclear site, it is determined that none will compromise the operation and the long-term integrity of the DGR. Evidence to support this conclusion is summarized below.

- The erosion expected due to glacial activity is likely to total only about 100 m over the next 1 Ma.
- Permafrost depths are not expected to exceed 60 m at the Bruce nuclear site.
- Groundwater recharge beneath the glacier at the Bruce nuclear site will not penetrate farther than the Silurian Salina Group.

- The Bruce nuclear site lies within the stable interior of the North American craton, an area characterized by low rates of seismicity. Seismic events recorded in the region are M<5.

- A PSHA revealed that the surface ground motions are expected to be less than 60.1 % g for annual exceedance frequencies of 10^{-5} (reference case) and 10^{-6} (extreme case).

- The potential for fault rupture or reactivation is extremely low given the location, seismic history and neotectonic evidence of the Bruce region.

- Volcanic activity is not expected to influence the Bruce nuclear site at timeframes relevant to DGR safety.

Shaft long-term stability was assessed through a series of geomechanical modelling scenarios. Conclusions are summarized below.

- Most rock damage in the shaft is caused during the excavation phase of the project.

- The shafts will be backfilled at the end of the operational period. Consequently, the long-term shaft stability will not be an issue as the seals provide stability to the shafts over the long- term. The depth of damage, for all load combinations after 1 Ma, exceeds the shaft radius (by a maximum of 28%) only in the case of the very weak Cabot Head Formation. Otherwise, the depth of damage is typically in the range of 60% to 70% of the shaft radius or less.

- Seismic shaking and glacial loading are practically inconsequential for the EDZ and performance of the shafts.

A comprehensive suite of analyses have been performed to date to test the repository design and the Cobourg against the challenges imposed by stress, material strength degradation, fracture generation, seismic loading, pore pressure effects, and multiple glacial cycles over a period of 1 Ma. Conclusions are summarized below.

- The caverns will be stable during construction and operation, requiring only standard support. They will suffer increasing degradation over 60 ka as the long-term strength is reached.

- For time-dependent strength degradation under in-situ stress conditions and assuming a long-term strength of 45 MPa (40% UCS), no breakouts are predicted with yielding along the bedding planes in the roof and the floor limited to a depth of approximately 2 m.

- Gas and pore pressure variations within the caverns do not have a significant effect on damage around the caverns or the breakout depth. The preferential direction for potential hydraulic fracturing is horizontal, along the bedding planes, perpendicular to the vertical minor principal stress. Under the assumption of a high gas generating rate (resulting in maximum gas pressure of 15 MPa), bedding-parallel fractures may propagate up to 5 m beyond the cavern walls. However, the gas pressures, in all analyzed cases, will not generate hydraulic fractures that can result in gas release into the biosphere.

- Multiple glacial events and associated loading/unloading cycles are expected to cause failure of the pillars between the caverns and cavern collapse eventually. The number of glacial cycles that will cause pillar collapse and the timing of the pillar collapse depend on the long-term strength of the Cobourg Formation. Even using a conservative assessment for the Cobourg long-term strength of 45 MPa, the caverns will stay open for at least 100 ka. For a realistic assumption of the long-term strength of the Cobourg using 72 MPa (65% UCS), the pillars and the caverns are expected to remain stable even after 1 Ma.

- In the event of total collapse under the assumption of lower-bound strength of 45-MPa (40% UCS), rubble that accumulates inside the caverns as a result of collapses during multiple loading/unloading cycles will eventually arrest further propagation of the caved region due to volume increase. A steady state is reached when glacial cycles cause no further expansion of the damaged or caved regions. Reasonable assumptions indicate full pillar load capacity for 7 to 8 glacial cycles. Importantly, the models predict that the steady state is reached prior to propagation of the caving related damage into the Blue Mountain shale, the lowest unit of the shale cap rock. Therefore, all damage remains contained within the Cobourg under all loading conditions.

- The 3D panel-scale analysis shows that deformation of the cap rock due to potential complete pillar collapse, when assuming a lower-bound long-term strength of 45 MPa (40% UCS) for the Cobourg, will cause no or insignificant damage in the cap shales including the Blue Mountain shale. Thus, the repository-induced damage is contained within the Cobourg under all loading conditions.

- The analyses show that the seismic effect on cavern stability is relatively small. Seismic shaking causes some additional unravelling of already fractured rock mass, but no new damage is predicted irrespective of the probability level of the seismic events.

4.6 Geoscience Summary

This chapter provides an assessment of the Bruce nuclear site with respect to its geologic suitability for implementation of OPG's proposed DGR concept. This assessment is supported by a number of specific geoscience reports commissioned by the NWMO as part of the overall Geosynthesis program described above. In addition to these studies, the Geosynthesis integrates the results of the Bruce nuclear site Geoscientific Site Characterization activities that comprise detailed site investigations including drilling programs, field testing, laboratory analyses and geophysical surveys (NWMO11c, NWM11k).

Chapter 3 outlines seven key hypotheses regarding site attributes and characteristics that, if satisfied, will provide confidence that the geologic setting of the Bruce nuclear site is suitable to host the DGR. These hypotheses are generally adopted, in some fashion, by radioactive waste programs internationally. The scientific support that can be developed for the hypotheses provides regulators, the scientific community and other stakeholders with multiple lines of evidence to allow them to judge site suitability. The seven hypotheses and the supporting evidence are presented below.

4.6.1 Predictable: Near-Horizontally Layered, Undeformed Sedimentary Shale and Limestone Formations of Large Lateral Extent

- The occurrences of individual bedrock formations, facies assemblages, marker horizons, and major minerals, and the distribution of hydrocarbons and karst, are predictable and traceable at the site-scale (Section 4.1.2.2). Comparing the Paleozoic bedrock stratigraphy encountered in the DGR boreholes to that derived from an assessment of historic oil and gas well records demonstrates traceability at the local scale (e.g., Texaco #6 well) and indicates a high degree of consistency with the regional stratigraphic framework as discussed in Section 4.1.2.2 herein and Section 3.13 of the DGSM (NWMO11k).

- The thickness and orientation of bedrock formations encountered beneath the Bruce nuclear site are highly consistent as indicated by the dataset shown in Tables 3.1 and 3.2 of the DGSM (NWMO11k). Within an area of approximately 1.5 km^2 enclosing the DGR footprint, information derived from the deep drilling and coring program confirms that Ordovician formation thickness variations are on the order of meters (Table 4-2 herein). Formation dips within the same chronostratigraphic sequence are uniformly 0.59° +/- 0.08° (≈10 m/km) to the southwest towards the Michigan Basin (Section 4.1.2.2).

- The results of the 2D seismic reflection survey (19.7 km of data collected) provide evidence for the traceable nature of the bedrock stratigraphy beneath the site (WATTS09). The inclined drilling and coring of DGR-5 and DGR-6 targeted potential subvertical faults or fault zone structures in proximity to the DGR footprint.

Continuous core retrieved from both boreholes showed no evidence of faulting or stratigraphic offset through the target interval as discussed in Section 4.1.2.3 herein and in Section 3.11.4 of the DGSM (NWMO11k).

- Evidence supporting vertical fault displacement or the occurrence of steeply oriented linear and elongate hydrothermally dolomitized reservoirs within the Ordovician carbonate rocks is absent. No proximal deep-seated fault system was identified during the 2D seismic survey as discussed in Section 4.1.2.3.

- As discussed in Section 4.1.1 and shown in Figure 4-4, mapped faults are not known to penetrate Paleozoic sedimentary rocks younger than Ordovician age within the regional study area (ARMSTRONG10). This is consistent with the results of the detailed fracture mapping study, which found no evidence for complex fault structures or shear zones in the exposed bedrock proximal to the site (NWMO11ab), and is also consistent with the results of the 2D seismic survey described above (WATTS09).

4.6.2 Seismically Quiet: Comparable to Stable Canadian Shield Setting

- The Bruce nuclear site is located within the tectonically stable interior of the North American continent, which is characterized by low rates of seismicity. No earthquake exceeding magnitude 5 has been observed in the Regional Monitoring Area in 180 years of record. The maximum earthquake within the 150 km radius study area is an M4.3 event at 99 km from the site (15 km north of Meaford, Ontario) with a focal depth of about 11 km. This is consistent with the seismic hazard information provided in the 2005 National Building Code of Canada (NBCC05), as discussed in Section 4.5.2.1.

- A neotectonic remote-sensing and field-based study that analysed Quaternary landforms for the presence of seismically induced soft-sediment deformation concluded that the Bruce nuclear site has not likely experienced any post-glacial tectonic activity as discussed in Section 4.5.2.1 (NWMO11v). No evidence has been found for the presence of structural features that would indicate a higher seismic hazard near the Bruce nuclear site than that estimated from the regional rate of earthquake occurrence (NWMO11ab, NWMO11k).

- The micro-seismic monitoring network installed and commissioned in August 2007 confirms the lack of low-level seismicity (>M1.0) within the vicinity of the Bruce nuclear site, implying no seismogenic structures or faults within or in close proximity to the DGR footprint (Section 4.5.2.1).

- Based on the results of a PHSA performed for the Bruce nuclear site, the far field/regional seismic sources are the dominant contributors to the hazard for the site at ground level. The estimated surface bedrock peak ground motions are 18.7 and 60.1 % g for events of annual probabilities of 10^{-5} and 10^{-6}, respectively (Section 4.5.2.1, NWMO11w).

- Seismic analysis of a DGR cavern using ground motions of 10^{-5} and 10^{-6} annual probability events reveals that seismic shaking would not induce damage to the host rock other than dislodging already fractured rock mass around the opening.

4.6.3 Multiple Natural Barriers: Multiple Low-Permeability Bedrock Formations Enclose and Overlie the DGR

- The sedimentary sequence underlying the Bruce nuclear site comprises 34 near horizontally layered, laterally continuous bedrock formations (Section 4.1.2.1). Within the Ordovician sediments that host and enclose the proposed DGR are numerous units characterized as aquicludes that posses extremely low rock mass permeabilities. The host Cobourg Formation has a very low horizontal hydraulic conductivity (K_H) ≈ 10^{-14} m/s. The overlying > 200 m of Ordovician shales (3 formations) have rock mass horizontal hydraulic conductivities <10^{-13} m/s. The underlying 150 m of Ordovician carbonates (5 formations) have K_H values ranging from ≈10^{-15} to 10^{-10} m/s. Above the Ordovician sediments, the Silurian sediments have K_H values, which are on the order of <10^{-11} m/s. These values are presented in Section 4.4.1 herein and in Section 4.9 of the DGSM (NWMO11k).

- The Appalachian Basin has gas traps below the Marcellus black shale that reach more than 70% of the overburden stress. The Marcellus black shale is also overpressured throughout the northern Appalachian Basin, leaving no doubt about its effectiveness as a regional seal (Section 4.1.2.3). In a similar manner, the underpressured nature of the Ordovician shales beneath the Bruce nuclear site (see Section 4.4.4.1) indicates that this sedimentary package represents a long-lived and stratigraphically controlled cap rock seal (NWMO11y).

- Other site-scale observations which provide further evidence for the long-term barrier integrity of the Ordovician shale cap rock include: sealed fractures filled with calcite, gypsum/anhydrite, and/or halite (e.g., Figure 4-15 in Section 4.1.2.2), low formation hydraulic conductivities (Section 4.4.1), a low degree of thermal maturation, which inhibited the pervasive development of NHFs and commercial hydrocarbon accumulations (Section 4.1.1.2), and compartmentalization of the minor hydrocarbon phases present (Figure 4-16 in Section 4.1.2.2).

- No seismically-imaged faults are interpreted to have breached the top of the Upper Ordovician shale-dominated sedimentary package (e.g., Figure 4-18 in Section 4.1.2.3) (WATTS09).

- No geochemical evidence has been found for the infiltration of glacial or recent meteoric recharge water into the host or bounding formations. The stable water isotopes (^{18}O and ^{2}H) indicate that the maximum depth of glacial meltwater penetration is 328.5 mBGS (reference depth in DGR-1/2) within the Salina A1 carbonate aquifer (Section 4.3.2.3). Further, the results of numerical simulations – paleohydrogeology – provide insight into long-term groundwater system performance, and indicate: 1) that glacial perturbations do not alter the governing solute transport mechanisms within the deep groundwater system; and 2) that single and multiple glaciation scenarios, when modelled using regional and site specific parameters, do not result in the infiltration of glacial meltwater into the deep groundwater system (Section 4.4.4.2).

4.6.4 Shallow Groundwater Resources are Isolated: Near-Surface Groundwater Aquifers are Isolated from the Deep Saline Groundwater System

- Regionally, the hydrogeochemistry of the Michigan Basin defines two distinct groundwater regimes: i) a shallow bedrock system containing potable groundwater at depths above 200 m; and ii) an intermediate to deep saline system characterized by elevated TDS (>200 g/L) and distinct isotopic signatures (Section 4.3.1; NWMO11q). A similar relationship is observed at the site-scale where a shallow potable water zone is defined down to approximately 170 mBGS (Section 4.3.2).

- Groundwater resources in the vicinity of the Bruce nuclear site are obtained from shallow overburden or bedrock wells extending to depths of ca. 100 m into the permeable Devonian carbonates (NWMO11k). At increasing depth groundwater becomes brackish and then saline (non-potable) as discussed in Section 4.3.2, and yields decrease. This would prevent or discourage deep drilling for water resources.

- Evidence of modern karst is observed to a depth of approximately 180 mBGS below the Upper Silurian (NWMO11z) as discussed in Section 4.1.2.2. Conditions necessary to generate karst connections to the shallow groundwater system do not exist within the intermediate or deep groundwater systems.

- Groundwater modelling illustrates that the Guelph Formation is the upper boundary for vertical radionuclide transport from the repository, whether by advection or diffusion; water-borne radionuclides would not reach the shallow groundwater system at the Bruce nuclear site through the far-field even after millions of years (Section 4.4.4.1).

- Observed abnormal hydraulic heads in the Ordovician and Cambrian rocks and high vertical hydraulic gradients strongly suggests: i) extremely low rock mass hydraulic conductivities at formation scale; and ii) that vertical transmissive connectivity across bedrock aquitards/aquicludes is highly unlikely (Section 4.4.4.3 herein and Section 4.12 of NWMO11k).

4.6.5 Geomechanically Stable: Selected DGR Limestone Formation will Provide Stable, Virtually Dry Openings

- Precedent construction experience with the excavation of underground openings in southern Ontario reveals that excavated openings in the Ordovician shale and Ordovician limestone are mostly dry and stable as discussed in Section 4.2.2 herein and in Section 4.1 of the Regional Geomechanics report (NWMO11n).

- The laboratory testing of the Cobourg Formation core rock samples reveals a high strength argillaceous limestone with an average UCS value of 113 MPa (Section 4.2.2). These rock strength conditions compare favourably with other sedimentary formations considered internationally for long-term radioactive waste management purposes, as discussed in Chapter 7 (see Table 7.1 therein) of the Geosynthesis (NWMO11c).

- The fact that no borehole breakouts were observed in the deep DGR boreholes over a 24 month timeframe provides a constraint on the possible range of in-situ stress magnitudes beneath the Bruce nuclear site. At the repository horizon, the range of stress ratios is estimated to be: σ_H/σ_V from 1.5 to 2.0; σ_h/σ_V from 1.0 to 1.2 (Section 4.2.3). Observed minor borehole deformation strongly suggests that the orientation of maximum horizontal stress is similar to that of the Michigan Basin, a NE to ENE direction (Section 4.2.4).

- 3D numerical modelling results suggest that due to the shaft's vertical geometry and the confinement created by the shaft backfill, glacial loading has only a minor effect on the EDZ along the shaft. Similarly, pore pressure and seismic shaking will not significantly increase the predicted damage zone around the shaft. The maximum extent of the damage zone is less than 1.28 times the shaft radius. Otherwise, the depth of damage is typically 0.7 times the shaft radius or less (Section 4.5.4.2).

- Numerical simulations of repository evolution illustrate, under varied long-term rock mass properties and loading scenarios (i.e., glacial ice sheet, seismic ground motions and repository gas pressure), that the barrier integrity of the enclosing Ordovician bedrock formations is unaffected (Section 4.5.4.3).

4.6.6 Contaminant Transport is Diffusion Dominated: Deep Groundwater Regime is Ancient Showing No Evidence of Glacial Perturbation or Cross-Formational Flow

- Horizontal hydraulic conductivities (K_H) within the Cobourg Formation (DGR host rock), the overlying Ordovician shales (Georgian Bay, Blue Mountain and Queenston formations, and the Collingwood Member), and underlying Ordovician limestones and dolostones (Sherman Fall, Kirkfield, Coboconk, Gull River, and Shadow Lake formations) are extremely low ($\approx 10^{-15}$ to 10^{-10} m/s) (Section 4.4.1 herein and see also Section 4.9 of NWMO11k). Vertical hydraulic conductivities (K_V) within the same formations are lower (Section 4.4.4.3). Such conditions are consistent with a diffusion dominated regime.

- The effective diffusion coefficient (D_e) for HTO in the Ordovician shales is on the order of 10^{-12} m^2/s, and in the carbonates 10^{-13} to 10^{-12} m^2/s. D_e values obtained with HTO are on average 1.9 times greater than D_e values obtained with an iodide tracer. This difference is attributed to the influence of anion exclusion in lowering the tracer-accessible porosity for iodide (Section 5.3.5 of NWMO11c). The low D_e values, coupled with the low hydraulic conductivities of the Ordovician sediments, indicate that solute migration is diffusion dominated in the deep groundwater system (Section 4.3.2.4).

- The occurrence of isotopically distinct types of methane and helium in separate zones (one zone in the Upper Ordovician shale and another zone in the Middle Ordovician carbonates) demonstrates that there has been little to no cross-formational mixing (advective or diffusive) while these gases were resident in the porewater. The sharp isotopic gradients observed in both the methane and the helium in all DGR boreholes near the Cobourg Formation-Sherman Fall Formation contact, and the lack of apparent mixing of the respective solutes, suggests that a barrier to solute migration is present at that horizon (Section 4.3.2.3).

- The radiogenic $^{87}Sr/^{86}Sr$ ratios in the Middle and Upper Ordovician porewater are interpreted to result from a combination of water-rock interaction, in-situ ^{87}Rb decay, and diffusive transport upward from the shield. These mechanisms suggest extremely long residence times (Section 4.3.2.3).

- The chemistries of the deep brines indicate that they were formed by evaporation of seawater, which was subsequently modified by fluid-rock interaction processes (Section 4.3.1). The Cl/Br and Na/Cl ratios, as well as the stable water isotope data for the site, suggest that the deep groundwater system contains evolved ancient sedimentary brines at, or near, halite saturation. The nature of the brines, in particular the high salinities and the enriched ^{18}O values (enriched in ^{18}O with respect to the GMWL) of the porewaters, indicate that the deep system is isolated from the shallow groundwater system and that the porewaters have resided in the system for a very long time (Section 4.3.2.3).

- Illustrative hydrogeochemical modelling suggests that the currently measured natural tracer (^{18}O, Cl) profiles could evolve by diffusion from baseline conditions (evaporated seawater composition) in the time frame of approximately 300 Ma (Section 4.3.3).

4.6.7 Natural Resource Potential is Low: Commercially Viable Oil and Gas Reserves are Not Present

- No commercial oil hydrocarbon accumulations were encountered during site characterization activities as discussed in Sections 4.1.2.2 and 4.5.3.1. No structural, lithological, chemical or hydrological evidence suggests that the Bruce nuclear site is proximal to an ancient HTD system as discussed in Sections 4.1.2.2, 4.1.2.3 and 4.4.4.3.

- An average TOC content of the Upper Ordovician shales of less than 1.0% (Figure 4-16 and see also discussion in Section 4.5.3.1), the recognition of low thermal maturity throughout the regional study area which indicates that these sedimentary rocks only reached the lower threshold of the oil window as discussed in Section 4.1.1.2 (LEGALL81, OBERMAJER96, NWMO11y), and the absence of natural gas shows during drilling of the DGR boreholes (NWMO11k) argues against the likelihood of commercial accumulations of either thermogenic or biogenic shale gas beneath the Bruce nuclear site (NWMO11y).

- Lateral traceability between the Bruce nuclear site boreholes and other proximal dry wells (e.g., Union Gas #1 and Texaco #6), demonstrates that locally around the Bruce nuclear site (~7 km radius), no pockets of oil or gas hydrocarbon are likely to exist, as discussed in Section 4.5.3.1 herein and in Section 3.13 of the DGSM (NWMO11k).

- A transition from fresh to saline groundwater is recorded through the shallow and intermediate hydrogeological systems with saline groundwater dominating below 200 mBGS depth within the Silurian Salina F Unit (Section 4.3.2). A transition into more permeable rock occurs in the lower Ordovician and the underlying Cambrian sandstone (830 mBGS). The porewater at the repository depth (680 mBGS) is not potable (TDS >200 g/L) and this extremely low permeability bedrock formation (hydraulic conductivities <10^{-13} m/s) cannot yield groundwater (e.g., Section 4.3.2). This combination of extremely high salinities and low hydraulic conductivities in the rock surrounding the proposed repository depth would discourage deep drilling for groundwater resources.

- Sphalerite (Lucas and Georgian Bay formations), marcasite (Kirkfield Formation and Cambrian), and pyrite (entire Paleozoic interval) are present in trace amounts within the host rock and secondary vein infillings as described in Section 3.10 of the DGSM (NWMO11k). These occurrences are not associated with any commercially exploitable base metal accumulations. MVT mineralization occurs in the Middle Silurian dolostones in southern Ontario as a minor diagenetic constituent but no commercial MVT deposits have been found within southern Ontario as discussed in Section 10.2.1 of the Regional Geology report (NWMO11m).

- The Salina salt does not represent a commercial resource because it has been dissolved and removed beneath the Bruce nuclear site through natural processes in the Paleozoic.

Thus, given all the information summarized above that supports the key hypotheses, thegeological setting at the Bruce nuclear site is suitable to support the development of a DGR for L&ILW in the Cobourg Formation.

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5. WASTE INVENTORY

5.1 Introduction

This chapter summarizes the physical, radiological and chemical characteristics of the wastes and the containers to be emplaced in the DGR. More detailed information is presented in the Reference L&ILW Inventory report (OPG10a).

5.2 Waste Classification

Radioactive wastes to be accepted by the DGR are classified as solid low level or solid intermediate level. The classification is as described below, and is consistent with CSA N292.3 (CSA08a).

Low Level Waste (LLW) consists of non-fuel waste in which the concentration or quantity of radionuclides is above the clearance levels and exemption quantities established by the Nuclear Substances and Radiation Devices Regulations (SOR/2000-207), and which contain primarily short-lived radionuclides (half-lives shorter than or equal to 30 years). LLW normally does not require significant shielding for worker protection during handling and storage. OPG LLW typically consists of: incinerator ash; compacted waste; bulk and drummed non-processible wastes; some low activity ion-exchange (IX) resins and filters from secondary side reactor process systems; and system components such as heat exchangers, feeder pipes and steam generators.

Intermediate Level Waste (ILW) consists of non-fuel waste containing significant quantities of long-lived radionuclides. ILW often requires shielding for worker protection during handling. OPG ILW typically consists of primary side and moderator IX resins and filters; irradiated core components; and reactor fuel channel wastes from refurbishment activities.

The L&ILW are generated from a variety of activities. For the purposes of this report, the wastes have been divided into two broad categories: operational wastes (which includes all L&ILW from operation and maintenance of the reactors and their associated facilities) and refurbishment wastes (which includes component waste from major refurbishment projects, such as pressure tubes, calandria tubes, end-fittings, steam generators, and associated hardware). A third general category, decommissioning waste (which includes waste from the final dismantling of reactors and facilities) is not included in this assessment.

The DGR will not accept used fuel or recognizable fuel fragments. The DGR also excludes liquid wastes, except for small amounts of incidental liquids that are inevitably associated with the solid wastes.

5.3 Waste Types and Categories

A wide variety of waste types are generated by nuclear generating stations. OPG currently tracks about 70 different waste types. However, many of these are small volume items, or have similar properties to other waste types. Therefore, for purposes of describing the DGR waste inventory, these waste types have been grouped into about 20 waste categories[1]. Table 5-1 and Table 5-2 provide more detailed descriptions of various L&ILW categories tracked for the DGR (OPG10a).

Table 5-1: LLW Categories

Waste Category	Description
Bottom Ash	Heterogeneous ash and clinker from waste incineration.
Baghouse Ash	Fine homogeneous ash from waste incineration.
Compact Bales	Generally compactible solid LLW, for example, empty waste drums, rubber hoses, rubber area floor matting, light gauge metals, welding rods, plastic conduit, fire blankets and fire retardant material, metal cans, insulation, ventilation filters, air hoses, metal mop buckets and presses, electric cable (<1/4" diameter), lathe turnings, metal filings, glass, plastic suits (Mark III/IV), rubbers, Vicraft hoods, rubber gloves.
Box Compacted	Same as compact bales.
Non-Processible Boxed	Solid LLW that is non-compactible or has contact dose greater than 2 mSv/hr, for example, heavy gauge metal (i.e., beams, IX vessels, angle iron, plate metal), concrete and cement blocks, metal components (i.e., pipe, scaffolding pipes, metal planks, motors, flanges, valves), wire cables and slings, electric cables (>1/4" diameter), Comfo respirator filters, tools, paper, plastic, absorbent products, laboratory sealed sources, feeder pipes.
Non-Processible Drummed	Generally small, granular or solidified LLW, for example, floor sweepings, cleaners and absorbents (e.g., Dust Bane, Stay Dry), metal filings, glassware, light bulbs, bitumenized LLW.
Non-Processible Other	Large and irregularly shaped objects such as heat exchangers, Encapsulated Tile Holes (ETHs), shield plug containers, and other miscellaneous large objects (e.g., fume hoods, glove boxes, processing equipment).
Low Level / ALW Resin	Spent low level IX resin arising from light water auxiliary systems, and/or Active Liquid Waste (ALW) treatment systems.
ALW Sludge	Sludge from Bruce two-stage ALW Treatment System.
Steam Generators	Steam generators removed from service.

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[1] For handling purposes, waste has been divided into four "waste handling groups" as described in Section 5.4, based on size, mass and handling features.

Table 5-2: ILW Categories

Waste Category	Description
Moderator Resin	Spent IX resin arising from moderator purification systems.[a]
Primary Heat Transport (PHT) Resin	Spent IX resin arising from PHT purification systems.[a]
Miscellaneous Resin	Spent IX resin and activated charcoal arising from station auxiliary systems (e.g., heavy water upgraders).[a]
CANDECON Resin	Spent IX resin from chemical decontamination process for nuclear heat transport systems.
IX Columns	Spent IX resin mainly arising from Pickering PHT purification system, comes as package with steel container.
Irradiated Core Components	Various replaced core components, notably flux detectors and liquid zone control rods.
Filters and Filter Elements	Filters and filter elements from various station process systems.
Retube - Pressure Tubes	Fuel channel waste from large scale retube.
Retube - Calandria Tubes	Fuel channel waste from large scale retube.
Retube - Calandria Tube Inserts	Fuel channel waste from large scale retube.
Retube – End-Fittings	Fuel channel waste from large scale retube.

Note:
a. These ILW resins often occur mixed together in the same container in varying proportions.

5.4 Waste Containers and Packages

There are currently in excess of 100 different waste container types that have been used for storage of L&ILW at the WWMF since it went into service in the mid-1970s. For the purposes of this PSR, containers of similar design have been grouped and only containers typical of those found in each waste category are described.

The combination of wastes plus container is defined as a waste package. Some waste packages currently stored at WWMF meet DGR waste acceptance criteria (see Section 5.5) and are considered DGR-ready. Others will require some waste conditioning, additional decay time, and/or container overpacking or shielding.

For the reference inventory forecast, it is assumed that all ash bins, low level resin boxes, ALW sludge boxes, and 10% of drum racks, will be placed in a standard LLW container overpack at the time of retrieval to form a repository waste package.

Steam generators, and if necessary heat exchangers, would be segmented to meet the size and weight limits of the DGR main shaft cage. They would be grouted to stabilize the contents, cut into sections, and have seal plates welded on the ends. Some steam generators may be processed to recycle the inactive steel components, and the residual parts with all the radioactivity transferred to the DGR in LLW non-processible or similar containers; however, the reference forecast conservatively assumes segments.

ILW resins are stored in steel resin liners. Depending on the dose rates of the resin liners, they would be placed into disposable concrete cylindrical shield overpacks, which ensure that the resulting dose rates do not exceed the DGR waste acceptance criteria. The reference overpack options are a 250 mm thick concrete shield that holds two resin liners, a 350 mm shield that holds two resin liners, and a short 350 mm shield with a steel insert that holds one resin liner. It is expected that approximately one-third of resin liners will not require a concrete shield.

ILW irradiated core components, filters and filter elements and IX columns are presently stored in long tile-hole-equivalent (T-H-E) liners within ICs, i.e., IC-2s and IC-18s. OPG's reference plan is that this existing ILW will be removed from the T-H-E liners and repackaged into smaller containers i.e., alternative tile-hole-equivalent liners (ATHELs) for later transfer to the DGR. These would be placed into disposable concrete cylindrical shield overpacks similar to the ones used for ILW resin liners, which ensure that the resulting dose rates do not exceed the DGR waste acceptance criteria. It is assumed that post-2018 newly arriving ILW of this type will be placed directly into smaller containers (e.g., ILW shields or ATHELs) for later transfer to the DGR.

The tile hole liners and ETH liners, which contain similar wastes except different activity levels, will be transferred to the DGR as-is.

As part of reactor lifecycle management activities, the current assumption is that replaced feeders will be cut into suitable lengths and packaged in non-processible (box) containers. Retube wastes will be emplaced using two sizes of shielded containers: RWC(PT) for pressure tubes, calandria tubes, and calandria tube inserts, and RWC(EF) for end-fittings.

The various representative DGR container types along with their associated container tracking codes and waste handling groups are listed in Table 5-3 and Table 5-4 (OPG10a). The container codes are used to facilitate identification of the container characteristics. For handling purposes, the containers have been divided into four groups based on size, mass and handling features. These groupings have been used for logistics simulation, developing waste package transfer methods and determining emplacement room sizing and layouts. Further explanation of the groupings can be found in Section 6.4.

Table 5-3: Representative Primary LLW Package Types

Waste Category	Container Type	Container Code	Waste Handling Group
Operational Wastes	Ash Bin (Old) – Bottom Ash	AIB02[a]	A
	Ash Bin (New) – Bottom Ash	AIBN[a]	A
	Drum Rack – Baghouse Ash	DRACK[a]	A
	Ash Bin (New) – Baghouse Ash	AIBN[a]	A
	Compactor Box	B25	A
	Bale Rack	BRACK	A
	Non-Processible Bin (47" high)	NPB47	A
	Non-Processible Bin	NPB4	A
	Drum Rack – Non-Processible Drums	DRACK[b]	A
	Drum Bin	DBIN	A
	Shield Plug Container	SPC	B
	Heat Exchanger	HX	B
	Encapsulated Tile Hole	ETH	D
	Low Level Resin Box (90")	RB90[a]	A
	Low Level Resin Pallet Tank	RTK	A
	ALW Sludge Box	NPBSB	A
Refurbishment Wastes	Steam Generator Segments	SGSGMT	D

Notes:

a. All containers of this type assumed to be overpacked in type BINOPK containers.

b. Some containers of this type assumed to be overpacked in type BINOPK containers.

Table 5-4: Representative Primary ILW Package Types

Waste Category	Container Type	Container Code	Waste Handling Group
Operational Wastes	Resin Liner	RL	C
	Resin Liner Overpack	RLOPK	C
	Resin Liner – 250 mm Concrete Shield	RLSHLD1	D
	Resin Liner – 350 mm Concrete Shield	RLSHLD2	D
	Resin Liner – 350 mm Concrete Shield + Steel Insert	RLSHLD3	D
	ATHEL Waste Package – 350 mm Concrete Shield	ATHELSHLD	D
	ILW Shield	ILWSHLD	C
	Tile Hole Liner	THLSTG3	C
Refurbishment Wastes	Pressure Tubes	RWC(PT)	D
	Calandria Tubes	RWC(PT)	D
	Calandria tube Inserts	RWC(PT)	D
	End-Fittings	RWC(EF)	D

5.5 Waste Acceptance Criteria

All LLW and ILW will be shipped or transferred to the DGR Facility in waste packages that meet the DGR waste acceptance criteria.

The DGR waste acceptance criteria have been developed to ensure that the wastes emplaced in the DGR are within the bounds of the safety assessment, design basis and regulatory requirements. The criteria are summarized in Table 5-5.

Table 5-5: Summary of Waste Acceptance Criteria

Criteria	Summary Description
Waste characterization	- physical, chemical, radiological characteristics of each package
Documentation	- waste packages must be tracked in OPG's waste tracking database with waste characteristics, dose rates, description of contents, etc. - verified load statements - supplemental info such as radiological surveys, chemical analyses, loading checklists - notes on package design documentation, such as drawings, technical specifications, design requirements, etc. - transfer documents for wastes subject to additional controls
Acceptable waste package designs	- all DGR waste package designs must be approved
Condition of waste container	- no significant rusting - sound structural integrity - no leakage - no wobbling or tilting
Mass limits	- 35 Mg, subject to maximum design limit for each waste package type
Size limits	- must fit within internal dimensions of the DGR cage
Containment	- wastes and contamination shall be contained during handling - all containers shall have lids
Venting	- where the potential for gas build-up exists and containers are not designed to withstand the pressure, the containers shall be vented
Identification/labelling	- containers bar-coded with OPG's waste tracking database tracking number on two adjacent vertical sides - additional information including gross mass, dose rate, and significant non-radiological hazards to be marked on packaged with lettering at least 25 mm high
Stackability	- stable, self supporting stack of up to 6 m high - use of standard footprints strongly encouraged
Handling	- conventional material handling equipment such as forklifts with loads of up to 35 Mg
Fire resistance	- non-combustible containers
Dose rate limits	- 2 mSv/hr on contact with external surface of waste package or shielding - 0.1 mSv/hr at 1 m from transportation package - exceptions approved by responsible health physicist
Radionuclide composition	- package amount must be reported for H-3, C-14, Cl-36, Co-60, Sr-90, Zr-93, Nb-94, Tc-99, I-129, Cs-135, Cs-137, U-235, U-238, Pu-239, Pu-240, Pu-241
Contamination limits	- removable surface contamination on package exterior to be less than 4 Bq/cm^2 beta-gamma and 0.4 Bq/cm^2 alpha when averaged over 300 cm^2
Heat load limits	- no restriction if less than 0.01 W/m^3 of waste package external dimensions - up to 10 W/m^3 by prior notification and approval for special cases
Waste form	- solids only - sludges must have slump of less than 150 mm
Residual liquids	- generally must be less than 1% free liquid by volume - bulk IX resins must be less than 5% free water by volume
Gas generation	- must not generate toxic gas on exposure to water
Excluded wastes	- reactive wastes, polychlorinated biphenyl (PCB) wastes, pathological wastes, ignitable wastes - explosives, corrosives, compressed gases - used nuclear fuel and recognizable fuel fragments - high thermal Co-60 sources
Special notice wastes	- wastes containing significant levels of Occupational Health and Safety Act (OHSA90) designated substances - leachate toxic wastes
Chelating agents	- must be less than 1% by weight of package
Petroleum oils	- must be less than 1% by weight of package

5.6 Waste Characterization Program

Most L&ILW is inherently heterogeneous, with considerable variability both across waste categories, and also from package to package within a waste category. OPG has therefore supported a waste characterization and tracking program for many years. The characteristics of various waste types have been identified, and information recorded on waste packages in an electronic database (see Section 5.7).

The reference methodology for characterizing radioactive wastes is based on the use of gamma dose rates associated with each waste package to derive certain marker radionuclide inventories (usually Co-60 and Cs-137), and scaling factors to calculate the inventory of other radionuclides of interest. The scaling factors are derived either from experimental data on OPG waste samples, from theoretical arguments (such as activation analysis and computer modelling, comparison with incinerator stack emission data for volatile species, etc.), or from the ratio of these nuclides in used fuel. This process is consistent with international best-practice (IAEA09, ISO07).

The result of the waste characterization is a description of the radionuclide and chemical composition of the waste in a typical waste package for each of the DGR waste categories identified in Table 5-1 and Table 5-2. Furthermore, the radionuclide composition is decay-corrected from time of receipt at WWMF.

5.7 Waste Tracking

Waste containers and inventories stored at WWMF are presently tracked using OPG's Integrated Waste Tracking System electronic waste tracking database (ANDERSON05). This system, or a similar one, will be adopted for the DGR, so that waste packages will be tracked with respect to their location within the DGR. This system will contain information on the characteristics of each package, and will have the ability to produce reports on the waste inventory within the DGR at any time.

5.8 Waste Volumes

The amount of waste and number of packages projected over the life of OPG's nuclear program is calculated based on the existing inventory tracked in OPG's waste tracking database and a future waste receipt projection.

Based on the existing plus projected inventory, it is estimated that approximately 53,000 packages representing a total emplaced volume of approximately 200,000 m^3 will be sent to the DGR. About 80% of the emplaced volume is LLW. Note that while refurbishment waste only makes up about 10% of the emplaced volume, it accounts for more than 60% of the radionuclide inventory at 2062.

Note also that the “emplaced volume” is greater than the “as-stored volume”, which is the volume of the containers in which the waste is presently stored, due to the extent of overpacking and disposable shielding used for DGR-ready packages. These are both larger than the “net waste volume” which is the internal volume within the containers available for waste. The waste volume breakdown for the reference forecast is given in Table 5-6.

Table 5-6: Waste Volumes in Reference Forecast (Rounded)

	Operations LLW	Operations ILW	Refurbishment L&ILW	Total
Net waste volume (m^3)	95,100	9,300	11,200	115,600
As-stored volume (m^3)	135,000	13,500	21,700	170,200
Emplaced volume (m^3)	154,700	27,600	21,700	204,000

Table 5-7 provides a more detailed forecast of the numbers and volumes (by container type) for operational and refurbishment L&ILW for the DGR, based on the planning assumption of refurbishment of all reactor units (except Pickering A) at or near the end of their initial life, with operation for a further 30 years after refurbishment. The projection includes overpacking of wastes as described in Section 5.4. Note that, as shown in Table 5-7, two waste types “non-processible bins” and “LLW container overpacks” account for approximately 50% of the overall emplaced volume.

The waste volume forecast is subject to changes to the nuclear operating and refurbishment program plans, and to changes in technologies and practices such as improvements to waste processing or repository storage technology. For example, this forecast does not take into account OPG’s recent decision not to refurbish Pickering B. However, approximately half of the projected waste volume is already stored at the WWMF site. In addition, the projection is based on recent actual waste generation and processing rates. Therefore, within the context of the given nuclear operating and refurbishment plan, the overall waste volumes are expected to be similar to this forecast.

Table 5-7: DGR Reference Forecast L&ILW Package Inventory

Container Type	Number of Containers	Emplaced Volume (m^3)	Dimensions LxWxH (m)	Empty Mass (kg)	Avg. Full Mass (kg)	Comments
Bale Racks	1,383	4,702	2.29 x 1.22 x 1.2	150	1400	-
Compactor Boxes	6,135	17,177	1.84 x 1.12 x 1.3	486	2,722	-
Non-Processible Bins	24,164	73,483	1.96 x 1.32 x 1.19	360	1,460	Volume/mass based on NPB47 container
Drum Racks	2,903	9,870	2.29 x 1.22 x 1.2	150	1,490	6 drums per rack
Drum Bins	4,615	12,922	1.96 x 1.32 x 1.03	290	1,450	6 drums per bin
Low Level Resin Pallet Tanks	2,085	5,627	1.24 x 1.24 x 1.68	320	2,000	Dimensions/mass do not include overpack
LLW Container Overpacks	3,212	27,303	2.54 x 1.78 x 1.88	1,591	Max 5,400	Overpacking for 1,100 ash bins, 80 low level resin boxes, 1,709 ALW sludge boxes, and 323 drum racks
Shield Plug Containers	26	309	3.0 x 1.8 x 1.8	13,000	26,000	-
Heat Exchangers	98	2,775	Various e.g. 2 OD x 4.6 OL	n/a	10,000 – 30,000	Assume 25% of 98 heat exchangers will be segmented in half
Encapsulated Tile Hole	66	504	1.5 (OD) x 4.6 (OL)	n/a	25,000	-
Steam Generator Segments	512	8,387	1.8-3.6 (OD) x 2.0-4.3 (OL)	n/a	25,730	Mass does not include grout
Resin Liners	286	858	1.63 (OD) x 1.8 (OL)	795	4,545	Dimensions/mass does not include sacrificial pallet
Resin Liner Overpacks	400	1,640	1.68 (OD) x 1.91 (OL)	1,450	6,000	Dimensions/mass does not include sacrificial pallet
Resin Liner 250 mm Shield	646	10,467	2.2 (OD) x 4.25 (OL)	17,760	26,850	Two resin liners per shield
Resin Liner 350 mm Shield	164	3,295	2.4 (OD) x 4.45 (OL)	27,060	36,150	Two resin liners per shield
Resin Liner 350 mm Shield with Steel Insert	140	1,925	2.53 (OD) x 2.74 (OL)	24,420	28,965	One resin liner per shield
ATHEL Waste Package – 350 mm Concrete Shield	300	4,140	2.53 (OD) x 2.74 (OL)	20,900	23,500	One ATHEL waste package per shield
Tile Hole Liners	201	176	0.61 (OD) x 3.4 (OL)	450	2,000	Dimensions/mass does not include rack
Retube Waste Containers	1,353	13,298	1.70 x 3.35 x 1.92	29,200	33,500	Volume/mass based on Bruce A RWC(EF) container
ILW Shield	3,952	5,137	1.0 (OD) x 1.7 (OL)	2,015	2,290	Replaces T-H-E waste packages
Total (rounded)	**52,600**	**204,000**	-	-	-	-

Note:
Outer Diameter (OD); Outer Length (OL)

5.9 Radionuclide Inventory

Radionuclide activity concentrations in operational and refurbishment L&ILW are presented in the Reference L&ILW Inventory report (OPG10a). In that report, they are described as follows: typical as-received packages at WWMF (or DGR); total as-stored inventory at WWMF at 2018 (the assumed in-service date for the DGR); and total inventory in DGR at 2062 (the assumed date for completion of DGR decommissioning).

The estimated total radionuclide inventory in operational and refurbishment L&ILW at 2062 is summarized in Table 5-8. The values are based on the L&ILW characteristics given in Appendix B of the Reference L&ILW Inventory report (OPG10a), and the projected L&ILW volumes calculated for each year (past and historical) with decay-correction. The results for the assumed repository decommissioning date of 2062 indicate that the total radioactivity will be dominated by H-3, C-14, Nb-94, and Ni-63.

The majority of radionuclides are relatively short lived. For a waste storage site servicing a fixed number of reactors, the overall radionuclide inventory of L&ILW approaches a steady state value where new radionuclides coming in are largely off-set by decay of previously stored radionuclides. Note that while the overall radionuclide inventory for L&ILW approaches a steady state value, the percentage of longer-lived radionuclides in the waste will gradually increase due to the faster decay of the shorter-lived ones.

Figure 5-1 displays the time dependence of the projected operational L&ILW radionuclide inventory. With the gradual shutdown of reactor units, less new waste inventory is being received and overall inventory decreases relatively quickly due to decay of H-3 which is the dominant radionuclide in LLW. ILW decays more slowly because the dominant radionuclide is the much longer lived C-14.

Figure 5-2 displays the time dependence of the projected refurbishment L&ILW radionuclide inventory. For refurbishment waste, the peak inventory is achieved at 2020 (the forecast end of station refurbishment activities) followed by decay of the short-lived radionuclides, such as Fe-55 and Co-60. Some refurbishment waste packages may require a period of decay before being loaded into the DGR. After a few hundred years, Ni-63 is largely reduced and the total DGR inventory is dominated by C-14 and Nb-94 and, eventually Zr-93. The sharp peaks are created by the influx of new refurbishment waste associated with discrete refurbishment events followed by the rapid decay of short-lived Zr and Nb species (e.g., Zr-95 and Nb-95).

Table 5-9 is a summary of waste volume fractions handled and dose rates associated with the major operational L&ILW types at WWMF up to 2005. The data were obtained from dose measurements at the surface of the various L&ILW storage containers as-received at the WWMF. Note that these measurements are based on storage containers without significant shielding. The data is especially relevant for controlling worker dose during the operational phase of the repository. Waste packages transferred to the DGR will contain additional shielding as needed in order to meet the waste acceptance criteria (Table 5-5).

The uncertainties associated with the waste-type specific inventories are described in the Reference L&ILW Inventory report (OPG10a). The reference waste-type specific inventories are typically log-mean values or conservative estimates of as-received values at WWMF. Often, the inventories have large variability between packages of a given type. However, the total repository inventory is less uncertain because it is based on the reference waste-type inventories, decay-corrected and summed over a large number (often thousands) of packages. The impact of uncertainty in the inventory is handled in the safety assessments in various ways. In the operational safety assessment (Chapter 7), packages are typically assumed to be at as-received (i.e., no decay) or at waste acceptance criteria limit dose rates, with some calculations further considering ten times higher inventories for single packages. In the postclosure safety assessment (Chapter 8), the effects of ten times higher total inventory for all radionuclides is evaluated as a sensitivity case.

Table 5-8: Estimated L&ILW Radionuclide Inventory at 2062 (Bq)

Nuclide	Half-life (a)	Operations LLW	Operations ILW	Refurb. L&ILW	Total
Ag-108m	1.3E+02	3.3E+07	1.0E+09	2.0E+13	2.0E+13
Am-241	4.3E+02	5.5E+10	2.2E+11	2.1E+12	2.4E+12
Am-242m	1.5E+02	5.1E+07	0.0E+00	2.3E+09	2.4E+09
Am-243	7.4E+03	6.8E+07	1.7E+08	2.9E+09	3.1E+09
Ba-133	1.1E+01	7.1E+08	0.0E+00	0.0E+00	7.1E+08
C-14	5.7E+03	1.4E+12	5.4E+15	6.6E+14	6.1E+15
Cf-252	2.6E+00	1.2E+06	0.0E+00	0.0E+00	1.2E+06
Cl-36	3.0E+05	5.4E+08	7.4E+08	1.4E+12	1.4E+12
Cm-243	2.9E+01	0.0E+00	0.0E+00	2.7E+09	2.7E+09
Cm-244	1.8E+01	2.7E+09	7.0E+10	2.2E+11	2.9E+11
Co-60	5.3E+00	1.7E+11	3.5E+12	9.0E+14	9.0E+14
Cs-134	2.1E+00	5.6E+07	3.1E+10	3.1E+06	3.1E+10
Cs-135	2.3E+06	4.3E+06	1.3E+08	2.3E+08	3.6E+08
Cs-137+Ba137m[a]	3.0E+01	1.3E+13	9.4E+13	5.4E+11	1.1E+14
Eu-152	1.3E+01	3.7E+07	1.5E+12	1.2E+09	1.5E+12
Eu-154	8.8E+00	7.1E+09	1.2E+11	3.2E+09	1.3E+11
Eu-155	5.0E+00	5.1E+07	1.7E+09	3.3E+08	2.1E+09
Fe-55	2.7E+00	3.8E+10	3.8E+11	5.5E+13	5.5E+13
H-3	1.2E+01	8.5E+14	1.5E+14	4.8E+12	1.0E+15
I-129	1.6E+07	1.2E+06	1.3E+08	1.0E+06	1.3E+08

Nuclide	Half-life (a)	Operations LLW	Operations ILW	Refurb. L&ILW	Total
Ir-192m	2.4E+02	0.0E+00	4.9E+07	1.1E+10	1.1E+10
Mn-54	8.6E-01	0.0E+00	0.0E+00	2.7E+02	2.7E+02
Mo-93	3.5E+03	0.0E+00	4.5E+08	1.0E+12	1.0E+12
Nb-93m	1.4E+01	0.0E+00	2.9E+10	9.2E+12	9.2E+12
Nb-94	2.0E+04	2.2E+10	1.2E+11	4.6E+15	4.6E+15
Ni-59	7.5E+04	2.1E+09	3.6E+11	3.6E+13	3.6E+13
Ni-63	9.6E+01	2.4E+11	3.9E+13	3.9E+15	3.9E+15
Np-237	2.1E+06	3.2E+06	1.1E+07	1.2E+08	1.3E+08
Pb-210	2.2E+01	3.2E+10	0.0E+00	0.0E+00	3.2E+10
Pt-193	5.0E+01	0.0E+00	3.1E+09	1.1E+13	1.1E+13
Pu-238	8.8E+01	8.5E+09	2.7E+10	4.6E+11	5.0E+11
Pu-239	2.4E+04	2.2E+10	7.7E+10	8.2E+11	9.2E+11
Pu-240	6.5E+03	3.0E+10	1.1E+11	1.2E+12	1.3E+12
Pu-241	1.4E+01	6.8E+10	1.6E+12	3.0E+12	4.7E+12
Pu-242	3.8E+05	3.2E+07	1.0E+08	1.2E+09	1.3E+09
Ra-226	1.6E+03	3.8E+09	0.0E+00	0.0E+00	3.8E+09
Ru-106	1.0E+00	3.0E+06	1.5E+08	0.0E+00	1.5E+08
Sb-125	2.8E+00	3.4E+08	1.8E+11	3.9E+11	5.7E+11
Se-79	3.8E+05	1.5E+06	4.5E+06	1.3E+10	1.3E+10
Sm-151	9.0E+01	1.0E+07	3.2E+08	1.7E+09	2.0E+09
Sn-119m	8.0E-01	0.0E+00	0.0E+00	2.4E+01	2.4E+01
Sn-121m	5.5E+01	0.0E+00	5.9E+11	7.7E+13	7.8E+13
Sn-126	2.1E+05	2.3E+07	7.0E+08	1.2E+07	7.4E+08
Sr-90+Y90[a]	2.9E+01	3.0E+12	4.2E+13	9.3E+12	5.4E+13
Tc-99	2.1E+05	5.2E+07	8.4E+08	6.0E+10	6.1E+10
U-232	7.2E+01	4.9E+06	0.0E+00	2.3E+08	2.3E+08
U-233	1.6E+05	6.6E+06	0.0E+00	3.1E+08	3.2E+08
U-234	2.5E+05	3.6E+07	1.1E+08	1.3E+09	1.4E+09
U-235	7.0E+08	5.6E+05	1.9E+06	2.1E+07	2.3E+07
U-236	2.3E+07	6.4E+06	2.1E+07	2.5E+08	2.8E+08
U-238	4.5E+09	4.2E+09	1.4E+08	1.7E+09	6.0E+09
Zr-93	1.5E+06	1.6E+06	6.7E+11	2.1E+14	2.1E+14
Totals	**-**	**8.7E+14**	**5.7E+15**	**1.1E+16**	**1.7E+16**

Notes:
a. Activity listed is total for parent plus progeny in secular equilibrium.
b. 0.0E+00 indicates value is zero or not significant.

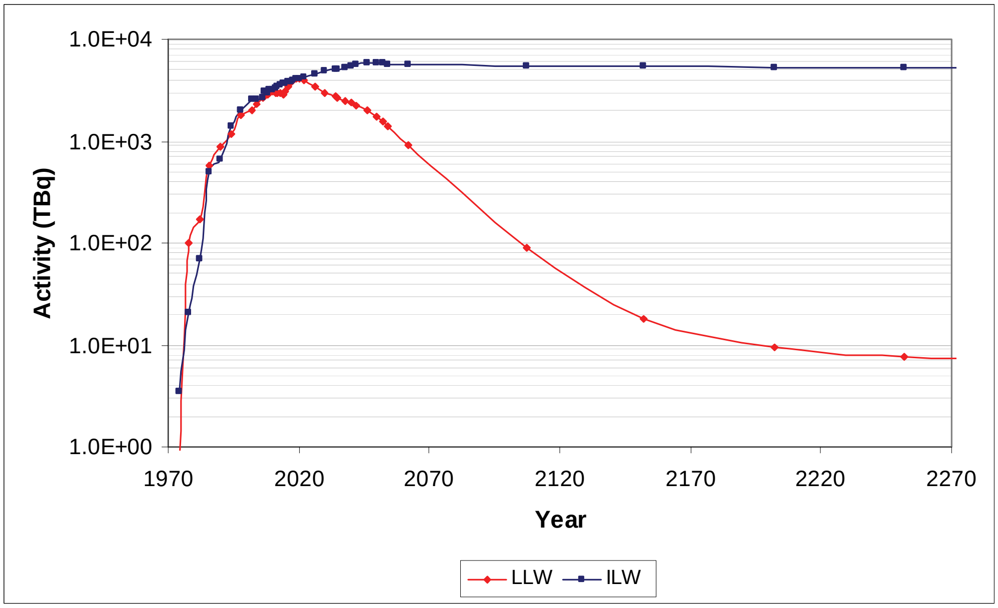

Figure 5-1: Time Dependence of Total Activity for Operational L&ILW

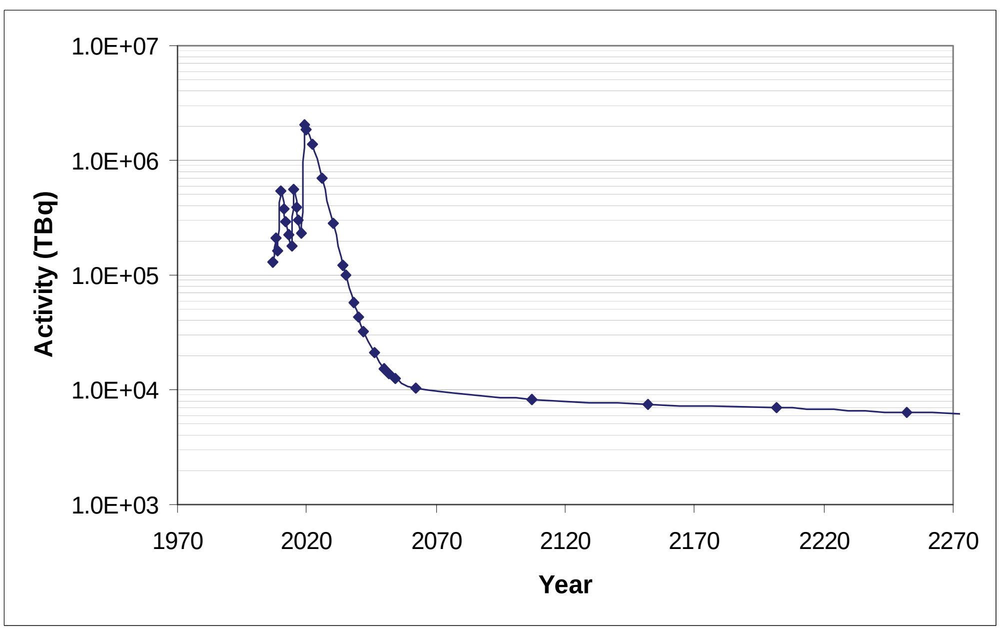

Figure 5-2: Time Dependence of Total Activity for Refurbishment L&ILW

Table 5-9: Summary of Waste Volume Handled at WWMF by Waste Category and Unshielded Dose Rate[a]

Contact Dose Rate (mSv/hr)	As-rec'vd Volume (m^3)	Overall LLW & ILW %	LLW								ILW						
			TOTAL LLW %	Ash %	Bales %	Box Comp %	Incin. %	Non-pro %	Non-pro Drum %	LL/ ALW Resin[b] %	TOTAL ILW %	CAN-DECON Resin %	ILW Resin %	Tile Hole Liners %	IX Column %	Filters %	Other ILW[c] %
< 0.01	209,754	76.2%	77.2%	9.5%	24.7%	57.9%	91.1%	64.1%	70.8%	86.2%	5.6%	2.4%	7.0%	2.9%	2.7%	1.7%	0.0%
0.01 - 0.05	28,830	10.5%	10.8%	5.6%	25.4%	26.8%	5.9%	15.5%	13.8%	13.8%	0.9%	0.0%	0.8%	0.7%	3.1%	1.2%	0.0%
0.05 - 0.1	10,997	4.0%	4.1%	14.7%	12.7%	8.8%	1.4%	6.9%	4.9%	0.0%	3.0%	7.1%	3.5%	0.0%	1.6%	1.1%	0.0%
0.1 - 0.2	6,397	2.3%	2.3%	25.4%	9.7%	3.4%	0.7%	3.7%	2.4%	0.0%	2.1%	0.0%	2.6%	0.0%	1.2%	1.5%	0.0%
0.2 - 0.5	6,770	2.5%	2.5%	38.1%	11.6%	1.7%	0.5%	4.5%	2.2%	0.0%	5.0%	4.8%	6.1%	0.0%	4.5%	2.8%	0.0%
0.5 - 1	3,338	1.2%	1.2%	5.0%	7.5%	0.8%	0.2%	2.0%	1.6%	0.0%	4.3%	0.0%	4.5%	4.4%	4.2%	6.0%	0.0%
1 - 2	2,391	0.9%	0.8%	1.3%	4.9%	0.5%	0.3%	1.2%	1.0%	0.0%	3.7%	0.0%	3.6%	4.1%	3.9%	7.1%	0.0%
2 - 10	3,380	1.2%	0.98%	0.4%	3.1%	0.1%	0.0%	1.9%	2.9%	0.0%	22.6%	23.8%	21.4%	57.2%	15.8%	22.9%	0.0%
10 - 50	1,369	0.5%	0.11%	0.0%	0.4%	0.0%	0.0%	0.2%	0.4%	0.0%	28.5%	35.7%	31.4%	19.2%	18.7%	21.0%	18.3%
50 - 100	548	0.2%	0.0%	0.0%	0.0%	0.0%	0.0%	0.0%	0.0%	0.0%	13.7%	23.8%	13.4%	5.8%	12.2%	13.5%	24.4%
100 - 500	349	0.13%	0.0%	0.0%	0.0%	0.0%	0.0%	0.0%	0.0%	0.0%	8.2%	2.4%	4.9%	5.6%	21.7%	16.4%	37.2%
500 - 1000	51	0.02%	0.0%	0.0%	0.0%	0.0%	0.0%	0.0%	0.0%	0.0%	1.4%	0.0%	0.7%	0.0%	7.1%	3.5%	3.3%
1000 - 5000	22	0.01%	0.0%	0.0%	0.0%	0.0%	0.0%	0.0%	0.0%	0.0%	0.6%	0.0%	0.1%	0.0%	3.1%	1.3%	5.7%
> 5000	985	0.36%	0.0%	0.0%	0.0%	0.0%	0.0%	0.0%	0.0%	0.0%	0.4%	0.0%	0.0%	0.0%	0.2%	0.0%	11.1%
Max. Dose Rate (mSv/hr)		-	-	5	40	3	4	30	90	0.03	-	200	1600	300	8000	5000	700000

Notes:

a. Data is as of December 2005. Volume is prior to any waste conditioning at WWMF.

b. Includes ALW resin and sludge.

c. All other ILW e.g., core components.

5.10 Physical and Chemical Characteristics of Key Radionuclides

The following provides a brief review of the characteristics of some key radionuclides, ordered by half-life, based on their initial amount of radioactivity and/or their importance in the safety assessment.

a) Co-60 (5.3 year half-life). Co-60 is produced by activation, mostly of steel components. In the DGR, it is primarily embedded in the stainless steel retube end-fittings, where it is very non-volatile. However, activated steel corrosion products containing Co-60 are widespread at low levels across the L&ILW, but notable on the moderator and CANDECON resins, where the Co-60 would be likely present primarily as particulate or other non-volatile forms. Co-60 is important for operational safety due to its relative abundance and its high-energy gamma. Along with Cs-137, it is used as an easy-to-measure marker or key nuclide for application of scaling factors for difficult-to-measure nuclides.

b) Tritium (12.3 year half-life). Tritium (H-3) is produced in CANDU reactors largely through neutron interactions with the heavy-water coolant and moderator. In the DGR, it will be present primarily in LLWs. The dominant chemical form will be HTO; HT will also be present, largely due to consumption of water in anaerobic corrosion processes. These forms are both volatile. Tritium is a beta emitter.

c) Fe-55 (2.7 year half-life) and Ni-63 (96 year half-life). Fe-55 and Ni-63 are produced by activation, mostly of steel. These are primarily present in the DGR embedded within retube end-fittings where they are not mobile. However, as activated steel corrosion products, they are more widely distributed at low levels across the L&ILW, notably as particulate or as soluble species. Ni-63 is a beta emitter; Fe-55 is an X-ray emitter.

d) Cs-137 (30.2 yr half-life). Cs-137 is a fission product and is widely distributed in operational L&ILW as surface contamination on and within materials, as well as concentrated on IX resins. It is also present in refurbishment wastes as surface contamination on materials. Cesium is classified as "semi-volatile". Cs-137 is important for operational safety due to its relative abundance, its high-energy gamma and its longer half-life (relative to Co-60). Along with Co-60, it is used as an easy-to-measure marker or key nuclide for application of scaling factors for difficult-to- measure nuclides.

e) C-14 (5,730 year half-life). C-14 is produced by activation of various precursors. It is primarily present in the DGR in the moderator IX resins, where the dominant form is likely to be as carbonate species sorbed onto the resins from the moderator water. This form can be readily released on contact with water. A significant amount of C-14 is also present within the pressure tubes. C-14 is observed to be released from L&ILW packages as activated methane or CO_2 gas, primarily due to degradation or radiolysis processes. C-14 is a beta emitter.

f) Cl-36 (300,000 year half-life). Cl-36 is also present in the DGR primarily as an activation product with the Zircaloy pressure tubes. Although not readily released from the Zircaloy itself, once released it is a soluble and mobile species in water. Cl-36 is a beta emitter.

g) Nb-94 (20,000 year half-life) and Zr-93 (1,500,000 year half-life). Nb-94 and Zr-93 are primarily present through activation of Zircaloy. These are not soluble, and within the DGR will be primarily present embedded in the pressure tubes. Nb-94 is relevant because of its high-energy gamma. Zr-93 is a beta emitter (with a low-energy gamma emitting daughter, Nb-93m), and it is relevant because it will be the most abundant very-long-lived radionuclide in the DGR. These are non-volatile.

h) Long-lived alpha emitters (e.g., Np-237, Pu-239, Pu-240, Pu-242, and various uranium isotopes). These are formed by neutron activation and subsequent decay of the natual uranium in the nuclear fuel. They are widely distributed at very low levels as contamination in most waste types. They are important to long-term safety, due to their very long half-lives which can range from thousand of years into millions of years.

One important radiological characteristic of the L&ILWs emplaced in the DGR is that they are non-fissile. Used fuel, and recognizable fuel fragments, are not accepted. The amount of fissile radionuclides is small, and is dispersed over the DGR volume. In particular, the total mass of Pu-239 and Pu-241 is about 0.4 kg, and plutonium is not in pure form in any location. Criticality is not a credible scenario.

5.11 Chemical Inventory

The main chemical (non-radioactive) components of the operational and refurbishment L&ILW are summarized in Table 5-10. The inventories are based on reference waste compositions and volumes detailed in the Reference L&ILW Inventory report (OPG10a).

Most of the wastes emplaced are normal industrial materials as described in Section 5.12. However, the waste contain varying amounts of chemicals or elements that can be hazardous. These include asbestos (originally used as insulating material in some stations); heavy metals like uranium, cadmium, mercury, chromium and lead; and certain organic materials such as polycyclic aromatic hydrocarbons, chlorinated benzenes and phenols, and dioxins and furans produced in the incinerator and trapped in the ash.

The uncertainties associated with the chemical compositions are presented in the Reference L&ILW Inventory report (OPG10a). The compositions are directly measured or known for some waste streams (e.g., ash, resins), and estimated for others (e.g., non-processible wastes). Specific chemicals or elements of interest (e.g., lead, uranium, asbestos) were given particular attention, including review of WWMF package records and consultation with station staff.

Table 5-10: Estimated L&ILW Key Element/Chemical Inventory (kg)

Element/ Species	Operations LLW	Operations ILW	Refurb. L&ILW	Total
Aluminum	2.8E+05	3.8E+03	6.6E+02	2.8E+05
Antimony	3.2E+03	2.0E+00	2.2E+01	3.2E+03
Arsenic	2.8E+02	1.2E+01	1.3E+02	4.3E+02
Barium	9.4E+03	1.6E+02	1.1E-02	9.6E+03
Beryllium	1.1E+02	2.1E+01	5.2E-03	1.3E+02
Bismuth	5.4E+00	5.2E+00	5.6E-02	1.1E+01
Boron	1.5E+03	5.2E+03	2.4E+00	6.7E+03
Bromine	1.3E+02	4.2E-01	4.5E-02	1.3E+02
Cadmium	1.1E+04	1.9E+01	7.9E-01	1.1E+04
Calcium	3.5E+05	4.1E+03	2.3E+01	3.5E+05
Cerium	1.3E-01	8.2E-02	7.1E-02	2.8E-01
Cesium	5.5E-01	2.1E-01	1.6E-02	7.7E-01
Chlorine	8.2E+04	4.9E+03	2.6E+00	8.7E+04
Chromium	4.1E+05	3.6E+04	5.4E+05	9.8E+05
Cobalt	3.4E+02	2.2E+01	2.8E+02	6.4E+02
Copper	3.3E+06	4.0E+03	3.0E+03	3.4E+06
Fluorine	0.0E+00	1.3E+02	2.4E+00	1.3E+02
Gadolinium	0.0E+00	5.4E+03	6.7E+01	5.5E+03
Hafnium	0.0E+00	0.0E+00	2.6E+02	2.6E+02
Iodine	6.6E+01	1.1E-01	8.8E-03	6.6E+01
Iron	7.9E+06	9.0E+05	1.1E+07	2.0E+07
Lead	1.5E+06	2.8E+02	3.8E+00	1.5E+06
Lithium	4.5E+01	5.9E+03	1.3E-02	5.9E+03
Magnesium	7.2E+04	9.1E+02	4.7E+00	7.3E+04
Manganese	6.8E+05	6.2E+03	1.6E+05	8.5E+05
Mercury	6.8E+01	2.9E-01	8.7E-02	6.9E+01
Molybdenum	2.2E+02	4.8E+01	9.3E+02	1.2E+03
Nickel	3.0E+04	4.5E+04	1.6E+06	1.7E+06

Element/ Species	Operations LLW	Operations ILW	Refurb. L&ILW	Total
Niobium	1.0E+02	0.0E+00	1.2E+04	1.2E+04
Phosphorus	1.1E+05	3.3E+03	6.0E+02	1.1E+05
Potassium	1.1E+04	1.5E+03	8.7E-02	1.3E+04
Rubidium	2.4E-01	0.0E+00	1.4E-01	3.8E-01
Scandium	2.3E+01	5.6E-02	5.6E-01	2.3E+01
Selenium	8.1E+01	4.9E+00	1.8E-01	8.6E+01
Silicon	3.2E+06	9.4E+04	7.7E+03	3.3E+06
Silver	5.1E+00	9.7E-01	1.2E+00	7.3E+00
Sodium	2.1E+05	1.2E+04	9.3E-02	2.2E+05
Strontium	3.2E+03	3.3E+01	1.7E-01	3.2E+03
Sulphur	2.0E+05	3.0E+05	3.1E+00	5.0E+05
Tellurium	2.0E+02	0.0E+00	6.6E-02	2.0E+02
Thallium	2.4E-01	2.8E-01	2.3E-02	5.4E-01
Thorium	5.5E+00	1.8E+00	1.1E-01	7.7E+00
Tin	1.4E+02	1.6E+01	2.4E+03	2.5E+03
Titanium	1.5E+05	3.3E+01	8.8E+01	1.5E+05
Tungsten	1.2E+00	1.0E+01	1.3E+02	1.5E+02
Uranium	3.4E+02	2.4E+01	1.4E+02	4.9E+02
Vanadium	9.0E+01	4.3E+00	9.5E+02	1.0E+03
Zinc	1.5E+05	2.0E+03	1.6E+01	1.5E+05
Zirconium	7.4E+02	1.2E+00	6.0E+05	6.0E+05
Asbestos	3.0E+05	0.0E+00	0.0E+00	3.0E+05
EDTA	0.0E+00	4.8E+04	0.0E+00	4.8E+04
PAH	3.4E+00	0.0E+00	0.0E+00	3.4E+00
Cl-Benzenes & Cl-Phenols	2.8E+00	0.0E+00	0.0E+00	2.8E+00
Dioxins & Furans	9.3E-02	0.0E+00	0.0E+00	9.3E-02
PCB	1.3E-01	0.0E+00	0.0E+00	1.3E-01

Notes:
- Does not include full amount of common elements, especially carbon, hydrogen, oxygen and nitrogen.
- 0.0E+00 indicates value is zero or not significant.
- Ethylenediaminetetraacetic acid (EDTA).
- Polycyclic aromatic hydrocarbons (PAH).

5.12 Physical and Chemical Characteristics of the Bulk Material Inventory

The bulk material compositions in LLW, ILW and refurbishment L&ILW are presented in Table 5-11, Table 5-12, and Table 5-13. They are based on characterization data, bulk densities, and waste volumes of the various L&ILW categories, in the forecast described in the Reference L&ILW Inventory report (OPG10a).

The total estimated inventory of steel in L&ILW is 9.3×10^6 kg (without steam generators). The total organic component, consisting of cellulose, plastic materials, and IX resins is estimated to be 2.2×10^7 kg.

The bulk material composition of the container materials, and container surface area, are described in the Reference L&ILW Inventory report (OPG10a).

The total inventory of steel in operational L&ILW package materials, including the iron in concrete shield rebar is estimated to be 2.7×10^7 kg. The estimated weight of concrete shielding for operational L&ILW is 3.5×10^7 kg.

The total estimated inventory of steel in refurbishment containers (including rebar in concrete) and the steam generator shell is 1.7×10^7 kg. The total weight of concrete in refurbishment waste is estimated to be 3.2×10^7 kg.

With time, the wastes and their containers will corrode and degrade; some materials within years, while some will take hundreds of thousands of years. The various metals will corrode into oxides, minerals, and inorganic salts consistent with the local saline, reducing chemical environment. The organics will degrade into simpler organic materials under slow microbially-mediated reactions, ranging from simple volatile species like methane to recalcitrant bitumen-type compounds. These processes are considered in Chapter 8, Postclosure Safety Assessment.

Table 5-11: Inventory of Bulk Materials in LLW

Waste Type	Ash (kg)	Cellulose (kg)	Rubber (kg)	Plastics (kg)	Resins[a] (kg)	Bitumen (kg)	Other Organic (kg)	Carbon Steel (kg)	Stainless Steel (kg)	Other Metal (kg)	Concrete (kg)	Other Inorganics (kg)
Bottom Ash	1.2E+06	-	-	-	-	-	-	-	-	-	-	-
Baghouse Ash	1.4E+05	-	-	-	-	-	-	-	-	-	-	-
Compact Bales	-	4.9E+05	1.2E+05	6.4E+05	-	-	5.2E+04	2.6E+05	-	-	-	-
Box Compacted	-	4.0E+06	9.9E+05	5.2E+06	-	-	4.2E+05	2.1E+06	-	-	-	-
Non-Pro	-	2.2E+06	2.6E+05	6.4E+05	-	1.9E+05	2.6E+05	1.3E+06	1.4E+06	3.4E+06	7.7E+05	3.6E+06
Non-Pro Drummed	-	4.9E+05	9.4E+04	2.4E+05	-	-	3.3E+05	4.7E+05	4.7E+05	-	2.8E+05	1.3E+06
Non-pro Other	-	-	-	-	1.6E+04	-	-	4.8E+03	-	-	-	-
Low Level /ALW Resin	-	-	-	-	1.5E+06	-	-	-	-	-	-	-
ALW Sludge	-	-	-	-	-	-	-	-	-	-	-	4.0E+06
TOTAL	**1.3E+06**	**7.2E+06**	**1.5E+06**	**6.7E+06**	**1.5E+06**	**1.9E+05**	**1.1E+06**	**4.1E+06**	**1.9E+06**	**3.4E+06**	**1.1E+06**	**8.9E+06**

Note:
a. Resin weight does not include bound water (approx 40% by weight) or interstitial water.

Table 5-12: Inventory of Bulk Materials in ILW

Waste Type	Total Mass (kg)				
	Resins	Carbon Steel	Stainless Steel	Inorganics	Plastic
ILW Resin (PHT, Moderator, Misc., CANDECON)	3.7E+06	-	-	-	-
IX Columns	1.9E+05	4.0E+05	-	-	-
Filters and Filter Elements	-	5.0E+05	9.2E+04	7.4E+04	9.8E+04
Irradiated Core Components	-	1.3E+04	4.8E+02	-	-
TOTAL	**3.9E+06**	**9.1E+05**	**9.2E+04**	**7.4E+04**	**9.8E+04**

Note:
ILW resin weight does not include bound water (approx 40% by weight) or interstitial water.

Table 5-13: Inventory of Bulk Materials in Refurbishment L&ILW

Waste Type	Total Mass (kg)				
	Zircaloy	Carbon Steel	Stainless Steel	Other Metals	Concrete
Pressure Tubes	4.4E+05	-	-	-	-
Calandria Tubes	1.7E+05	-	-	-	-
Calandria Tube Inserts	-	-	2.1E+04	-	-
End-Fittings	-	-	2.3E+06	-	-
Steam Generators	-	8.4E+06	-	2.8E+06	1.9E+06
TOTAL	**6.1E+05**	**8.4E+06**	**2.3E+06**	**2.8E+06**	**1.9E+06**

6. FACILITY DESCRIPTION

6.1 General

6.1.1 DGR Requirements

The requirements for the DGR Facility are detailed in the Project Requirements document (NWMO10b). The major requirements that influence the design are listed below.

1. The DGR shall be able to safely accept and emplace all L&ILW resulting from the operations and refurbishment of OPG-owned or operated reactors, including that currently in storage at OPG's licensed facilities.

2. The closed repository, including shaft seals, and the surrounding geosphere shall passively contain and isolate the radioactive waste so as to protect the environment, and the health and safety of persons.

3. The design capacity is nominally 200,000 m^3 of packaged L&ILW.

4. The facility shall be capable of being operated for 100 years (including waste emplacement, preclosure monitoring and decommissioning periods).

5. The facility shall be capable of operating with a throughput of not less than 24 LLW packages or 2 ILW packages per 8-hour shift.

6. The facility shall be located on OPG-retained lands on the Bruce nuclear site close to OPG's WWMF to minimize waste transfer distances.

7. The facility shall meet all regulatory requirements (see Chapter 1).

8. The width of an emplacement room pillar shall have a dimension that is no less than twice the average effective width of adjacent emplacement rooms.

9. The underground DGR facilities shall maintain a minimum off-set of 100 m from any deep borehole.

6.1.2 Applicable Regulations, Standards and Codes

Table 6-1 lists the regulations and major standards and codes applicable to the design and operation of the DGR. A more detailed listing is given in Project Requirements (NWMO10b).

Table 6-1: Regulations, Standards and Codes

Code or Standard	Applicability
Management System Requirements for Nuclear Power Plants CSA N286-05	Management systems
National Building Code of Canada	Surface buildings and structures, fire protection
National Fire Code of Canada	Fire protection systems
Occupational Health and Safety Act (Ontario) – Construction Projects Regulations (Reg 213/91)	Surface buildings and structures
Occupational Health and Safety Act (Ontario) – Industrial Establishments Regulations R.R.O. 1990 (Reg. 851)	Surface buildings and structures
Occupational Health and Safety Act (Ontario) – Mines and Mining Plants Regulations R.R.O. 1990 (Reg. 854)	Shafts, hoists, repository, fire protection, ventilation requirements
General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants/Material Standards for Reactor Components for CANDU Nuclear Power Plants CSA N285-08 Boiler, Pressure Vessel, and Pressure Piping Code CSA B51	Pressurized systems
Non-Rail-Bound Diesel-Powered Machines for Use in Non-Gassy Underground Mines CSA M424.2	Non-rail-bound diesel-powered machines
Concrete Materials and Methods of Concrete Construction/Test Methods and Standard Practices for Concrete CSA-A23.1 and CSA –A23.2	Surface buildings and structures
Design of Steel Structures CSA S16	Surface buildings and structures
OPG Radiation Protection Requirements Nuclear Facilities (OPG01b)	Radiation zoning and protection
Ontario Electric Safety Code	Electrical systems
Workplace Electrical Safety CSA Z462	Electric arc flash
Use of Electricity in Mines CSA-M421	Lightning protection
Installation Code for Lightning Protection Systems CSA B72-M87	Lightning protection

6.2 Surface Buildings and Infrastructure

The DGR surface buildings and associated infrastructure are located in three main areas:

- Main shaft area – provides intake ventilation and primary access to the underground repository for transfer of waste packages, personnel, equipment and materials;
- Ventilation shaft area – location where the waste rock from the development of the repository is brought to surface, provides a second egress and conveys the air discharged from the repository; and
- WRMA – location where rock excavated during underground construction of the DGR will be stored.

An artist's rendering of the DGR surface facilities in the vicinity of the two shafts including the access roadway to the WWMF is shown in Figure 6-1. Descriptions of various aspects of the design are provided in the following sections. The general layout of the surface facilities is shown in Figure 6-1 and drawing 11T1076-C-SK-1 (see Chapter 17). The underground layout is shown in Figure 6-2.

6.2.1 Main Shaft Area

The main shaft area includes the following key structures and services (see Figure 6-1 and Figure 6-2):

- Main shaft headframe;
- WPRB;
- Maintenance and storage area;
- Compressor building;
- Intake fans and heater house; and
- Offices, main control room and amenities building.

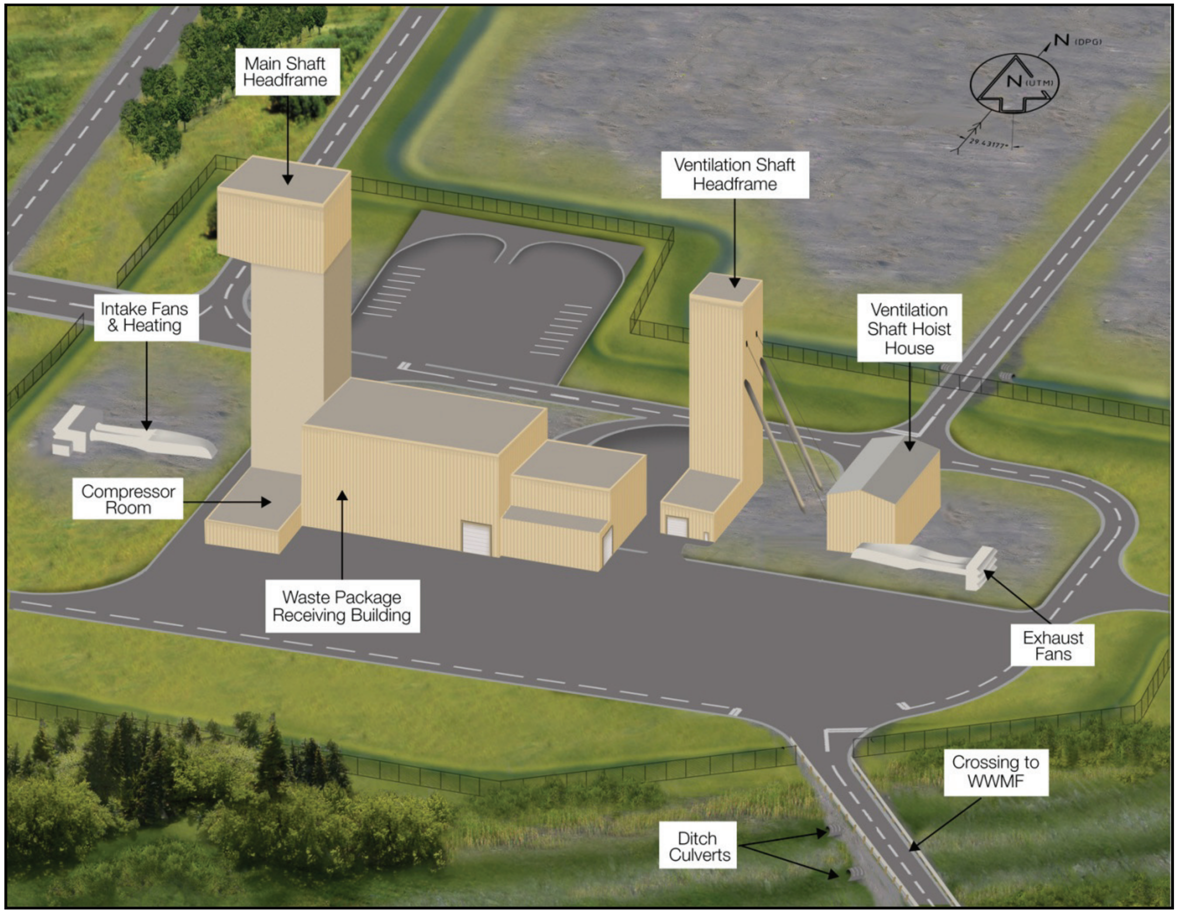

Figure 6-1: DGR Surface Facilities

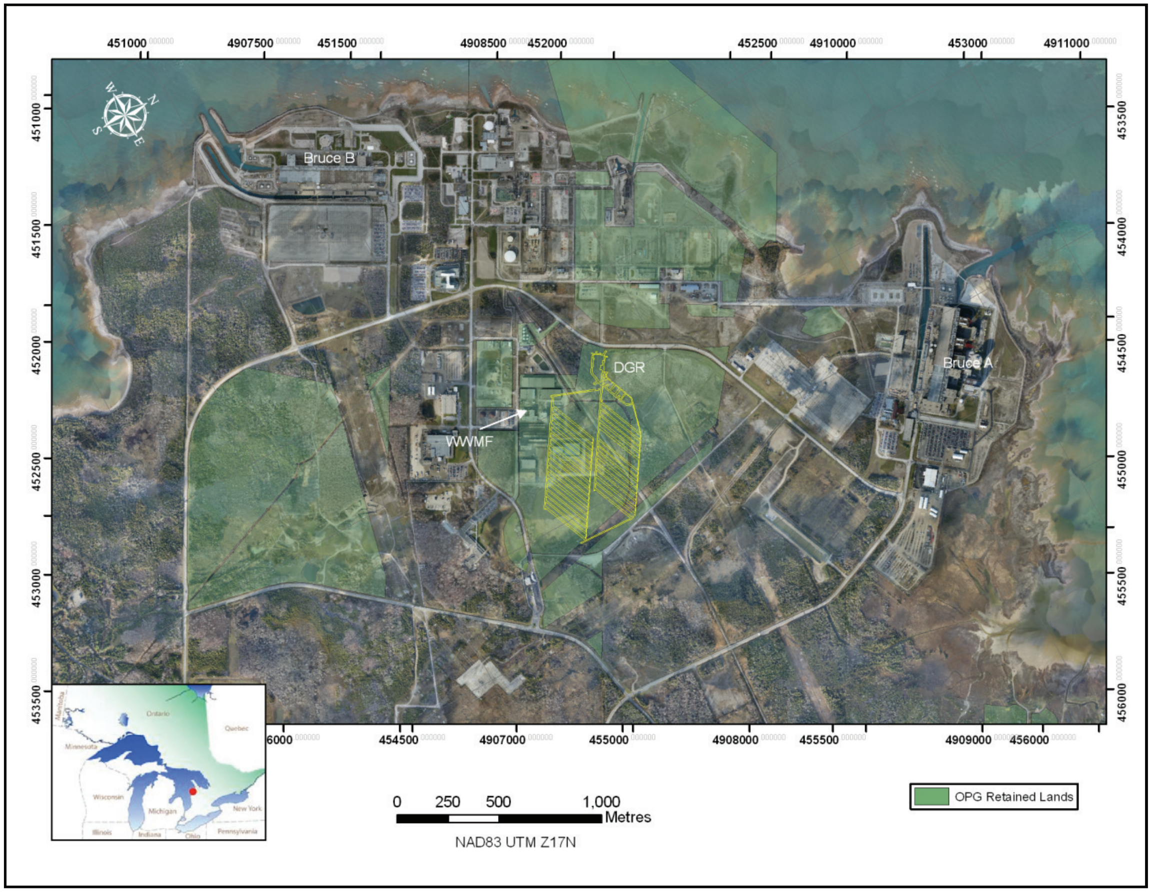

Figure 6-2: Layout of DGR Underground Facilities

6.2.1.1 Main Shaft Headframe

The main shaft provides primary access to the underground repository and the main shaft headframe houses hoisting equipment to lower and raise conveyances for transporting personnel, equipment and waste packages.

The main shaft headframe is a reinforced concrete structure, nominally 60 m high with a nominal plan area of 225 m^2 (15 m x 15 m). This headframe contains a tower mounted 4.27 m diameter main and 1.4 m diameter auxiliary Koepe friction hoists installed in the permanent condition as shown in Figure 6-3. The concrete headframe design is such that, with a planned maintenance system in place (see Chapter 10), the structure will not require major refurbishments for the 100-year design life of the DGR. The concrete structure provides the necessary structural support for the large, heavy-duty Koepe friction hoist and provides insulation of the equipment and personnel working within the headframe during winter conditions.

The headframe contains the 4.2 m diameter deflection sheaves for the main Koepe hoist head ropes, arresting gear for retarding the conveyances in the event of overwind and overhead crane beams for maintaining and installing the conveyances. Stairs and intermediate floors and platforms are provided for access and maintenance requirements. An elevator has been included to service the various floors in the headframe and to provide access to the hoist room.

The main shaft hoist room is located at the top of the main shaft headframe and has nominal external dimensions of 15 m x 22 m with a height of 12.5 m. The hoist room has a 7 m overhang to facilitate hoisting of major components for the main Koepe hoist from ground surface to the hoist level. A 50-tonne overhead travelling crane mounted in the hoist room is used to hoist the equipment and maintenance supplies to the top of the headframe. This hoist room houses all the controls and electrical equipment necessary to operate the hoist along with a local operating station.

The main headframe design incorporates shaft sinking requirements such that there is no requirement for a temporary sinking headframe during construction. Using the same structure for both sinking and operations will optimize the construction timing and provide a more efficient transition from sinking to operational configuration (see Chapter 9).

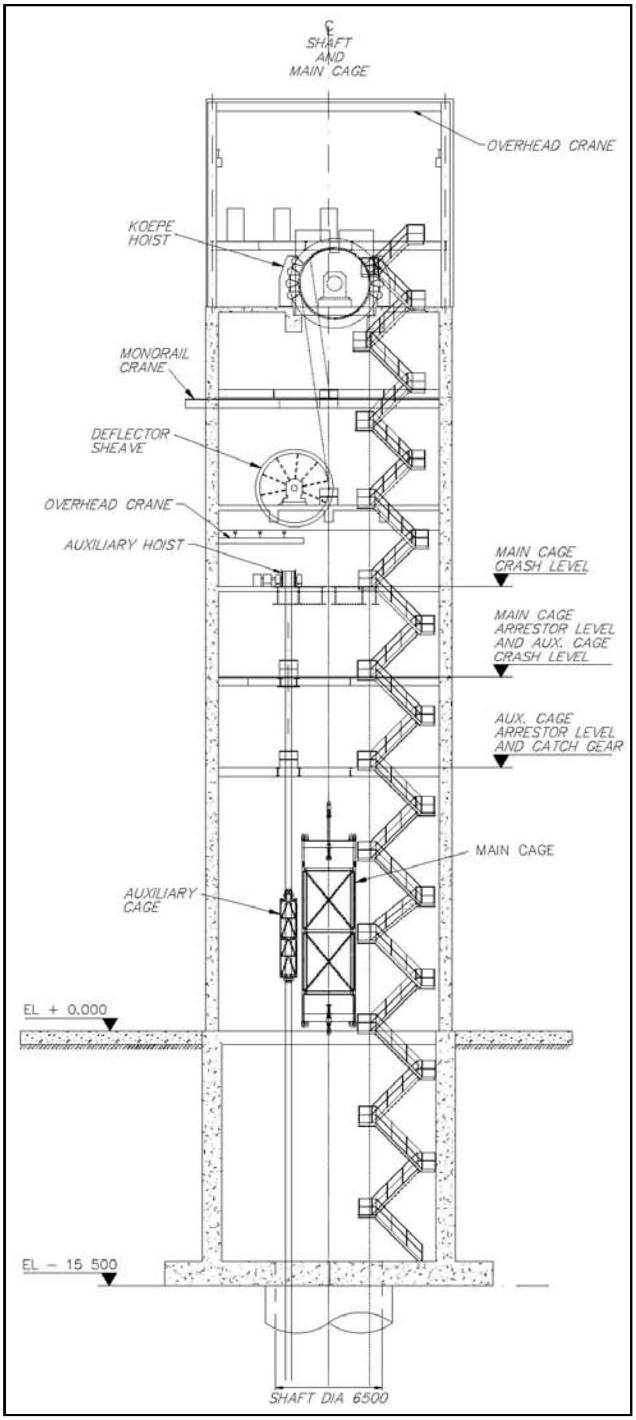

Figure 6-3: Main Shaft Headframe

6.2.1.2 WPRB

The WPRB receives waste packages from the WWMF and stages them for transfer onto the main shaft cage. The WPRB is connected to the main shaft headframe. The WPRB area is nominally 40 m long by 25 m wide, constructed as an insulated and clad steel frame structure. The layout of the WPRB is shown in Figure 6-4 and includes the following features:

- Waste package staging area;
- Unloading bay and truck dock;
- 40-tonne overhead crane for handling heavy waste packages that are not forkliftable; and
- Access to the adjoining maintenance and storage area.

The main open area of the WPRB has a package staging area with room for 24 Group A (see Section 6.4.1) packages stacked two high and is based on the number of bin-type packages anticipated to be transferred during an 8 hour shift. Localized shielding is incorporated into the WPRB wall design adjacent to the staging area, as required, to protect workers in adjacent offices and main control room in accordance with OPG radiation protection requirements.

An unloading bay is located on the end of the WPRB away from the shaft. Trucks or forklifts enter parallel to the common wall with the maintenance and storage area and can exit the building through the opposite side.

A covered dock is located perpendicular to the unloading bay and beside the maintenance and storage area.

The WPRB floor is equipped with rail up to the shaft to facilitate main shaft cage loading and unloading. There is a side-switched area with a two-cart capacity and the two parallel tracks extend out through a rollup door into the maintenance and storage area. A second switch is located towards the maintenance and storage area so that carts can be moved to the package loading area without approaching the shaft. Two power reels provide power and control signals to the self-propelled rail carts.

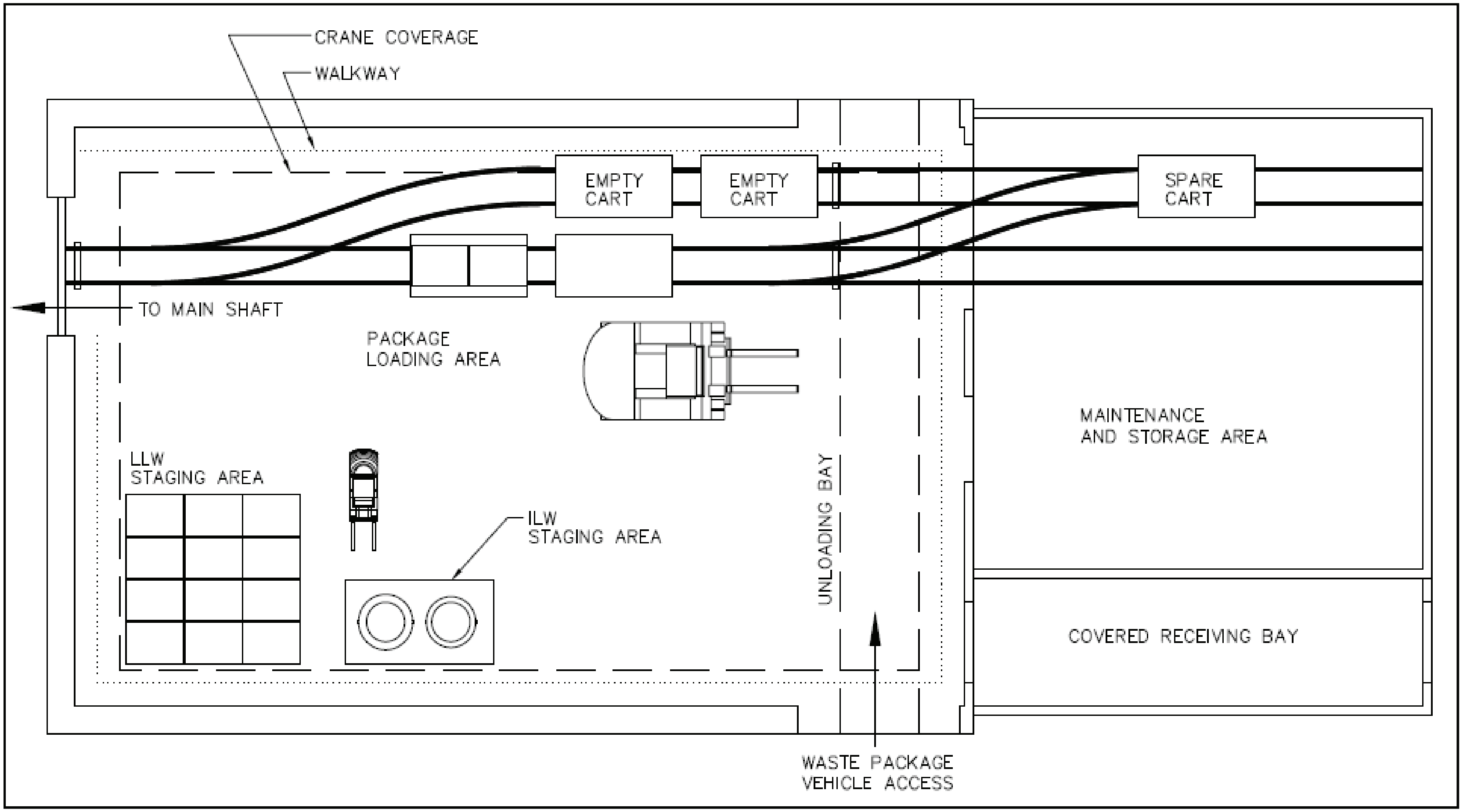

Figure 6-4: Layout of WPRB

6.2.1.3 Maintenance and Storage Area

The maintenance and storage area adjoined with the WPRB is used for minor repairs and preventative maintenance tasks for the shaft components and the equipment used within the WPRB (see Figure 6-4). The area is part of the WPRB steel clad building with access from within the WPRB, as well as, door access along the east wall.

6.2.1.4 Compressor Building

The compressor building located close to the main shaft houses two compressors that provide compressed air for surface and underground maintenance. In the event of an underground emergency, these compressors will be used to provide breathing air to the underground refuge stations.

The steel framed metal clad compressor has a nominal footprint of 9 m x 10 m. The building will be designed to act as an acoustic enclosure. Each compressor is capable of providing compressed air of 30 m^3/min at 830 kPa. One compressor will normally operate with the other compressor on stand-by.

6.2.1.5 Intake Fans and Heater House

The function of the surface intake fans is to provide the required airflow for the DGR. The fans are designed to deliver the maximum anticipated flow at any point through the life of the DGR (see Section 6.3.8). Two fans of equal specification are located at the intake of the heater house and will include silencers as required.

The function of the electric surface heaters is to raise the ambient temperature of winter air drawn in by the intake fans to a temperature of 5°C so that services within the main shaft and headframe are not subject to adverse temperature conditions. The footprint of the heater house is nominally 7 m x 10 m and interfaces with the intake plenum.

6.2.1.6 Offices, Main Control Room and Amenities Building

The offices, main control room and amenities building is a steel framed, insulated and clad structure adjacent the north side of the main shaft headframe and WPRB. The approximate size of the building is nominally 25 m x 25 m and is two-storeys high. The main control room is equipped with computing, control, and monitoring equipment to marshal all signals and data transmitted from surface and underground.

The amenities area is equipped with change room/locker facilities, lunch room and a training/visitors room. Radiological badging and work control are also managed in the amenities building. A parking area is provided to the north of the building to receive DGR staff and visitors. Other facilities provided include a lamp room, mechanical areas and storage.

6.2.2 Ventilation Shaft Area

The ventilation shaft area includes the following key structures:

- Ventilation shaft headframe and collar house;
- Ventilation shaft hoist house; and
- Exhaust fan building.

The ventilation shaft exhausts the repository ventilation air and is used as a second egress.

6.2.2.1 Ventilation Shaft Headframe and Collar House

The ventilation shaft, apart from exhausting the repository ventilation air, is primarily used during construction to remove waste rock generated from the repository development activities. During the operations phase, the ventilation shaft is the prime conduit for repository services (service water, power, communications, etc.) and is used for emergency secondary egress.

The ventilation shaft headframe is nominally a 43 m high, insulated and clad steel structure (see Figure 6-5). The headframe is designed so that the structure will not require major refurbishments during the 100-year design life of the DGR. The headframe design incorporates the sinking and permanent requirements with minimal modifications required to change over from sinking to permanent condition (see Chapter 9).

The headframe includes two 3.7 m diameter sheaves for hoisting the two shaft conveyances (skip[1]-skip during construction or skip-cage during operations). The headframe also includes the bale and skip arresting gear for retarding the conveyance in the event of overwind and overhead crane beams for maintaining and installing the conveyances. Stairs, intermediate floors and platforms are provided for access and maintenance requirements.

[1] Skip is a large open-ended bucket that is used to move waste rock from underground to surface (see Figure 6-11). During construction there would be two skips used to transport waste rock.

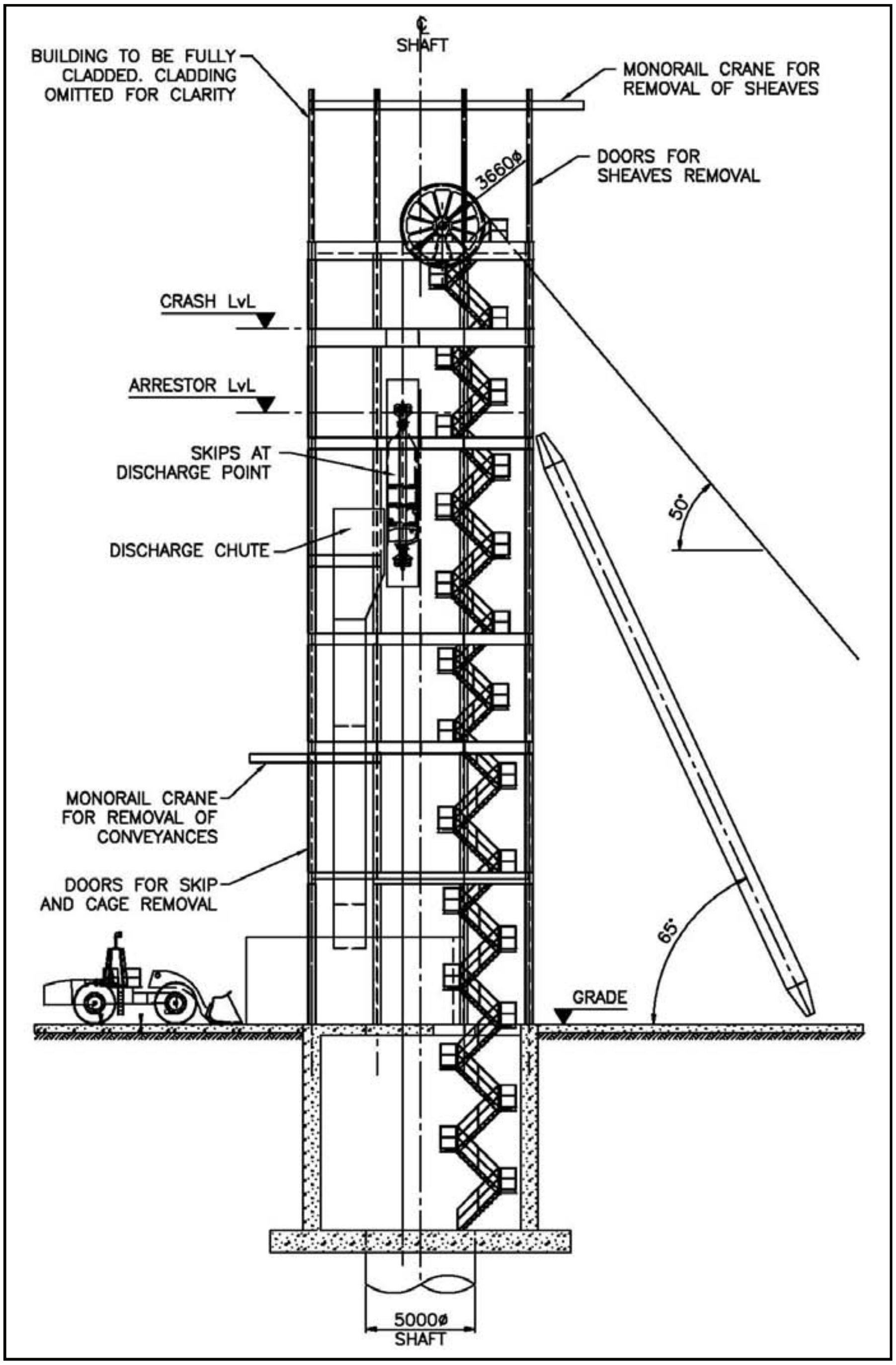

Figure 6-5: Ventilation Shaft Headframe

The insulated and clad steel-framed collar house is used for general maintenance and storage of shaft hardware and equipment spares. The internal dimensions of the collar house are nominally 15 m x 10 m and 5 m high. The building contains electrical panels, lighting, roll-up door for access of equipment, an overhead crane, equipment and materials required for maintenance personnel.

6.2.2.2 Ventilation Shaft Hoist House

The ventilation shaft hoist house is nominally 13 m x 24 m and 11.5 m high and constructed as an insulated and clad steel frame structure. This building is required for both the sinking and operations phases and houses a 3.66 m diameter double drum hoist for hoisting the sinking bucket during shaft development and dual conveyances during DGR development and operations (skip-skip or skip-cage configurations). The building contains all the electrical equipment and control station, roll up doors for access and 8-tonne overhead crane for installation and maintenance of the hoist.

6.2.2.3 Exhaust Fan Building

The main exhaust ventilation fans are located near the ventilation shaft at the repository level (see Section 6.3.8.3). Surface based exhaust fans are installed in the exhaust fan building to help to draw the exhaust air through the ventilation shaft plenum. The building is steel with cladding.

6.2.3 WRMA

Waste rock generated as a result of excavation of the shafts and at the repository level (see Chapter 9) is managed on the DGR site in the WRMA. Approximately 832,000 m^3 of rock will be managed over the long-term at the WRMA. This volume excludes the 80,000 m^3 of soil and rock material from shaft development that will be temporarily stored within the WRMA, but eventually reused on-site in the construction of berms and roadways.

The quantities of rock materials to be excavated for the repository (shafts, access tunnels, emplacement rooms, etc.) are given by material type in Table 6-2.

The WRMA is located adjacent to the surface facilities (see drawing H333000-WP404-10-042-0001) and provides an area for: (i) temporary management of the overburden, shales and dolostones during construction (approximately 2 ha) and; (ii) long-term management of argillaceous limestones (approximately 9 ha). The overall footprint of the WRMA, including the stormwater management system, is approximately 17 ha.

Table 6-2: Estimated Quantities of Excavated Materials

Material Type	Approx. Depth	Volume (m^3)	
		In-Situ	Bulked
Overburden	0 – 20 m	1,400	2,000
Dolostone and Shale	20 m – 410 m	34,300	48,000
Shale	410 m – 660 m	21,200	29,700
Limestone	660 m – 840 m	594,200	832,000
	Total	**651,100**	**911,700**

The 9 ha waste rock pile is constructed with 2.5:1 slopes to ensure stability. In order to prevent ponding of water on the top of the 15-m-high argillaceous limestone pile, the top of the pile is graded (see drawing H333000-WP404-10-042-0003, Chapter 17).

6.2.4 Shared Services

6.2.4.1 Electrical Supply and Emergency Power

Class 4 electrical power is supplied to the facility by a 13.8 kV high voltage transmission line.

The main shaft hoist, ventilation shaft hoist and large power distribution transformers are major loads and supplied at 13.8 kV. The large distribution transformers will step down 13.8 kV to 600 V. The 600 V system supplies the surface main control centres (MCCs). The surface MCCs feed:

- Intake fans;
- Exhaust fans;
- Air compressors;
- Overhead electric cranes;
- Maintenance and storage area, office and amenities building; and
- Small power distribution transformer for lighting, receptacles, and other facility service loads at 110 VAC.

Lightning protection is installed in accordance with the applicable regulations (see Table 6-1).

The connected loads are given in Table 6-3 for both surface and underground power users. The total connected load for the facility is estimated to be approximately 16,360 kVA.

An emergency power system using diesel generators, complete with load bank, is installed to assure safety in the event of a grid power failure. An emergency generation capacity of approximately 1,750 kW (which consists of multiple generators providing the required load with additional capacity) is required to serve the site loads that are essential for personnel safety. In the event that one of the generators is not operable, the remaining generators can supply the emergency load requirements. It is not intended to maintain operational power requirements following a grid power failure.

Table 6-3: Electrical Power Loads

Surface		Underground	
Main shaft hoist	3,300 kVA	Shaft sump pumps [a]	80 kVA
Ventilation shaft hoist [a]	1,700 kVA	Dewatering pumps [a]	250 kVA
Main shaft auxiliary hoist [a]	160 kVA	Underground ventilation fans	540 kVA
Intake fans and heaters	6,500 kVA	Jumbo drilling machines	450 kVA
Exhaust fans	240 kVA	Rock bolting machines	220 kVA
Air compressors [a]	430 kVA	Shotcrete machine	100 kVA
Lighting and misc low power equipment (50% emergency [a])	100 kVA	Lighting and misc low power equipment (50% emergency [a])	60 kVA
Misc. 600V loads	350 kVA	Misc. 600V loads	200 kVA
10% growth factor	1,300 kVA	20% growth factor	380 kVA
Total Surface Connected	14,080 kVA	Total Underground Connected	2,280 kVA
Total (Surface & Underground):	16,360 kVA		

Notes:
a. Indicates loads that are connected to the emergency power system. Not all of the connected loads are expected to be live concurrently.

The emergency power system is located at the surface electrical substation and will feed equipment through the cables and switchgear used for normal operations. The loads served by the emergency power system are:

- Ventilation shaft hoist at a reduced speed of 3 m/s, which may be used as the second egress to remove personnel from underground to surface;
- Main shaft auxiliary hoist;

- Main shaft Koepe friction hoist brakes and controls to allow for controlled lowering of the cage via gravity and brakes and not by the motor;
- Sump and dewatering pumps
- One air compressor; and
- Emergency lighting and communications.

The emergency power system will automatically supply power to critical components within 30 seconds of a grid power outage. Specialized controls and switchgear are used to initiate the start up of the generators and shed non-critical loads following a power outage, as well as allow an uninterrupted switchover when the supply grid is re-energized. Inspection and maintenance programs will be implemented to ensure the reliability of the emergency power system (see Chapter 10).

6.2.4.2 Communications Systems

The DGR communications system includes the surface and underground infrastructure required for:

- Telephones;
- Wireless radios;
- Business network; and
- Process control network.

The communications system does not include the signal transmission infrastructure for fire detection/suppression and hoist control. These two systems utilize dedicated signal transmission infrastructures. However, outputs from these systems are accepted by the DGR communications system for inclusion on the operator's screens in the main control room.

The communications infrastructure uses a fibre optic network with cable supplied in both the main shaft and the ventilation shaft for redundancy. This equipment is used for local switching and as an overall link to the main OPG network.

Ethernet-based IP telephone (i.e., Voice over Internet Protocol (VoIP)) technology is used for both surface and underground telephone service. It is connected via fibre optic link, for access to external lines and OPG internal phone network. Hard-wired emergency phones are installed at the surface main control room, at the main shaft and ventilation shaft stations, and at each refuge station. The emergency phone system is connected by twisted pair cable which is installed in a ring between each of the phones. These phones are intended for emergency communications in the event of failure of the other voice communications systems (VoIP phones and radio). Because the system uses a separate and isolated infrastructure, it provides additional redundancy to emergency communications and is not affected by loss of electrical power.

Wireless voice coverage is provided for the underground repository, main and ventilation shafts, and the surface main control room using “leaky feeder” technology. Leaky feeder is a simple and robust analogue system that, for example, utilizes a coaxial antenna cable. This cable is hung throughout the underground tunnels using hangers and can easily be removed from emplacement rooms as they are filled with waste containers. A separate channel is provided for in-shaft work, as this is required to ensure uninterrupted communications between shaft workers and the hoist control operator, particularly during maintenance and shaft inspections.

Although alerts of fire or other emergency conditions are made via the radio system, the primary system of notification to the underground DGR is the stench gas system, which is described in Section 6.8.2.2.

The leaky feeder radio system is also used to carry monitoring signals from remotely installed instrumentation. The leaky feeder carries cable modem terminal services connectivity to support Ethernet data communications requirements in remote areas of the underground installation.

The business network provides access for business computers to e-mail, internet and other network services at all appropriate locations at surface and underground. The process control network carries all signals for monitoring and controlling systems at the DGR both on surface and underground. This network is provided in the key locations where connectivity between instrument and equipment is required. Main process network switches are installed in the main control room.

6.2.4.3 Control and Monitoring Systems

The DGR Facility has a main control room. The main control room operator can view custom-configured control screens that display equipment and system status and allow inputs to be executed through a mouse/keyboard interface. The operator can also monitor key areas through the use of closed circuit video monitors.

In the off-shift hours, selected main control room functions are transferred to the WWMF main control room, which is continually staffed, allowing an operator to monitor the facility and respond to any alarms.

Shaft hoisting operations are controlled from the respective control terminals. Hoisting operations can be automated or controlled manually. A certified hoist operator is on-site at all times that the hoists are in operation.

The following underground equipment is monitored and controlled from the main control room:

- Sump and dewatering pumps;
- Power distribution facilities including motor starters and some switchgear; and
- Ventilation fans and air heaters.

The following equipment is only monitored in the main control room because this equipment either does not require control or is controlled locally:

- Fire detection and suppression systems;
- Uninterruptible power supplies (status monitoring);
- Water quality monitoring, as required;
- Air quality monitoring, as required;
- Ground support monitoring, as required; and
- Hoist system monitoring.

The fire detection and suppression system report into the main control room but are monitored and controlled by a separate and isolated infrastructure.

The control and monitoring system allows for connection and activation of alarm devices to notify personnel of abnormal or unsafe conditions. Alarm notification devices are used within the main control room and, as necessary, underground.

The DGR Facility has video monitoring systems throughout the surface facilities. Closed circuit cameras are used and the video data is carried over the business network to the main control room where the operator monitors these areas through the use of multiple screens. There are also closed circuit cameras specific to the hoisting system that feed to the respective hoist control for monitoring.

6.2.4.4 Diesel Fuel Storage

During facility operation diesel fuel will not be stored on surface at the DGR site, with the exception of the diesel tank for emergency power generators. Diesel fuel and lubrication oils are provided by supply truck from WWMF to the DGR Facility, as needed. Totes are filled at the WWMF fuelling station and brought to the WPRB for transfer underground to the diesel fuel bay and for use by underground equipment.

The diesel fuel storage tank for the emergency diesel generators allows for 48 hours of operation of the diesel generators at 35% loading. The storage tank is an above ground double-walled tank and connected directly to the generators.

Both surface and underground (see Section 6.3.10.2) fuel storage areas are provided with sufficient sump capacity to collect accidental spillage that could occur during fuel transfer or leakage from any tanks or pipes. Berms are constructed as needed to ensure that any spillage of fuel or lubricant is retained within the storage and refuelling areas. Space for only a single piece of mobile equipment is provided in the underground diesel fuel bay to reduce any risk of a fire incident.

6.2.4.5 Potable and Service Water

The potable and service water used at the surface facilities is supplied through the existing infrastructure at the Bruce nuclear site.

6.2.4.6 Sewage System

All human effluent collected from the surface facilities is collected and discharged into the existing sewage system at the Bruce nuclear site.

6.2.4.7 Road Connection to WWMF

A crossing is provided for direct access between WWMF and the DGR over the abandoned railway right-of-way. The crossing features a connecting two-lane road situated on a fill embankment over the existing ditches and railway. Concrete-surrounded corrugated-galvanized steel elliptical culverts (approximately 3 m wide, 2 m high) are used to accommodate the existing water flow in the railway ditches.

The 20 m width embankment accommodates wide road lanes (4 m minimum), shoulders (1.5 m minimum), walking area (2 m on each side), and adequate space for snow storage (1.5 m minimum) during winter operations, and concrete barrier (1 m) on both sides of the road.

Excavated material from shaft sinking and lateral development is used for embankment fill material.

6.2.4.8 Stormwater Management System

All stormwater run-off from the DGR site as well as any groundwater pumped to surface from underground shaft sumps is directed via ditches to the stormwater management pond for treatment to remove suspended solids. The pond discharge water is directed into a ditch that ultimately discharges into Lake Huron.

The area around the surface facilities (shafts and headframes, WPRB, fan buildings, etc.) is approximately 5.6 ha. To control stormwater within this area, a stormwater management system has been designed which includes a perimeter drainage ditch, two oil/water separators and an intermediate settling pond (see drawing H333000-WP404-10-042-0001, Chapter 17). Overland flow drains directly to the perimeter drainage ditch. Subdrains and catch basins are used around the building areas to facilitate effective drainage, discharging to the perimeter ditch. Pumped water from the main shaft and the ventilation shaft is directed to oil/water separator No.1 and then released into the perimeter ditch. All of the stormwater collected by the surrounding drainage ditch is released from a single outlet into the intermediate settling pond. The drainage water in the settling pond is directed to oil/water separator No.2 and then released to the WRMA perimeter ditch which ultimately discharges into the stormwater management pond.

Stormwater run-off from the WRMA is collected in a network of trapezoidal drainage ditches around the perimeter of the WRMA and is then directed to the stormwater management pond. Drainage ditches constructed around the perimeter of the WRMA are trapezoidal and vegetated to minimize erosion (see drawing H333000-WP404-10-042-0001, Chapter 17).

Rainfall run-off volumes from the two aforementioned areas are summarized for the 6 hour, 25 mm and the 1:100 year events in Table 6-4. The assumed run-off coefficients are also tabulated. As the surface facilities area is predominantly paved, the run-off coefficient is correspondingly higher than for the unpaved WRMA.

Table 6-4: Rain Run-Off Volumes

DGR Area	Area (ha)	Stormwater Management Parameters	Storm Event	
			6 hour, 25 mm	1:100 year
WRMA	17.0	Rainfall (m^3)	4,250	24,620
		Run-off (m^3)	2,848	20,188
		Run-off Coefficient	0.67	0.82
Surface Facilities Area	5.6	Rainfall (m^3)	1,400	8,148
		Run-off (m^3)	1,134	7,741
		Run-off Coefficient	0.81	0.95

The primary function of the stormwater management pond is to control total suspended solids (TSS) / turbidity prior to discharge. The stormwater management pond discharges into an existing drainage channel near the northern portion of the WRMA. The pond consists of:

- A retention area for settling of particles (to a size of approximately 0.02 mm);
- An extended storage area for larger storm events; and
- A low permeability base (e.g., composite or natural) with a protective cover (granular material).

The stormwater management system is designed with capacity to:

- Retain the 6 hour, 25 mm storm for a period of 24 hours; and
- Safely pass the 1:100 year storm event without overtopping of the embankments and erosion of the outlet system.

To stop water discharge from the management pond in the unexpected event that contaminants in the discharge water exceed acceptable limits or general discharge needs to be halted due to downstream issues, a gate is installed on the outlet. This gate is controlled manually and will normally remain in the open position.

6.3 Underground Facilities

An isometric view of the underground arrangements at the repository level (nominally 680 m below shaft collar level) is shown in Figure 6-6. Two panels of waste emplacement rooms are located to the east of the main and ventilation shafts. The emplacement rooms are nominally 250 m in length and arranged parallel to the assumed direction of the major principal horizontal in-situ stress of east-northeast. Stress direction will be confirmed following shaft sinking and room orientation may be modified, as required, to suit in-situ stress conditions. Panel access and exhaust ventilation tunnels run parallel to one another perpendicular to the emplacement rooms. End walls or bulkheads, are constructed at the end of the emplacement rooms where they meet with the exhaust ventilation tunnel. These end walls allow for the installation of ventilation regulators and there are access doorways for egress when rooms are empty.

A services area (see Figure 6-7) is constructed around the two shafts and contains refuge stations to ensure personnel safety in the event of any underground incidents such as fire or spills. The services area also contains sanitary facilities, a lunch room, maintenance shop, diesel fuel bay, electrical and instrumentation services, geotechnical area and stores.

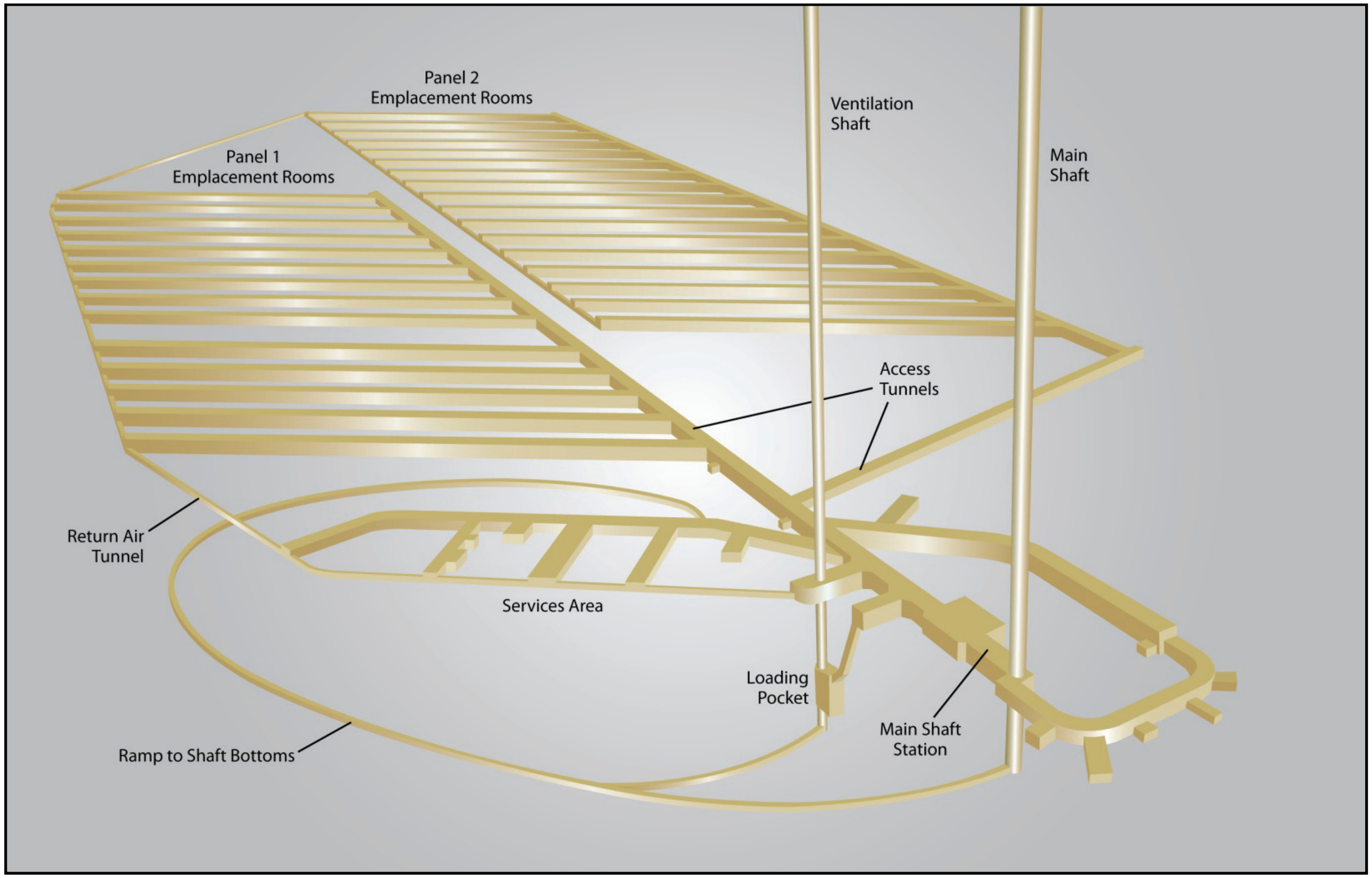

Figure 6-6: Isometric View of the Repository Level

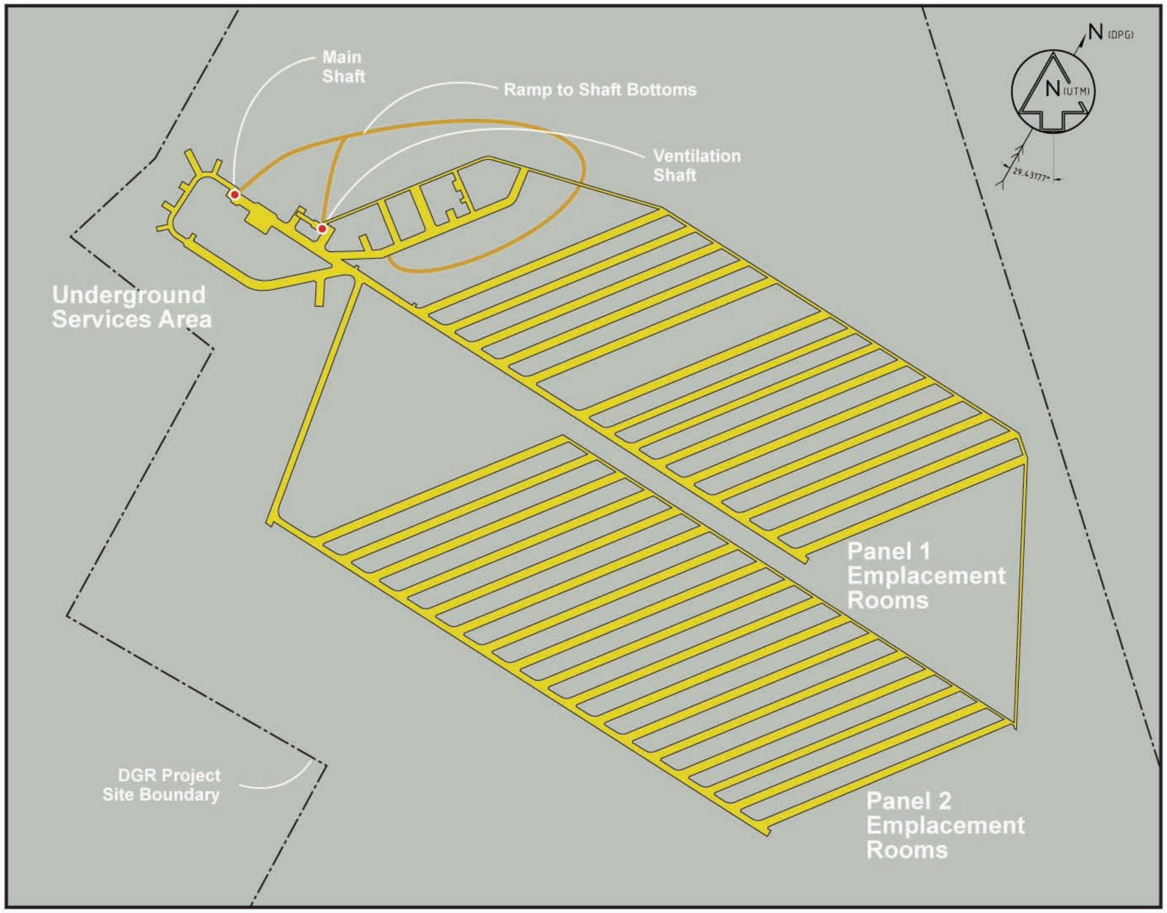

Figure 6-7: DGR Base Case Layout

Panel 1 has fourteen rooms and Panel 2 has seventeen rooms. The panels are connected by the ventilation exhaust tunnel providing flow-through ventilation to the ventilation shaft (see Section 6.3.8.2). Panel 2 is to be filled first primarily with LLW materials and is the furthest away from the shaft area. The furthest nine rooms of Panel 1 are filled next, consisting of a mix of LLW and ILW rooms. The closest five rooms are filled primarily with ILW materials with three of the rooms being configured for rail and gantry crane capabilities.

6.3.1 Main Shaft

6.3.1.1 Layout

The main shaft is nominally 720 m deep and the cross-sectional dimension and layout is dictated by the external dimension of the main shaft cage which is 5.6 m long by 3.0 m wide. The main shaft finished, or inside, diameter is 6.5 m and the shaft configuration is split into three parallel compartments as shown in Figure 6-8 and described below:

- Main shaft cage compartment which is the largest and in the centre of the shaft;
- Main shaft cage counterweight compartment to the south of the main shaft cage compartment; and
- The auxiliary cage and auxiliary cage counterweight compartment to the north of the main shaft cage.

The main shaft cage compartment will contain the cage conveyance as well as the following:

- Fibre optics and communications;
- Hoist signalling cables; and
- Fire detection.

The main shaft cage counterweight compartment will contain the counterweight as well as the following:

- One service water line; and
- Two slick lines (concrete for shaft construction only).

The auxiliary cage and counterweight compartment will contain the cage and counterweight as well as the following:

- One dewatering water line;
- One compressed air line; and
- Power feeds.

This configuration provides separation of power and communications as well as facilitates the installation of the conductors off the top of the main shaft cage.

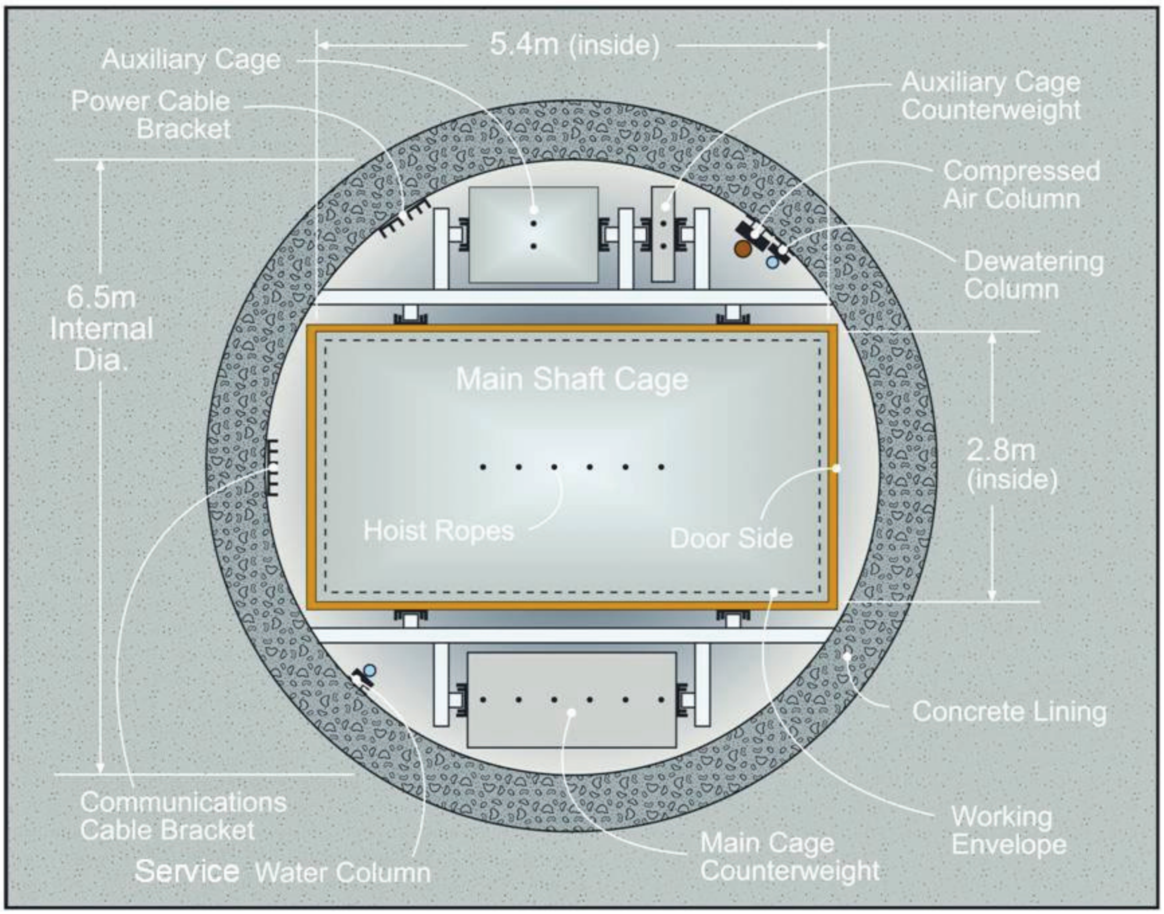

Figure 6-8: Cross Sectional View of Main Shaft

6.3.1.2 Shaft Liner

The main shaft contains a concrete liner designed for the varying conditions from the shaft collar to the shaft bottom. The liner is a key component to the support of the shafts, as well as, controlling water inflow into the shaft.

6.3.1.3 Main Shaft Hoisting

The hoisting duties for the main Koepe friction hoist are:

- Transfer of heavy equipment into and out of the repository level; and
- Transfer of L&ILW waste packages into and possibly out of the repository level.

The hoisting duty for the auxiliary Koepe friction hoist is to provide daily transfer of small non-nuclear material and personnel to and from the repository level.

The main Koepe friction hoist can, in the event of an emergency, be used to transport personnel. At no time will radioactive waste be transferred in the main shaft cage while personnel are being concurrently transferred in the auxiliary cage under normal operating conditions.

The expected duties for the two hoists are determined based on the heaviest loads to be transferred in each cage during operations and are as follows:

- Main Koepe friction hoist - waste packages including transport rail cart up to 44.0 tonnes; and
- Auxiliary Koepe friction hoist - material transport of 1.27 tonnes total or six persons per deck of the auxiliary cage (see Section 6.3.1.7).

A fundamental requirement of the hoisting systems is that the systems are stable and safe in the event of an equipment malfunction or power failure.

Both Koepe friction hoists are of a cage-counterweight configuration and are comprised of the following elements:

Main Koepe arrangement:

- Six-rope direct driven main friction hoist;
- Head rope deflection sheave cluster consisting of six sheaves on a common spindle;
- Single deck cage;
- Counterweight;
- Six head ropes; and

- Four tail ropes.

Auxiliary Koepe arrangement:

- Two-rope gear driven auxiliary friction hoist;
- Double deck cage;
- Counterweight;
- Two head ropes; and
- Two tail ropes.

6.3.1.4 Main Koepe Friction Hoist

The hoist consists of a 4.27 m diameter, fabricated steel drum with friction inserts mounted around the circumference of the drum, in which the wire ropes run. The friction between the ropes and these inserts transfers the driving and retarding torque to operate the hoisting system. A set of ropes is attached to the top of the cage and run over the hoist drum to the counterweight on the other side.

To maintain a reasonable internal shaft diameter, the ropes on the main shaft cage side are moved horizontally by a set of deflection sheaves 12.5 m below the Koepe drum (see Figure 6-3), thus providing an angle of wrap greater than 180°. A set of tail ropes are connected to the bottom of the cage and run down to a loop below the lowest shaft station and then back up to attach to the bottom of the counterweight. This type of hoist has a high load capacity since multiple ropes are used to share the load.

The hoist drum will have two integral machined steel discs on the outside of the drum cheeks, against which multiple disc brake units are mounted. These multiple brake units provide redundancy, and are designed such that only one set of brake units on one disc is required to safely stop the hoist, with the second complete set of brake units on the other disc as spare capacity. The brake controls are fully dynamic and emergency braking is achieved at controlled and ramped retardation rates to avoid any shock loads being applied to the shaft conveyances as required by Section 214 of Reg. 854. In the event of any power failure, the braking system acts in a fail safe manner to bring the hoist to a controlled stop through the use of hydraulic accumulators.

The main mechanical and structural components of the hoist (drum shaft, drums, bearings, etc.) are designed to provide a 100-year service life.

Koepe friction hoists are a well-proven technology within the world-wide mining industry for personnel, material, and rock hoisting at hoist trip frequencies and speeds well in excess of those proposed for the DGR hoists. By design, the maximum out-of-balance load will be half of the maximum cage payload. The minimum permitted factor of safety under Section 228(13) of Reg. 854 for this style of hoist is 5.5 for each head rope. The current ropes considered for the main Koepe hoist permit the loss of up to two ropes before the minimum rope safety factor is exceeded.

The multi-head rope configuration does not require the use of timber guides, and allows the shaft conveyances to run on steel guides. However, design considerations have also retained a form of single rope safety devices for the cage conveyance (wedge dogs), which are suitable for steel guides and are over and above the requirements of Section 232(6) of Reg. 854. Use of steel, rather than timber, guides in the shaft also has the benefit of removing flammable mass from the main shaft.

6.3.1.5 Main Shaft Cage and Counterweight

The main shaft cage has an external floor plan of 5.6 m long by 3.0 m wide to accommodate the largest waste package and various types of mobile development equipment. The payloads to be transferred are defined as having a maximum footprint of 2.65 m wide by 5.2 m long.

The height of the cage is determined by the longest disposal-ready waste package, which are the resin liner shields at 4.7 m long. Accounting for the transfer rail cart and handling clearances, the main shaft cage is nominally 7 m in height.

The cage top transom is designed to support the draw bar complete with chase blocks and sockets for the six hoisting ropes, as shown in Figure 6-9. For clarity, Figure 6-9 does not show the cage side and roof steel plate cladding. The bottom arrangement is designed to support the draw bar complete with chase block and socket for the four tail ropes.

The main shaft cage counterweight has a design envelope of 9.4 m long x 2.2 m wide by 1.0 m deep. It contains a series of ballast plates spread over four levels and weighing a total of approximately 33 tonnes. The top transom caters for the six head ropes and the bottom transom allows four tail ropes to be attached via their respective attachments.

6.3.1.6 Auxiliary Koepe Friction Hoist

The auxiliary Koepe friction hoist is a two-rope configuration. The ropes make a 180° wrap around the 1.4 m diameter Koepe drum and connect to the counterweight in a similar manner to that described above for the main Koepe friction hoist, but without the

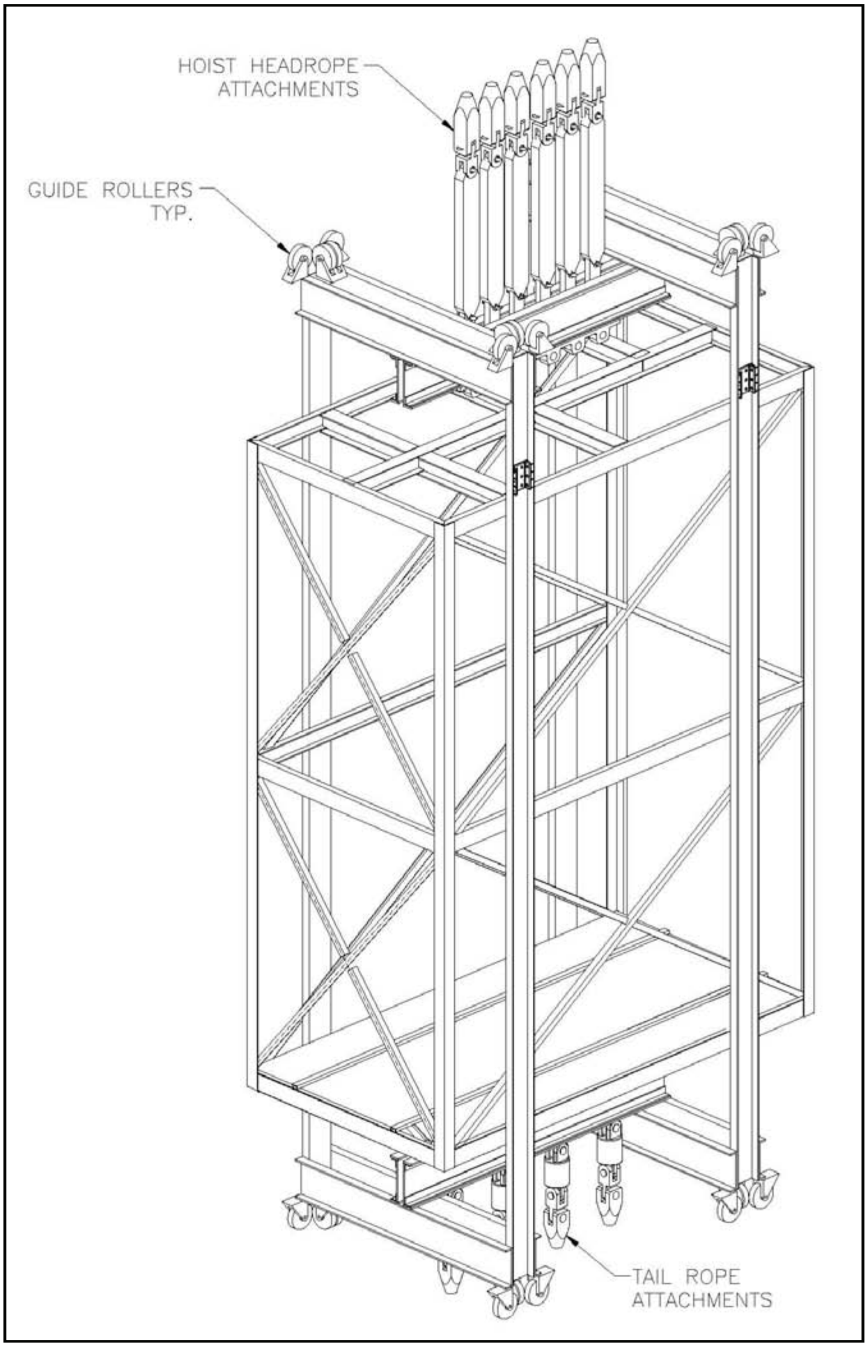

Figure 6-9: Isometric View of Main Shaft Cage

need for deflection sheaves. The drum has dual disc brakes with spring applied, hydraulically released brake disc callipers.

The hoist is situated on a level in the main shaft headframe below the deflection sheaves for the main Koepe friction hoist. Arrangement of the head and tail ropes is similar to that of the main Koepe friction hoist.

6.3.1.7 Auxiliary Cage and Counterweight

The two-deck auxiliary cage is approximately 6 m high with an internal floor plan of approximately 1.0 m by 1.4 m to transport personnel to the repository level. Each deck transports up to six people at a design mass of 90 kg per person. The cage is clad with perforated plate and each deck has lockable folding access doors.

The cage top transom is designed to support the draw bar complete with sockets, hydraulic adjusting links and chase blocks for the two hoisting ropes. The bottom transom is designed to support the draw bar complete with socket and swivel for the two tail ropes.

The counterweight for the auxiliary cage is approximately 4 m tall with a nominal plan dimension of 1.0 m by 0.3 m. The counterweight is designed to carry a series of ballast plates split over two decks. The top transom is designed to support the draw bar complete with sockets for the two hoisting ropes. The bottom transom is designed to support the draw bar complete with socket and swivel for the two tail ropes.

6.3.2 Ventilation Shaft

6.3.2.1 Layout

The ventilation shaft is nominally 745 m deep with the diameter set by exhaust ventilation air flow and construction requirements (see Chapter 9), requiring an internal finished shaft diameter of 5 m. The shaft is split into compartments as follows:

- Cage and skip compartment; and
- Compartment for upcast ventilation.

The cage and skip compartment also contains:

- One dewatering water line;
- One service water supply line;

- One compressed air line;
- Power feeders;
- Fibre optics and communications;
- Hoist signalling cables; and
- Fire detection.

The compartment for upcast ventilation also includes the following:

- One shaft dewatering line from shaft bottom to the repository level; and
- Two slick lines (for construction concrete/shotcrete only).

The ventilation shaft is equipped with one set of main steel buntons, which divides the shaft into the aforementioned compartments. Two sets of guides for each conveyance are fixed to stub buntons off the main buntons (see Figure 6-10). The buntons are fixed to the shaft concrete lining with a system of steel inserts which are designed to allow horizontal alignment.

Additionally, the open compartment contains a ladderway section down from the repository level to shaft bottom with landings to provide a second access to the loading pocket (for rock loading of the skip) and the shaft bottom.

6.3.2.2 Shaft Liner

The shaft liner design for the ventilation shaft will be similar to that described in Section 6.3.1.2 for the main shaft.

6.3.2.3 Ventilation Shaft Hoisting System

During operations, the following duties are required of the ventilation shaft hoist:

- Emergency egress for personnel; and
- Provision to remove waste rock, if required, after start of operations.

These duties are met using a double drum hoist. As the hoist is the same as that for shaft sinking, the hoist is rated on the largest expected duty of 12 tonne payload required for shaft sinking equipment.

A fundamental requirement of the hoisting system is that the system is stable and safe in the event of equipment malfunction or power failure.

The ventilation hoisting system will consist of the following elements:

- Ventilation shaft hoist;
- Two ropes – one for each conveyance;
- Two conveyances (skip and cage); and
- Two sheaves mounted in the ventilation shaft headframe.

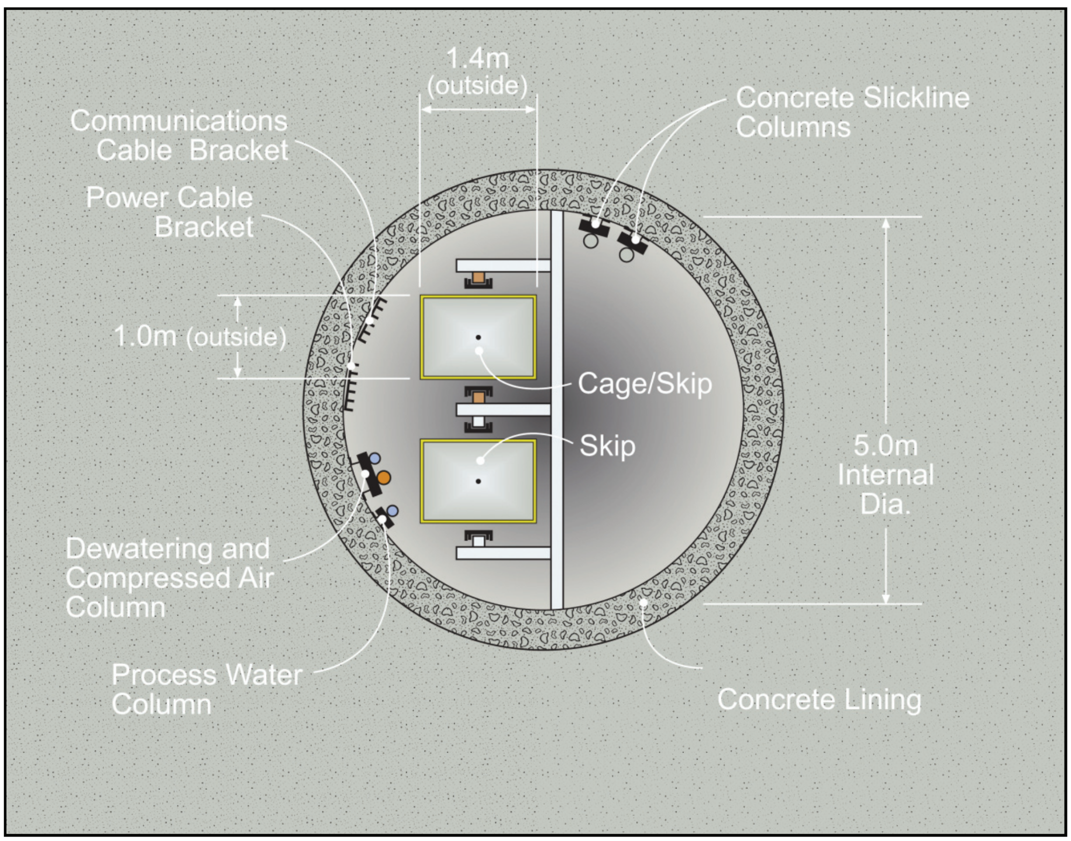

Figure 6-10: Cross-Sectional View of Ventilation Shaft

The ventilation shaft hoist is a 3.66 m double drum configuration with the two ropes passing over two sheaves mounted within the ventilation shaft headframe. The hoist is housed in a separate hoist house near the headframe and is driven by a direct coupled motor. The hoist consists of two separate drums mounted on a single drum shaft. Either one or both of the drums can be decoupled from the drum shaft through a mechanical clutch arrangement mounted on the outside of each drum. This enables the hoist to be used in a single drum configuration during operations since it is expected that the skip will not be used and will normally be stored inside the headframe.

Each drum accommodates one rope, which is coiled on and off the drums with the rope 'dead end' being secured to the drum assembly. Bolted onto each drum is a brake disc onto which spring applied, hydraulically released disc brake callipers act. The brakes and clutches of the drums are interlocked electrically, hydraulically and mechanically through the hoist's drive brake and clutch levers to prevent the withdrawal of a clutch without the relevant drum's brakes being applied.

A sophisticated drive control and safety system is part of the hoist and control of the hoist can be either fully automatic or manual from a local hoist operator's console in the hoist house.

6.3.2.4 Ventilation Shaft Conveyances

During the operations phase, the ventilation shaft will be equipped with a skip and single deck cage arrangement (see Figure 6-11). Refer to Section 9.3.7.2 for the configuration of conveyances during the construction phase. The single deck cage in the bale is designed for seven persons. The cage is nominally 3 m high with a floor plan of approximately 1 m wide by 1.5 m long.

6.3.3 Shaft Safety Systems

Hoist rope stretch is not a concern for the ventilation or auxiliary hoisting systems as the loads applied at any given time are not significant. However, to prevent movement of the main shaft cage when heavy loads are placed in or removed from the cage, a cage chairing system is installed at the collar in the main shaft headframe and at the main shaft DGR station. This system is required to keep the cage locked in position to prevent either upward or downward movement as a result of changing stretch in the hoist head ropes. The chairs are mounted on the shaft station and are moved into the shaft using hydraulics to lock both top and bottom of a load bearing member of the cage structure. The chairing system is interlocked with the main Koepe friction hoist control system to ensure that the chairs cannot be deployed until the cage has stopped in the correct position at the station and has permitted the chair system to operate. This cage locking system will minimize any differential motion between the cage floor rails and the collar rails. The rail system configuration for package transport requires

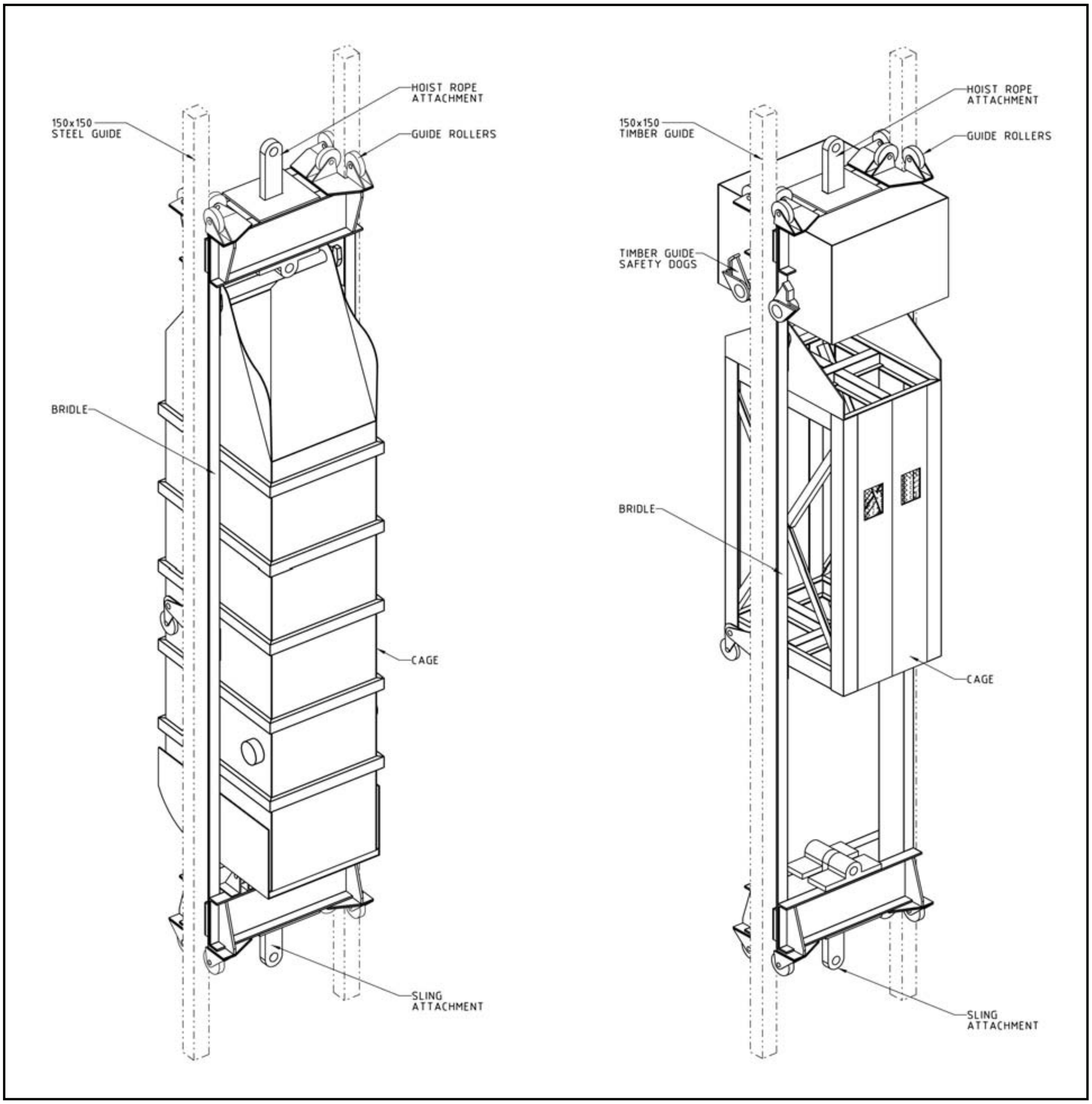

Figure 6-11: Permanent Skip and Bale with Cage

closer tolerances than would be required for a rubber-tired transport system. Such systems are common in mine cage systems with and without rails and are used effectively to provide safe conditions for heavy material movement into and out of the mine cages. Figure 6-12 shows a typical chairing arrangement for the main shaft cage.

Arresting gear, for retarding the conveyances in the event of overwind, are installed in the headframe. Refer to Figure 6-13 for arresting details, which show an identical system in a shaft bottom configuration. The conveyance, if travelling beyond its prescribed travel limits and failing to be stopped by travel limit switches mounted in the shaft, will impact upon the arrestor frame and force the frame to move along its fixed guides, deforming the arrestor strips. The kinetic energy of the moving conveyance is converted into strain energy by deformation of the arrestor strips, thereby stopping the conveyance in a controlled manner.

Guide safety systems are employed in both the main and ventilation shafts. The use of steel wedge dogs for the main shaft is described in Section 6.3.1.4 above and will also be employed for the skip compartment of the ventilation shaft. In the personnel cage compartment of the ventilation shaft, timber guides and safety dogs are used since the cage is supported on a single rope. Safety dogs are devices that are automatically deployed in the event of failure of the cage rope connection and dig into the timber guides to stop the cage from free-falling to the shaft bottom.

6.3.4 Underground Shaft and Services Area

The underground shaft and services area is laid out with the concept of locating both the main and ventilation shafts in close proximity and “clustering” the service or ancillary rooms close to the shafts (see Figure 6-14). The ancillary areas are located in such a way as to provide a degree of isolation from the movement of the waste packages.

The design has been developed based on rock mass data obtained via deep exploration boreholes. The underground layout and design may need to be adjusted to suit rock mass conditions as determined through in-situ investigations at the repository level.

Adjacent to the main shaft, there are two electrical rooms, one for 13.8 kV switchgear and the second for step-down transformers, motor control centres and communications/instrumentation. This location provides for the establishment of the permanent underground electrical infrastructure during initial development.

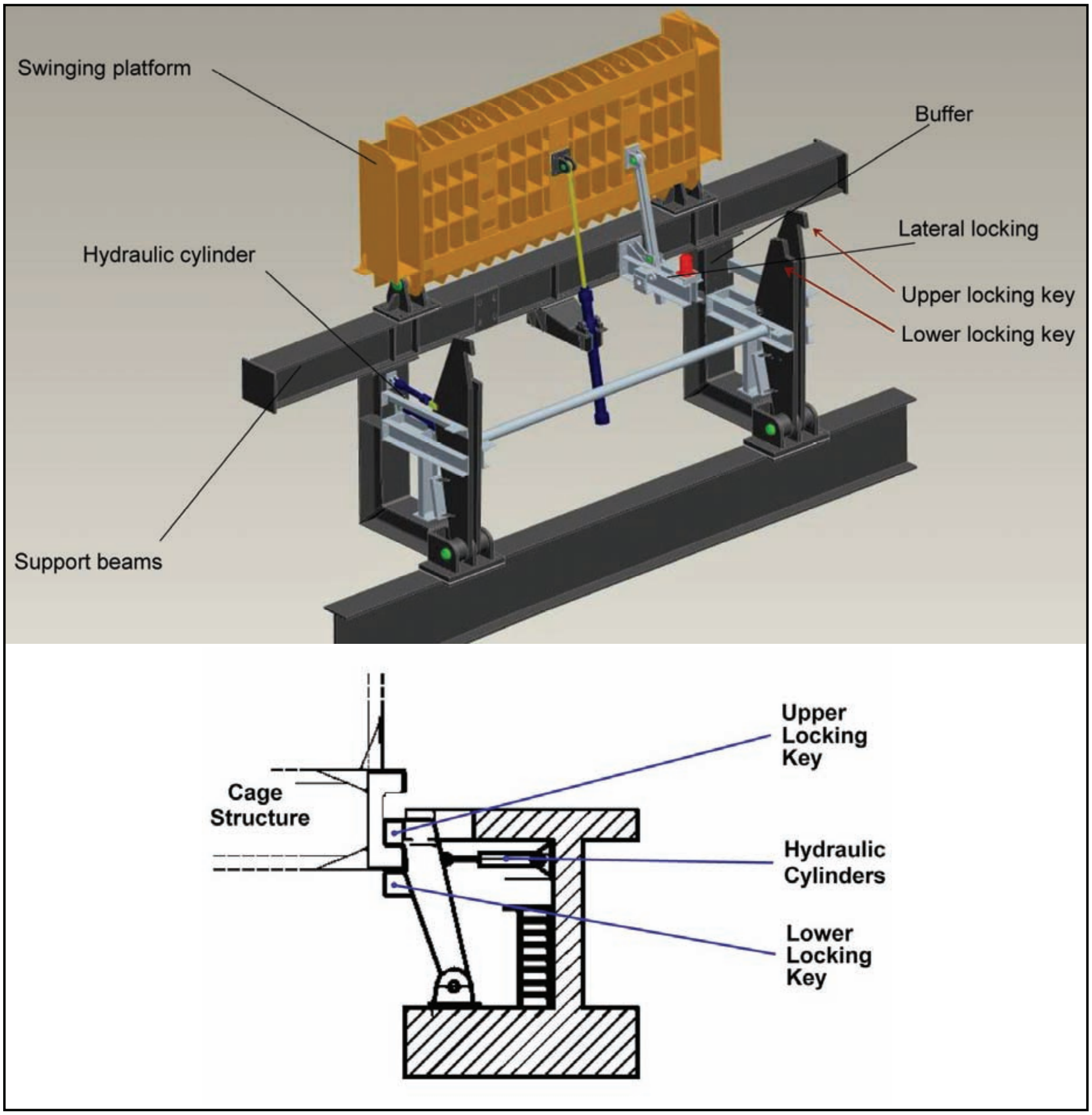

Figure 6-12: Typical Cage Chairing Arrangement

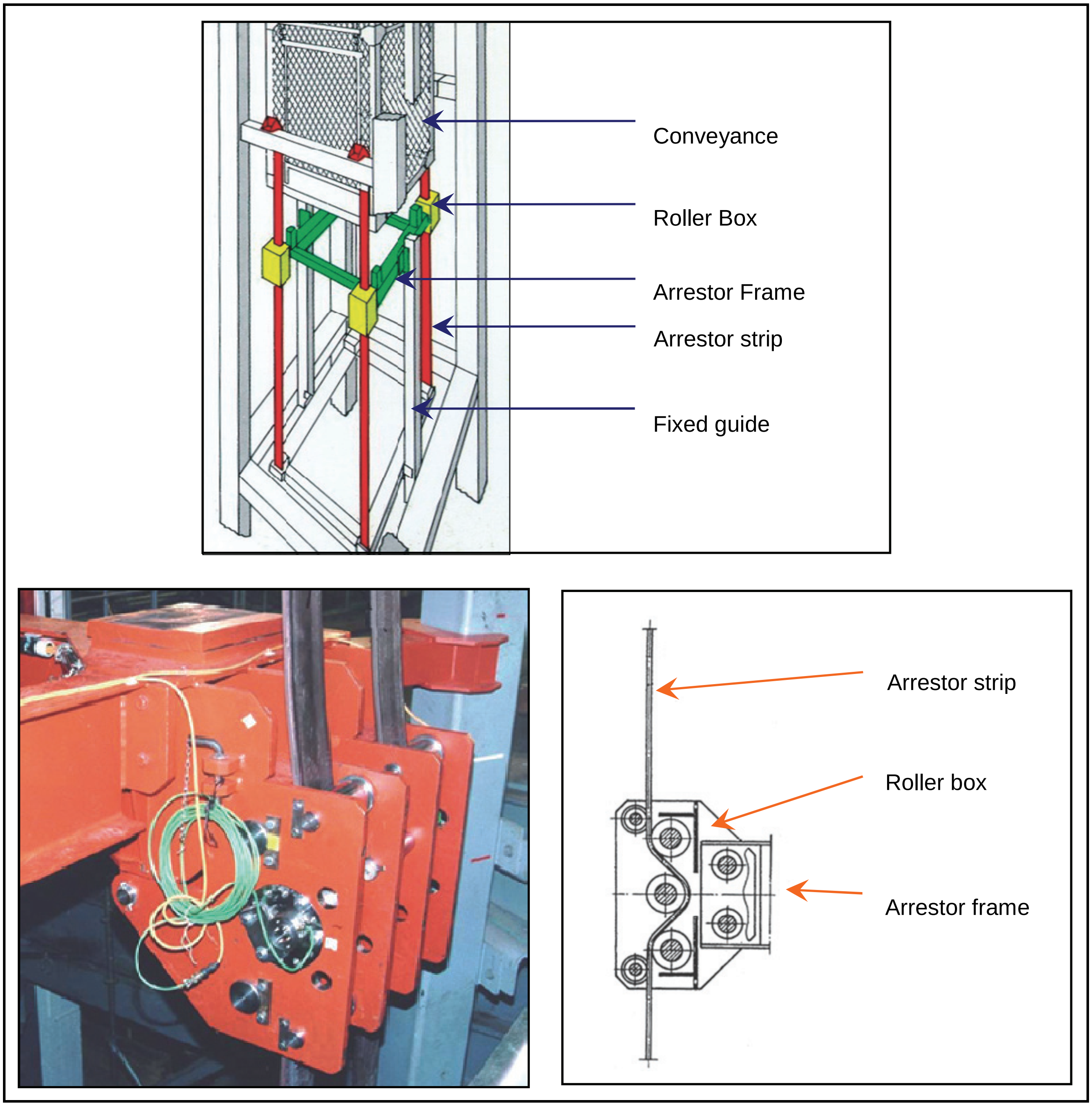

Figure 6-13: Conveyance Arrestor System

There is a combined office, lunch room and refuge station accessible from the main shaft (see Figure 6-14). The area is capable of seating approximately 24 people. A sanitary facility is located beside the office/lunch room area. A geotechnical area and stores is positioned in the south services tunnel as the activities are not related to the waste packages and workers are not required to travel past any waste package emplacement activities to reach these areas. This location is also a low equipment travel area for improved pedestrian travel.

The maintenance shop, underground diesel fuel bay are located along the maintenance access tunnel. This location provides isolation from the waste package handling route, while providing ready access to the equipment for servicing requirements. These rooms are also close to the exhaust ventilation shaft, which facilitates routing of the exhaust ventilation from the rooms direct to the ventilation exhaust tunnel. A second sanitary facility is located off of the maintenance access tunnel.

The area around the ventilation shaft includes the decommissioned rock handling facilities from the development and construction stage and ventilation fans for primary exhaust ventilation.

The main access-ways consist of the access tunnel from the main shaft, Panel 1 access, south access, and Panel 2 access tunnels. All of these tunnels have slightly differing dimensions. Figure 6-15 and Figure 6-16 show the main shaft access and south panel access which are most typical of the dimensions and services.

6.3.5 Emplacement Rooms

The emplacement room dimensions have been determined based on the waste package emplacement requirements as described in Section 6.5.3. Figure 6-17 and Figure 6-18 show the profile and dimensions of two typical emplacement room profiles.

An optimal emplacement room length of 250 m was selected by considering the following factors:

- Health and safety considerations, i.e., difference in egress time from rooms, radiation protection considerations, and operational time span for a given room;
- Ability to place the repository within the lower member of the Cobourg formation;
- Capital cost; and
- Operating cost.

Refer to Section 6.5.3 for detail on emplacement room and waste allocation.

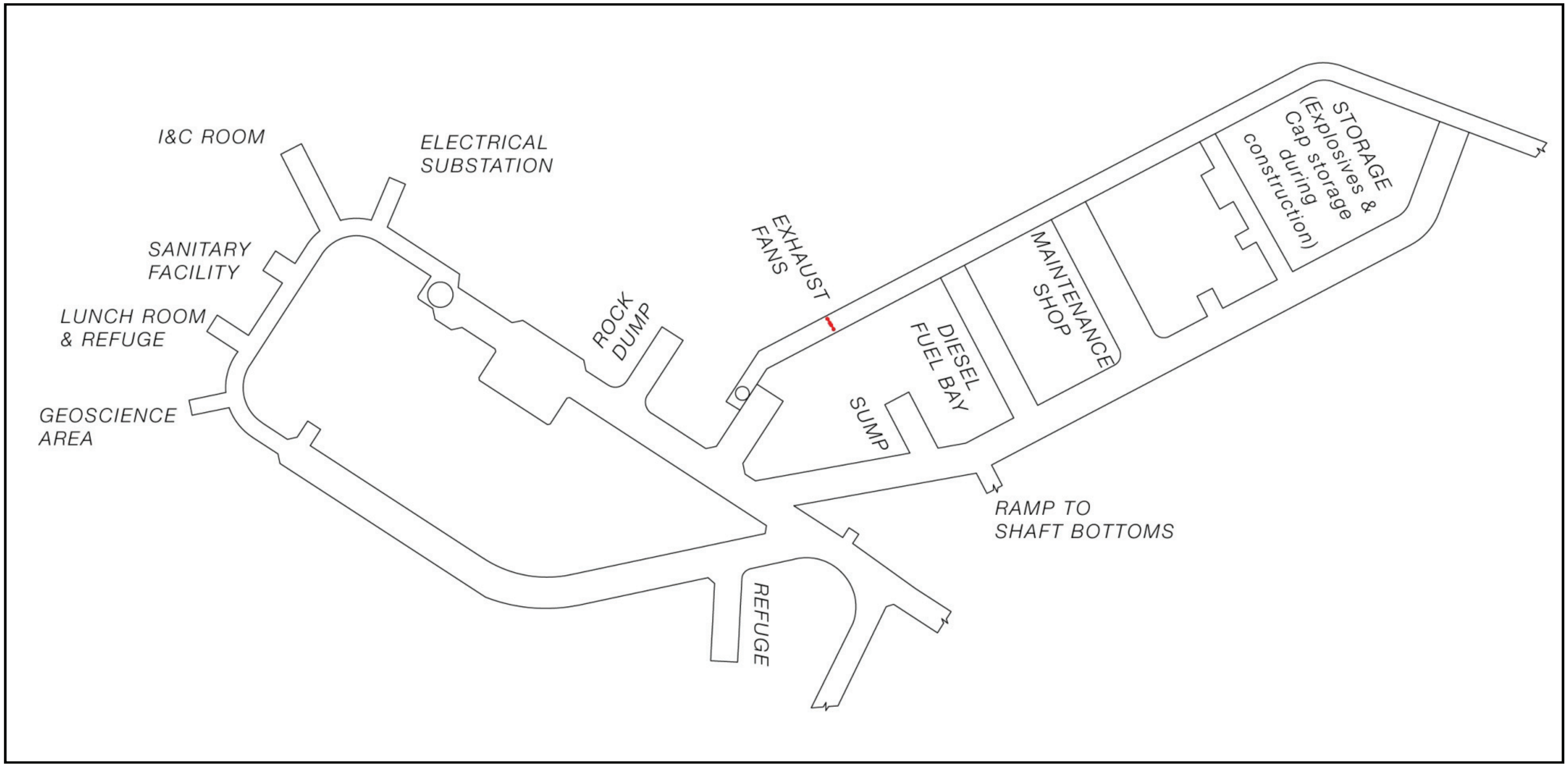

Figure 6-14: Underground Services Area

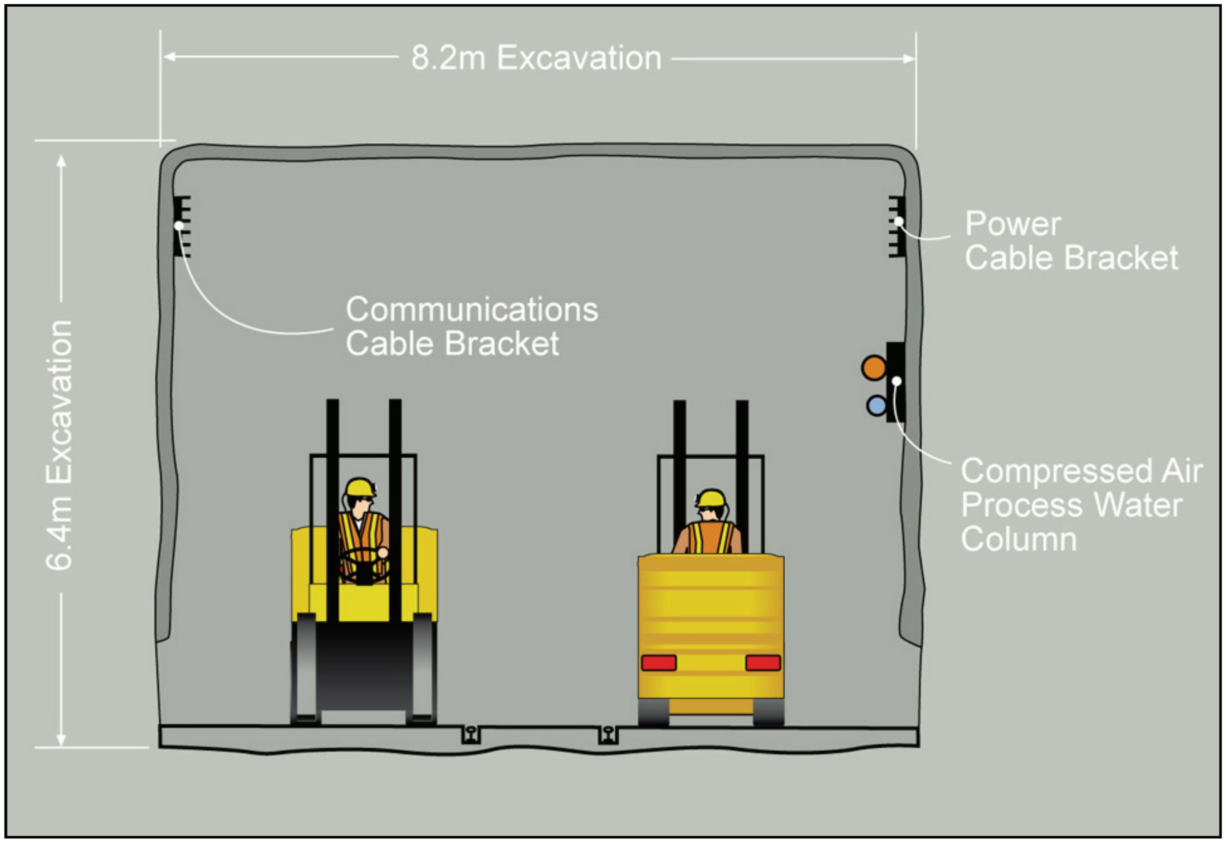

Figure 6-15: Main Access Tunnel Showing Typical Services

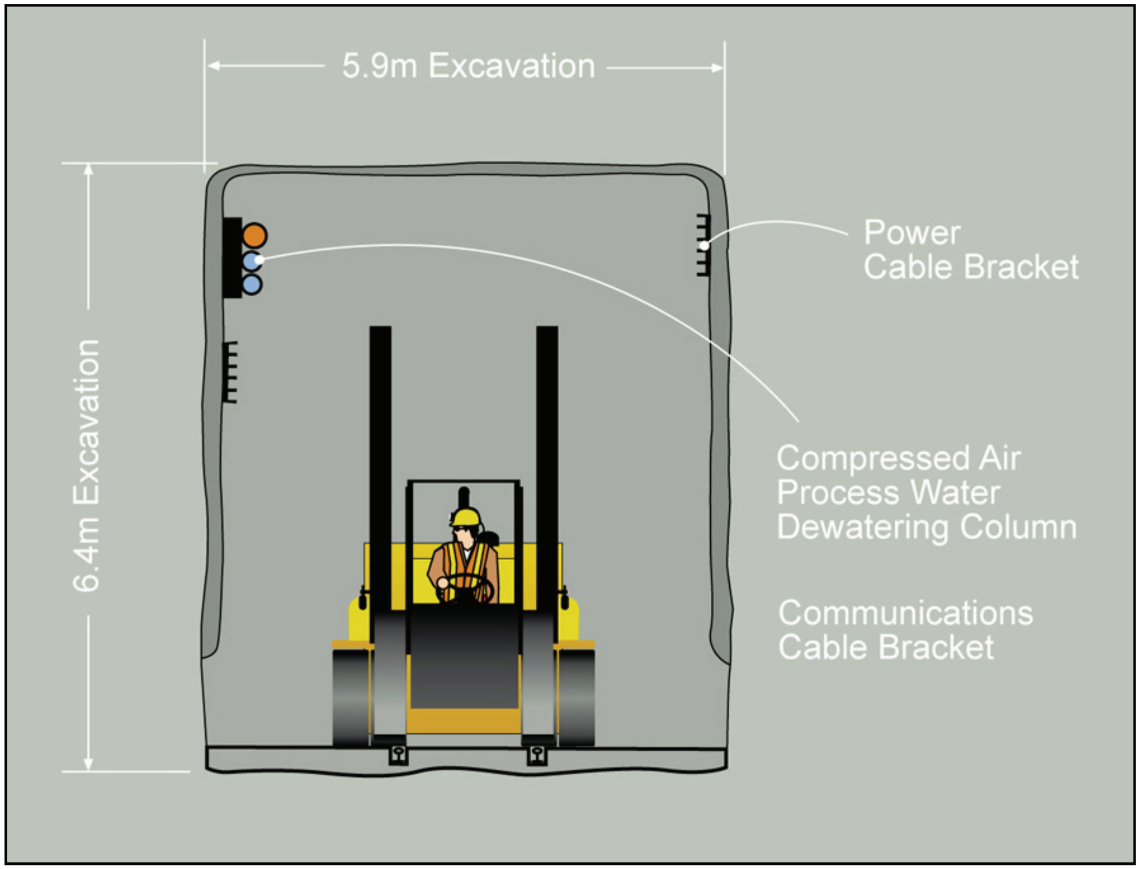

Figure 6-16: Access Tunnel Showing Typical Services

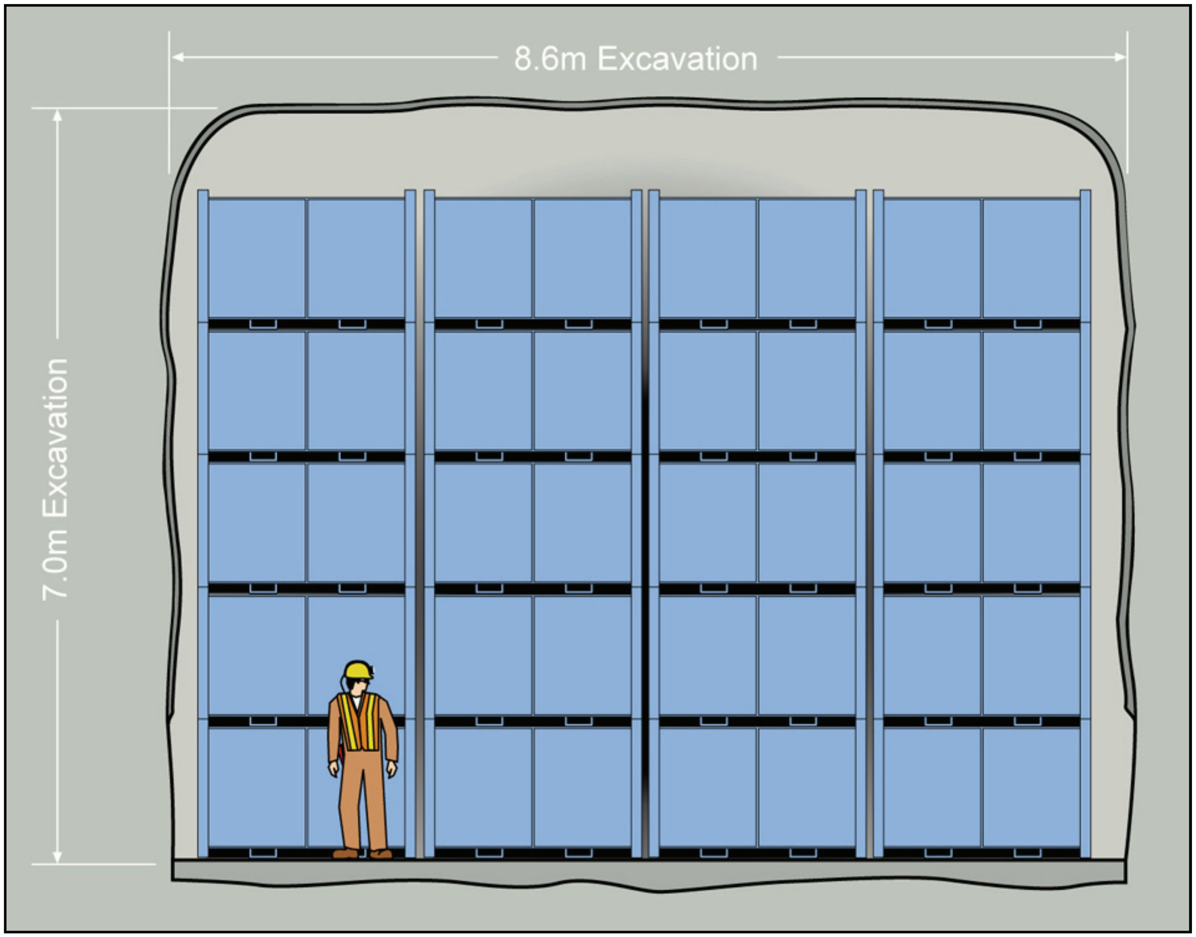

Figure 6-17: Emplacement Room Section View – Bin Type Waste Packages

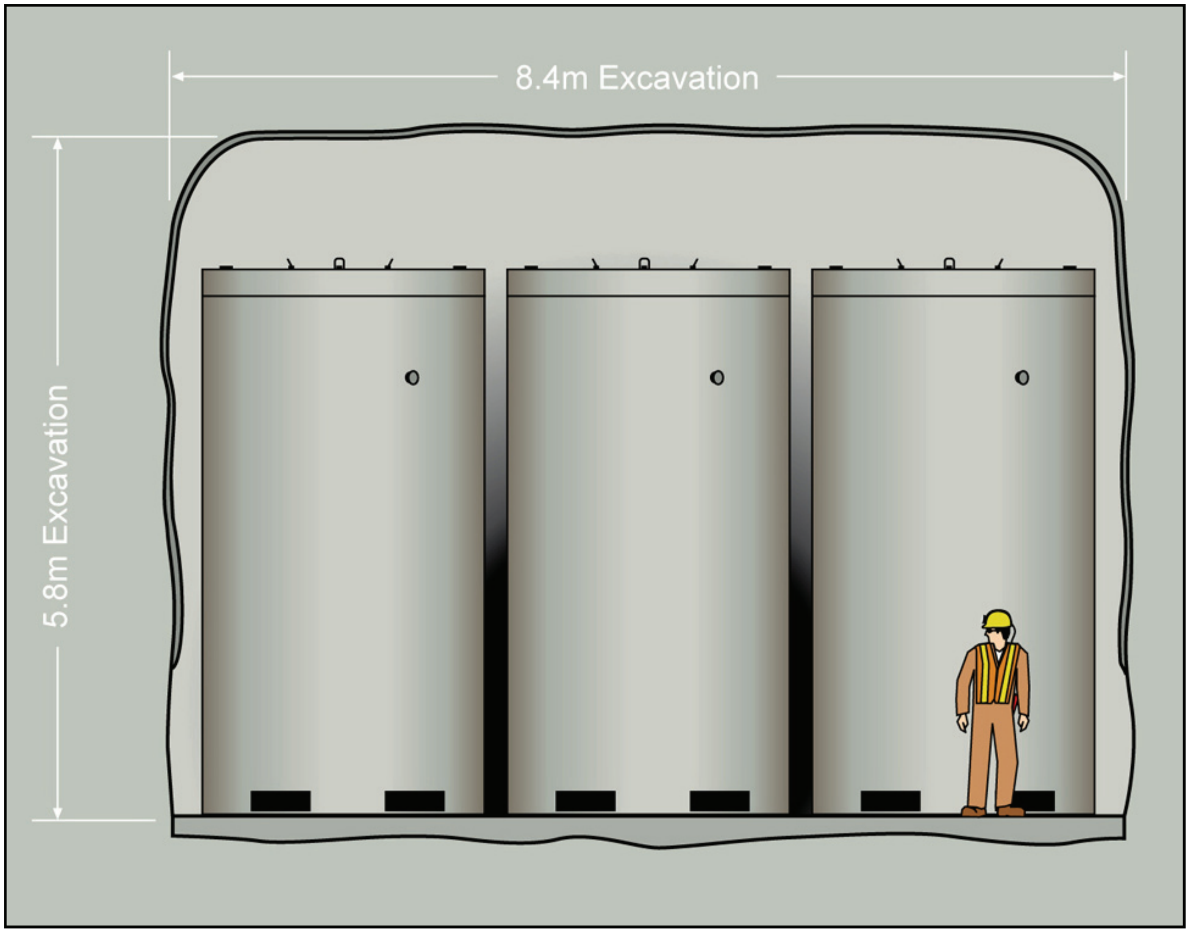

Figure 6-18: Emplacement Room Section View – Resin Liner Type Waste Packages

6.3.6 Ramp to Shaft Bottoms

A ramp providing access to both shaft bottoms and the loading pocket is shown in Figure 6-6 and Figure 6-7.

Reasons for including a ramp are listed below.

- Provides access for personnel and equipment to the shaft bottoms versus climbing down from the repository level for operation and maintenance requirements. (Note: a ladder-way is installed from the DGR level to the shaft bottom, providing an alternate means of egress from the loading pocket area and shaft bottom).
- Provides efficient shaft bottom sump configuration and the ability to construct suitably sized sumps for each shaft at significantly lower cost than "in-shaft" sump configurations.
- Servicing of the shaft bottom pumps from the ramp as opposed to from inside the shaft; thus making it safer, simpler and less labour intensive to install and maintain.
- Provides access to the main shaft bottom for servicing of tail-rope monitoring sensors and other hardware at this location.
- Provides significantly increased sump capacity in the ramp and shafts (approximately 8,500 m^3) to mitigate unforeseen water inflow events prior to water levels reaching the repository level.

6.3.7 Loading Pocket

To facilitate the excavation of the repository, a rock handling system is constructed as part of the initial development following shaft sinking activities. The loading pocket is connected to the repository level through a waste rock raise (see Chapter 9). The loading pocket is inactive during the operation of the DGR.

6.3.8 Underground Ventilation

The reliable delivery of fresh air to the underground workplaces is critical for the health and safety of workers. This air supply is used to maintain safe working conditions through all stages of the DGR life. The total volume of air supplied to the DGR varies based on the nature of work being performed, the number of active and non-active rooms and will be periodically adjusted throughout the life cycle of the facility.

Ventilation air will be supplied to the facility to ensure that:

- There is breathable air available for all underground personnel;

- Contaminants are diluted and removed;
- Personnel are not exposed to levels of noxious gases that exceed regulatory limits;
- Levels of explosive gases do not exceed explosive limits; and
- Temperatures within the DGR are maintained such that it remains safe and acceptable for both personnel health and infrastructure integrity.

The current reference ventilation system design as described in Section 6.3.8.1 is a "flow-through" system with fresh air moving via excavated tunnels and rooms and returned to the exhaust ventilation shaft via the exhaust ventilation tunnels running perpendicular to the emplacement rooms.

6.3.8.1 Ventilation System and Operation

The DGR is designed to be a "flow-through" ventilation system. Ventilation flow throughout the facility is facilitated primarily by the action of maintaining the underground facility under a negative pressure such that air flows from the main shaft (acting as the fresh air intake) and through the repository level to the ventilation shaft (acting as the exhaust route). This is achieved through the application of a pressure differential between the main and exhaust shafts by the operation of the underground exhaust fans.

While designed as a pull-type ventilation system, low-pressure fans will be used to deliver a controlled air volume from the surface intake to the collar of the main shaft so that the main exhaust fans do not cause a "negative pressure" condition in the main shaft headframe. The fresh air supply fans deliver air at a volume and pressure such that positive pressure is imparted to the main shaft headframe. This positive pressure ensures that should there be an incident at the WPRB or main shaft headframe, potentially contaminated air is not sent down the shaft and through the repository level.

Fresh air enters the facility through the heater house by the action of the surface intake fans and is delivered to the main shaft through the intake plenum. The heater house contains an electric heating system, typically used between November and April, to ensure air enters the main shaft at a temperature (5°C) such that services within the main shaft are not affected by cold surface ambient air temperatures.

Following distribution of the fresh air underground and collection of the return air through the exhaust ventilation tunnels to the base of the ventilation shaft, the return air returns up the ventilation shaft and through the exhaust plenum by the action of the underground exhaust fans (see Figure 6-19).

The main shaft and ventilation shaft are located approximately 85 m from each other (with the intake and exhaust structures being approximately 160 m apart). The shafts are located relative to each other so that the upcast ventilation shaft is generally downwind of the air intake at the main shaft, taking into account the prevailing wind directions at the Bruce nuclear site. Positioning of the large main shaft complex, which includes the WPRB, between the ventilation exhaust fan discharge and fresh air intake also assists in dispersing and diluting the concentration of any contaminants in the air that may flow in the direction of the intake.

The distribution of the air underground is controlled by the main exhaust fans located underground at the ventilation shaft on the repository level and regulators at the ends of the emplacement rooms. Figure 6-20 shows how the downcast fresh air exits the main shaft at the repository horizon and supplies ventilation to the shaft services areas and the access tunnels to the east and west of the main shaft station.

Fresh air is directed from the main shaft to the access tunnels through the use of booster fans located adjacent the shaft (see Figure 6-20). A portion of the air may be ducted to reduce the quantity and velocity of air flowing unconstrained in the services tunnel to acceptable levels.

The balance of the fresh air flows freely across the main shaft station to the ventilation shaft to provide for diesel equipment and personnel unloading the main shaft cage and the staging area.

The criterion for air distribution to the emplacement rooms is that the airflow direction shall be from areas of low potential of contamination to areas of greater potential contamination. To maintain this, fresh air is taken along the access tunnels to the emplacement panels with the return air from the panels removed through the exhaust tunnels. The Panel 2 exhaust tunnel connects to the Panel 1 exhaust tunnel as shown in Figure 6-7. High pressure exhaust fans are located at the end of the Panel 1 exhaust tunnel at the ventilation shaft. These high pressure fans keep the exhaust tunnel under negative pressure, encouraging the fresh air to travel outwards along the access tunnels.

6.3.8.2 Ventilation System Capacity

The total airflow to be delivered underground is determined by the amount of diesel equipment in operation underground and the amount of air to be delivered to various rooms and facilities to ensure safe working conditions. The following describes the required amount of airflow during construction and operations phases, as well as, during a potential repository expansion scenario. The repository expansion scenario sets the required maximum airflow capacity of ventilation system of 130 m^3/s.

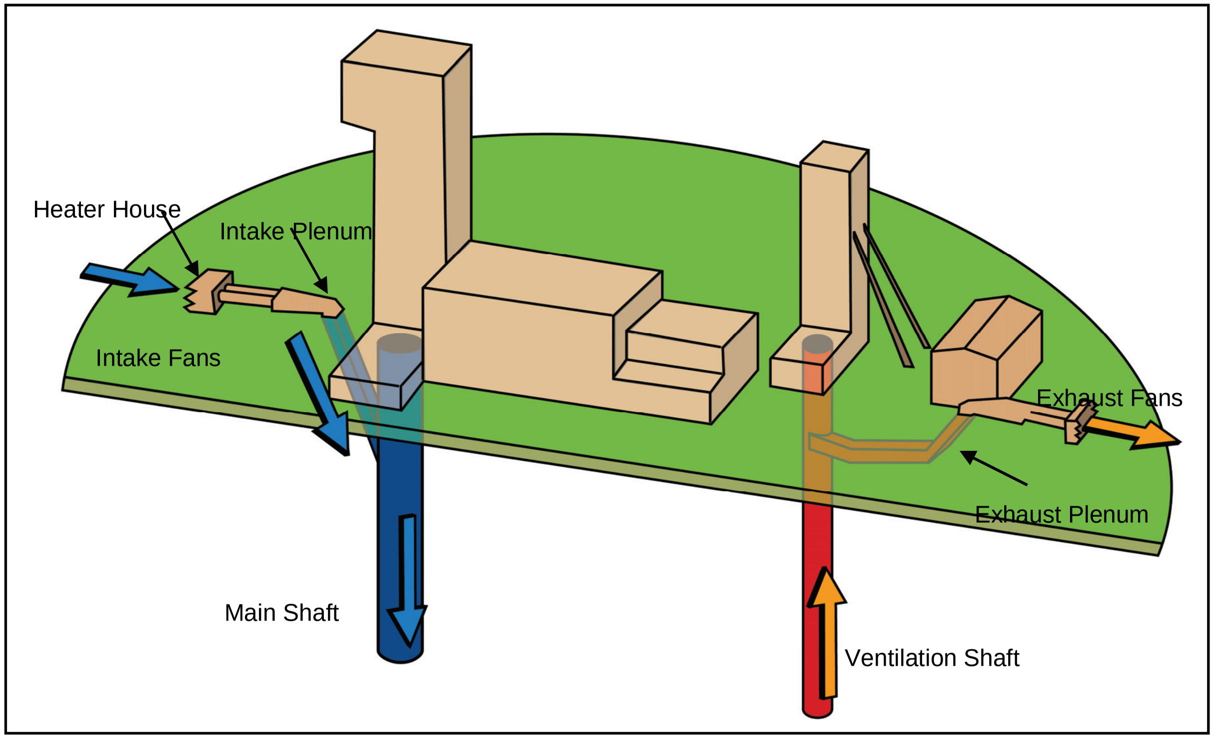

Figure 6-19: Primary Surface Ventilation System

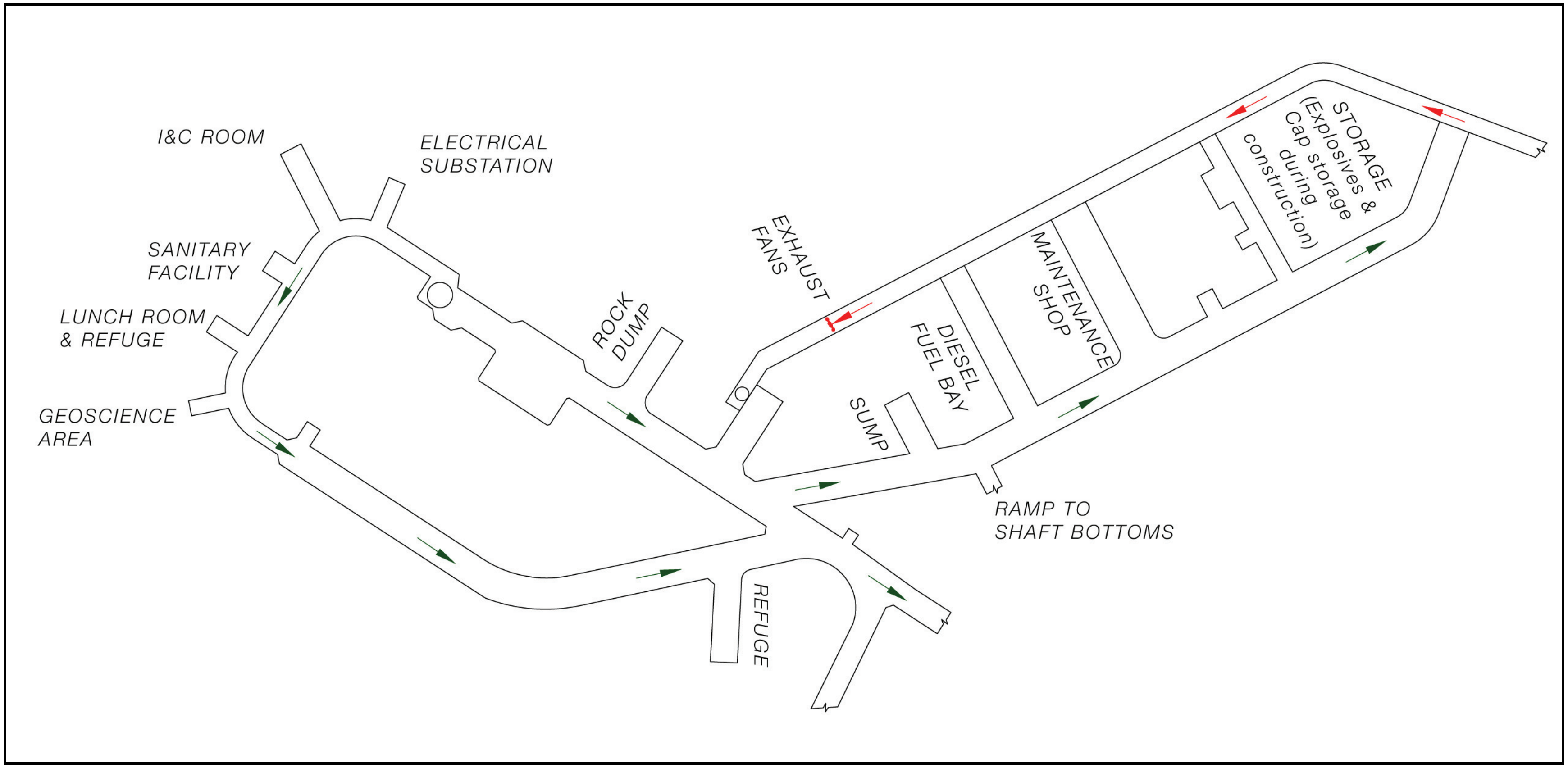

Figure 6-20: Underground Ventilation Distribution System

To be conservative in estimating the maximum required airflow in the ventilation system, it was assumed that construction activities could be re-initiated at the end of the operating life to support repository expansion. In this case, it is assumed that 5 emplacement rooms will not be isolated by closure walls and will require continuous flushing. The expected maximum airflow through the DGR Facility is as follows:

- Construction diesel equipment = 102 m^3/s;
- Maintenance shop = 12 m^3/s;
- Underground diesel fuel bay = 11 m^3/s; and
- Five emplacement rooms filled with waste and end walls but no closure wall in place = 5 m^3/s.

The addition of each of these gives a total maximum airflow through the DGR of 130 m^3/s.

The airflow requirements of the operations phase are met with the above system capacity. The airflow requirements will fluctuate depending on the number of active rooms, the number of rooms with end walls, flow requirements for off-gas removal and the equipment being used. Airflows of 85 – 120 m^3/s are required during the operations phase.

6.3.8.3 Operations Ventilation

During operations, each emplacement room will be either empty, active or filled. Each of these stages requires a different approach to ventilation.

Upon the completion of emplacement room construction, there will be a period of time before active emplacement commences. During that time, it is planned that the empty emplacement rooms will not be ventilated. These rooms are considered "confined spaces" (Reg. 854, Part XII) and access to non-active empty rooms prevented.

Unventilated empty rooms will therefore require:

- Installation of a barricade at the entrance to the room;
- Adequate signage indicating entry is prohibited; and
- A procedure for re-entry (e.g. inspection of regulator, air monitoring, ground inspection, etc.) that meets acceptable atmospheric conditions (Reg. 854 Section 294) and developed health and safety guidelines; see Figure 6-21 for example of a non-active empty room.

Upon commencement of emplacement operations, fresh air requirements are managed through the use of airflow regulators at the end of the emplacement rooms and variable speed exhaust fans at the ventilation shaft. Typical arrangements are shown in Figure 6-22 and Figure 6-23.

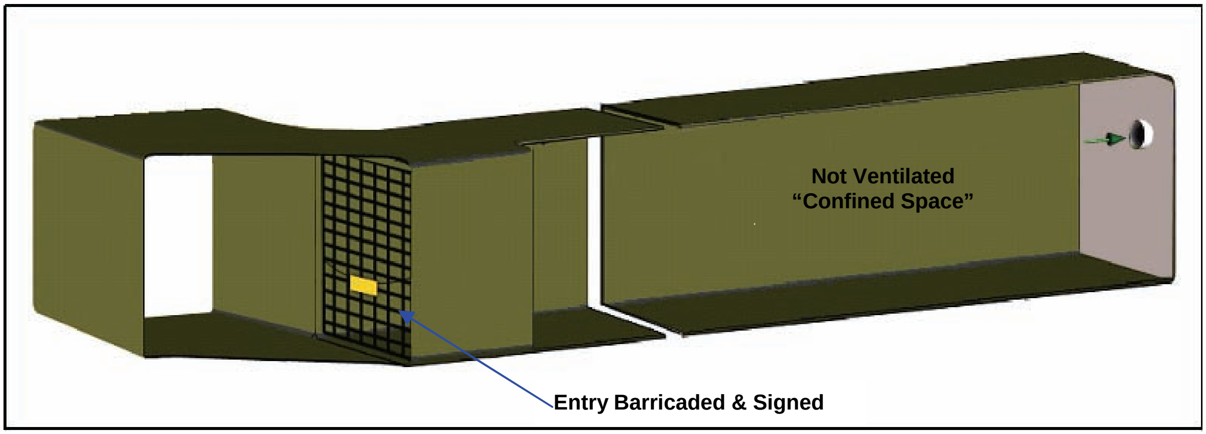

Figure 6-21: Non-Active Empty Emplacement Room

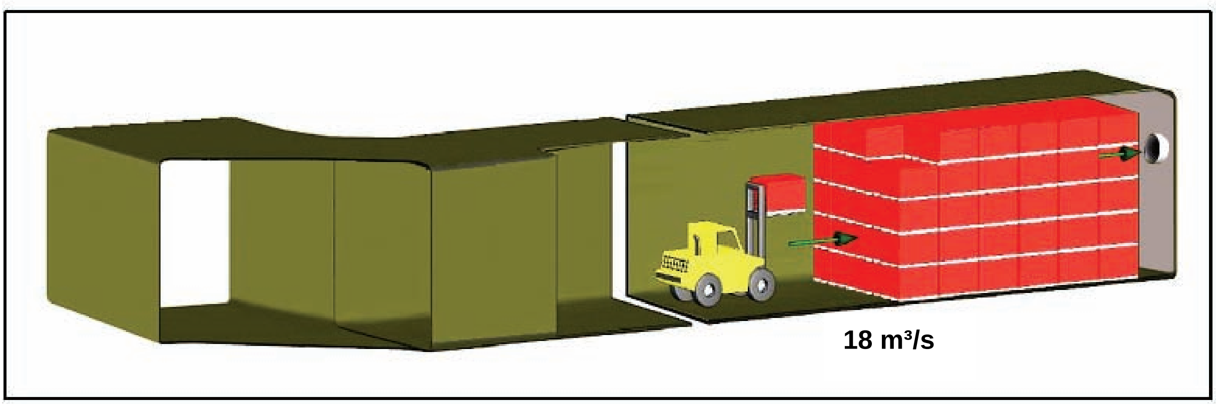

Figure 6-22: Active Emplacement Room – Typical LLW

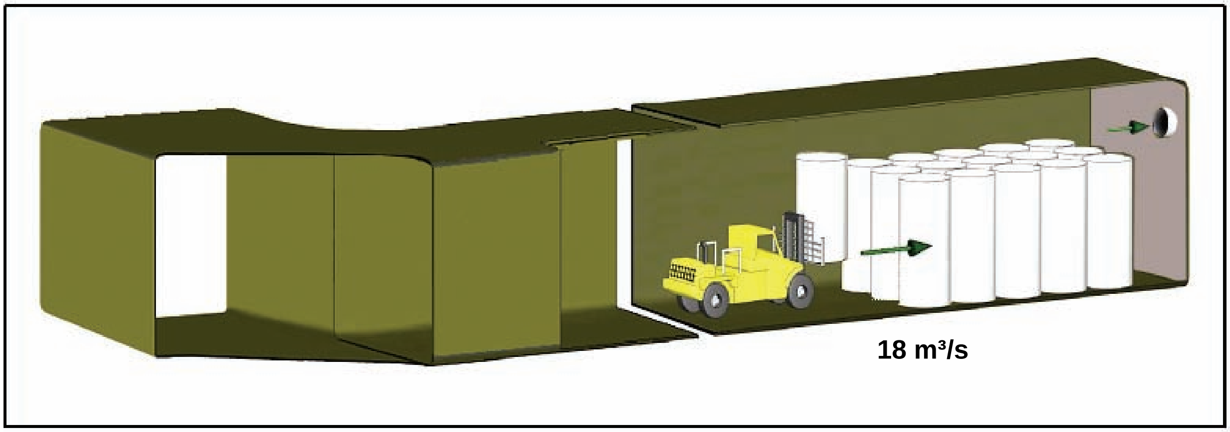

Figure 6-23: Active Emplacement Room – Typical ILW and Heavy LLW

Following emplacement activities, a filled emplacement room will be monitored while adjacent rooms are being filled. During this period of time, the room is continuously ventilated at a much reduced flow rate to enable one air exchange every 2 to 4 hours, or approximately 1 m^3/s. Continuous flushing provides the benefits listed below.

- Maintenance of temperature and humidity levels. A non-continuously flushed room may allow humidity to rise, which may accelerate the corrosion of ground support and waste packages.
- Minimal opportunity for gases to collect or concentrate.
- Monitoring of contaminants in the exhaust flow is continuous.
- Timely reaction to fire - continuous smoke detection available, smoke from any part of room can be detected.

Regulating the airflow in filled emplacement rooms is the same as with active rooms. The regulators restrict the amount of air flowing to meet requirements.

An end wall will be constructed, as required, at the entrance to each emplacement room to provide worker protection from radiation from the waste packages in the room, prevent people from entering the room and/or to control ventilation airflow (see Figure 6-24 and Figure 6-25).

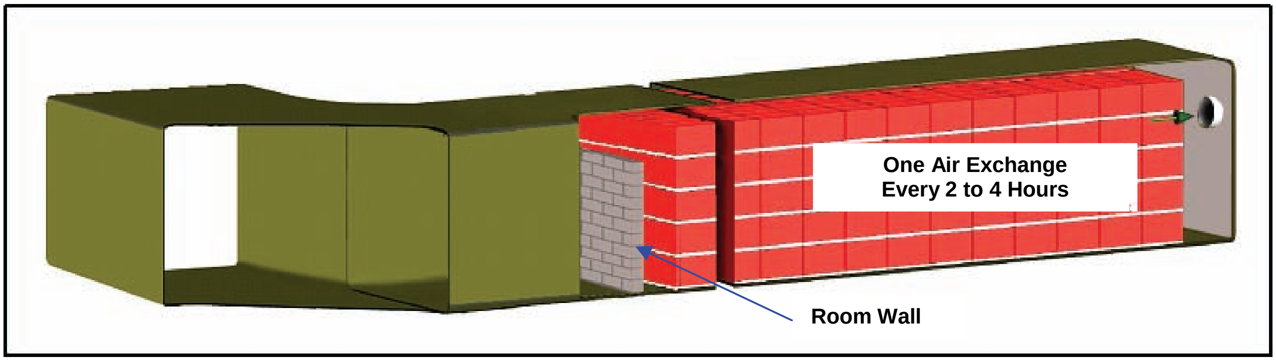

Figure 6-24: Filled Emplacement Room – Typical LLW

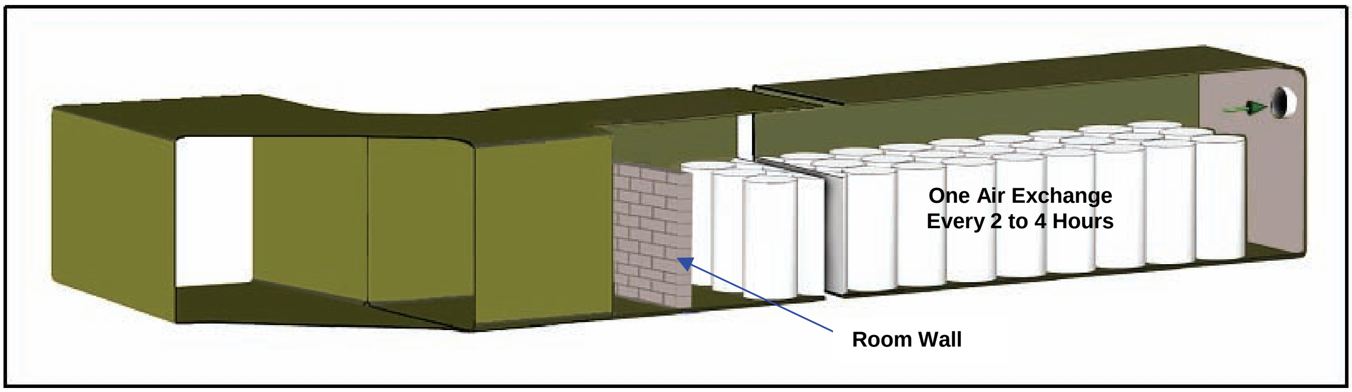

Figure 6-25: Filled Emplacement Room – Typical ILW and Heavy LLW

6.3.8.4 Intake Fans and Electric Heaters

The function of the surface intake fans is to provide the required airflow for the DGR at a pressure that overcomes the losses across the heater house, silencers and intake plenum.

The fans also generate pressure sufficient to impart a positive pressure into the main shaft headframe while providing enough pressure to a neutral point, (point at which pressure from intake fans and pressure created by the exhaust fans equal zero), in the main shaft below the plenum. The fans are designed to deliver the maximum required flow at any point through the life of the DGR.

The two intake fans are of equal specification (each nominally 1.8 m diameter, 112 kW axial fan), arranged to operate in parallel to provide the required flow and are located on the outlet side of the heater house. An additional fan, of equal specification, is available as a standby.

The operating point of the intake fans is expected to change throughout the life of the facility. As such, each fan is operated through a Variable Frequency Drive (VFD), which changes the rotational speed of the fan hub and the fan operating point. In concert with the exhaust fans, use of VFDs provides the added benefit of permitting the ventilation system to be reduced when no one is underground.

At the DGR site, surface temperatures will fall below freezing at various times during the year. Heating is applied to the air intake for the main shaft to a minimal level of 5°C to prevent freezing of services in the shaft.

During the normal heating season period, the heaters operate as required when inlet temperatures fall below the heater set point (less than 5 °C).

A surface refrigeration plant to cool the air before being delivered to the main shaft for transport underground is not included in the reference design. Heat flow modelling considered the application of heat loads at various points through the facility. The primary heat loads considered were as follows:

- Autocompression;
- Rock strata;
- Diesel equipment;
- Electrical equipment;
- Concrete and shotcrete placement; and
- Retube waste containers.

Modelling was undertaken for both construction and operation periods of the DGR, considering surface climatic conditions at yearly average temperature (6.1 °C), average maximum temperature for the warmest month (24.0 °C) and average minimum temperature for the coldest month (-11.3 °C). The modelling considered the location at which the highest temperatures are likely to occur within the facility during both construction and DGR operation periods in respect to the design criterion of keeping below 28.5 °C wet bulb globe temperature. It was found that:

- Construction: the design criteria would be exceeded when surface conditions reach 26.4 °C dry bulb and 21.0 °C wet bulb which happens for around 138 hours per year; and

- Operation: the design criteria would be exceeded when surface conditions reach 28.5 °C dry bulb and 22.8 °C wet bulb which happens for around 47 hours per year.

Based upon the modelling undertaken and the results, particularly with regard to temperatures expected at underground locations, it was ascertained that a permanent surface refrigeration system is not required for the facility. If necessary, construction or operating activities can be curtailed. However, space has been allocated in the surface layout in proximity to the heater house complex for a bulk air cooler and refrigeration plant, if required, in the future.

6.3.8.5 Exhaust Fans

There are two sets of exhaust fans and their function is to ensure that the DGR operates as a "pull" type of ventilation system. The main fans are located underground in the ventilation exhaust tunnel at the repository level. A second set of fans is located on surface to draw the upcast air through the plenum as opposed to the ventilation headframe. The plenum intersects the ventilation shaft below the collar and extends along the subsurface to the surface exhaust fan inlets. It is likely that water will condense out of the exhaust air during certain periods throughout the year as it enters and travels along the plenum. The plenum is designed such that the water drains along the plenum and is collected.

The two main exhaust fans at the repository level are of equal specification (each nominally 1.7 m diameter, 131 kW axial fan), arranged to operate in parallel and located at the exit of the exhaust plenum. An additional fan, of equal specification, acts as a standby for a total of three fans. Noise mitigation is attained through the application of acoustic baffle type silencers on the outlet of the exhaust fans.

As with the intake fans, the exhaust fans are operated through VFDs to provide the variation of ventilation requirements throughout the DGR life, as well as, reduce ventilation flow when the DGR is off-shift.

6.3.9 Monitoring of Underground Structures

A geotechnical instrumentation and monitoring program will be developed for the DGR to assess performance of openings and rock support systems. The program will be developed on the basis of geotechnical data collected from boreholes prior to start of construction, as well as, data collected during construction (see Chapters 3 and 9). The monitoring program will include, as a minimum:

- Pillar convergence monitoring extensometers;

- Rock bolt and shotcrete load cells;

- Multi-point borehole extensometers to measure roof convergence; and
- Tape or laser extensometer arrays to measure convergence of rock openings.

6.3.10 Underground Services

Underground services, for the most part, are linked to the shared services as described in Section 6.2.4 above. The DGR communications and controls, as described in Sections 6.2.4.2 and Section 6.2.4.3, respectively, are integrated systems between surface and underground. The following services are further described specific to the underground configurations:

- Electrical;
- Diesel fuel;
- Service and potable water;
- Dewatering; and
- Sewage.

6.3.10.1 Electrical Services

Power is fed down the two shafts at 13.8 kV and terminates at the 13.8 kV electrical substation on the repository level. Shaft power cables are redundant, fire resistant high tensile that meet the Insulated Cable Engineers Association guideline's safety factors and are approved for shaft use by CSA and Mines and Aggregates Safety and Health Association (MASHA).

The DGR 13.8 kV electrical substation distributes 13.8 kV to portable Mine Power Centres (MPCs) and the repository level double ended substation. Two portable MPCs are installed at appropriate positions in the DGR and moved as operations retreat back towards the shaft. Portable MPCs are used to minimize voltage drops over long distribution distances in underground openings and step the voltage down from 13.8 kV to 600 V. The 600 V system is used to power the emplacement room lighting skids. The double ended substation powers large 600 V loads and feeds the underground MCC. Among other electrical power users, the underground MCC will feed:

- Sump pumps;
- Dewatering pumps; and
- Small power distribution transformers for lighting and receptacles.

6.3.10.2 Diesel Fuel

For the underground repository, an underground diesel fuel bay is included at the services area. It provides fuelling for both construction and operations stages. The 5,400 L diesel fuel storage, comprised of two 2,700 L steel fuel totes, is an integrated unit with built-in leak containment and fire suppression system. The dispensing unit is air-powered with retractable hoses.

The underground diesel fuel storage unit is recharged as required using the 2,700 L forkliftable metal fuel totes. These totes are filled at the existing WWMF fuelling station and delivered to the DGR via the WPRB. The totes are loaded onto rail carts using the light duty forklift and transferred in the main shaft cage. Fuel totes will never be transferred in the main shaft cage at the same time explosives or waste packages are delivered underground.

6.3.10.3 Potable and Service Water

Potable water is transferred underground in bottles or jugs and provided to all personnel in the underground areas for both drinking and hand washing. Bottled water is available at various locations including the lunch room. The underground hand washing stations are similar to those used in mining operations, where the washing stand is integrated with a small reservoir, pump, and water heater. The reservoir is filled using typical 18.5 L drinking water jugs.

Service water is primarily required for the construction phase of the project to supply water for drilling, dust suppression and equipment wash down. During operations, use of service water is limited as it is important to limit moisture at the repository to minimize the potential for condensation within the ventilation shaft.

Service water is supplied underground using a heavy-wall steel pipeline in the ventilation shaft. A spare column of equal specification is installed in the main shaft for use as a backup during maintenance or in the event of failure in the ventilation shaft. Both shafts are equipped with automatic shut-off valves at the surface in the event that the water column fails. At the base of the shaft, a pressure reducing valve is used to reduce the static pressure to a safe working pressure. Steel pipes distribute the water throughout the repository with down-pipes provided at regular intervals to provide access for hose connections.

6.3.10.4 Underground Dewatering

Water from the main shaft, ventilation shaft, Panel 1, Panel 2 and the shaft and services area is collected in sumps at each of these locations and pumped to the dewatering sump. Periodic sampling of the ventilation sump water to test for tritium concentrations will performed. If required, the water at the ventilation sump could be placed into totes and brought to surface for treatment. The dewatering sump is located off of the ramp slightly below the repository level near the ventilation shaft. Water is pumped to the surface via a positive displacement pump through the ventilation shaft discharge column. A back-up discharge column is also provided in the main shaft.

To minimize potential contamination, the underground maintenance shop and the underground diesel fuel bay are each equipped with an isolated containment sump. These sumps are suitable for containing any accidental fluid spills, such as fuel, oil, or engine coolant and any captured fluids are pumped into a tote on the repository level and transferred in the main shaft cage to surface for appropriate treatment.

The dewatering system consists of several separate sumps. The combined storage capacity of the shaft bottoms and the shaft bottom ramp could be used for emergency and temporary water storage in the remote event of a major water in-rush.

Each sump is equipped with redundant water level instruments, which transmit the level of water in the sump to the main control room. The pumps are arranged to run automatically, but may also be manually started from the main control room or locally at the sump.

The sump design takes into consideration the need to remove collected sediments and thus all sumps are accessible and maintainable. A sump being cleaned will need to be pumped empty and the incoming line will be locked out as required. To handle sediment material, manual cleaning via pressure washer and industrial vacuums will be used as appropriate.

The main shaft and ventilation shaft bottoms are connected by a cross-cut to the ramp from the repository level, at which a sump is constructed to collect any groundwater ingress. Each shaft bottom sump is equipped with one operating and one stand-by submersible pump configured to allow quick change over (i.e., turn of a switch). The ventilation shaft bottom sump also collects any condensation that forms within the shaft.

Sump pumps run once the sump level reaches its sump “live” capacity. In the remote event of an in-rush scenario, a single main and ventilation shaft pump would direct flow to the dewatering sump where the two positive displacement pumps work in parallel to transfer water to the surface. The design capacity for the pump flow required for the main shaft, ventilation shaft and dewatering sump pumps is based on the amount needed to accommodate a 15 L/s flow in addition to the normal flow of nominally 2 L/s.

Any groundwater coming from the emplacement rooms gravity drains into the Panel 1 and Panel 2 sumps where the fluid is pumped out with a submersible pump to the dewatering sump. Because of the very low permeability of the Cobourg Formation host rock, underground openings are expected to be dry with little or no inflow.

The pumps at the dewatering sump are installed on permanent concrete foundations. The two pumps are positive discharge plunger pumps which are ideally suited to this high-static head application. One pump normally operates with the second acting as a stand-by in the event of failure of the first pump or to supplement pumping capacity to clear any excessive short-term inflow of water. Each pump is rated to pump 11 L/s. The full pumping capacity with both pumps operating in parallel is 22 L/s.

The submersible pumps at the other sump locations are typical to industrial and mining applications. These relatively small and rugged pumps have integral motors, frames, and inlet screens. The pumps are simply lowered directly into the sump using an overhead winch and are easily replaced, if required.

6.3.10.5 Sewage

For sewage in the underground areas, toilets will be provided at the sanitary facilities. These "mine toilets" are typical to underground mining applications and use compressed air to function as simple, small-scale sewage treatment plants. This allows the self-contained toilet/reservoir units to function for approximately 18 months before a fluid clean-out is required. These will be forkliftable and will be taken to surface for the clean-out work to be completed.

6.4 DGR Waste Package Inventory

As described in Chapter 5, the DGR inventory consists of operational and refurbishment L&ILW from OPG-owned or operated nuclear facilities. This includes both existing stored wastes at the WWMF as well as L&ILW arising from future nuclear reactor operations and refurbishment projects. The total projected emplaced waste package volume is approximately 200,000 m^3 with a total of about 50,000 packages.

All waste packages delivered to the DGR will be required to meet the waste acceptance criteria described in Section 5.5. All packages will have lids and will be free of loose contamination.

For handling purposes, the L&ILW inventory has been divided into four categories or groups based on size, mass and handling features (see Table 6-5). These categories have been used for developing waste package transfer methods and determining emplacement room sizing and layouts.

Table 6-5: Summary of Waste Handling Groups

Group Name	Group Title	DGR Handling Method
A	Bin-Type Waste	Rail Carts and Light Duty Forklifts
B	Heavy Non-Forkliftable	Rail Carts and Cranes
C	Light ILW	Rail Cart and Light Duty Forklifts
D	Heavy Forkliftable	Rail Cart and Heavy Duty Forklifts

In the following subsections, the processes for retrieving waste packages from the WWMF (waste received from operating stations will be received at the WWMF prior to transfer to the DGR), and processing these waste packages, if necessary, to put them in "DGR-ready" state, are discussed. This would be conducted under the operating licence of the WWMF and are only described here for context and completeness.

6.4.1 Group A – Bin-Type Wastes

Group A containers consist of LLW in bin-type containers. These include low level resin pallet tanks, drum and bale racks, compactor boxes, drum bins, non-processible (non-pro) bins, etc. See Chapter 5 for details of the Group A package inventory.

It is assumed that all of the ash bins (old), ash bins (new), drum racks - baghouse ash, ash bins (new) - baghouse ash, low level resin boxes (90"), ALW sludge boxes and approximately 10% of the drum racks - non-processible drums are to be overpacked in the standard container overpack. For planning purposes, there are anticipated to be approximately 3,200 overpacked bin-type packages.

6.4.2 Group B – Heavy Non-Forklift

The Group B packages are relatively heavy and are not appropriate for handling using a forklift. The types of waste represented in this group are described below.

6.4.2.1 Shield Plug Containers

The shield plug containers represent the smallest quantity of any of the waste package types within the inventory. They are large and heavy, at approximately 26 tonnes. They are not stackable and must be handled using a crane. Removable shielding and specialized lifting hardware, if required, will be installed at the WWMF during retrieval from storage.

These specialized containers utilize removable shielding panels during the package retrieval process at the WWMF, which is included in the total mass of the package. It is planned to keep the shield plug containers in storage for some time after the DGR is in service to allow additional decay and further reduction in dose rates to allow safe transfer into DGR without excessive amounts of shielding.

6.4.2.2 Heat Exchangers and Heat Exchanger Segments

Prior to transfer from the WWMF, the heat exchangers may need to have protuberances (e.g., nozzles, supports) cut off and any openings so created welded closed with a seal plate. This will be done to improve the stacking qualities of these items in the underground emplacement room. In preparation for transfer at the WWMF, lifting lugs may need to be affixed to the exterior of the heat exchangers to allow them to be lifted by overhead crane.

For planning purposes, it has been assumed that all heat exchangers will have the same dimensions with the most common size from the Pickering NGS (2.0 m diameter by 4.57 m long) taken as representative. It has also been assumed that 25% of the heat exchangers will exceed the shaft cage dimensional limitations and they will, thus, be grouted to stabilize the contents and cut into sections prior to receipt at the DGR.

6.4.3 Group C – Light ILW

The packages in Group C are similar in size and handling features as Group A, but are ILW waste and fewer in number.

6.4.3.1 Unshielded Resin Liners

Resin liners are vessels of carbon steel, stainless steel or carbon steel in stainless steel overpacks used to store spent ion exchange resins. These resins are considered ILW and normally require shielding to allow safe handling.

A portion of the resin liners in storage at WWMF prior to the DGR operational phase may have dose rates such that they can be safely handled without shielding. When the mobile crane is used to retrieve one of these liners out of an IC, it will be lowered into an awaiting sacrificial pallet. A light duty forklift will then be used to load them onto the flat-bed trailer used to transfer them to the DGR.

The remaining resin liners currently in in-ground storage and future resin liner packages that require shielding will be placed in one of three shields (see Section 6.4.5.2).

6.4.3.2 Tile Hole Liners

The tile hole liners are a steel tube, which has dimensions of 3.0 m long by 0.61 m in diameter. The mass of the grouted liner will be approximately 2.0 tonnes.

The tile hole liner is equipped with lifting brackets, which will be used to lift the container from the tile hole liner with a mobile crane. Because of the liners' narrow profile, they will be placed horizontally on stackable racks at the WWMF for transfer to the DGR. Each of the stackable racks will hold two liners.

6.4.3.3 ILW Shields

After the start of DGR operations, it is assumed that the waste materials previously destined for the in-ground T-H-E liners would then be placed in ILW shields. These shields, which have yet to be designed in detail, are anticipated to be relatively compact in size and lightweight; at 1.0 m diameter by 1.7 m high with a full mass of 2.3 tonnes. Integral forklift pockets will allow the light duty forklift to be utilized for handling. Other alternative container designs, such as the ATHELs to be used for repackaging existing T-H-E liner wastes are also being considered.

6.4.4 Group D – Heavy Forkliftable Waste Packages

The Group D packages are large, heavy, and are handled using a heavy duty forklift as described below.

6.4.4.1 ETHs

ETHs are classified as LLW and have dimensions of 4.6 m tall by 1.5 m in diameter with a mass of about 25 tonnes. ETH packages are comprised of an outer cylindrical steel pipe that encapsulates the waste-filled tile hole that was once in the ground.

The ETH package has features that allow lifting by forklift. The forklift pockets are an integral part of the outer steel shell. The ETH may be transferred to the DGR by flatbed truck or heavy duty forklift. The ETH package will be in a vertical orientation throughout the entire transfer process.

6.4.4.2 Resin Liner Shields

To safely handle some of the resin liners that are in storage prior to the operations phase of the DGR they must be placed in shields. The dose rates from resin liners will vary and thus shielding requirements will also vary. The expected dose rates on resin liners were examined and a suite of standard shield designs was developed. All shields will be appropriate for handling using a heavy duty forklift.

The shield 1 is a cylindrical concrete shield, which can accommodate two stacked resin liners. Where a shield 1 does not provide sufficient shielding for the given dose rate of a resin liner, a shield 2 or a shield 3 may be used. Depending on the existing packaging arrangement, there will either be one or two resin liners placed in these shields (shields 3 and 2 respectively) to meet the waste acceptance criteria weight requirements.

The shells from Quadricells are existing shields that were used before in-ground storage of these packages was employed. The concrete shells contain two resin liners and are considered to be disposal-ready. These concrete shells are similar to a shield 1, but are 4.6 m tall. These shells will require a sacrificial pallet to facilitate movement with the large forklifts and set the height of many of the excavations.

6.4.4.3 Retube Waste Containers

Retube wastes are associated with the reactor refurbishment process. To date, two types of shielding containers have been used for the retube wastes from Bruce A reactors: one, designated for volume reduced components (pressure tube, calandria tubes and calandria tube inserts); and one slightly narrower and longer package, designated for uncut end-fittings. The stackable containers are a steel-concrete-steel construction with a maximum full mass of 35 tonnes.

6.4.4.4 Steam Generator Segments

The steam generators are too large and too heavy to transfer whole into the DGR. Thus, it is assumed that they are segmented prior to transfer to the DGR. A reference approach for segmenting the steam generators has been developed. However other methods of size reduction may also be employed.

In the reference approach, each steam generator will be filled with low-density grout to stabilize the internal parts, then cut into sections using a diamond wire saw. Each segment will be sealed with a plate welded to each cut end. These plates will serve a dual purpose of increasing the shielding of the grouted segment, and providing a flat surface to aid stacking in the emplacement rooms. Forklift pockets and other elements to aid stability during stacking will be welded onto one seal plate on each segment to facilitate safe lifting and transfer of these segments.

The internal dimensions of the main shaft cage are the defining restraints for the size of large segmented sections of the steam generators. The masses and dimensions account for the attachment of steel plates to seal the cut ends and forklift pockets.

The base of the segments will be fitted with forklift pockets, which will be designed to not only withstand the loads imposed during lifting and handling, but also the loads due to stacking of segments in the emplacement rooms.

6.5 Transfer Equipment and Emplacement Operations

The DGR is capable of receiving, inspecting, tracking, handling, and emplacing all operational and refurbishment L&ILW from OPG-owned or operated reactors.

6.5.1 Methods of Surface Waste Package Handling

6.5.1.1 General

All waste packages retrieved from WWMF will be transferred in a DGR-ready state on flat-bed transporters, covered transporters, or forklifts to the WPRB. The packages will be inspected to ensure that damage has not occurred in transfer and confirmed that waste acceptance criteria have been met.

Once the package has been deemed acceptable, packages will be off-loaded by forklift or overhead crane and placed into the staging area, if required, prior to being loaded onto rail carts for shaft transport. The WPRB will be arranged with off-loading facilities for both flatbed trailers and covered trailers.

All waste package shields will be generally designed to ensure that dose rate limits are not exceeded. However, there may be some packages on which the dose rate limit is exceeded. Packages with high dose rates may require spot shielding or temporary shields to meet the waste acceptance criteria as part of a specific ALARA plan to protect workers.

All waste packages will be transferred into the main shaft cage by means of a rail-based transfer cart. Empty carts will move into the waste package loading area of the WPRB where forklifts or the overhead crane will place packages on the cart deck. The packages will be secured on the cart, as required, to ensure that the load will remain stable while the cart is moved into and out of the cage and while the cage is in motion.

The following sections describe the process for handling each group type of waste packages at the WPRB.

6.5.1.2 Group A – Bin Type Waste

The bin-type LLW packages are transferred from the WWMF and off-loaded and stacked in a staging area on the incoming side of the WPRB by a light duty forklift, or directly to an empty rail cart. The DGR is capable of receiving and transferring at least 24 of these packages in one 8-hour shift.

6.5.1.3 Group B – Heavy Non-Forklift Waste

All Group B packages are large and heavy items, which are handled by crane as they are not able to be moved by forklift. They are transferred on a flatbed truck to the DGR and off-loaded into the staging area (if required) by the WPRB overhead crane. The same overhead crane in the WPRB is used to place these waste packages on a rail cart for transfer into the DGR. To ensure stability of the cylindrical heat exchanger packages, they are loaded horizontally onto a custom-designed saddle that is pre-installed on the rail cart.

6.5.1.4 Group C – Light ILW

The Group C packages are alike in overall size and mass to the Group A packages and thus the handling methodology is also similar. The light duty forklift is used to off-load these packages from the transfer vehicle, if not transferred by forklift directly from the WWMF, and also to place them onto rail carts. Because they are generally higher in dose rate, the staging area will only be used at the WPRB when it is necessary, with most packages being loaded directly onto the rail carts.

6.5.1.5 Group D – Heavy Forkliftable Waste

The Group D packages are handled in the WPRB using the heavy duty forklift. This same forklift may be used to transfer these packages from the WWMF, or they will arrive on a low-deck flatbed trailer.

6.5.2 Shaft Handling

The following describes the shaft handling description for all waste.

6.5.2.1 Capacity

The main shaft cage has a defined capacity for the total size and mass of waste packages transferred within it on any given trip between surface and the repository horizon. The deck is sufficiently large for several of the smaller packages to fit side-by-side as long as their size and weight do not exceed the main shaft hoisting capacity.

6.5.2.2 Groups A, B, C and D

The general process of shaft handling for all rail cart-based packages is the same. After the package or packages are properly loaded and secured on the cart, the steps to transfer the rail cart to the cage and lower the cage to the repository horizon are summarized below.

1. The main shaft cage is chaired at the collar shaft station, hoist brakes are set and the hoist "locked-out", the cage door and station gate are opened, and the rail stop is removed.

2. If the cage contains an empty rail cart, the rail switch is closed, the cart is unlocked from the cage, the electrical tether is connected, and the cart is moved out to the empty cart siding.

3. The rail switch is opened, ensuring that the section of rail between the loaded rail cart and the main shaft cage is clear.

4. The loaded rail cart is traversed into the cage.

5. The electrical tether is disconnected (note that the cart brakes are automatically applied).

6. The rail cart is locked in the cage as required, and the loaded cage is inspected to verify it is ready for travel.

7. The cage door and station gate are closed and the station rail stop is replaced, releasing the hoist control interlocks.

8. The cage is lowered to the underground station. This will involve the hoist operator unchairing the cage and lowering it at the designated velocity.

The steps in unloading the cage at the underground station are similar to the reverse of the loading process.

1. The underground station and unloading area are prepared for receipt of a loaded rail cart. This includes opening the rail switch in addition to preparing an empty cart at the empty cart siding for the return cage trip.

2. The cage is chaired and the hoist brakes applied and locked out. With this complete, the interlocks allow the station door and cage door to be opened and the rail stop to be removed.

3. The securing mechanism to lock the cart in place in the cage is released and the electrical tether connected.

4. The rail cart is moved out of the cage to the package off-loading area. The rail tether is disconnected from the full cart and connected to the empty rail cart.

5. The rail switch is closed and the empty cart moved onto the cage. The tether is disconnected, the cart locked in the cage and inspected.

6. The cage door and station gate are closed, and the rail stop is replaced, allowing the cage to be unchaired and hoisted to surface, repeating the process.

In addition to the slow speed of the rail cart, hydraulic or mechanical dampers are used to prevent any damage to the cage, rail cart, or package in the event that the operator fails to slow the cart at the end of travel.

6.5.3 Underground Transfer and Emplacement in Rooms

To minimize worker dose rates, rooms will be generally filled starting at the furthest distance from the shaft and working back in the direction of the main shaft. This reduces the time operators spend driving past rooms containing waste packages.

6.5.3.1 Room Profiles and Waste Package Allocation

The DGR contains the currently projected inventory of waste packages within 29 of the 31 emplacement rooms in Panel 1 and Panel 2. The remaining two rooms are planned to cater for possible contingencies (e.g., potential growth in waste package quantities or dimensions or failure to achieve assumed packing efficiencies in emplacement rooms).

The clearances around the waste packages required for operational considerations are summarized in Table 6-6.

Packages are stacked based on the limitations of the package construction and available headroom in the respective emplacement room, whichever is lower.

The bin-type wastes (Group A) will normally be kept in rooms separate from ILW packages. However, heat exchangers, ETHs, steam generator segments, and shield plug containers may be emplaced in the same rooms as ILW packages. A summary of emplacement room profiles and the types of waste packages that will be placed in each room type is provided in Table 6-7.

Profile 1 Rooms

The majority of emplacement rooms are Profile 1 emplacement rooms for Group A bin-type waste packages because of the relatively large number of bin-type waste packages.

Profile 2 Rooms

The Profile 2 room is designed to accept tall waste packages and specifically the resin liner shells from Quadricells and the ETHs. These packages require a minimum room height of 6.4 m.

Table 6-6: Summary of Clearances Used in the Emplacement Room Design

Location	Reason for Clearance	Minimum Clearance (mm)
Package to emplacement room side wall.	Ground support protuberances.	300
Package to package (front, back and sides).	Practicality of package manoeuvring at final emplacement location.	50
Top of package to roof – Group A waste.	Practicality of package manoeuvring at final emplacement location.	750
Package to roof clearance – all except Group A waste.	Practicality of package manoeuvring during transport in emplacement room.	1,150
Emplacement room entrance. (Unusable length from room entrance to first row of packages, measured at centre line)	Clearance to construct end wall.	8,000
End of emplacement room. (From last row of packages to wall.)	Room to allow ventilation airflow.	1,000
Wall to package clearance for gantry crane equipped rooms.	Clearance for "legs" of gantry crane.	1,000
Top of package to excavated roof clearance - gantry crane rooms.	This allows room for crane and hook, and rigging.	3,000

Table 6-7: Summary of Waste Package Allocations to Emplacement Rooms

Room Profile	Waste Type	Groups	Specific Waste Types	Rail / Gantry
P1	LLW	A	Bin-Type Waste	No
P2	LLW/ILW	D	Steam Generator Segments ETHs Tile Hole Liner Racks Shielded and Unshielded Resin Liners	No
P3	ILW	C,D	Shielded and Unshielded Resin Liners	No
P4	ILW	C	ILW Shields ATHELs Retube Waste Containers	No
P5	ILW	D	Steam Generator Segments	No
P6	LLW/ILW	B,D	Shield Plug Containers Heat Exchangers	Yes

Profile 3 Rooms

The Profile 3 emplacement room is used exclusively for the emplacement of resin liners. Both shielded and unshielded resin liners are placed in these rooms.

Profile 4 Rooms

The Profile 4 room is used for retube waste containers and ILW shields. The 6.5 m height allows for stacking of two-high retube waste containers and three-high ILW shields.

Profile 5 Rooms

The Profile 5 room is used for steam generator segments and this room type is 8.4 m wide and 6.2 m high.

Profile 6 Rooms

The Profile 6 emplacement rooms are connected to the embedded rail network and receive Group B heavy non-forkliftable packages. Because these rooms remain open for an extended period of time, the room infrastructure and package integrity will be designed and controlled accordingly. A second set of embedded rails are also provided in these rooms for the gantry crane.

6.5.3.2 Group A Waste Package Transfer and Emplacement

At the underground shaft station, these packages are removed from the rail carts using a light-duty forklift and placed in the staging area or taken directly to the emplacement room. There is capacity for 6 packages at the underground staging area, and its use will be minimized.

To maximize the use of available space and ensure stability of the stacked packages within the rooms, each row of packages will only contain one type of package. A typical stacking arrangement for Group A packages within the emplacement rooms is shown in Figure 6-26.

Stacking of Group A packages at the DGR is very similar to the arrangements currently used in the LLSBs at the WWMF. Packages are stacked a minimum of three high (e.g., LLW overpacks) and do not exceed stacking five high (e.g., drum and bale racks, drum bins).

Figure 6-26: Perspective View of Emplaced Group A (Bin-Type) Waste Packages

6.5.3.3 Group B Waste Package Transfer and Emplacement

The Group B packages on the rail cart are off-loaded from the cage and proceeds via rail directly into the Profile 6 emplacement room. As these rail carts are self-powered using an electrical tether cord, several electrical connections are provided en route to the emplacement room.

At the emplacement room, the package is lifted off the rail cart by the gantry crane and stacked on the floor of the room. Heat exchangers are stacked in a pyramid-shaped pile two high (a row of three on the emplacement room floor and a row of two sitting on top). Support frames are placed on the floor of the emplacement room to ensure that the bottom row cannot roll, providing a stable base for the top row of packages. While stacking the top row of heat exchangers, a remotely releasable sling arrangement is used.

It is not possible to stack shield plug containers due to their mass (28.6 tonne maximum) and the shape of the container. The top of the container is not flat, so smaller boxes cannot be stacked on top. Since there are only a small number of these packages, these stacking limitations will have minimal effect on the overall packing efficiency of the DGR. The gantry crane is used to off-load the rail cart in the emplacement room.

6.5.3.4 Group C Waste Package Transfer and Emplacement

Unshielded Resin Liners

There is capacity for two unshielded resin liners on the rail cart. On arrival at the station, these packages are off-loaded from the cart by a light duty forklift and taken to the emplacement room.

ILW Shields

In future, wastes that are currently being stored in the T-H-E liners may be stored in the proposed new ILW shields. These shields will be small, light-weight and stackable. On arrival at the underground station, a light duty forklift removes these packages from the rail cart and stacks them in the staging area or takes them directly to the emplacement room.

ILW shields are stacked up to three high in a manner that will maximize utilization of the available space in the emplacement room.

Tile Hole Liner Rack

The tile hole liners are smaller cylinders than the resin liners or the ETHs, and therefore there is more flexibility in the method of emplacement. However, because of their slender aspect ratio (ratio of 5:1), they are placed on a rack that is compatible with inter-stacking of multiple racks. These racks are off-loaded from the rail cart at the shaft station and transferred directly by the forklift to the emplacement room.

In the emplacement rooms the racks are stacked three high (two double level racks and one single level rack) for a total stack height of five racks.

6.5.3.5 Group D Waste Package Transfer and Emplacement

Steam Generator Segments

All steam generator segments are transferred in the shaft using the rail cart. The package is off-loaded using a heavy duty forklift and transferred directly to the emplacement room.

ETH Liners

ETH liners are large (4.6 m high), heavy, cylindrical packages. As they are emplaced on their ends, nothing will be stacked on them. Positioning in emplacement rooms is achieved in a similar manner to the resin liner shields that contain two liners each. Tile hole liners are emplaced in the same room as ILW wastes.

6.5.3.6 Shielded Resin Liners

The resin liner shields are only be transferred one per cage trip on a rail cart. They are off-loaded at the shaft station by the heavy duty forklift and transferred directly to the appropriate emplacement room.

Shielded resin liners are not stacked in the emplacement rooms. For design purposes it is assumed that rows of resin liners contain only the same package types. However, the P2 and P3 profile emplacement rooms contain alternating groups of shielded (and unshielded) resin liner types. A rendered image of resin liners in an emplacement room is provided in Figure 6-27.

Retube Waste Containers

The retube waste containers are off-loaded from the rail cart by the heavy duty forklift and transferred by forklift to their emplacement rooms.

Figure 6-27: Perspective View of Emplaced Resin Liner Shield

Retube waste containers are heavy but are stacked on top of each other as they currently are at WWMF. There are two types of containers: the end fitting container and the pressure tube container.

6.6 Waste Package Retrieval

The materials that are placed in the DGR are considered waste and the need for retrieval is not anticipated. However, in the unlikely event that any waste package(s) would need to be retrieved from a room following emplacement, retrieval can be achieved.

A specific plan for retrieving the package(s) would be developed. First, the position of the waste package(s) to be retrieved will be identified using the waste tracking system and the number and type of packages that will have to be moved to access the identified waste package will be determined. Alternative locations, which may be temporary or permanent, for the packages will be identified. They could be relocated to another room, which is partially filled or empty. This new location could be suitable as a permanent location for these packages.

The retrieval concept would be carried out by one of two methods depending on the status of the room.

1. For an open room, packages would be removed using the reverse of the initial emplacement procedure. In most instances this would involve using the same equipment (forklifts, rail carts, etc.) that had been used to originally emplace the waste packages.
2. For a waste-filled room that is isolated by an end wall, the ventilation fan system for that room would need to be re-established and run for adequate time to purge the room of any noxious or other gases and to ventilate the room. The packages would be recovered in the same manner as for an open room. If a gantry crane is required for retrieval, then this equipment would be re-installed after the end wall has been opened.

Although it would be possible to remove waste packages from a room without excessive difficulty, it is expected that the retrieval procedure would be relatively slow to complete to ensure worker safety at all times. If any waste packages were required to be moved to surface, they would be handled in the reverse way to which they were moved underground.

6.7 Hazardous Materials and Waste

6.7.1 Storage

A number of materials that are explosive or flammable in nature are required to construct and operate the DGR Facility. These include diesel fuel and lubricants to operate mobile equipment and explosives for rock excavation (construction phase only). All materials will be stored and handled according to the Workplace Hazardous Materials Information System (WHMIS).

Above ground, lubricants and materials used to clean up spills are stored in the maintenance and storage area adjoining the WPRB. A 'waste' bin is provided at the main shaft area for temporary, but immediate disposal of any materials used for spills clean-up.

During facility operations, it is not expected that explosives will be required. Special projects requiring miscellaneous rock excavation would have specific procedures in place with day-of-use delivery of explosives.

6.7.2 Conventional and Hazardous Waste Management

Conventional and hazardous waste is produced during the operation phase of the DGR Facility. These will consist of consumable materials, namely rags and coveralls used in maintenance and clean-up operations, solids generated from underground sanitary facilities and other miscellaneous wastes. The projected range of conventional and hazardous wastes produced annually by the DGR during the operations phase is shown in Table 6-8.

All consumable waste materials are collected in waste bins or totes located at the main shaft area, both on surface and underground. These bins are transferred to the WWMF at regular intervals for processing, if required, and disposed of in accordance with WWMF practices.

6.7.3 Sewage System

As described above in Section 6.3.10.5, the use of toilet facilities specifically designed for underground use minimizes the amount of human effluent for disposal. The units are forkliftable and transferred to surface for proper treatment.

Grey water from hand washing stations is collected via gravity drain into tanks. This water is pumped out as required to totes to be transferred to surface.

Table 6-8: Projected Range of Annual Output of Waste

Waste Material	Projected Range of Annual Output
Low Level Radioactive Waste (generated during OPG operations)	50 m^3
Oils and grease	400 – 600 L
Batteries	90 – 135 kg
Solvents, Paints, etc.	25 – 50 L
Domestic waste	3,000 – 5,000 kg
Sanitary waste	1,000 – 1,500 kg
U/G sump discharge	40,000 – 60,000 m^3

6.8 Fire and Life Safety

6.8.1 Fire Safety

Fire suppression and detection system are designed in accordance with the requirements of the National Building Code of Canada and the National Fire Code of Canada for surface structures, and Reg. 854 for underground facilities.

The design and operation of the DGR Facility is such that the risk of a fire occurring is minimized. Features of the DGR that lower the risk of fires include:

- Independent third party review of the fire protection design;
- Implementation of the Nuclear Waste Management Division (NWMD) Fire Protection Program (refer to Chapter 10) and fire hazard analysis;
- Minimized use of combustible materials. Specifically, the shafts are steel construction and timber construction underground is limited to shaft guides in the ventilation shaft;
- Fire resistant cabling is used;
- Waste materials contained within non-combustible metal or concrete containers complete with lids;
- There is no diesel fuel storage on surface during operations;

- Underground, the amount of diesel fuel stored is minimal, and the underground diesel fuel bay is a dedicated room with appropriate separations and fire protection systems;

- Diesel fuel is transported underground in approved containers following appropriate procedures with no other packages, materials or personnel allowed in the conveyance while fuel is being transported; and

- No explosives will be stored underground during the operational phase of the DGR.

6.8.2 Fire Detection and Alarming

6.8.2.1 Surface Facilities

For the surface facilities, fire detection and alarming will be in accordance with the National Building Code of Canada and the National Fire Code of Canada. Fire detection is achieved using smoke detectors and manual pull stations in all surface buildings and activation/flow alarms on all automatic fire suppression systems. Carbon monoxide detectors are installed in the WPRB, main shaft headframe and emergency diesel generator building because of the use of diesel powered equipment in these areas. Audible and visual alarm signals, similar to those currently in use at the WWMF, are activated when alarm levels are reached. Alarm signals are routed to a fire panel that will transmit all alarm signals to the DGR main control room on surface. Alarm signals will also be transmitted to an underground monitoring terminal and the main control room at the WWMF for monitoring during DGR off-shift periods. It is possible to identify which sensor has detected an alarm condition and its location.

6.8.2.2 Underground

Underground, fire detection will be achieved using smoke and carbon monoxide detectors at key points in the facility. The points include:

- All underground infrastructure rooms situated in the shaft and services area;

- The exhaust regulators exiting each emplacement room (whether empty and awaiting start of emplacement operations, during emplacement operations, or full of waste packages);

- In the DGR air intake plenum at the exit from the heater house; and

- At the discharge of the main exhaust fans at entry to the upcast ventilation shaft.

This provides levels of redundancy so that any failure of one instrument will not enable a fire to remain undetected. All regulator monitors are located such that they are accessible from the exhaust ventilation tunnel for ease of maintenance. All instruments that are behind a closure wall will be disconnected prior to installation of the closure wall.

All underground instrumentation signals are displayed locally and transmitted to a fire panel underground. The fire panel outputs all status and alarm signals to the DGR main control room, an underground monitoring terminal and WWMF via the instrumentation network. It is possible to identify which sensor has detected an alarm condition and its location in the repository. The underground monitoring terminal displays both surface and underground alarms signals. If alarm levels are reached, audible and visual alarms, similar to those currently used at the WWMF, are automatically activated.

However, in underground mining-type environments, audible alarms may not be fully effective on their own due to the nature of the environment. The "stench gas" system used in Ontario mines is well proven. A stench gas is a foul-smelling but safe gas that is injected into the downcast ventilation air stream, which quickly and effectively reaches workers in all parts of the facility. Therefore, once a fire condition has been detected, the stench gas releases into the intake plenum at the main shaft.

In addition to the visual and audible alarms and the stench gas system, all workers underground carry personal radios that communicate over the leaky feeder communication system. In the event of an emergency, a call will be put out over the radio system from the surface main control room to provide an additional means of alerting personnel of an emergency.

6.8.3 Fire Suppression

6.8.3.1 Surface Facilities

Fire suppression for surface facilities is achieved through the use of fire extinguishers, water sprinkler suppression systems, and fire hose systems.

All surface facilities are equipped with hand-held fire extinguishers that are mounted on clearly demarcated boards. These fire extinguishers are located at regular intervals throughout the buildings and are selected to suit the potential fire hazards for the location.

Fire suppression systems are installed in buildings as listed in Table 6-9. At this stage, the reference fire suppression systems are for the general coverage required for buildings. During detailed design, each building will be evaluated and zoned such that areas within each building may have different fire suppression requirements due to the potential fire hazard or to protect equipment and personnel.

A fire water main is installed around the surface facility complete with fire hydrants. The fire water main is connected to the existing Bruce Power fire water system. Water for all water-based fire suppression systems will be supplied by the fire water main.

Fire fighting services to the DGR Facility is provided by the Bruce Power ERT, similar to the practice at the WWMF.

Table 6-9: Surface Buildings Reference Fire Suppression Systems

Building	Reference Fire Suppression
Ventilation shaft headframe	Sprinkler system
Ventilation shaft hoist house	Hoses / fire extinguishers
Ventilation shaft exhaust fans	Fire extinguishers
Main shaft headframe (including main & auxiliary Koepe friction hoist rooms)	Sprinkler system
Electrical room	Automatic dry chemical system
Amenities building	Sprinkler system
Offices	Sprinkler system
Main control room	INERGEN system (INERGEN is a clean-agent fire suppression system that is people-safe and environmentally friendly)
WPRB	Sprinkler system
Heater house & intake fans	Fire extinguishers
Air compressor plant	Hoses / fire extinguishers
Electrical substation	Fire extinguishers
Emergency diesel generator	On board 'Ansul' system for diesel engines Automatic dry chemical system for building

6.8.3.2 Underground

The principle for underground fire suppression is that there are no water sprinklers or fire hose systems installed near the emplacement rooms as their use could create a large volume of contaminated water that would have to be collected and treated before release from the DGR Facility. In addition, the use of water for fire suppression would introduce high levels of humidity, which could negatively affect the long-term integrity of waste packages, structures and ground support. A dry standpipe and hose will be available at the main shaft station, if required. Underground, fire suppression is achieved by methods that are described below.

Hand-held fire extinguishers are mounted on clearly demarcated boards in or close to all rooms in the shaft and services area. At any workplace that is not a fixed location (e.g., maintenance shop) workers must have a fire extinguisher available and close at hand.

A foam-based suppression system is provided for the maintenance shop and diesel fuel bay. Because the maintenance shop and diesel fuel bay are adjacent to each other, a single fixed pipe foam system is installed to provide coverage to both rooms. This meets the requirements of a fire suppression system for flammable liquid storage areas and service garages as stipulated in Section 28(2) of Reg. 854.

To fight a fire in a waste-filled emplacement room, fire doors or temporary barrier walls will be placed across the entire cross-section of the access and return air drifts to isolate the oxygen supply to the fire so that it will burn out (note, personnel are not present downstream of the barrier wall). The regulator can be closed off remotely from the DGR main control room once the mine rescue team have assessed the situation, accounted for all personnel underground and indicated it is safe to do so. Installing fire suppression equipment in the emplacement rooms would be ineffective due to the size of the rooms and the storage arrangement of packages. In addition, it would not be practical to maintain fire suppression equipment installed inside the emplacement rooms once it is filled with waste packages and an end wall is erected.

A portable, skid mounted dry chemical system is provided to aid mine rescue teams in fighting fires underground. The dry chemical system is stored underground and moved into place using a forklift when required.

All diesel equipment are equipped with automatic, foam-based fire suppression systems and are triggered on detection of any fire on the vehicle.

In addition to fire suppression systems, fire doors are installed at the main shaft station, the maintenance shop and the diesel fuel bay in compliance with Section 39 of Reg. 854. A normally open, manually operated fire door is installed on each side of the main shaft station. These doors are closed by the mine rescue team if required. The maintenance shop and diesel fuel bay each have a fire rated, overhead, roll-up door that automatically closes when a fire is detected. Each of these roll-up doors have a fire rated personnel door installed adjacent to it to ensure personnel are not trapped in the room if the overhead door is closed.

6.8.4 Emergency Ventilation Controls

Ventilation fans and regulators underground are controlled remotely from surface at the main control room or manually at the ventilation shaft fans and emplacement rooms. For safety reasons, no alteration or disruption to the ventilation system will occur until all underground workers are accounted for and the mine rescue team has assessed the situation. If, after reviewing the situation with appropriate personnel, it is decided that adjusting the ventilation system will aid in the rescue of personnel or controlling the fire, underground fans and regulators will be operated from surface. However, standard procedure is to leave the ventilation system untouched in an emergency situation.

6.8.5 Refuge Stations

The first line of protection for underground workers is the refuge station. There are two refuge stations in the shaft and services area; one as part of the underground lunch room and offices and the other close to the Panel 1 and Panel 2 access tunnels. There are always two escape routes for any worker to reach a refuge station in the shaft and services area.

Although the flow-through ventilation system provides egress from both sides of the emplacement rooms, once emplacement activities begin, the rooms become dead-end from an egress perspective. As such, portable refuge stations are positioned at the far end of the panel access tunnels providing workers with routes to two refuge stations from the panels (see Figure 6-28). As the emplacement panel is filled, the portable refuge stations retreat with the emplacement activities, providing the shortest route from active emplacement rooms.

Each refuge station is sized to accommodate the maximum expected number of employees and visitors underground and is designed to be compliant with the requirements of Section 26 of Reg. 854. Fire clay is stored in each refuge station and will be used to seal doors from the inside to prevent the ingress of smoke and gases during a fire. Refuge stations are equipped with a communication line to surface, a compressed air supply (with a secondary built-in scrubber air exchange unit) and a supply of bottled potable water.

Refuge stations at the shaft and services area are designed to accommodate up to 25 persons, allowing for about 15 underground workers during construction or 10 during operation, plus any potential visitors.

6.9 Emergency Response

Emergency response could be required following a fire, rock fall; or radiological contamination.

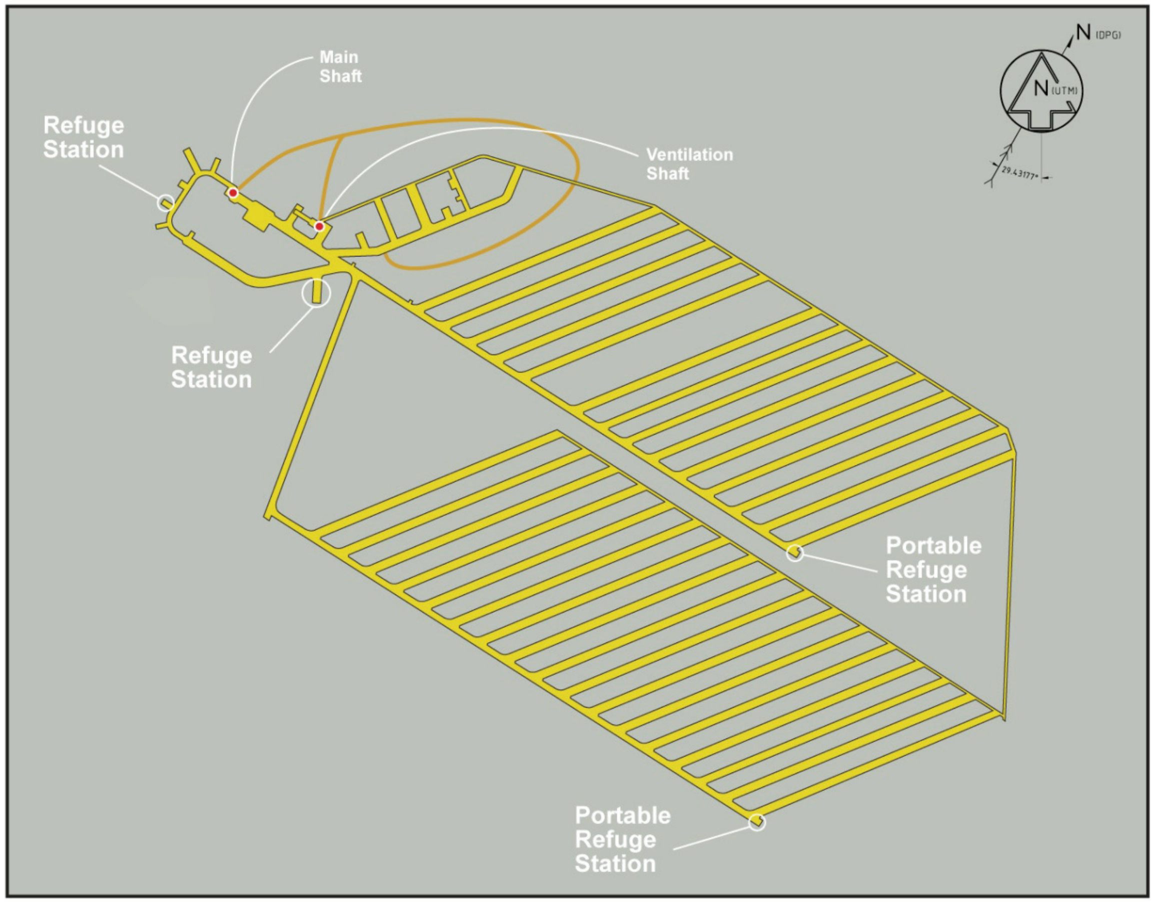

Figure 6-28: Location of Refuge Stations

The DGR requires an emergency response mine rescue team (MRT) to respond to fire and rock fall events. The MRT is provided with special training from the Ontario Mine Rescue division of MASHA, with mandatory refresher courses six times per year. The DGR MRT will have its own equipment so that it can immediately respond to a fire or other non-radiological emergency. MASHA requires a back-up team to be on-site before the first team is sent underground, and a third team must be on-site before the second team can go underground. Although the DGR will provide two teams, reliance on the neighbouring mines in the region will be necessary. This network is the basis upon which mine rescue works with any team from any mine in Ontario being available if required.

DGR personnel will be expected to respond to radiological contamination events as described in Section 6.9.3.

Management of emergency response is controlled from the main control room.

6.9.1 Fire

The following events have been grouped together under a fire event as they will have the same emergency response procedure:

- Fire;
- Explosion;
- CO alarm; and
- Explosive gas monitor alarm.

6.9.1.1 Surface

In the event of a fire alarm on surface, all personnel will evacuate the buildings to the nearest assembly area outside. The ERT will be dispatched to the site to evaluate the situation and fight the fire if required.

During a fire event on surface, personnel underground will be instructed to report to the refuge stations to ensure all personnel are accounted.

6.9.1.2 Underground

Immediately on initiation of a fire alarm (or carbon monoxide or explosive gas alarm), the stench gas system is deployed and all workers will report to a refuge station.

The main control room operator will call out the MRT and the nearest off-site support unit, with whom the DGR is affiliated. No workers leave the refuge stations until the MRT has determined that it is safe to do so, either by extinguishing a fire or by identifying a safe route to whichever shaft is in the fresh air supply and uncontaminated by any combustion products.

6.9.2 Rockfall

In the unlikely event of underground workers trapped by a rock fall or other extraordinary event (e.g., any shaft conveyance event that renders the personnel cage inoperable), management of the facility will coordinate the response and utilize the MRT to assess the situation and recommend a recovery strategy depending on the circumstances.

6.9.3 Radiological Contamination Event

In the event of a contamination event, clean-up of such an event will follow OPG's Radiation Protection Program. At surface, clean-up would be consistent with the practices currently employed at the WWMF. Underground, personnel would report to the refuge stations and the appropriate clean-up conducted according to the requirements of the Radiation Protection Program.

6.10 Zoning

6.10.1 Radiological Control

OPG's Radiation Protection Requirements (OPG01b) comply with the Nuclear Safety and Control Act (NSCA97) and its associated regulations. OPG's Radiation Protection Requirements (OPG01b) apply the intent of the principles and recommendations established by the ICRP. They also take into account the knowledge gained through OPG's long experience in designing, constructing and operating a nuclear-electric generation program.

A key practice in maintaining control of radiation exposure and contamination is through the use of zoning. The following excerpt from OPG's Radiation Protection Requirements defines the zones that will be applied to the DGR Facility:

> *5.1.3 Zone 1*
>
> *Zone 1 is a clean area which is not a radiological zone and may be considered the equivalent of a normal public access area.*
>
> *5.1.3.1 Zone 1 shall not contain radioactive sources other than those found in normal*

industrial establishments, or those specifically approved for use in applications such as training and demonstrations.

5.1.3.2 Fixed contamination levels in Zone 1 shall not exceed the established contamination limit for Zone 1 surfaces [4.8.2]. No detectable loose contamination shall be permitted in Zone 1 [4.8.2].

5.1.3.3 Zone 1 shall have a very low probability of cross-contamination from adjacent areas and shall have a low general radiation background, not exceeding the established limit [4.7.1.1].

5.1.4 Zone 2

Zone 2 is a radiological zone that is normally free of contamination but is subject to infrequent cross-contamination due to the movement of personnel and equipment from contaminated areas.

5.1.4.1 Zone 2 is normally free of radioactive sources other than those found in normal industrial establishments, or those specifically approved.

5.1.4.2 Zone 2 shall have a low general radiation background.

5.1.4.3 All materials being moved from Zone 3 to Zone 2 should be monitored[2].

5.1.4.4 Where appropriate, local containment systems shall be used when radioactive systems in Zone 2 are opened or leaking.

5.1.4.5 If local containment systems are not used, a rubber area shall be established when radioactive systems in Zone 2 are opened or leaking, and it shall be removed promptly when work on the system is complete.

All areas of the DGR associated with the handling of radioactive waste are designated as Zone 2. These include the crossing from WWMF to the WPRB, the WPRB, shafts and the underground areas. Office and amenities areas at the DGR are designated Zone 1. Figure 6-29 shows the different zones for the DGR surface facilities. A Zone 1 and Zone 2 boundary is located within the amenities area for the movement and tracking of personnel.

As all areas underground (i.e., below the shaft collars) are Zone 2, access to the lunch room underground will require the use of the whole body and small article monitors.

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[2] Paragraph 5.1.4.3 does not apply to the DGR Facility.

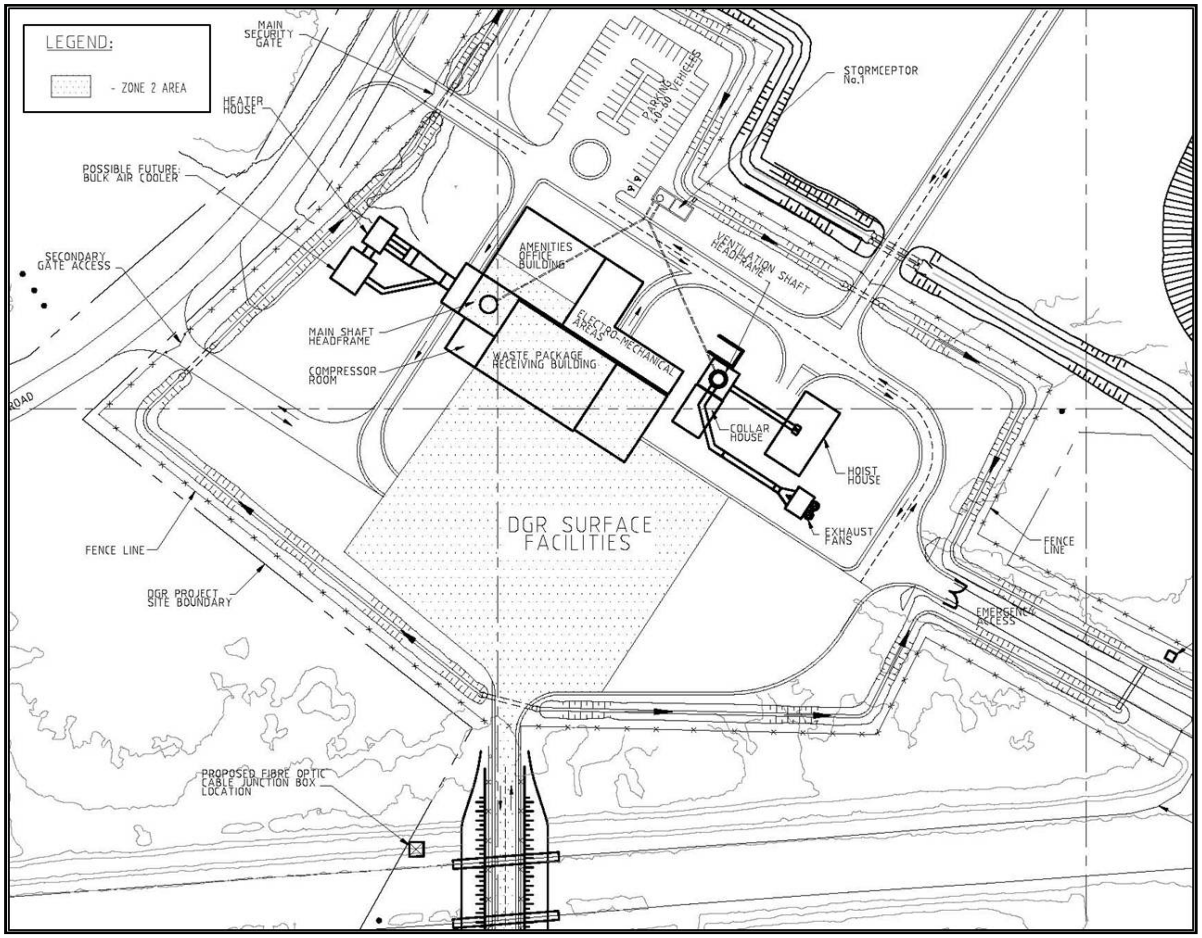

Figure 6-29: Surface Radiological Zones

6.10.2 Decontamination

Routine decontamination of personnel and equipment is not anticipated since one of the key DGR waste acceptance criteria requirements is that there shall be "no loose contamination" on waste packages. In the event that decontamination underground is required, the following facilities will be provided:

- The maintenance shop will contain materials and equipment that can be used to decontaminate forklifts or other mobile equipment that are discovered to be contaminated underground;
- Materials will be provided next to the whole body monitor underground that will be used to contain contamination so that personnel may be transported to surface to the decontamination facility;
- The refuge stations will be equipped with radiation protection equipment for monitoring and decontamination of staff in the event of contamination; and
- Detailed procedures for decontamination underground will be developed.

6.11 Radiation Monitoring

Radiation monitoring will be provided at the DGR Facility to ensure radiation levels in air and water are consistent with regulatory limits. Routine air monitoring would take place at strategic locations underground to confirm air concentrations of radionuclides are acceptable at various work locations (e.g., shaft service area, access tunnels, active emplacement rooms). Routine air monitoring would be performed at key exhaust air points (e.g., at the regulators in waste-filled rooms and at the ventilation shaft exhaust location at surface).

Water sampling and testing would take place at strategic locations underground and at surface to confirm water concentration of radioactivity is acceptable. The underground waste water, which would be delivered to surface in totes, would be sampled and analyzed for radioactivity, as necessary, to identify proper treatment and disposal.

6.12 Underground Air Quality Monitoring

Air quality underground will be monitored to ensure that the health and safety of personnel within the repository is not compromised. The monitoring system will ensure:

- Levels of noxious and explosive gases do not exceed regulatory limits (Section 294, Reg. 854); and
- Airflows remain adequate for the equipment or activity in active work areas.

Airflow, CO and NO_2 measurements are taken at the ventilation shaft. Explosive gas monitors will also be installed to monitor a range of potential gases, including methane and hydrogen. Instrumentation measuring airflow, temperature, relative humidity, etc. will be installed at the main shaft. Emplacement room exhaust regulators will be equipped with combustible gas monitors to monitor a range of potential gases, including methane and hydrogen. All measurements will be monitored remotely on surface at the main control room and will also be available to be monitored underground.

6.13 Access Tunnel Closure Walls

After a group of rooms have been filled with waste packages and following a period of monitoring, closure walls will be constructed in the access and exhaust ventilation tunnels to fully isolate this group of rooms. The underground space behind the closure walls will not be ventilated and all services terminated. These closure walls are designed to limit release of tritiated air, natural and waste-generated methane, and other off-gases from waste packages (e.g., H_2 and CO_2), as well as potentially contaminated water. In the remote event that explosive gases build up behind the closure wall and an explosion occurs, the air blast from the explosion will be contained by the closure wall. The conceptual design for the closure wall is shown in Figure 6-30.

There will be a series of closure walls constructed to coincide with the emplacement activities to isolate groups of waste. The first walls could be erected approximately five years following the filling of Panel 2 with waste to allow for monitoring. Given that the Panel 2 will be filled within a five-to-ten year period, the first closure walls will be erected approximately 15 years after start of operations to isolate all rooms in Panel 2. The next series of walls could be erected fifteen years later and would isolate nine rooms in Panel 1, which are furthest from the shafts. The final walls would be erected at end of repository operations to isolate the last five rooms in Panel 1, which are located closest to the shafts.

The closure wall shown in Figure 6-30 would consist of mass concrete within the access tunnel. Grout holes would be drilled through the concrete into the surrounding host rock for provision of high pressure consolidation / contact grouting. This closure wall resists pressure through friction between the concrete plug and the rough surface of the access tunnel along the entire length of the seal.

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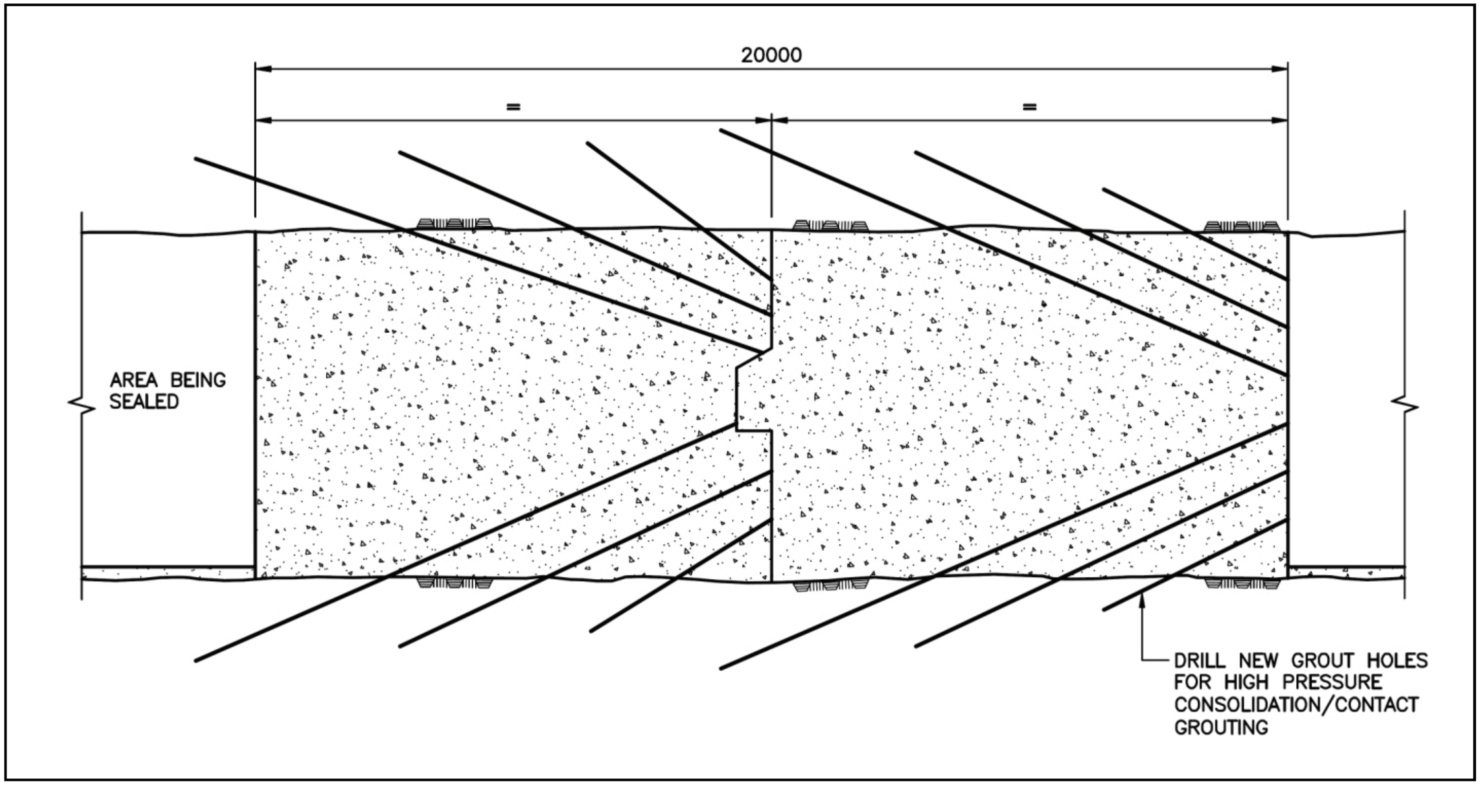

Figure 6-30: Conceptual Design for Access Tunnel Closure Wall

6.14 Shaft Seal

The arrangement of the shaft sealing system, selected components, their relative location and construction for the shafts is described in Chapter 13, Preliminary Decommissioning Planning.

7. PRECLOSURE SAFETY ASSESSMENT

This chapter provides a summary of the DGR radiological and non-radiological safety assessment for humans during the operational period (normal and accident scenarios).

The scope extends from arrival of the waste packages at the DGR site boundary to emplacement underground, and from the start of operations until decommissioning begins. Conventional safety during site preparation and construction is covered in Chapter 9 (Section 9.4.9). The Preliminary Conventional Safety Assessment report provides conventional safety assessment information for site preparation, construction and operations (NWMO11ac). The postclosure safety assessment is covered in Chapter 8.

7.1 Assessment Context and Criteria

7.1.1 Context

This preliminary assessment is based on the reference waste inventory described in Chapter 5, and the facility design of the DGR described in Chapter 6.

The WWMF operating experience over the past 40 years is also an important context for this DGR operational safety assessment. Many of the waste packages to be emplaced within the DGR are currently handled, transferred and stored in the WWMF. WWMF operation has demonstrated that these waste packages can be safely handled and stored. Worker and public dose rates have been consistently below regulatory limits. The WWMF routinely operates contamination free.

7.1.2 Criteria

The radiological criteria are the same criteria as those used at the WWMF, and are provided in Section 7.1.2.1 for normal operating conditions and for accidents. The assumed preliminary DRL for the DGR are also given in Section 7.1.2.1. The non-radiological criteria are given in Section 7.1.2.2.

7.1.2.1 Radiological Protection

Radiological Protection of Public and the Workers – Normal Operating Conditions

During operation, radiation protection criteria currently applied to other nuclear facilities, specifically the WWMF, are applicable to the DGR as well. General radiological protection requirements are in accordance with the Radiation Protection Regulations (SOR/2000-203) promulgated under the Nuclear Safety and Control Act (NSCA97).

Doses resulting from the DGR operation will be within the regulatory dose limits and will be kept ALARA. The CNSC regulatory dose limits for the public and the Nuclear Energy Workers (NEWs) are shown in Table 7-1.

Table 7-1: CNSC Effective Dose Limits

Person	Period	Effective Dose (mSv)
NEW, including a pregnant NEW	One-year dosimetry period Five-year dosimetry period	50 100
Pregnant NEW	Balance of the pregnancy (after the licensee is informed of the pregnancy)	4
A person who is not a NEW	One calendar year	1

The dose rate targets for the DGR preclosure period, derived from Table 7-1 for a member of the general public or a non-NEW, are as follows:

- ≤ 0.5 μSv/hr at the DGR fence, based on the CNSC annual dose limit of 1 mSv for a member of the public, over a maximum of 2000 hours per year occupancy for non-NEWs; and
- ≤ 10 μSv/year at the Bruce nuclear site boundary, based on year round occupancy – this dose rate target is 1% of the CNSC annual dose limit of 1 mSv for a member of the public.

These dose rate targets are consistent with those used at the WWMF (OPG06a). In addition, the dose rates listed below are adopted for this assessment.

- Worker dose rate target (exposure control level) of ≤10 mSv/year (OPG06c).
- Derived Air Concentration (DAC) limits as listed in Table 7-2, also referred to as Maximum Permissible Concentrations in air, for workers for radionuclides relevant to normal operations (OPG01b). These correspond to a dose of 20 mSv/year for 2000 hour exposure.
- Maximum dose rate outside emplacement rooms or outside WPRB of 25 μSv/hr. This is adopted from OPG Radiation Protection Requirement (Section 4.7.2.1, OPG01b) for external dose rate limit outside a long-term storage structure.

- A maximum dose rate of 2 mSv/hr at contact from waste package exterior surface was assumed in this assessment. Most packages will be at or less than this dose rate, after decay and package shielding.

Table 7-2: DACs

Radionuclide	$\mu Ci/m^3$ [a]	Bq/m^3
HTO [b]	10	3.7E+05
$^{14}CO_2$	20	7.4E+05
Particulate β/γ [c]	0.003	1.2E+02
Particulate α [c]	10^{-5}	0.33

Notes:
a. OPG Radiation Protection Requirements (OPG01b).
b. DAC for tritiated water includes an additional 50% intake through skin.
c. Assumes particulate contains unidentified long-lived β/γ or α emitter.

Preliminary DRLs

DRLs are the limits set for radionuclide releases so that releases occurring from a nuclear facility will not result in dose to individual members of the public exceeding the dose limits set by the CNSC. These limits are derived from statutory public dose limits and radionuclide transport pathway models.

Since the DGR is adjacent to the WWMF and has similar waste characteristics and similar location of release sources, for the purpose of this PSR, the preliminary DRLs for the DGR are assumed to be the same as those for the WWMF (OPG03b). As such, the preliminary DRLs presented below in Table 7-3 are those established for the WWMF. These DRLs have been established as though each radionuclide was the only one being emitted.

Since the Bruce nuclear site is a multi-facility site, the total exposure of individual members of the public due to releases from various source facilities must be considered. Therefore, the releases from the DGR must be a small fraction of the DRL for each radionuclide group.

Table 7-3: Preliminary DRLs for the DGR

Radionuclide or Radioactive Release Group		DRL (Bq/year)	Operational DRL (Bq/period)
Air	Tritium (H-3) (HTO) [a]	1.4E+17	2.7E+15 Bq/week
Air	Carbon-14 (C-14)	4.6E+15	8.9E+13 Bq/week
Water	H-3 (HTO) [a]	2.1E+15	1.7E+14 Bq/month
Water	C-14	1.7E+12	1.4E+11 Bq/month
Water	Gross beta, gamma [b]	1.2E+11	9.6E+09 Bq/month

Notes:
a. Assume all H-3 being in the form of HTO.
b. Cs-137 was identified as the limiting radionuclide (Table 30 of OPG03b).

Radiological Protection of Public and the Workers – Accidents

The radiological doses from radionuclide releases and direct radiation, either to members of the public at the Bruce nuclear site boundary or to workers, following an event involving abnormal operating conditions or a credible accident during the entire DGR operational period, must not exceed 50 mSv for the workers and 1 mSv for the public.

The prevention, mitigation and accommodation of abnormal and credible accident conditions are a consideration in the facility design and planned operations.

Radiological Protection of Non-Human Biota – Operational Period

Aquatic and terrestrial biotas receive radiation doses from exposure to radioactivity in the atmosphere, surface water, soil, and groundwater. Criteria for assessing the potential impact of the DGR project on non-human biota are presented in the Radiation and Radioactivity Technical Support Document (Section 8.1.1.2, NWMO11e). Calculated radiation doses to the biota during the DGR operational period are also given in the Technical Support Document (Section 8.3.3, NWMO11e). In summary, doses to aquatic and terrestrial biota were calculated to be at least two orders of magnitude below the criteria.

Predicted radiological effects on non-human biota are included in the Malfunctions, Accidents and Malevolent Acts Technical Support Document (NWMO11ad). For the operations phase bounding scenarios, all doses are lower than the applicable standard.

7.1.2.2 Non-Radiological Protection

Radioactivity is the primary concern related to the L&ILW placed in the repository. However, the waste packages also contain potentially hazardous elements such as lead that are present in the waste in appreciable amounts (e.g., as waste shielding material), as well as other elements, such as cadmium, that are present in small amounts (see Chapter 5). As the wastes are solid and are contained in lidded packages, these elements do not present a hazard during normal operation. However, the assessment of potential impacts of these non-radiological hazardous elements in the wastes under accident conditions is presented in this report.

Non-radiological conventional safety aspects of DGR operation are considered in the Preliminary Conventional Safety Assessment report (NWMO11ac) and are not discussed here.

The scope of this section is, therefore, limited to presenting the specific criteria used to assess protection of the public and the workers from potentially hazardous non-radiological components in the waste.

Non-Radiological Protection of Public - Accidents

The only credible exposure pathway to public is via inhalation of airborne contaminants. For inhalation pathway, the maximum potential public exposure to non-radiological species due to various accident scenarios conservatively assumes the public to be located at the closest point on the Bruce nuclear site boundary, with the plume directed towards this location, with no mitigation measures for this period.

The public exposure criteria adopted for non-radiological species under accident scenarios are the U.S. Department of Energy (DOE) Protective Action Criteria (PACs). The PACs are developed and maintained by the U.S. DOE / National Nuclear Security Administration. The PACs are a comprehensive set of short-term public exposure guidelines based on U.S. Environmental Protection Agency Acute Exposure Guideline Levels or American Industrial Hygiene Association one-hour Emergency Response Planning Guidelines where available, and on internal Temporary Emergency Exposure Limits otherwise.

PAC 1 values are the maximum concentration in air below which it is believed nearly all individuals could be exposed without experiencing other than mild transient adverse health effects or perceiving a clearly defined, objectionable odour. PAC 1 values are adopted as public exposure criteria for DGR accidents in this report, taking into account that the accidents are unlikely, and that the analysis is conservative (e.g., public was assumed to be at the nearest Bruce nuclear site boundary). Table 7-4 provides the PAC 1 values for the relevant non-radiological species (USDOE10).

Table 7-4: Public Exposure Screening Criteria for Short-Term Inhalation of Non-Radiological Species in Wastes

Non-Radiological Species	PAC 1 Criteria [a] (µg/m^3)
Asbestos	50
Antimony	500
Arsenic	300
Barium	1,220
Beryllium	3.5
Cadmium	30
Chromium [b]	25
Cobalt	60
Copper	220
Lead	150
Manganese	3,000
Mercury [c]	250
Nickel	600
Selenium	600
Strontium	125,000
Uranium [d]	600
Zinc	3,000
Zirconium	10,000
Dioxins / Furans [e]	1.5

Notes:

a. PAC 1 criteria based on lowest of element or oxide form; expressed in terms of element content.
b. based on 20% CrO_3 (Cr-VI); PAC-1 values are 5 as Cr-VI; 1,000 as Cr; 10,000 as Cr_2O_3 (Cr-III).
c. as mercury vapour; Hg_2O is not stable.
d. as U, UO_2, U_3O_8.
e. as 2,3,7,8-tetrachlorodibenzo-p-dioxin.

The PAC 1 values are the lower of the elemental or oxide values, except where the oxide listed is notably unstable, since the DGR wastes are non-reactive and have generally been in contact with air or water for an extended period. The chemical form is particularly relevant for Cr. The dominant source of Cr in the wastes is as an alloying element in stainless steel, so it is present primarily as Cr (i.e., Cr-0) or Cr_2O_3 (i.e., Cr-III), and the PAC 1 value for Cr is used as default (lower than Cr_2O_3). The possible presence or formation of CrO_3 (i.e., Cr-VI) is considered by assuming 20% of the Cr is released as Cr-VI, based on the fraction of Cr-VI typically seen in stainless steel welding fumes and on the fraction in ashes from Cr-0 or Cr-III sources (e.g., SERAGELDIN09).

Non-Radiological Protection of Workers - Accidents

For potential impact from short-term accident exposure, the worker exposure criteria to non-radiological species are based on concentrations that are Immediately Dangerous to Life and Health (IDLH), provided by the U.S. National Institute for Occupational Safety and Health (NIOSH) (NIOSH05). The IDLH value is an atmospheric concentration that poses an immediate threat to life or would cause irreversible or delayed adverse health effects or would interfere with an individual's ability to escape from a dangerous atmosphere.

Table 7-5 shows the IDLH values for the non-radiological species of interest. Currently, there are no IDLH values for asbestos, strontium, or dioxins and furan, so the impact of exposure to these species was not assessed for workers.

IDLH values are generally based on 30-minute exposure. This is longer than the time that workers would need to leave an area, reach a refuge station underground, or acquire protective equipment.

Table 7-5: Worker Exposure Criteria for Short-Term Inhalation of Non-Radiological Species in Wastes

Non-Radiological Species	IDLH (µg/m^3)
Asbestos	N/D
Antimony	50,000
Arsenic	5,000
Barium	50,000
Beryllium	4,000
Cadmium	9,000
Chromium	25,000
Cobalt	20,000
Copper	100,000
Lead	100,000
Manganese	500,000
Mercury	10,000
Nickel	10,000
Selenium	1,000
Strontium	N/D
Uranium	10,000
Zinc	500,000
Zirconium	25,000
Dioxin/Furan	N/D

Note:
N/D Criteria for workers have not been developed by NIOSH.

Non-Radiological Protection of Non-Human Biota – Operational Period

For non-human biota, impacts during the DGR operational period are considered in the Terrestrial and Aquatic Environment Technical Support Documents (Section 8 in both NWMO11d and NWMO11g).

The Malfunctions, Accidents and Malevolent Acts Technical Support Document considers the potential effects to aquatic and terrestrial biota in the event of conventional accidents such as spills (Sections 5.4.1 of NWMO11ad).

7.2 DGR Waste Packages

The repository will contain L&ILW from operation and refurbishment of OPG-owned or operated nuclear reactors. The wastes are or will be emplaced in a variety of steel and concrete waste containers and overpacks. The waste and container amounts and characteristics are described in Chapter 5. There are approximately 20 waste categories, and a total of about 53,000 containers. These containers are approximately 80% LLW and 20% ILW.

About 70% of these containers will already be in storage at the WWMF at the time of DGR start-up, assumed to be 2018. Most of these will be transferred into the DGR over the first 5-10 years of operation. The remainder of the containers will be generated by the stations during future operation, and transferred over the balance of DGR operation. The container throughput will, therefore, be significantly reduced during the subsequent operation. Wastes already stored at the WWMF will have decreased in radioactivity due to decay during the storage interval; however, the inventories of future arisings may only have decayed within the normal station time frame for shipment of wastes off-site.

Although the specific waste package handling schedule has not been defined, for purposes of this safety assessment, the reference schedule listed in Table 7-6 is adopted. The basic rationale for this schedule is:

- Most stored LLW and ILW are transferred from WWMF into DGR in the first 5 years of DGR operation;
- Most heavy, more complex and higher dose rate waste packages are transferred in the next 5-15 years; and
- Future waste arisings transferred directly from stations to the WWMF and then to the DGR.

This results in essentially three phases of operation. At the end of each of these phases, a section of the DGR would have been filled and could be isolated with closure walls. The decision of when to install the closure walls will be made at the time, and may involve a period of several years of monitoring first. Timing of closure wall construction is discussed in Section 6.13.

It is emphasized that this schedule is adopted for the purpose of quantitative safety assessment. It is a relatively aggressive schedule initially, as the actual initial transfer rates are likely to be lower.

Table 7-6: Assumed Emplacement Schedule for Waste Package Handling at DGR

Room Profile [a]	Waste Packages	Number of Filled Emplacement Rooms		
		2018-2022 (5 years)	2023-2037 (15 years)	2038-2052 (15 years)
P1	Bin-type LLW	12	3	1
P2	Non-processible other (such as ETHs) ILW resins	1	0	0
P3	ILW resins	2	3	0
P4	ILW shields Retube waste	1	2	0
P5	Steam generator segments	0	1	1
P6	Shield plug container	0	0	2
Unassigned		1	0	1

Note:
a. Emplacement room design; see Section 6.5.3.1 and Table 6-7.

7.3 DGR Waste Inventory

The radionuclide inventory in the DGR will increase each year due to the emplacement of more packages, but it will decrease due to decay. Filled rooms will remain ventilated and accessible until a decision is made to install closure walls and isolate a set of rooms or a panel. For the nominal schedule given in Table 7-6, closure walls could be constructed during or after 2023, 2038 and 2052, when the respective areas of the DGR have been filled.

For safety assessment purposes, the projected DGR inventory with the reference schedule is illustrated in Table 7-7 at the end of 2018, 2023, 2038 and 2052. Note that the waste inventory of emplacement rooms isolated by closure walls (i.e., unventilated rooms) was not included. For clarity, the inventory prior to the closure walls is referred as "ventilated inventory". The inventory estimates were based on the Reference

L&ILW Inventory report (OPG10a), and on the assumed emplacement schedule. The actual emplacement rate may be different. The inventories in Table 7-7 have been decay-corrected.

The H-3 and C-14 ventilated inventory in the emplacement rooms is highest at the end of the initial emplacement period (2023), when most of the L&ILW has been emplaced, these rooms are still ventilated, and there has been the least amount of decay.

Table 7-7: Ventilated H-3 and C-14 Inventory in Repository in Reference Schedule

Inventory	2018 (Bq)	2023 (Bq)	2038 (Bq)	2052 (Bq)
H-3 LLW Inventory	5.2E+14	3.2E+15	4.7E+14	9.2E+13
H-3 ILW Inventory	0.0E+00	4.6E+14	2.2E+14	1.2E+13
H-3 Total Inventory	**5.2E+14**	**3.6E+15**	**6.9E+14**	**1.0E+14**
C-14 LLW Inventory	1.6E+11	1.3E+12	2.9E+11	9.1E+10
C-14 ILW Inventory	0.0E+00	3.6E+15	2.8E+15	6.5E+13
C-14 Total Inventory	**1.6E+11**	**3.6E+15**	**2.8E+15**	**6.5E+13**
Notes: Based on assumed emplacement schedule starting in the beginning of 2018, decay-corrected. Closure walls isolate previously emplaced inventory after 2023, 2038 and 2052.				

The inventories in the WPRB will generally be small, because the WPRB is not intended for storage as packages will be transferred directly to the main shaft cage and then down to the repository. However, there will be capacity for some temporary staging of waste packages if necessary. A bounding case is considered where the WPRB holds 24 LLW packages and 2 ILW resin packages (which each hold 2 ILW resin liners) and that these packages contain high H-3 and C-14 inventories. The corresponding maximum WPRB inventory is listed in Table 7-8.

Table 7-8: Maximum H-3 and C-14 Inventory in the WPRB

WPRB	H-3 Inventory (Bq)	C-14 Inventory (Bq)
LLW in staging area	1.8E+13	1.6E+10
ILW in staging area	4.8E+12	3.2E+13
Note: Assumes 24 LLW and 2 ILW packages with high H-3 and C-14 inventories present.		

7.4 Radiological Safety during Normal Operations

7.4.1 Hazard Identification- Exposure Pathways

Normal operating conditions include routine operations as well as maintenance. The public and workers could be exposed during normal operations via three main paths:

- Air or water emissions from the DGR - members of the public;
- Air emissions from the DGR - workers; and
- External radiation- workers and members of the public.

Each of these potential exposure routes and associated impacts are assessed below.

7.4.2 Radiological Assessment of Air and Water Emissions from DGR on Public

7.4.2.1 Source Terms - Air and Water Release Rates

Airborne Release

The dominant releases from WWMF are H-3 and C-14; releases of particulates and Iodine-131 are much smaller (e.g. OPG10b). WWMF experience includes both stored wastes and fresh wastes. The dominant releases from DGR are, therefore, expected to also be H-3 and C-14.

Particulate releases are not expected at the DGR because: (1) there are no waste conditioning processes at the DGR (e.g., no incinerator as at WWMF); (2) all the waste packages arriving at the DGR are closed with lids; and (3) external loose contamination will be checked prior to acceptance at the DGR.

During DGR waste package handling and storage until emplacement rooms are closed, H-3 and C-14 can be released as water vapour or gas due to off-gassing from waste packages, because the containers are closed but not generally sealed air-tight. While the LLW containers are stacked, the weight of most of the containers is spread across the stacking frame at the four corners of the container, rather than directly on the lid. Therefore, any slowly generated gases can be released from the containers.

H-3 released from LLW is expected to be primarily in oxide form (i.e., HTO), since the LLW is in closed but not sealed waste containers with aerobic conditions and little radiolysis. WWMF measurements have confirmed that the airborne water-soluble organic bound H-3 component is small (OPG99). H-3 released from ILW also exists in elemental form (i.e., HT) due to higher radiolysis and more tightly sealed containers that can support anaerobic corrosion. A study found about 30% of airborne H-3 was in elemental form as HT in ILW resins, with a small amount as methane (KINECTRICS03). Most C-14 release comes from ILW resins. In ILW resins, C-14 is released mostly in the CO_2 form (KINECTRICS03). For this preclosure safety assessment, all H-3 was assumed to be released as HTO, and all C-14 as $^{14}CO_2$. These have the highest inhalation dose coefficients for the common release forms.

Since the DGR will mostly contain the same waste packages as the WWMF, relevant experience at WWMF was used to relate measured air release rates to corresponding H-3 or C-14 inventories in ILW or LLW. The estimated airborne release rate per unit inventory based on WWMF experience is provided in Table 7-9 (see Box 1 and Box 2).

Table 7-9: Waste Package Fractional Airborne Release Rates

	H-3 (Bq/year per Bq)	C-14 (Bq/year per Bq)
LLW	4.2E-03	5.4E-02
ILW	4.3E-04	5.0E-04

These fractional release rates are approximate, since the WWMF values may be influenced by environmental factors such as wind pumping, which would not apply in the DGR. For comparison, Douche estimates 1%/year for H-3 from L&ILW (DOUCHE07), and Bracke and Miller estimate 0.07%/year for C-14 from the Asse repository (BRACKE08). Table 7-9 also shows that the fractional release rate for H-3 or C-14 in ILW is at least a factor of 10 lower than that in LLW, which is consistent with the more tightly sealed ILW containers. Retube waste containers would have negligible releases, since the corrosion rate of the retube materials is very slow and these containers are seal welded.

The total airborne release rate from the DGR was estimated by multiplying the fractional release rate (Table 7-9) by the ventilated inventory in the DGR (Table 7-7 and Table 7-8), and is provided in Table 7-10.

For comparison, the average monitored emissions to air from the WWMF in 2009-2010 are approximately 5×10^{13} Bq/year for H-3 and 4×10^{9} Bq/year for C-14 estimated from quarterly operations reports (OPG09a, OPG09b, OPG10b). These WWMF emissions are a small fraction of the total releases from the Bruce nuclear site, which was 1.4×10^{15} Bq for H-3 and 2.5×10^{12} Bq for C-14 for 2009 (BP10). The estimated DGR H-3 air emissions are initially comparable to those from the WWMF as the inventories are similar, but decrease with time due to decay and as panels are isolated by closure walls in the DGR. The estimated DGR C-14 air emissions are larger than those currently estimated from WWMF, due to the larger inventory in the DGR and to the conservative estimate of C-14 off-gassing rate. In all cases, the DGR estimates are much smaller than the preliminary DGR DRL of 1.4×10^{17} Bq/year for H-3 and 4.6×10^{15} Bq/year for C-14 (Table 7-3).

BOX 1: Fractional Airborne Release Rate Estimate for H-3 and C-14 in LLW

The airborne release rate of H-3 and C-14 from LLW was determined based on the information collected in 1999 and 2000 (OPG00c, OPG07c). The concentrations of H-3 and C-14 inside the LLSBs at the WWMF were measured at various sampling locations and in the exhaust air. At the same time, the air flow rates into and out of the LLSB were measured.

The tables below give the estimated annual emissions from LLSBs for H-3 and C-14 respectively. The average annual release rate from all LLSBs is 8.83 x 10^{12} Bq/year for H-3 and 9.91 x 10^{10} Bq/year for C-14. The H-3 and C-14 inventories in the LLSBs in 2000 were 2.09 x 10^{15} Bq and 1.82x10^{12} Bq respectively, according to recent estimates from OPG's waste inventory database. The fractional release rate from LLW was, therefore, estimated as 4.2x10^{-3} /year for H-3 and 5.4x10^{-2} /year for C-14.

Estimated Annual H-3 Emissions from the LLSBs

LLSB Number	Annual Emission Rate based on October 1999 Test (Bq)	Annual Emission Rate based on March 2000 Test (Bq)	Annual Emission Rate based on June 2000 Test (Bq)
1	1.48E+12	3.37E+11	6.96E+11
2	2.15E+12	8.88E+10	2.15E+12
3	8.14E+11	3.18E+11	2.53E+11
4	2.44E+12	1.26E+12	2.85E+12
5	3.33E+11	1.15E+12	1.77E+12
6	2.37E+12	1.92E+12	2.67E+12
7	Not In Service	2.22E+10	1.41E+12
Total	**9.59E+12**	**5.10E+12**	**1.18E+13**

Estimated Annual C-14 Emissions from the LLSBs

LLSB Number	Annual Emission Rate based on October 1999 Test (Bq)	Annual Emission Rate based on March 2000 Test (Bq)	Annual Emission Rate based on June 2000 Test (Bq)
1	59.2E+09	14.8E+09	27.8E+09
2	11.1E+09	0.9E+09	2.9E+09
3	7.0E+09	1.8E+09	1.2E+09
4	12.6E+09	1.0E+09	2.5E+09
5	4.8E+09	3.1E+09	8.9E+09
6	59.2E+09	36.3E+09	37.3E+09
7	Not In Service	0.03E+09	4.9E+09
Total	**153.9E+09**	**57.9E+09**	**85.5E+09**

BOX 2: Fractional Airborne Release Rate Estimate for H-3 and C-14 in ILW

The fractional airborne release rate for H-3 and C-14 from ILW was based on emissions from the ICs at the WWMF. Air exchange can occur between the IC and the surrounding environment, depending on a variety of factors but particularly wind pumping (OPG00c). Prior to 2000, modelling of the ICs showed that there were approximately 1000 Headspace Air Turnovers (HATs) per year (OPG00c). After 2000, the air exchange rate was reduced by up to a factor of 100, due to modification of the gaskets in the ICs (OPG03c). This reduced exchange rate was used in the fractional release rate calculation below.

The total concentrations of H-3 and C-14 in the headspace gas and on the resins of several resin liners were measured (KINECTRICS03). The fractional release rate of H-3 from an IC-12 resin container can, therefore, be estimated as, for example, 2.03×10^{-4}/year for IC-1207 based on:

$$\frac{\begin{array}{c}\text{Concentration of } H\text{-}3 \\ \text{in Headspace gas}\end{array} \times \dfrac{\text{Headspace in IC}}{\text{Number of Liners}} \times \text{HATs}}{\begin{array}{c}\text{Concentration of } H\text{-}3 \\ \text{in resins}\end{array} \times \text{Amount of resins per liner}} = \frac{1.31 \times 10^{6}\ Bq/m^{3}\text{air} \times \dfrac{5m^{3}\text{air}}{4\text{liners}} \times \dfrac{1000/year}{100}}{2.70 \times 10^{10}\ Bq/m^{3}\text{resins} \times 3m^{3}\text{resin/liner}}$$

The fractional release rate from ILW was, therefore, estimated as 4.3×10^{-4}/year for H-3 and 5.0×10^{-4}/year for C-14. This does not apply to sealed ILW containers such as retube waste containers.

Estimated H-3 Fractional Release Rate from IC Resin Liners

Resin Liner	H-3 Concentration in Headspace (Bq/m^3)	Resin Activity at Measurement Time (Bq/m^3)	Fractional Release Rate (/year)
IC-1207	1.31E+06	2.70E+10	2.03E-04
IC-1209	6.36E+06	2.90E+10	9.19E-04
Log Mean	**2.88E+06**	**2.80E+10**	**4.32E-04**

Estimated C-14 Fractional Release Rate from IC Resin Liners

Resin Liner	C-14 Concentration in Headspace (Bq/m^3)	Resin Activity at Measurement Time (Bq/m^3)	Fractional Release Rate (/year)
IC-1809	2.66E+06	4.40E+09	1.69E-03
IC-1826	7.11E+07	1.25E+12	1.59E-04
IC-1834	1.60E+05	4.24E+09	1.06E-04
IC-1208	4.43E+07	2.00E+11	9.27E-04
IC-1220	2.72E+07	9.70E+10	1.17E-03
Log mean	**8.17E+06**	**5.38E+10**	**4.99E-04**

Table 7-10: Maximum Estimated Airborne Releases during Normal Operation (Nominal Schedule)

Location	Waste Category	H-3 2018 (Bq/year)	H-3 2023 (Bq/year)	H-3 2038 (Bq/year)	H-3 2052 (Bq/year)
WPRB (max) [a]	LLW	7.4E+10	7.4E+10	7.4E+10	7.4E+10
	ILW	2.1E+09	2.1E+09	2.1E+09	2.1E+09
Ventilation Shaft	LLW	2.2E+12	1.3E+13	2.0E+12	3.9E+11
	ILW	0.0E+00	2.0E+11	9.5E+10	5.2E+09
Total DGR	-	2.3E+12	1.4E+13	2.2E+12	4.7E+11
Percentage of Airborne DRL (%)	-	0.002	0.01	0.002	0.0003
Location	**Waste Category**	**C-14 2018 (Bq/year)**	**C-14 2023 (Bq/year)**	**C-14 2038 (Bq/year)**	**C-14 2052 (Bq/year)**
WPRB (max) [a]	LLW	8.6E+08	8.6E+08	8.6E+08	8.6E+08
	ILW	1.6E+10	1.6E+10	1.6E+10	1.6E+10
Ventilation Shaft	LLW	8.8E+09	7.0E+10	1.6E+10	5.0E+09
	ILW	0.0E+00	1.8E+12	1.4E+12	3.2E+10
Total DGR	-	2.6E+10	1.9E+12	1.4E+12	5.4E+10
Percentage of Airborne DRL (%)	-	0.0006	0.04	0.03	0.001

Note:

a. Assumes 24 LLW and 2 ILW packages with high inventories are present in WPRB.

Waterborne Release

During normal operation of the DGR, there will be no direct route for waterborne release, as the containers have little free water. Some H-3 and C-14 will escape from waste packages by off-gassing or air exchange as airborne release, as described in the previous section. Consistent with similar emissions at the WWMF, these are mostly released as HTO and $^{14}CO_2$.

Some of these gases may be dissolved in or collected in condensate water, which will be collected in the WPRB or repository sumps during DGR operations or may condense near the ventilation system exhaust.

WPRB Water

The WPRB sumps are not expected to be a routine source of contaminated water, because the building will be actively ventilated (limiting concentration and condensation), and there will be no activities involving routine use of water within the waste package handling area.

Underground Condensate Water

Normally the DGR will be a dry facility during operations, as the host rock has very low permeability and as there will be few other sources of moisture underground. Even the moisture from diesel equipment will be low due to the limited amount of activity at any one time during operations. It is, therefore, expected that there will be little condensation in the underground (including the rooms) or the ventilation shaft most of the time. This is consistent with experience excavating the cooling intake tunnels at the Darlington nuclear station, which are also excavated in Cobourg limestone, but at a much shallower depth (OH86).

However, during certain periods throughout the year, some water may condense in the ventilation shaft and outcast plenum. This condensation will be dependent upon the surface air temperature and humidity, the extent of underground activities (generating moisture), the airflow rate through the repository, and the ventilation shaft and exhaust system design.

For example, during the warmest month of the year, where the average maximum temperature is 24°C, the moisture capacity of the air is about 19 g/kg (i.e., at 100% relative humidity). During the coldest month of the year at an average minimum temperature of -11°C, the moisture capacity of the air is 1.6 g/kg. At a ventilation air flow of 100 m^3/s, the air mass flow rate is about 120 kg/s, so the moisture carrying capacity of the air ranges from about 190 to 2300 g/s. At 80% relative humidity at surface, the air enters with 80% of the above amount, and therefore has capacity for an additional 40 to 400 g/s moisture before it reaches saturation at 100% relative humidity.

This can be compared with the diesel equipment water generation rate of about 16 g/s at 250 kW power during operations. Other underground water vapour sources are estimated to be 5 to 50 g/s, assuming that water usage is generally limited and based on the low permeability of the host rock. Comparing with the general air moisture capacity above, it can be seen that condensation is possible during colder winter days and humid summer days. The extent can, however, be controlled in part through increasing air flow rate and reducing underground activities during peak periods. Based on a simple ventilation model, and considering seasonal and day/night changes in surface conditions and activities, the moisture condensation rate within the ventilation shaft and exhaust plenum was estimated to be up to 100 m^3/year.

Waterborne release from the underground will be dominated by this condensate water collected in the ventilation shaft sump, because the other sump waters will be from generally uncontaminated areas, and are expected to have essentially negligible radioactivity. A condensate water amount of 100 m^3/year from the ventilation shaft was used in the waterborne release calculations.

H-3 Release in Condensate Water

Airborne H-3 is conservatively assumed to be entirely present as HTO. Assuming that the water vapour and the water condensate are in equilibrium, the H-3 levels in the condensate can be determined. The H-3 level will vary depending on the air conditions (e.g., less outgassing at lower temperatures and humidities) and the phase of operation activities (e.g., amount of emplaced LLW); however, peak condensate H-3 level of around 10^6 Bq/L could occur.

For example, at a maximum H-3 airborne concentration of 10,000 Bq/m^3 in the ventilation shaft (rounded up from the calculated value of 9900 Bq/m^3, Table 7-16, Section 7.4.3.2) and an air moisture content in the ventilation shaft of 8 $g_{water}/kg^3{}_{air}$ (a representative value from the range of ventilation shaft conditions), the H-3 level in condensate would be (10,000 $Bq/m^3{}_{air}$) / [(8 g_{water}/kg_{air}) x (1.2 $kg_{air}/ m^3{}_{air}$) x (0.001 L_{water}/g_{water})] = 10^6 Bq/L. This volume and concentration of tritiated water is similar to that currently collected in LLSB and ILW sumps at WWMF.

The corresponding maximum waterborne H-3 release rate would be about 10^{11} Bq/year based on the estimated condensate amount (100 m^3/year). This release rate is much less than the preliminary DGR waterborne DRL of 2.1 x 10^{15} Bq/year (Table 7-3), i.e., the H-3 release rate is about 0.005% of the DRL.

C-14 Release in Condensate Water

The concentration of C-14 in the condensate water in the ventilation shaft is related to the airborne C-14 concentration according to Henry's law, which states that the equilibrium (maximum) concentration of dissolved gas in liquid is directly proportional to the concentration of the gas in the gaseous phase above the liquid. Henry's law relationship is expressed as follows:

$$C_{aq} = C_{gas} \times K_H \tag{7-1}$$

where:

C_{aq} = Concentration of the dissolved gas in solution (Bq/m^3)

C_{gas} = Concentration of the gas in the gaseous phase above the interface with the liquid (Bq/m^3)

K_H = Henry's law constant (dimensionless)

Airborne C-14 will be present in the ventilation shaft primarily as CO_2. The dimensionless Henry's law constant for CO_2 in water at 25°C is 0.832 ($C_{CO2,aq}/C_{CO2,gas}$) (PERRY07). The maximum $^{14}CO_2$ airborne concentration in the ventilation shaft was estimated to be about 1500 Bq/m^3, rounded up from the calculated value of 1330 Bq/m^3 (Table 7-16, Section 7.4.3.2). Using Equation 7-1, the equilibrium concentration of C-14 in the ventilation shaft condensate water was estimated to be about 1.2 Bq/L.

The maximum waterborne C-14 release rate based on 100 m^3/year of condensate would be about 10^5 Bq/year, which is much less than the assumed DGR waterborne DRL of 1.7×10^{12} Bq/year, i.e., < 0.0001% of the DRL.

Total Waterborne Release

The maximum waterborne release rates for H-3 and C-14 are summarized in Table 7-11. The maximum value would likely occur prior to closure of the first panel containing most of the currently stored WWMF wastes, within the first 5-10 years of operation.

Table 7-11: Estimated Maximum Waterborne Releases during DGR Normal Operations

Radionuclide	Maximum Waterborne Releases (Bq/year)	Percentage of Waterborne DRL (%)
H-3	1.1E+11	0.005
C-14	1.3E+05	<0.0001

7.4.2.2 Dose Impact Model

The impact of the airborne and waterborne releases on the public was assessed using two methods.

- Dose based on the DGR estimated release rate in comparison to the WWMF dose pathways model used for the DRL. This theoretical method is very conservative.
- Dose based on the DGR estimated releases in comparison to the Bruce emissions and the REMP dose. This empirical method is more realistic but still conservative.

DRL Pathways Estimate

The WWMF DRL model (OPG03b) uses the pathways illustrated in Figure 7-1 to estimate the dose impact to members of the public living in the vicinity of the site for a given release rate from WWMF. Exposure pathways considered are ingestion of water and food and incidental ingestion of soil; inhalation and immersion; and external radiation from soil and sediment. The effect of air dispersion is included using ADFs developed for the WWMF DRL calculations. The dispersion of waterborne releases is also included in the WWMF DRLs.

The WWMF DRLs are calculated for a variety of possible receptors at different locations around WWMF, and at different ages from infants to adults. For each of these receptors, the DRL pathways model estimates a dose per unit release rate of a given radionuclide from the WWMF.

For the DGR, the most conservative dose-per-unit-release rates for HTO and C-14 emissions were adopted from the DRL pathways model, as given in Table 7-12, even though this corresponds to different receptors. The potential DGR dose consequences to the public for air and water emissions were based on these dose-per-unit-release rates scaled to the DGR estimated release rates (Table 7-10 and Table 7-11) respectively.

Bruce REMP - Based Estimate

The Bruce nuclear site has a REMP, in which the actual concentrations of key radionuclides are measured in plants, milk, water and air around the site, and this is used to provide a more accurate estimate of the potential Bruce nuclear site impacts. Table 7-13 summarizes the measured H-3 and C-14 emission rates and corresponding dose rates for the Bruce nuclear site from the REMP (BP07, BP08, BP09, BP10).

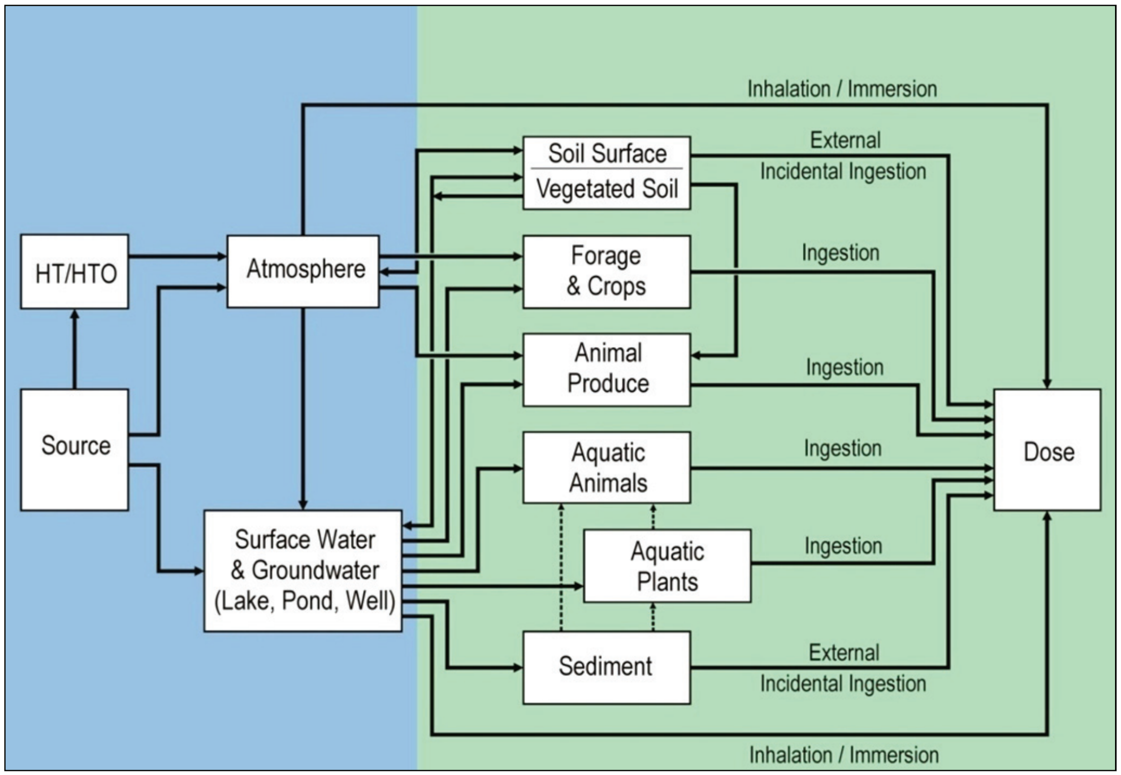

Figure 7-1: The Environmental Pathways Used in the WWMF DRL Dose Estimates

Table 7-12: Maximum Dose Conversion Factor from WWMF DRL Pathways Model

Radionuclide	Air Emissions (Sv/year per Bq/s)[b]	Water Emissions (Sv/year per Bq/s)[b]
HTO	2.3E-13	1.5E-11
C-14 [a]	6.8E-12	1.9E-08

Notes:
a. Assumed to be released as CO_2 in air and carbonate in water.
b. WWMF DRL model (OPG03b).

Table 7-13: Bruce Nuclear Site REMP Measured Emission Rates and Dose Rates

Year	Air Emission Rate (Bq/year)		Water Emission Rate (Bq/year)	
	H-3	C-14	H-3	C-14
2006	9.5E+14	1.2E+13	7.3E+14	7.6E+09
2007	1.6E+15	7.2E+12	1.3E+15	6.9E+09
2008	1.6E+15	5.4E+12	4.7E+14	4.3E+09
2009	1.4E+15	2.5E+12	6.3E+14	4.5E+09
Year	**Estimated Air Dose Rate (µSv/year)**		**Estimated Water Dose Rate (µSv/year)**	
	H-3	**C-14**	**H-3**	**C-14**
2006	3.9E-01	2.3E-01	1.2E+00	2.8E-02
2007	9.4E-01	3.5E-01	2.5E-01	2.7E-03
2008	1.4E+00	1.2E-01	3.2E-01	4.9E-03
2009	3.6E+00	2.2E-01	2.3E-01	6.2E-04

Notes:
- The REMP reports (BP07, BP08, BP09, BP10).
- The Bruce nuclear site measured emission rates include the station stacks, the WWMF and other monitored points, but not fugitive emissions.
- Estimated doses presented for the Bruce nuclear site are for the receptor (age and location) of maximum estimated dose in the REMP report for all relevant pathways:
  - Air-inhalation, air (external), soil (external), plants, and animals.
  - Water-ingestion, water (external), sediment (external), and fish.

The potential impacts on the public from the DGR were estimated by scaling the REMP-based doses by the ratio of the Bruce nuclear site and DGR emissions (Table 7-10 and Table 7-11).

For comparison purposes, the REMP-based dose to the public from releases of radionuclides from all facilities at the Bruce nuclear site has remained below 4 μSv/year for the past several years (BP07, BP08, BP09, BP10). The WWMF is a small contributor to these values.

7.4.2.3 Public Dose Results – Air and Water Emissions

Table 7-14 shows the peak estimated doses to the most exposed public groups due to the airborne and waterborne emissions from the DGR. These occur in 2023, based on the nominal emplacement schedule described in Table 7-6. In this emplacement schedule, most of the wastes have been transferred from the WWMF to the DGR by 2023, with the resulting releases and, therefore, potential public doses at a maximum. Doses at other years would be lower due to decay and as panels are closed off. A more delayed transfer rate of packages to the DGR would lead to a lower peak dose rate from the DGR.

The REMP-based method gives a more realistic estimate of the public dose, which is lower than that derived from the DRL pathways method. In either case, the results indicate very low doses to the public, similar to what would be calculated for WWMF for similar LLW and ILW radionuclide inventories. The doses are far below the CNSC regulatory limit of 1 mSv/year.

The assessment results indicate that there are no concerns with respect to exposure to members of the public during normal operations of the DGR.

7.4.3 Radiological Assessment of Air Emissions on Workers

7.4.3.1 Methodology

The primary source of radionuclides in the WPRB and underground DGR during normal operations is the slow release of H-3 and C-14 bearing gases from waste packages. The ventilation system is designed to ensure fresh air flows from areas of low potential of contamination to areas of greater potential of contamination, to keep the workers in a fresher air stream. The ventilation is also designed to ensure that the emplacement rooms are appropriately ventilated at all times, so that the levels of potentially noxious or hazardous gases (e.g., diesel fumes, methane or radon from the rock, H-3 or C-14 from waste package off-gassing) are controlled. Exposure pathways considered for workers were inhalation and skin absorption or immersion of H-3 and C-14 dispersed in air above ground and underground.

Table 7-14: Estimated Maximum Dose to the Public during DGR Normal Operation using Two Dose Estimate Models [a]

Source	DRL Pathways Dose Model [b]			REMP-Based Dose Model [c]		
	H-3 (µSv/year)	C-14 (µSv/year)	Total (µSv/year)	H-3 (µSv/year)	C-14 (µSv/year)	Total (µSv/year)
Dose from Air Releases	0.10	0.41	0.51	0.04	0.17	0.21
Dose from Water Releases	0.05	< 0.01	0.05	< 0.01	< 0.01	< 0.01
Total Dose	0.15	0.41	0.56	0.04	0.17	0.21

Notes:
a. Maximum occurs at around 2023, based on the reference emplacement schedule.
b. Based on a hypothetical receptor using the highest dose per release rates from exposure to air and water of H-3 and C-14 (calculated for receptors used in DRL calculations). This is a conservative approximation, as there is no single receptor location or age group which maximizes all these exposures.
c. Scaled based on 2009 Bruce nuclear site REMP report (BP10).

Impacts of radon generated from wastes and from surrounding host rock were assessed, and details of the results are given in the Radon Assessment report (NWMO11ae). In summary, radon is not expected to be present in the DGR in significant concentration on the basis of the measured low uranium/radium content of the rock and wastes. Radon will be specifically checked during construction, and then periodically during operation.

Airborne H-3 and C-14 concentrations were calculated using a compartment-based ventilation model. Figure 6-20 shows the underground ventilation flow distribution system. Table 7-15 lists the key ventilation assumptions. The ventilation rates were calculated for various underground locations based on the ventilation requirements for the DGR during the emplacement operations (Table 7-15).

Air dispersion calculations also require an estimate of H-3 and C-14 release rates at various locations (ventilation shaft, ventilation drift, emplacement rooms, and WPRB). Table 7-10 gives the estimated H-3 and C-14 release rates from the ventilation shaft and from the WPRB.

The release rate in an emplacement room was calculated based on the fractional airborne release rate (Table 7-9), and the H-3 and C-14 inventory in the room. The maximum H-3 ventilated inventory in any room (2023) was estimated to be 3.2×10^{14} Bq (from LLW room); the maximum C-14 ventilated inventory in any room (2023) was estimated to be 1.8×10^{15} Bq (from ILW room).

Table 7-15: Assumed Ventilation Basis during DGR Normal Operations

Location	Ventilation Basis
Empty emplacement rooms	None. Entry barricaded and signed to prevent entry without appropriate procedure.
Active emplacement room	18 m^3/s day; minimum 3 m^3/s night.
Filled emplacement rooms	About 1 m^3/s to ensure controlled air exchange. Partial wall installed at end of room to limit access and direct the ventilation.
Closed panel	None. Ventilation will be disconnected and full closure walls installed.
Total underground	85 - 120 m^3/s daytime, approximately half for night time.
WPRB	2.8 m^3/s, based on 0.5 air changes/hr and building volume of 20,000 m^3.

Air concentrations during day operations and during off-operating hours at night were then estimated using the following equations:

$$C_{Shaft} = QR_{shaft} \; / \; F_{Shaft} \tag{7-2}$$

$$C_{Room} = (Q_{Room} \times f_r) \; / \; F_{Room} \tag{7-3}$$

$$C_{WPRB} = QR_{WPRB} \; / \; F_{WPRB} \tag{7-4}$$

where:

C_{Shaft} = H-3 or C-14 concentration in shaft (Bq/m^3)

C_{Room} = H-3 or C-14 concentration in room (Bq/m^3)

C_{WPRB} = H-3 or C-14 concentration in WPRB (Bq/m^3)

QR_{shaft} = H-3 or C-14 release rate through ventilation shaft (Bq/s)

QR_{WPRB} = H-3 or C-14 release rate in WPRB (Bq/s)

F_{Shaft} = Ventilation rate in shaft (m^3/s)

F_{Room} = Ventilation rate in room (m^3/s)

F_{WPRB} = Ventilation rate in WPRB (m^3/s)

Q_{Room} = H-3 or C-14 ventilated inventory in room (Bq)

f_r = H-3 or C-14 fractional release rate (1/s), given in Table 7-9

7.4.3.2 Airborne H-3 and C-14 Concentration Results

Underground

The airborne H-3 and C-14 concentrations were calculated in the ventilation shaft and an emplacement room using Equation 7-2 and 7-3 respectively, for both the daytime and night time (i.e., off-operating hours). The airborne H-3 and C-14 concentrations in the ventilation drift were also calculated using Equation 7-2, but considering a decreased ventilation rate. The ventilation rate will be lower in the ventilation drift than in the shaft. Estimated H-3 and C-14 release rates in Table 7-10 were used for the calculation.

Maximum concentrations of these radionuclides would occur around the year 2023 just before the construction of closure walls in the access tunnel and ventilation drift of panel 2 as shown in Figure 7-2 and Figure 7-3 for the ventilation shaft. These maximum concentrations are also given in Table 7-16. For example, the maximum H-3 concentration in the ventilation shaft was calculated from the ratio of the release rate over the ventilation rate: (1.4×10^{13} Bq/year max release rate x 3.17×10^{-8} year/s) / (45 m^3/s nominal night time ventilation rate) = 9900 Bq/m^3. The H-3 and C-14 concentrations are much lower than the DACs (see Table 7-2).

The H-3 and C-14 concentrations in various emplacement rooms and the ventilation drift were also estimated for both the daytime and night time. The results are given in Table 7-16. The H-3 and C-14 concentrations are much lower than the DACs. In addition, these concentrations would occur only within the filled rooms or ventilation drift, which are normally not accessible by workers.

Table 7-16: Estimated Maximum Airborne H-3 and C-14 Concentrations

Location	Maximum H-3 Concentration (Bq/m^3)		Maximum C-14 Concentration (Bq/m^3)	
	Day [a]	Night [b]	Day [a]	Night [b]
Ventilation Shaft	4000	9900	550	1330
Emplacement Rooms	2400	14,000	1600	9600
Ventilation Drift	6500	19,000	890	2600

Notes:

a. Nominal ventilation rate of 110, 18 and 68 m^3/s in the shaft, room and ventilation drift respectively.

b. Nominal ventilation rate of 45, 3 and 23 m^3/s in the shaft, room and ventilation drift respectively.

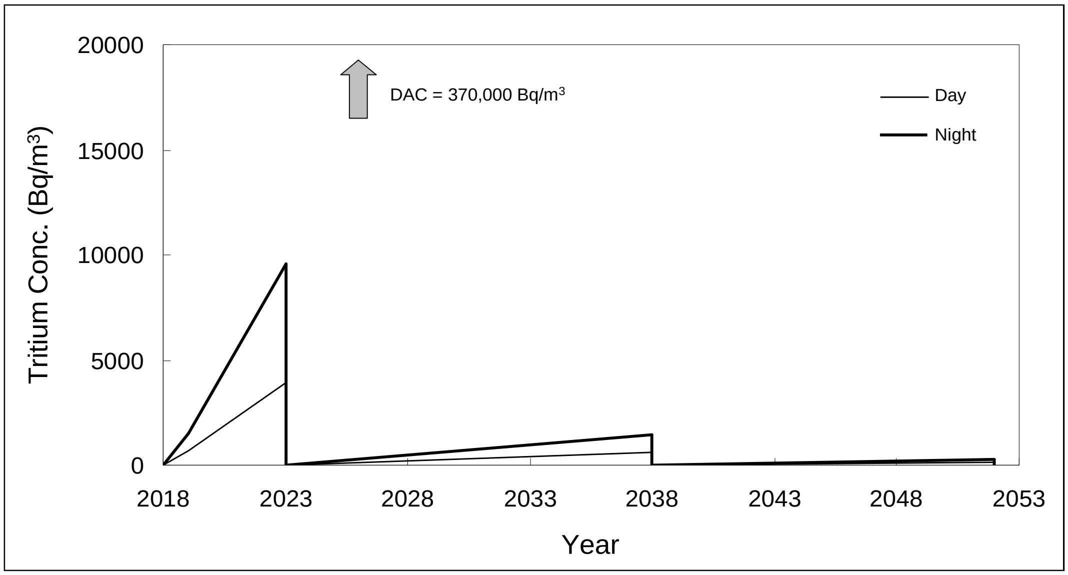

Figure 7-2: Estimated Airborne H-3 Concentrations in the Ventilation Shaft

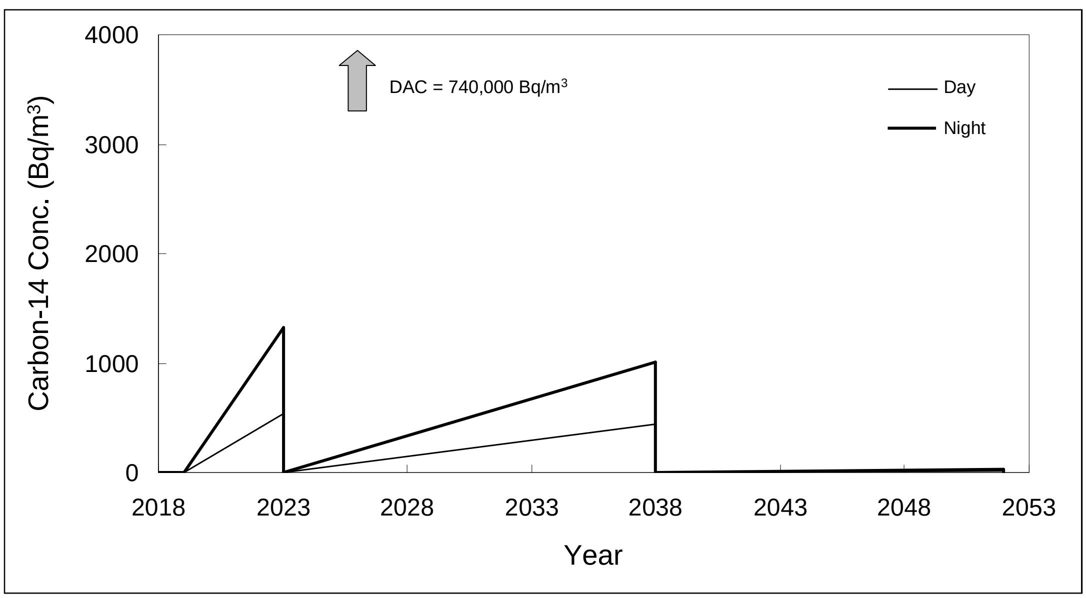

Figure 7-3: Estimated Airborne C-14 Concentrations in the Ventilation Shaft

Above Ground

The maximum H-3 and C-14 airborne concentrations at the WPRB were calculated, using Equation 7-4, to be about 900 and 200 Bq/m^3 respectively. These were based on the ventilation rate of 2.8 m^3/s, and the release rates given in Table 7-10. These concentrations were based on the bounding case whereby the WPRB temporarily holds 24 LLW packages and 2 ILW packages.

7.4.3.3 Estimated Worker Inhalation Dose

The dose coefficients, or estimates of dose per unit exposure to individual radionuclides are taken from CSA N288.1 (CSA08b). The inhalation dose coefficient for H-3 as HTO is 3.0x10^{-11} Sv/Bq for an adult; this value includes the contribution from skin absorption. For C-14 as CO_2, the inhalation dose coefficient is 1.2x10^{-11} Sv/Bq for an adult, and the immersion dose coefficient is 8.2x10^{-11} Sv/year per Bq/m^3. The worker inhalation rate is 1.6 m^3/hr (adult, moderate activity, USEPA97). This inhalation rate is also used in the accident assessment.

Underground

Table 7-17 gives the maximum estimated inhalation doses for DGR workers underground at three locations - the main shaft station, the ventilation drift, and the ventilation shaft. The main shaft station is a normally occupied area, which will routinely have one or two packages present. For example, assuming 2000 hours per year occupancy, 18 m^3/s nominal air flow rate, and 2 non-processible drums (H-3 concentration and package volume given in Table 7-30), the airborne H-3 concentration can be estimated from the ratio of annual H-3 release rate over the air flow rate:

$$\frac{2 \text{ x } (1.2 \text{ m}^3\text{/pkg}) \text{ x } (6.1\text{x}10^{11} \text{ Bq/m}^3 \text{ non-processible drum}) \text{ x } (0.0042\text{/year}) \text{ x } (3.17 \text{ x } 10^{-8} \text{ year/s})}{(18 \text{ m}^3\text{/s})} = 10.8 \text{ Bq/m}^3$$

The H-3 dose rate can then be calculated by multiplying the H-3 concentration by the inhalation rate, exposure time, and H-3 inhalation dose coefficient:

$$(10.8 \text{ Bq/m}^3) \text{ x } (1.6 \text{ m}^3\text{/hr inhalation}) \text{ x } (2000 \text{ hr}) \text{ x } (3\text{E-8 mSv/Bq}) = 0.001 \text{ mSv/year}$$

Similar calculation can also be done to estimate the dose rate for C-14.

At the other end of the repository, the ventilation drift and the ventilation shaft will collect all the off-gassed H-3 and C-14 and is a higher air concentration location – up to 6500 Bq/m^3 in the ventilation drift during day operations (Table 7-16) - although still much less than DAC. This area will normally not be occupied by workers. It was estimated that workers will spend about 260 hours/year in the drift or in the shaft conducting weekly/monthly inspections of the liner and shaft hoisting equipment, which corresponds to a dose of about 0.08 mSv/year. As Figure 7-2 shows, the air concentration will vary significantly over the operation of the DGR, and the dose rate will normally be less. The ventilation shaft conditions are monitored, and if necessary, worker exposure can be reduced through use of appropriate protective equipment and/or by adjusting air flow for the duration of each inspection to provide cleaner air.

Above Ground

The ventilation shaft exhaust is not a normally occupied area, and would have appropriate access controls (e.g., fencing) to limit exposure. Air concentrations would be less than in ventilation shaft, and much less than DAC.

The WPRB is a normally occupied area. Based on the estimated maximum airborne concentrations of H-3 (900 Bq/m^3) and C-14 (200 Bq/m^3) in WPRB, the total inhalation dose to a worker was calculated to be about 0.09 mSv/year, based on working 2000 hours/year in WPRB and assuming the WPRB contains the maximum inventory of staged packages for 100% of the time.

In summary, the estimated worker doses are all much less than the OPG's occupational dose target of 10 mSv/year, and the regulatory limit. Further mitigation can be addressed in the context of ALARA.

Table 7-17: Estimated Maximum Annual Inhalation Dose to a Worker

Location	Occupational Exposure Time (h/year)	H-3 Dose Rate (mSv/year)	C-14 Dose Rate (mSv/year)	Total Dose Rate (mSv/year)
WPRB	2000	0.08	0.007	0.09
Main Shaft Station	2000	0.001	<0.0001	0.001
Ventilation Shaft	260	0.05	0.003	0.05
Ventilation Drift	260	0.08	0.006	0.08

7.4.4 Assessment of External Radiation on Workers and Public

7.4.4.1 Methodology

Shielding calculations were carried out for workers handling representative LLW and ILW packages, specifically:

- Non-processible boxed waste constituting the highest volume of waste packages (about half of total operational LLW volume);
- Feeder pipes, a higher dose rate LLW;
- Moderator resin, a higher dose rate resin waste; and
- Retube-pressure tubes, a higher dose rate ILW.

The radionuclide concentrations in these representative waste packages are given in Table 7-18. These are based on "as-received" concentrations at WWMF, scaled to account for typical decay as shipped to DGR. The non-processible boxed wastes correspond to a package dose rate of about 0.03 mSv/hr at 1 m; the feeder pipe containers have a package dose rate of about 0.3 mSv/hr at 1 m, and the other waste concentrations correspond to a shielded package dose rate of about 0.1 mSv/hr at 1 m.

The bulk compositions of the wastes are given in Chapter 5. From this information, the element composition of the wastes was estimated for shielding calculations, and is given in Table 7-19. The characteristics of the waste packages are summarized in Table 7-20.

Dose rates were estimated for the following receptors:

- Workers located within and outside of the WPRB;
- Non-NEWs at the nearest DGR fence line, about 80 m from the WPRB; and
- Public at the nearest Bruce nuclear site fence line, about 1.1 km from the WPRB.

It is noted that non-NEWs working in the railway ditch area would potentially be exposed to waste packages during transport over the crossing, in addition to any staged LLW in the WPRB. However, packages in transit would only represent exposure for a brief period, and would be limited to one ILW or several LLW packages at most, whereas staged packages in the WPRB could include more packages and longer duration exposure. The specific exposure from packages during crossing has not been evaluated in detail. However, it can likely be addressed by simple measures such as extending the fence line around the crossing, possibly around 50 m each side. This will be addressed during detailed design.

Direct external radiation dose calculations were undertaken with MicroShield Version 8.02 (GROVE09). Skyshine doses were calculated to receptors at fence line locations using MicroSkyshine Version 2.10 (GROVE06). Some of the underground calculations were also carried out using Monte Carlo N-Particle (MCNP) code version 5.1.40 (LANL05); in particular to assess the influence of scattering along walls and ceilings.

Table 7-18: Radionuclide Activity in Representative Waste Packages for Dose Rate Calculations

Radionuclide [a]	Radionuclide Concentration (Bq/m^3)			
	Non-Processible Boxed [b]	Non-Processible Feeder Pipes [c]	Moderator Resin (Shielded) [b]	Retube-Pressure Tubes [d]
Reference Container	NPB47	NPHC	RSHLD2	RWC(PT)
Am-241	4.4E+05	9.4E+06	-	7.0E+07
Ce-141	-	-	1.2E+09	-
Ce-144	-	4.7E+03	5.3E+08	2.5E+00
Cm-244	7.9E+03	2.1E+06	-	-
Co-60	2.0E+07	2.0E+09	5.1E+10	4.0E+12
Cs-134	3.0E+05	2.4E+05	4.1E+08	1.1E+09
Cs-137+ Ba-137m	7.8E+08	1.8E+08	3.6E+08	6.8E+07
Eu-152	-	-	1.2E+09	3.1E-01
Eu-154	-	1.8E+07	6.4E+08	4.2E+03
Gd-153	-	-	5.8E+11	-
Mn-54	-	-	-	2.7E+08
Nb-94	-	-	-	2.3E+13
Nb-95	-	-	8.5E+08	1.4E-08
Pb-210	3.0E+06	-	-	-
Pu-238	5.4E+03	2.8E+06	9.5E+02	-
Pu-239	1.0E+04	5.8E+06	1.3E+03	4.3E+07
Pu-240	1.5E+04	8.0E+06	1.9E+03	5.9E+07
Pu-241	5.4E+05	6.2E+07	3.7E+03	7.5E+08
Ra-226	8.2E+04	-	-	-
Sb-124	-	-	3.1E+08	2.4E-06
Sb-125	7.8E+05	2.4E+06	9.9E+08	3.2E+10
Sn-119m	-	-	-	3.8E+08
Sr-90 + Y-90	1.5E+07	1.7E+09	2.8E+07	2.3E+10
Te-125m	-	-	-	1.0E+01
Zr-95	-	-	-	2.0E-01

Notes:

a. From Reference L&ILW Inventory report (OPG10a, Table B-1 to B-3); radionuclides do not include beta emitters such as H-3 and C-14 for external dose rate calculations.

b. As-received concentrations at WWMF.

c. As-received concentrations at WWMF, decayed by 15 years.

d. At-reactor-shutdown concentrations, plus 10 year decay before transfer to DGR.

Table 7-19: Elemental Composition of the Waste Categories for Shielding Analysis

Element	Weight %			
	Non-Processible Boxed	Non-Processible Feeder Pipes	Moderator Resin [a]	Retube-Pressure Tubes
Oxygen	6.7	-	48.7	
Carbon	4.5	1	36.6	-
Hydrogen	-	-	8.8	-
Sulphur	-	-	3.2	-
Nitrogen	-	-	2.0	-
Chlorine	1.5	-	-	-
Iron	83.8	99	-	-
Zirconium	-	-	-	97.5
Niobium	-	-	-	2.5
Silicon	2.2	-	-	-

Notes:
- <1% weight not included. Therefore, total weight may not add to100%.
a. Moderator resin includes 40 weight % bound water. Assumes mixed bed resins.

Table 7-20: Waste Container Characteristics

Dimensions	Non-Processible Boxed	Non-Processible Feeder Pipes	Moderator Resin		Retube-Pressure Tubes
			Stainless Steel Liner	Resin Liner Shield	
Container	NPB47	NPHC	RLSS	RLSHLD2	RWC(PT)
Length, m [b]	1.96	1.96	-	-	0.90
Width, m [b]	1.32	1.32	-	-	0.90
Height, m [b]	1.19	0.91	1.80	4.45	1.10
Diameter, m [b]	-	-	1.66	2.37 [a]	-
Material, mm	Carbon steel, 2.8	Carbon steel, 2.6	Stainless steel, 6	Concrete, 350	Steel-Concrete-Steel, 100-388-16
Waste Density, kg/m^3	227	500	850	-	970

Notes:
- Not applicable.
a. Based on OD of resin liner + concrete.
b. Inner dimensions for retube waste container; outer dimensions for other containers.

The external radiation dose rates were assessed for the following four scenarios:

- **Scenario 1** LLW in staging area within the WPRB;
- **Scenario 2** ILW in loading area within the WPRB;
- **Scenario 3** LLW in underground emplacement room; and
- **Scenario 4** ILW in underground emplacement room.

These scenarios were identified based on representative waste handling activities of LLW and ILW in WPRB and underground. Although not normally expected to be used for storage, a temporary staging area is available in the WPRB to store up to 24 LLW packages and 2 ILW resin shields. Retube waste (pressure tubes) will be transferred directly to the underground repository, through the package loading area on route to the main shaft headframe. Table 7-21 shows the details of the four scenarios.

For this preliminary assessment, the following assumptions were made for the WPRB:

- Steel clad construction, with external wall and roof thickness taken to be the shielding equivalent of 0.635 cm steel;
- Concrete shielding walls around the LLW staging area with thickness about 38 cm;
- An internal steel wall between the package loading area and general maintenance and storage area, with same thickness (0.635 cm) as the external wall; and
- No shielding on the forklift.

All dose points in MicroShield calculations were placed 1 m above the ground (i.e., centre of the body of a person) and along the centre line of the source. Figure 7-4 to Table 7-6 show the receptor locations at and around the WPRB for Scenario 1 (LLW packages staged), 2 (moderator resins) and 2 (retube wastes) respectively. Figure 7-7 to Figure 7-9 show the shielding configuration for Scenario 2, 3, and 4 respectively. To model multiple waste packages as single volume, the waste density (Table 7-20) is reduced to account for air space between multiple packages. The building walls were included as a shield in the calculations when applicable.

Table 7-21: External Radiation Exposure Scenarios during Normal Operations

Scenario	Description	Waste Packages
1 (LLW in LLW staging area within the WPRB)	• NEW handles the LLW packages inside the WPRB at 1 m from waste packages (receptor location 3). • NEW stands outside the WPRB external walls (receptor location 2). • NEW inside the office/control room (about 20 m from waste packages) is exposed due to the temporary storage of LLW packages (receptor location 1). • NEW works on the roof directly above the source (receptor location 4). • Non-NEW at the DGR fence line (80 m) is exposed due to LLW packages inside the WPRB (receptor location 6). • A member of the public at the nearest Bruce nuclear site fence line (1.1 km) is exposed due to LLW packages inside the WPRB (receptor location 5). • See Figure 7-4 for receptor locations.	• 24 LLW packages (either non-processible boxed or feeder pipes) • Container configuration: 3 (L) x 4 (W) x 2 (H) with 50 mm air gap between waste packages • 4 mm thick steel container • Density reduction factor of 0.956
2 (ILW in loading area within the WPRB)	• NEW handles the ILW packages (moderator resin or pressure tubes) inside the WPRB, at 2 m from waste package (receptor location 3). • NEW stands outside the WPRB external walls (receptor location 2). • NEW inside the office/control room (~25 m from source) is exposed (receptor location 1). • NEW works on the roof directly above the source (receptor location 4). • Non-NEW at the DGR fence line (80 m) is exposed due to ILW packages inside the WPRB (receptor location 6). • A member of the public at the nearest Bruce nuclear site fence line (1.1 km) is exposed due to ILW packages inside the WPRB (receptor location 5). • For moderator resin, see Figure 7-5 for receptor locations and Figure 7-7 for shielding configuration. • Figure 7-6 shows the receptor locations for pressure tubes.	**Shielded moderator resin packages** • 1 x 2 configuration. • 6 mm thick stainless steel for resin liner container material • 350 mm thick concrete for resin liner shield materials • Density reduction factor of 0.785 **Shielded retube waste (pressure tubes)** • 1 package 100 mm steel- 388 mm concrete-16 mm steel for retube waste containers materials

Scenario	Description	Waste Packages
3 (LLW in underground emplacement room)	• NEW drives a forklift to move LLW packages in the LLW emplacement room (2 m from source). • See Figure 7-8 for configuration.	• 20 LLW feeder pipe packages (4 packages side by side and stacked 5 high) • 4 mm thick steel container • Density reduction factor of 0.981
4 (ILW in underground emplacement room)	• NEW drives a large forklift to move ILW packages in the ILW emplacement room (2 m from source). • See Figure 7-9 for configuration.	• 3 shielded moderator resin packages • 6 mm thick stainless steel for resin liner container material • 350 mm thick concrete for resin liner shield materials

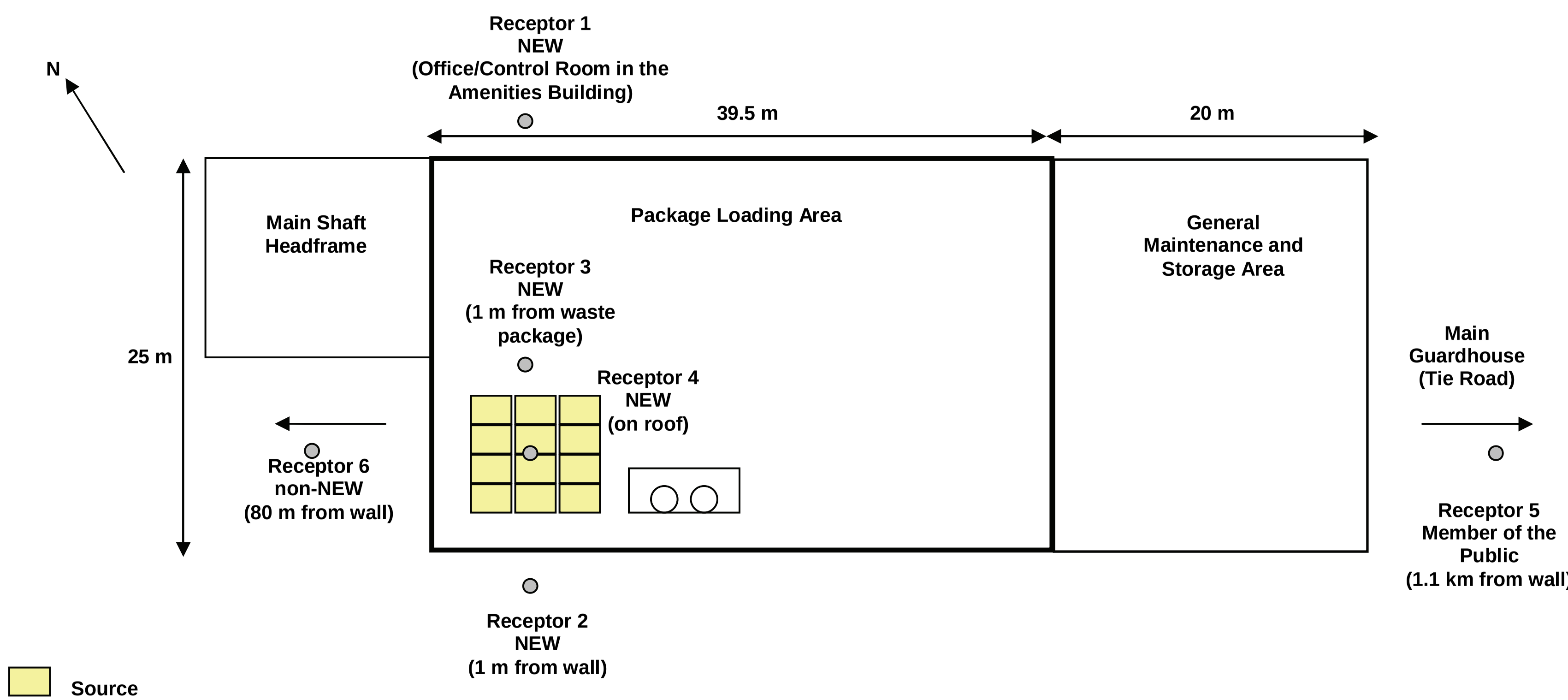

Figure 7-4: Receptor Locations Above Ground (Scenario 1)

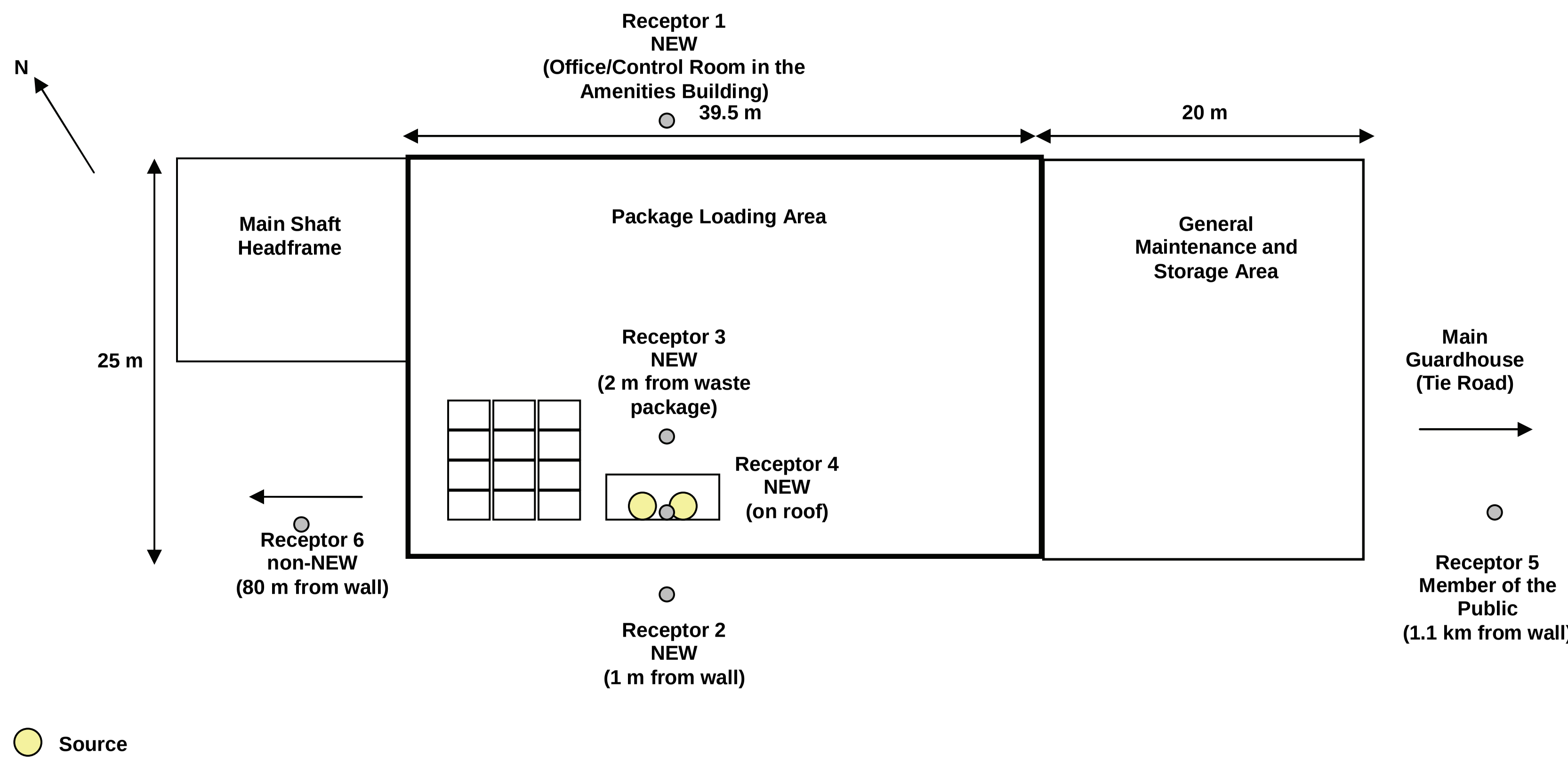

Figure 7-5: Receptor Locations Above Ground (Scenario 2, Moderator Resins)

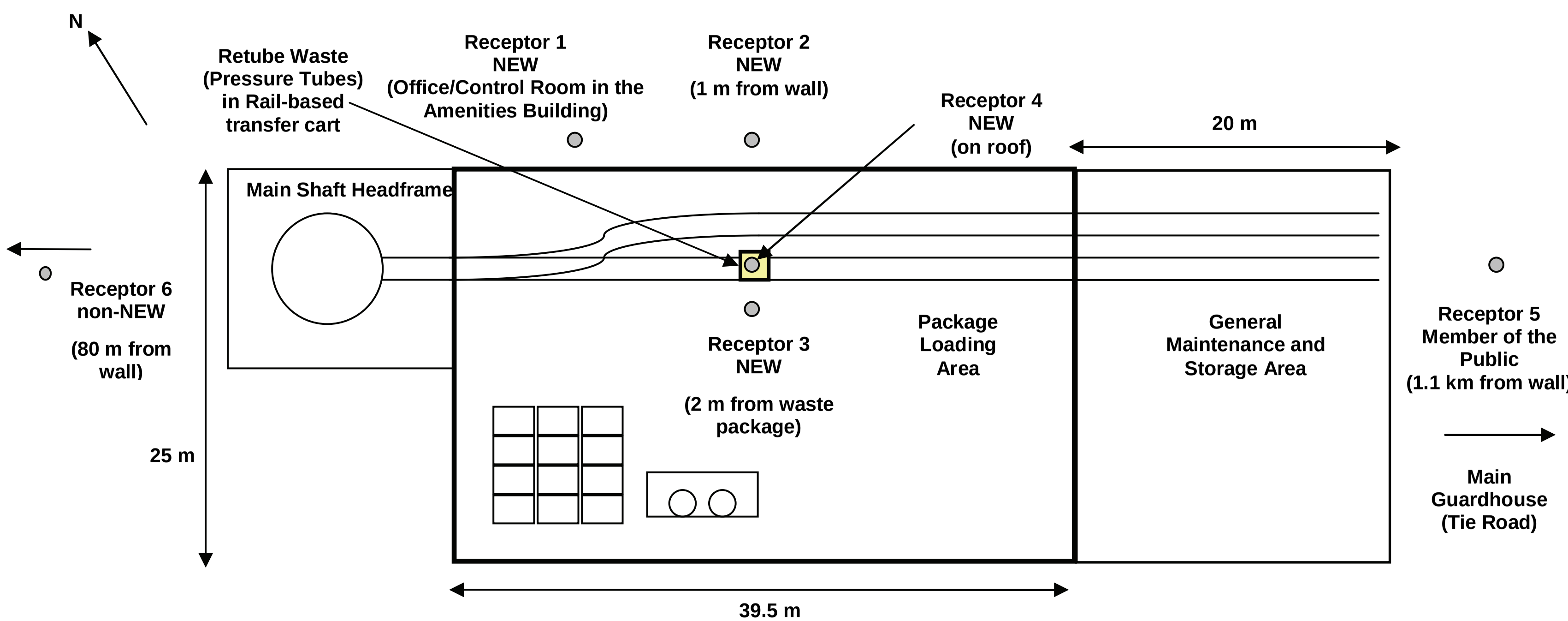

Figure 7-6: Receptor Locations Above Ground (Scenario 2, Retube Waste)

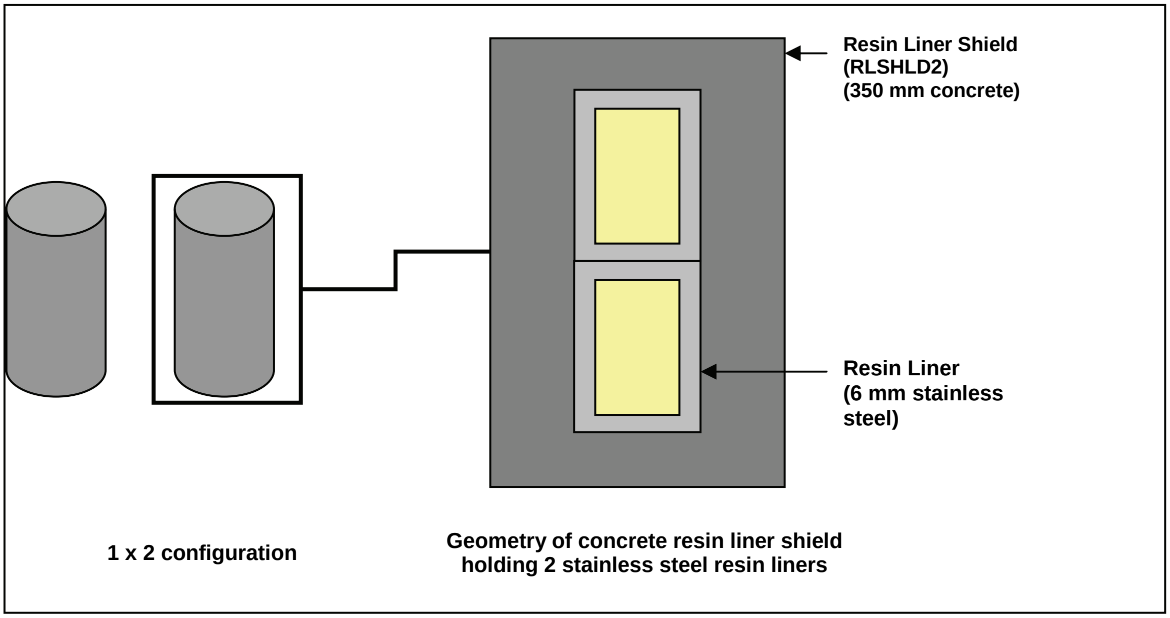

Figure 7-7: Configuration of Resin Liner Packages in Loading Area for Scenario 2

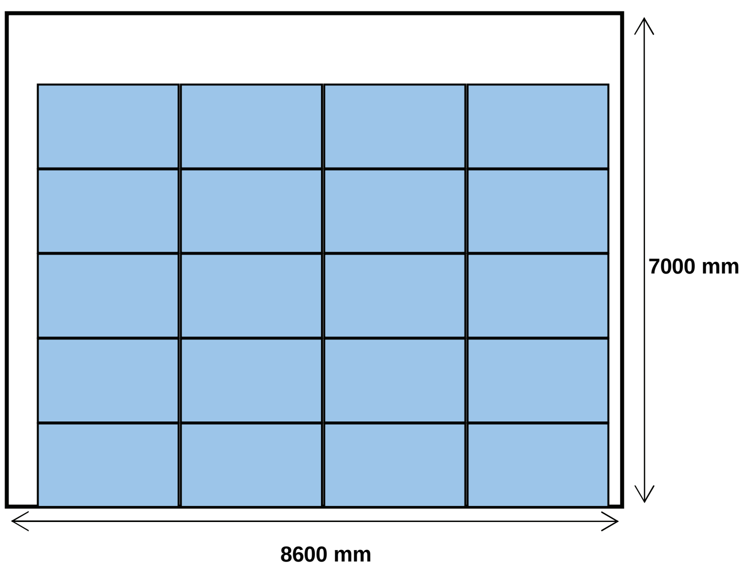

Figure 7-8: Configuration of LLW Bins in Emplacement Room for Scenario 3

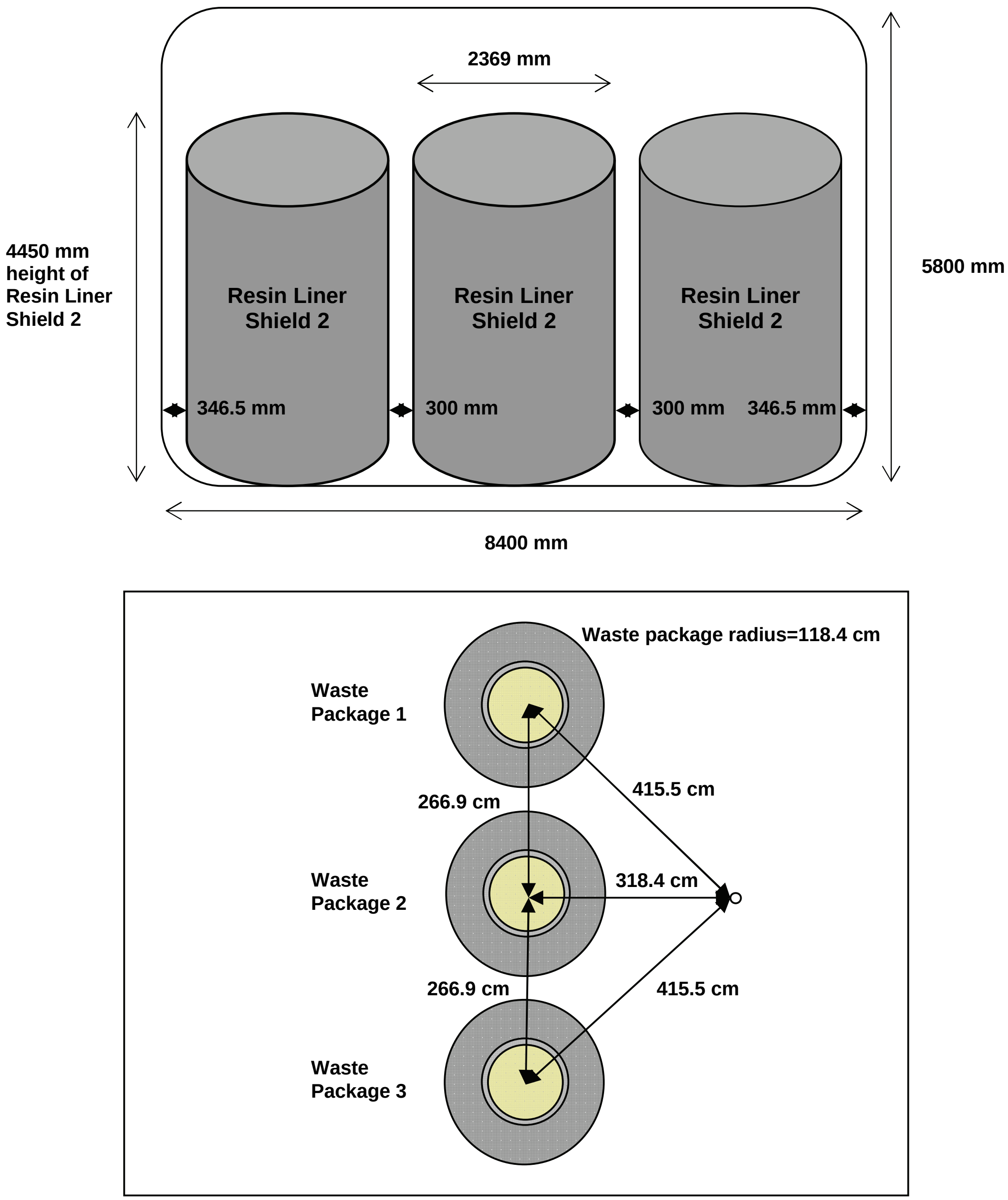

Figure 7-9: Configuration of Resin Liner Packages in the ILW Emplacement Rooms for Scenario 4

7.4.4.2 External Dose Results

<u>Estimated External Dose Rates for the Scenarios</u>

The results of shielding calculations for worker exposure are given in Table 7-22 for representative locations near waste packages.

The calculations indicate potentially high dose rates in the WPRB if there are many packages staged there, and in the emplacement room underground, if these locations contain high dose rate packages. However, the doses would be limited by the worker exposure time in these locations, and the actual package dose rates.

The results also show that a wall around the WPRB staging area similar in thickness to LLSB walls will need to be incorporated in the detailed design to ensure that the external dose rate outside of the WPRB remains below 25 µSv/hr (OPG Radiation Protection Requirements, Section 7.1.2.1) and that the dose rate in the office/main control room is below 10 mSv/year, if multiple packages are routinely staged within the WPRB.

In practice, the dose rates are likely to be much lower than those calculated here, because the assessment considered the maximum number of packages in the areas, and did not consider WWMF operating practice such as placing higher dose rate packages behind lower dose rate packages. More detailed assessments of dose rates, as well as the value of mitigating measures such as shielding or greater stand-off distances, can be considered as part of an ALARA assessment based on detailed design.

MCNP modelling was performed to determine the effects of backscattering from surrounding rock in the underground ILW emplacement room (Scenario 4). The result indicates that scattering along the walls, roof and floor increases the dose by about 5% at the worker location (forklift driver, 2 m away from waste packages). The MCNP results also indicate the overall dose rate to be a factor of 1.7 lower than those estimated using MicroShield, showing that the MicroShield dose results are conservative compared to the MCNP results.

Shielding calculations were carried out to estimate dose rates at the DGR fence line (about 80 m from the WPRB, receptor location 6) and nearest Bruce nuclear site fence line (about 1.1 km from the WPRB, receptor location 5). The receptor locations are indicated in Figure 7-4 to Figure 7-6. Even with the skyshine consideration, the dose rate to non-NEWs at the DGR fence line is below the dose rate target of 0.5 µSv/hr, and the dose rate to members of the public at the Bruce fence line is much smaller than the dose rate target of 10 µSv/year at the Bruce nuclear site boundary for members of the public.

Table 7-22: Worker External Dose Rates for Normal Operation

Scenario	Loc #	Location Description	Estimated Worker Dose Rate (mSv/hr)	Allowable Occupancy at Estimated Dose Rate [a] (hr/year)
1 (24 LLW in staging area within WPRB)	R3	Inside the package loading area (forklift driver moving waste packages,1 m away)	0.11 (non-processible boxed) 0.99 (feeder pipes)	90 (non-processible boxed) 10 (feeder pipes)
	R2	Standing outside the WPRB external walls [b]	2.9E-04 (non-processible boxed) 6.3E-03 (feeder pipes)	>2000 (non-processible boxed) 1590 (feeder pipes)
	R1	In the office/main control room (~20 m from source)	<1.6E-03 (non-processible boxed) <5.0E-03 (feeder pipes) [c]	>2000 (non-processible boxed) >2000 (feeder pipes)
	R4	On the roof directly above the source	3.6E-03 (non-processible boxed) 3.4E-02 (feeder pipes)	>2000 (non-processible boxed) 290 (feeder pipes)
2 (ILW in loading area within WPRB)	R3	Inside the package loading area (forklift driver moving waste packages, 2 m away)	0.23 (moderator resin) 0.048 (pressure tubes)	40 (moderator resin) 210 (pressure tubes)
	R2	Standing outside the WPRB [b]	1.4E-03 (moderator resin) 5.7E-03 (pressure tubes)	>2000 (moderator resin) 1750 (pressure tubes)
	R1	In the office/main control room (~25 m from source)	<5.0E-03 [c]	>2000
	R4	On the roof directly above the source	4.8E-03 (moderator resin) 7.7E-04 (pressure tubes)	>2000
3 (LLW in underground emplacement room)	-	Inside LLW emplacement room (forklift driver moving waste packages, 2 m away)	0.11 (non-processible boxed) 1.0 (feeder pipes)	90 (non-processible boxed) 10 (feeder pipes)
4 (ILW in underground emplacement room)	-	Inside emplacement room (forklift driver moving waste packages, 2 m away)	0.16 (moderator resin)	60

Notes:
a. Allowable occupancy without other mitigating measures, based on OPG occupational dose target of <10 mSv/year.
b. Based on concrete shielding wall around the staging area with thickness about 38 cm to ensure that the external dose rate outside of the WPRB is below 25 µSv/hr.
c. Office/main control room will be designed to ensure that workers in this location are exposed below 10 mSv/year dose target.

For example, the highest dose rate including direct and skyshine radiation is due to Scenario 1 (24 LLW boxes with feeder pipes in staging area within WPRB), and was calculated to be 0.03 µSv/hr at the DGR fence line and 9.4×10^{-6} µSv/hr (corresponding to 0.08 µSv/year) at the Bruce nuclear site fence line. The external dose rate including skyshine from non-processible boxed waste and ILW (moderator resins and retube wastes) is much smaller.

7.4.5 Assumptions and Uncertainty in Normal Operations Assessment

Table 7-23 summarizes the main uncertainties in assessing the impacts to the public and workers during normal operations, and how these have been addressed using conservative models and assumptions. See also Section 7.6, Contingency Planning.

7.5 Accident Assessment

The assessment focuses on accidents that are directly related to the wastes transferred to the DGR. Other hazardous events such as vehicle accidents or fires that could affect workers but do not affect waste packages, were not considered in this assessment. Also, conventional materials handling risks such as impact from a waste package are not assessed. These conventional hazards are addressed in the Preliminary Conventional Safety Assessment report (NWMO11ac).

7.5.1 Hazard Identification - Bounding Accident Scenarios

The hazard identification process was based on a systematic review of relevant site and facility features and processes in order to identify credible accident scenarios that could lead to harm. Events identified include the full range of events from possible to very low probability events. Hazard identification follows the following steps:

- Identification of sources of hazard;
- Identification of initiating events;
- Establishing potential hazardous events and identification of consequences;
- Identification and screening of accident scenarios; and
- Identification of bounding scenarios.

Table 7-23: Approach to Addressing Important Uncertainties in Normal Assessment

Exposure Pathways	Uncertainty	Approach
Air or Water Emissions on Public	Airborne H-3 and C-14 fractional release rate	• Based on WWMF experience; no credit for loss of airborne H-3 and C-14 over prior years; no credit for different conditions underground (no wind pumping). • Estimated release rates are comparable to those from other studies: o Estimated H-3 airborne release rate of 1%/year from the French medium activity waste (DOUCHE07). o Estimated C-14 airborne release rate of 0.07%/year, from drummed L&ILW in the Asse facility in Germany (BRACKE08).
	Airborne H-3 and C-14 release rate	• Conservative estimate based on maximum number of containers present with high H-3 and C-14 inventories.
	Waterborne H-3 and C-14 release rate	• Based on cautious assumptions with respect to equilibrium with underground water, and volumes of underground (sump) water. • The amount of moisture within the plenum and ventilation shaft was estimated to be 10 – 100 m^3/year. Upper end of the estimate was used to assess the release implications.
	Public dose rate	• Assumes that all H-3 is HTO. • Estimated by two methods, one based on DRL- pathway and the other based on measured REMP method. • The DRL-pathways was based on conservative assumptions with respect to air and water dispersion as well as the use of the most conservative calculated doses from all receptors. • Both methods yields very low annual dose.
Air Emissions on Worker	Airborne H-3 and C-14 concentrations	• Airborne concentrations in the repository were estimated based on maximum H-3 and C-14 inventory in any room and in any time. • Estimated maximum airborne concentrations are orders of magnitude below the DAC for a worker.
External Radiation on Worker and Public	External dose rate	• Based on maximum number of waste packages in a given location. • The waste packages assessed are higher dose rate packages, and no shielding was assumed for the workers. • Calculations were carried out with MicroShield, which has been found to be conservative relative to MCNP for the underground cases.

7.5.1.1 Step 1: Sources of Hazard

Sources of hazard are defined here as any material, equipment or process that has the ability to cause harm to a person through release of radiological/non-radiological species from wastes. Potential sources of hazard at the DGR are identified in Table 7-24. They are grouped into the following:

- Geology;
- Radioactive waste packages;
- Non-radioactive combustible materials;
- Heavy equipment; and
- Utilities.

Table 7-24: Potential Sources of Hazards at DGR Site

Group	Sources of Hazard
Geology	Methane
	Radon
	Aquifer [a]
	Stressed Rock
Radioactive Waste Packages	ILW Packages [b]
	LLW Packages [b]
	Gases including methane / hydrogen [c]
	Criticality
Non-Radioactive Combustible Materials	Diesel Fuel
	Electrical Cables
	Oil/ Lubricants
	Hydraulic Fluids
	Rubber (Tires)
Heavy Equipment	Forklift
	Transportation Truck
	Cranes
	Hoisting System
Utilities	Ventilation System
	Gas Cylinders
	Compressed Air
	Service Water
	Electrical Power

Notes:
a. Potential source of internal flooding of DGR.
b. Including combustible materials.
c. Gases generated from waste degradation.

7.5.1.2 Step 2: Initiating Events

A list of initiating events which could affect wastes at the DGR was then developed, based on considering the above sources of hazard, and three event categories:

- Operations initiating events – e.g., equipment failure, human error;
- Geotechnical initiating events – e.g., earthquake or rock fall; and
- External initiating events – e.g., severe weather conditions or aircraft crash.

The initiating event frequencies were also estimated within three broad categories:

- Possible events – annual frequency $>10^{-2}$;
- Unlikely events – annual frequency between 10^{-2} and 10^{-7}; and
- Non-credible events – annual frequency of $\leq 10^{-7}$.

The resulting list of events and their general frequency category is listed in Table 7-25.

Accident scenarios with an annual frequency of 10^{-6} or less are generally considered to be not credible. However, to accommodate the uncertainty in frequency estimates in this range, hazardous events with a frequency of 10^{-7} or less were considered non-credible. The risk from such accident scenarios was deemed to be acceptable, and they were screened out for further assessment.

Below are brief description of the initiating events, and their likelihood.

Mechanical/Equipment Failure

Equipment failure refers to any equipment malfunction other than cage fall and major vehicle accident that could potentially result in waste package breach or fire. This would include: forklift lift-mechanism break; cable or chain break; crane lift-mechanism break; electrical short or arcing; and compressed air pipe rupture. It is anticipated that minor failures could occur relatively frequently. However, the most likely consequence would be none on waste packages, or that the waste package would be stuck. Major failures while handling packages, including failures of safety features, would be much less likely. However, based on the large number of packages that could be handled, equipment failure is considered as a possible initiating event.

Table 7-25: Summary of the Initiating Events Considered

	Initiating Events	Frequency [a]
Operations Initiating Events	Mechanical/equipment failure	Possible
	Human error causing: • LLW package drop/hit • ILW package drop/hit [b] • Indoor fire • Inadequate package shielding	 Possible Unlikely Unlikely Unlikely
	Major vehicle accident	Unlikely
	Container failure	Unlikely
	Power failure (both grid and backup)	Unlikely
	Cage fall	Unlikely
	Criticality	Non-credible
	Explosion	Non-credible
Geotechnical Initiating Events	Major earthquake	Unlikely
	Rock fall/rock burst	Unlikely
External Initiating Events	Severe weather conditions: • Severe rainfall • Severe snow/ice • Severe wind • Lightning strike • Tornado	 Unlikely Unlikely Unlikely Unlikely (headframe) Non-credible (waste package) Non-credible
	Flooding (above ground)	Unlikely
	Flooding (underground)	Unlikely
	External fire	Non-credible
	Aircraft crash	Non-credible
	Meteor impact	Non-credible

Notes:

a. Possible events were assessed to have an annual frequency of $>10^{-2}$ of occurring at the DGR; Unlikely events have an annual frequency of between 10^{-2} and 10^{-7}; Non-credible events have an annual frequency $\leq 10^{-7}$.

b. Less likely than LLW package due to the much fewer ILW packages handled at DGR.

The ventilation system could fail due to fan or damper electrical or mechanical problems. This would not affect the package integrity, but could allow the local build up of flammable gases or radioactive gases. However, these gases would take days (radioactive gases) to months (flammable gases) to build up to hazardous levels. Since the ventilation flow is driven through the tunnels and rooms by a simple negative pressure maintained by fans at the ventilation shaft, and since the underground area is monitored for flammable gases and radioactivity, it is not credible that they would build up to hazardous levels before being detected. Nonetheless, ventilation system failure is considered as an unlikely initiating event.

LLW and ILW Package Drop/Hit due to Human Error

An examination of the WWMF station condition records from 1998 to 2006 identified various human error related incidents, including several cases with minor damage to packages during handling. None of these cases led to package drop or breach. Over the DGR operating life, the largest risk of package drop is with the LLW packages due to their large number. Package drop is considered as a possible initiating event for LLW and an unlikely initiating event for ILW.

Indoor Fire due to Human Error

During the maintenance work indoor (e.g., welding and cutting), it is possible that a fire could break out locally. However, for the size of the WPRB, the nature of the activities in it, the generally low amounts of combustible material (e.g., steel and concrete construction of WPRB), and the building fire detection and suppression system, the indoor fire is considered as an unlikely scenario.

Inadequate Shielding due to Human Error

Waste packages transferred to the DGR should be within the DGR normal handling dose rate limits. There is a risk of human error in transferring of waste packages with inadequate shielding. In particular, ILW resin liners will be transferred to the DGR in either an unshielded liner, or in one of three types of shield packages, so the correct shield needs to be identified and used.

An examination of the WWMF station condition records from 1998 to 2006 did not identify any cases where waste packages were improperly placed in the wrong packages leading to undiagnosed and inadequate shielding. In addition, monitoring and handling of ILW packages requiring shielding will be a well defined activity using trained operators and operating procedures, and Electronic Personal Dosimeters (EPDs) worn by DGR staff will provide additional monitoring redundancy for preventing inadvertent exposure. Thus, inadequate shielding is considered as an unlikely initiating event.

Major Vehicle Accidents

OPG has been moving LLW and ILW packages for over 40 years with no major vehicle accidents. The DGR packages will be moved entirely within the WWMF/DGR sites, which are access-controlled whereby the surface is level-graded and smooth, with minimal traffic. Major vehicle accidents involving collisions and roll-over are unlikely, but could potentially lead to a fuel leak and fire, or an electrical fault and fire, or cause a waste package to drop. Major vehicle accident is considered as an unlikely initiating event.

Container Failure

Containers will be inspected before shipment to the DGR, and will be placed in overpacks if necessary. Therefore, corrosion which compromises the structural integrity of the packages with consequential failure is not expected during handling or the initial open-panel operation period. Chemical or microbial reactions within the package could also cause gas pressure build up and container failure if the package is sealed. However, packages that are particularly susceptible to gas generation are not sealed. Package failure is an unlikely initiating event.

Power Failure

The DGR does not require active safety systems. It has on-site emergency power generators, but mostly for personnel safety. The failure of both the on-site grid and the DGR emergency generators is unlikely. However, even if it occurred, the loss of all power would have to extend beyond a few days before there could be potentially significant accumulation of radioactive gases in the DGR due to lack of ventilation, and much longer before accumulation of flammable gases.

Cage Fall

In modern mines, hoist or cage failures leading to cage falls are very unlikely. Relevant accidents from the U.S. mining records since 1981 include one case in 2000 with hoist cable break and runaway rail car, and one case in 1982 involving hoist winch clutch failure and cage fall (USMSHA09, KECOJEVIC07). The one relevant Canadian accident was in 1982, and involved a fire in a headframe (PCS Allen mine) causing failure of hoist ropes and cage fall.

The DGR will be built with current best practices. In particular, the hoist and cage design includes the following features:

- Koepe friction hoist with 6 ropes, to permit the loss of up to two ropes before the minimum rope safety factor is exceeded;

- Multiple brakes on the rope drum;
- Counter weight;
- Emergency safety "dogs" (brakes) on the cage;
- Holding clamps or locks at the station levels; and
- Cage arrestor system in shaft bottoms for overwinds.

In addition, it is expected that there will be routine inspection of hoist safety system, and lower frequency of DGR shaft operation (e.g., single shift) compared with typical mines.

Therefore, cage fall is considered as an unlikely initiating event, and the consequences of such an accident are evaluated as a bounding scenario.

Criticality

A criticality accident is not credible, because there will be very little fissile material in the DGR.

Explosion

Explosives used for underground construction activities would not be present in the facility during normal operation when waste packages are being received.

There are no commercial active rail lines or natural gas pipelines within 1 km of the DGR site. The two propane tanks that fuel the WWMF incinerator are about 300 m distant.

There is the possibility of slow ingress of methane from the rock, or methane/hydrogen from anaerobic corrosion or degradation of the wastes. However, these are slow processes due to the low rock permeability and low gas content, and the slow rate of the anaerobic reactions. Furthermore, the DGR uses a simple ventilation system based on fans drawing negative pressure at the ventilation shaft, such that there is always a flow of air from the tunnels and filled rooms, preventing large buildup of gases. The possibility of extended failure of the power and/or ventilation system is very unlikely. Finally, there are no ignition sources within an emplacement room, even if there were ventilation failure, so the combined probability is not credible. On closure of a panel, the ventilation is stopped but blast-resistant closure walls are installed so that any accident within the closed panel would not impact the rest of the DGR Facility.

Therefore, explosion-related accidents capable of damaging the DGR are not credible.

Major Earthquake

Southwestern Ontario and the Bruce region lie within the tectonically stable interior of the North American continent, which is a region characterized by low rates of seismicity. The above ground structures at the DGR will be built to meet the National Building Code of Canada. A major earthquake (beyond design basis) is an unlikely event, and the likelihood of both significant failure of the WPRB plus packages present is very unlikely. However, the consequence of severe seismic activity - collapse of the WPRB onto packages temporarily staged within the building is conservatively considered as a bounding accident scenario.

The potential damaging effects of seismic activities are smaller underground than at surface, due to the lower shaking that typically occurs at depth and the robust underground design basis. That is, it would take a larger earthquake to cause damage underground, and this would be very unlikely. Geomechanical stability analysis shows little to no failure of the rock around emplacement rooms with a 10^{-5}/year or 10^{-6}/year frequency earthquake (Section 5.4, NWMO11t). Rock failure after a panel had been closed would not cause operational releases because of the panel closure walls. Although unlikely, the consequences of an earthquake are assessed via an accident assuming the breach of several packages, either from rockfall or collapse of a stack of packages due to the shaking.

Rock Fall/Rock Burst

The main hazardous event regarding the rock fall would be overbreak within the roof of an emplacement room occurring during excavation or operations. This could result in crushing and breaching of some containers. Rock fall prevention is part of the engineering design and excavation sequence for the underground drifts and emplacement rooms. Therefore, it is considered unlikely, particularly, given the low level of seismic activity within the Bruce region. Rock burst is also considered unlikely because of the in situ stress condition and the nature of the sedimentary rock. However, an underground breach scenario is assessed.

Severe Rainfall

Severe rainfall (e.g., due to a hurricane) is an unlikely event. This could cause surface flooding, which is addressed in the discussion of above ground flooding below.

Severe Snow or Ice

Severe snow or ice is not explicitly considered, because the consequence is similar to severe wind - any potential consequence would be bounded by a roof collapse of the WPRB. Severe cold weather conditions are also not explicitly considered, because the consequences would be similar to equipment failure. It is further likely that above ground transfer operations would be suspended under extreme weather conditions.

Severe Wind

Above ground structures constructed at the site will meet all building code requirements including those for wind load. Wind speeds above the design basis may cause damage; however, the primary effects include local failure, e.g., window breakage, loss of roof panels, and water penetration of the building. Complete failure of the building is unlikely.

Lightning Strike

The above ground structures will be designed with lightning protection. Lightning striking the structures is unlikely but credible. The consequence of such an event is bounded by the WPRB room fire or power outage accident scenarios.

The likelihood of lightning strike on packages during transit is considered not credible due to the low target area of the packages, and the restriction on transport from the WWMF to DGR when there is lightning risk in the area.

Tornado

Tornadoes are very localized severe wind events. Based on the WPRB footprint of approximately 0.001 km^2 and the reported annual frequency of approximately 1 tornado per 10,000 km^2 in southern Ontario (Section 2.5.4.8), it is not credible for the WPRB to be hit by a tornado. Wind-generated missile striking a waste package is also not credible, as it would require a hurricane or tornado, as well as the generation of a missile that hits the waste package.

Flooding – Above Ground

Coastal flooding is not credible because the DGR is about 1 km from the shoreline and several metres above the lake level (NWMO11b).

Severe rainfall (e.g., associated with a hurricane) is an unlikely event; however, release of contaminants from waste packages in the WPRB due to flooding is not credible, because the WPRB floor will be above the maximum flood level (NWMO11b).

Flooding - Underground

Underground flooding during operations is very unlikely.

- The DGR is designed to keep the shaft collars and any air intake/outflow points above the maximum flood level (NWMO11b).
- The host rock has very low permeability and porosity, and will be fully excavated before operations begin.
- The upper permeable formations will be contained by the shaft liner, and also the possible installation of a grout curtain. This liner will be inspected and maintained.
- The service water pipe into the repository has automatic shutoffs in case the pipe should break.
- The shaft sumps are pumped; these pumps are connected to backup power.

However, the consequences of such an accident have been considered and are assessed to be minor for worker and public safety.

- The DGR design incorporates a sump pump system with a capacity much larger than the expected normal in-seepage rate. Leak rates up to this capacity (approximately 20 L/s) can be directly handled with no impact.
- The DGR shaft structures below the repository level represent a large volume that would need to be filled before water could spill into the access tunnels and then the emplacement rooms. This capacity is about 8500 m^3, including shaft bottoms and access ramps. This capacity provides some time for the leak to be sealed or significantly reduced, and/or additional pump capacity to be brought onto site, or for the workers to safety exit the underground.

External Fire

There will be no large forest fires near the DGR Facility, as there is no forest in the vicinity, nor are there large diesel or propane tanks, or roads/rails within 1 km where large amounts of flammable materials are carried. Therefore, the risk of an off-site external fire affecting the waste packages is not credible.

Other potential fires in the vicinity of the DGR could be from: operations and maintenance within the WPRB; electrical faults; maintenance activities such as welding; or lightning strikes. No large external fire potentially affecting waste packages was identified underground, because combustible materials are avoided or minimized in waste package handling areas; underground fuel storage will be kept in an area separated from the waste packages transfer route and the rooms; and diesel fuel will not be moved simultaneously with waste packages.

The most credible potential fire that could occur near waste packages is from the diesel-fuelled equipment handling waste packages. This type of fire could be due to a vehicle accident or an equipment malfunction, and is considered as an unlikely initiating event.

A fire in a waste package may spread to adjacent packages when several packages are placed together, i.e., in the emplacement rooms or the WPRB staging area, and if the fire cannot be extinguished in a short period of time. Combustible LLW and unshielded ILW resins could burn. However, all outer packages and pallets are made with non-combustible materials, so the risk of fire spreading is small. The structural integrity of a shielded ILW package will not be compromised by small fires. However, the water content of the ILW packages could reach the boiling point resulting in steam release. These accident scenarios are assessed.

The risk of a self-ignited internal fire within a waste package is not credible for the following reasons: the radioactivity in flammable wastes is too low to heat the wastes; there are no chemically reactive wastes such as oxidizing agents; and the wastes do not have the right combination of combustible material, moisture and surface area to support spontaneous combustion.

Aircraft Crash

The Bruce region has low levels of general aviation – typically small non-commercial, non-military aircraft. The nearest fields are at Kincardine and Port Elgin airport, about 16 km distant. Using the U.S. DOE approach (USDOE06), the aircraft crash frequency can be estimated based on:

- Number of flight operations in area;
- Probability that an aircraft will crash during a flight operation; and
- Conditional probability that the aircraft crashes into the facility.

Based on the U.S. DOE approach (USDOE06), and since the local airports are small and distant, the risk of aircraft crash into the DGR can be estimated based on a general aviation crash rate of about 10^{-4} per square mile per year and a DGR structure footprint of 0.0004 square miles, giving 10^{-4} x 0.0004 ~ 4 x 10^{-8} per year risk of impact.

Therefore, an aircraft crash accidentally impacting on the above ground structures (WPRB and main or ventilation shaft headframe) is not credible, due to small footprint of the above ground structures and low levels of general aviation in the Bruce region.

Meteor Impact

Meteor impact is not credible due to the small footprint of above ground structures and the very low likelihood of large meteors (capable of damaging rock to 680 m) hitting the site.

7.5.1.3 Step 3: Hazardous Events and Consequences

The assessment then considered the potential consequences of each of the credible initiating events interacting with potential sources of hazard, in terms of either direct release of wastes or radioactivity, or of causing another event that could in turn release wastes. The ultimate consequences that could release wastes or radioactivity were:

- Fire causing waste packages to burn or to release radioactive volatile gases;
- Breach of waste packages; and
- Inadequate package shielding.

An example of some consequential events is given in Table 7-26. Approximate conditional probabilities for these consequences are also noted.

Table 7-26: Example List of Consequences or Conditional Events

Initiating Event	Consequence	Conditional Probability
Major Vehicle Accident	Vehicle fire	0.001 [a]
	Waste package drop	1 [b]
Small Fire (Vehicle, Equipment)	Waste fire – combustible waste, unshielded package	1 [b]
	Waste fire – combustible waste, shielded package	10^{-5} [c]
	Steam release from shielded package	1 [b]
	Waste fire – non-combustible waste	10^{-7} [d]
Waste Package Drop	Waste package breach – unshielded package	1 [b]

Notes:
a. From U.S. Department of Transport statistics (USDOT06).
b. Conservative assumption.
c. Very unlikely due to shielding, which provides thermal inertia.
d. Incredible for non-combustible waste.

7.5.1.4 Step 4: Identification and Screening of Accident Scenarios

Potential accident scenarios involving an initiating event, and potential consequences were considered. The result of this analysis is a list of specific accident scenarios. For example, a major vehicle accident while carrying LLW bins above ground (the initiating event), causing a vehicle fire (diesel fuel, rubber tires as a source of hazard), which then causes the LLW to burn, releasing radioactivity.

Based on the frequency of the initiating events and the conditional probability of the consequence(s), resulting accident scenarios which are non-credible were screened out. For example, a major vehicle accident while carrying a shielded ILW package (unlikely, < 10^{-2}/year) leading to a vehicle fire (0.001), leading to waste fire in shielded package (10^{-5}) is not credible (< 10^{-7}/year).

Combinations of events were also considered. Most combinations are not credible in terms of frequency, unless they have a common cause. Combinations of package fire/breach plus ventilation system failure are a particular possibility - for example, an earthquake could cause package drop plus loss of power leading to ventilation failure. In this case, worker exposure due to package drop could be higher (although the ventilation fans would still take a few minutes to stop spinning even after losing power). But public exposure would be less due to the stopping of ventilated release of contaminants, and so the failure of the ventilation system simultaneously during other accidents was not considered. Failure of the ventilation system itself was considered for workers.

7.5.1.5 Step 5: Selection of Bounding Accident Scenarios

The resulting credible accident scenarios can be categorized into the following accident types, which are described below.

- **Fire:** External fires may cause the content of some waste packages to ignite and burn, mainly LLWs and unshielded ILW packages. Shielded ILW packages are unlikely to ignite, but the heat from an external fire can cause release of steam and volatile species.

- **Container Breach (Low Energy):** Low-height or low-speed impacts resulting in some loss of containment. Waste packages are not crushed. Includes low-speed transfer vehicle accidents, and drops from heights lower than 4 m (USDOE07).

- **Container Breach (High Energy):** Drops or impacts that result in significant package failure. Includes drops from heights greater than 4 m (USDOE07), cage fall, and roof collapse.

- **Inadequate Shielding:** Inadvertent exposure of staff to high dose rate conditions.

- **Ventilation System Failure:** Loss of ventilation underground due to loss of power.

Within each of these accident types, bounding accident scenarios were identified. The bounding accident scenarios were based on the qualitative estimation of the magnitude of the consequences (not the likelihood) which, in turn, is a function of the waste category, the number of waste packages affected and the location of the hazardous event.

Table 7-27 and Table 7-28 list the set of bounding accident scenarios identified for above ground and underground accidents, respectively.

As a check, the list of potential accidents was compared with the accidents considered in the WWMF and the U.S. Waste Isolation Pilot Plant (WIPP) safety reports. No missing accidents or event combinations were identified.

In order to quantitatively assess the potential consequences of these accidents, specific waste packages needed to be identified. Therefore, the wastes were grouped into similar categories in terms of characteristics, and representative waste categories selected from each category.

- Ash LLW (spillable, not combustible, containing potentially chemically hazardous elements) – Bottom ash selected as these have the highest radiological inventory of ash waste packages.

- Combustible LLW (combustible) – Box compacted waste selected since these have higher package radiological inventory.

- Non-Processible/Other LLW (not readily spillable or combustible, largest volume of wastes) – Non-processible boxed waste selected as these are the largest volume of waste, and non-processible drummed waste as these have the highest LLW package radiological inventory.

- Resin/Filter ILW (spillable, potentially combustible) – Moderator resins selected as these have the highest radiological inventory (especially C-14 and H-3).

- Retube (not spillable, not combustible, activated metal) – End fittings selected as these have the highest radiological inventory.

Although retube waste packages are robust and designed not to fail under accident conditions, including drop from stacking height (OPG06a), they are considered in high energy breaches due to cage fall in the underground.

Table 7-27: List of Potential Accidents in the DGR Above Ground Operations

<table>
<tr><th>Accident Type</th><th>Bounding Scenario</th><th>Selected Waste Category</th><th>Number of Packages at Risk</th></tr>
<tr><td rowspan="10">Fire</td><td rowspan="4">Outdoor Waste Package Fire</td><td>Box Compacted</td><td>8</td></tr>
<tr><td>Non-Processible Boxed</td><td>8</td></tr>
<tr><td>Non-Processible Drummed</td><td>8</td></tr>
<tr><td>Moderator Resin (Unshielded)</td><td>1</td></tr>
<tr><td rowspan="6">Indoor Waste Package Fire</td><td>Box Compacted</td><td>24</td></tr>
<tr><td>Non-Processible Boxed</td><td>24</td></tr>
<tr><td>Non-Processible Drummed</td><td>24</td></tr>
<tr><td>Moderator Resin (Unshielded)</td><td>1</td></tr>
<tr><td>Combined LLW and ILW Packages</td><td>24 Non-Processible Drummed + 2 Moderator Resin (Unshielded)</td></tr>
<tr><td>Moderator Resin (Shielded)</td><td>1</td></tr>
<tr><td rowspan="13">Low Energy Breach</td><td rowspan="6">Outdoor Waste Package Breach</td><td>Bottom Ash (Old)</td><td>8</td></tr>
<tr><td>Box Compacted</td><td>8</td></tr>
<tr><td>Non-Processible Boxed</td><td>8</td></tr>
<tr><td>Non-Processible Drummed</td><td>8</td></tr>
<tr><td>Moderator Resin (Unshielded)</td><td>1</td></tr>
<tr><td>Moderator Resin (Shielded)</td><td>1</td></tr>
<tr><td rowspan="7">Indoor Waste Package Breach</td><td>Bottom Ash (Old)</td><td>24</td></tr>
<tr><td>Box Compacted</td><td>24</td></tr>
<tr><td>Non-Processible Boxed</td><td>24</td></tr>
<tr><td>Non-Processible Drummed</td><td>24</td></tr>
<tr><td>Moderator Resin (Unshielded)</td><td>1</td></tr>
<tr><td>Combined LLW and ILW Packages</td><td>24 Non-Processible Drummed + 2 Moderator Resin (Unshielded)</td></tr>
<tr><td>Moderator Resin (Shielded)</td><td>1</td></tr>
<tr><td>Other</td><td>Inadequate Shielding</td><td>Moderator Resin</td><td>1</td></tr>
</table>

Table 7-28: List of Potential Accidents in the DGR Underground Operations

Accident Type	Bounding Scenario	Selected Waste Category	Number of Packages At Risk
Fire	Waste Package Fire During Transfer	Box Compacted	1
		Non-Processible Boxed	1
		Non-Processible Drummed	1
		Moderator Resin (Unshielded)	1
	In Room Waste Package Fire	Box Compacted	2400
		Non-Processible Boxed	2400
		Non-Processible Drummed	2400
		Moderator Resin (Unshielded)	1200
		Moderator Resin (Shielded)	1
Low Energy Breach	Waste Package Breach During Transfer	Bottom Ash (Old)	1
		Box Compacted	1
		Non-Processible Boxed	1
		Non-Processible Drummed	1
		Moderator Resin (Unshielded)	1
		Moderator Resin (Shielded)	1
	In Room Waste Package Breach	Bottom Ash (Old)	3
		Box Compacted	4
		Non-Processible Boxed	5
		Non-Processible Drummed	5
		Moderator Resin (Unshielded)	4
		Moderator Resin (Shielded)	3
High Energy Breach	Cage Fall with Waste Package Breach	Bottom Ash (Old)	2
		Box Compacted	2
		Non-Processible Boxed	3
		Non-Processible Drummed	3
		Moderator Resin (Unshielded)	2
		Moderator Resin (Shielded)	1
		Retube- End Fittings	1
Loss of Ventilation	Ventilation System Failure	All Waste	-

7.5.2 Radiological and Non-Radiological Species of Potential Concern

The radionuclides and non-radiological species of potential concern were first identified through a conservative screening assessment, which took into account all species in the inventory in all waste categories present.

The screening assessment considered a non-fire and a fire scenario. For radionuclides, the groundshine (from waste spilled on ground) and inhalation pathways were considered for the non-fire scenario, while the inhalation pathway was considered for the fire scenario. For non-radiological species, the screening was based on a comparison of air concentrations to an acute occupational health and safety criterion representing the inhalation, ingestion, skin or eye contact or skin absorption pathways. All species tracked in the inventory were evaluated. Thirty-one radionuclides and nineteen non-radiological species were identified for further evaluation under various accidents scenarios for the selected waste categories.

- The 31 "screened in" radionuclides are: Am-241, C-14, Ce-141, Ce-144, Cm-244, Co-60, Cs-134, Cs-137, Eu-152, Eu-154, Fe-55, Fe-59, Gd-153, H-3, La-140, Mn-54, Nb-94, Nb-95, Pb-210, Pu-238, Pu-239, Pu-240, Pu-241, Ra-226, Ru-106, Sb-124, Sb-125, Sn-119m, Sr-90, Te-125m, and Zr-95.

- The 19 "screened in" non-radiological species are: asbestos, antimony, arsenic, barium, beryllium, cadmium, chromium, cobalt, copper, lead, manganese, mercury, nickel, selenium, strontium, uranium, zinc, zirconium, and dioxins/furans.

7.5.3 Methodology for Consequence Assessment

The potential bounding accidents were analyzed for consequences using simple models, consistent with the U.S. DOE methodology (USDOE94, USDOE05, USDOE07). For each accident scenario, the amount (or source term) of radionuclides and/or non-radiological species potentially impacting the receptor was calculated for the selected waste category and for each radionuclide and/or non-radiological species. The following equation presents the calculation of the source terms:

$$Q = MAR \times DR \times ARF \times RF \times LPF \quad (7\text{-}5)$$

where:

Q = Source term (Bq or µg)

MAR = Material at risk (Bq or µg) – Maximum amount of material present that may be acted upon with the potentially dispersive energy source

DR = Damage ratio – Fraction of MAR actually impacted by the accident condition

ARF = Airborne release fraction – Fraction of radioactive material actually impacted by the accident condition that is suspended in air

RF = Respirable fraction – Fraction of airborne particles that are in the respirable size range

LPF = Leakpath factor – Fraction of the release not attenuated along the leak path (e.g., by deposition as the contaminants move from within the container to the outside of the container)

7.5.3.1 Source Term Parameters

Material at Risk

The source term represents a reasonable maximum for a given process or activity (USDOE05, USDOE07). The Material At Risk (MAR) quantity is the total radionuclide/non-radiological species amount that can be exposed during the accident.

For all non-fire accident scenarios, the MAR can be calculated by multiplying the concentration of a given radionuclide or non-radiological species in the waste with the total amount of waste that can be affected (i.e., MAR = maximum number of packages that can be affected in bounding scenario x net waste volume per package x concentration of radionuclide/non-radiological species in waste).

The bounding scenarios evaluated in this report consider the maximum number of packages that could be at risk in an accident (Table 7-27 and Table 7-28).

Fire Release Rate & Duration

For all fire accident scenarios, the rate of release of contaminants is needed, and is calculated using the following equation:

$$QR = Q / T_{FD} \tag{7-6}$$

where:

QR = Source term release rate (Bq/s or µg/s) during fire accidents only

Q = Source term (Bq or µg) (See Equation 7-5)

T_{FD} = Fire duration (s), time taken to burn all affected waste.

The maximum fire duration is the time required to burn the combustible materials in the affected wastes (M x DR). Table 7-29 shows the total amount of combustible materials in wastes. DR values are given in Table 7-32. In practice, the fire duration could be less because the fire is extinguished; however, the rate of release during the burn period is the same. The maximum fire duration was calculated using Equation 7-7 to 7-9 below.

$$T_{FD} = (M \times DR) / BR \qquad (7\text{-}7)$$

$$BR = m''S \quad \text{for outdoor fire} \qquad (7\text{-}8)$$

$$BR_{O2} = m''S \times (O_{2s} / O_{2r}) \quad \text{for underground fire} \qquad (7\text{-}9)$$

where:

M = Dry combustible materials mass (kg)

BR = Burn rate (kg/s)

BR_{O2} = Oxygen limited burn rate (kg/s)

m'' = mass burn flux (kg/(m^2·s))

S = Exposed surface area of the fire (m^2)

O_{2s} = Rate of oxygen supplied for burning (m^3/s)

O_{2r} = Rate of oxygen required for burning (m^3/s)

The burn rate depends on the mass burn flux and the exposed burning surface area. The burn rate may also be limited by the availability of oxygen for an underground fire.

The mass burn flux is an empirical parameter derived by dividing experimental data on heat release rate per unit area for combustible materials (USNRC04). The mass burn flux ranges from around 0.01 kg/(m^2·s) for plywood; 0.015-0.03 kg/(m^2·s) for various common plastics, rubber and nylon; to 0.1 kg/(m^2·s) for particle board or short stacked wooden pallets. The mass burn flux of 0.026 kg/(m^2·s) (polystyrene) was used in the release rate calculation, as the resins are made up of polystyrene backbones and this is a mid-range burn flux value.

Table 7-29: Combustible Materials and Exposed Surface Area for Fire Calculations

Waste Category	Number of Packages	Combustible Materials (kg)	Exposed Surface Area (m^2)
Box Compacted	1	1960	2.1
	8	15,600	16.5
	24	46,900	24.7
	2400	4,690,000	3060
Non-Processible Boxed	1	340	2.6
	8	2720	20.7
	24	8170	31.1
	2400	817,000	3220
Non-Processible Drummed	1	360	2.6
	8	2880	20.7
	24	8640	31.1
	2400	864,000	2950
Moderator Resin (Unshielded)	1	1530	2.1
	1200	1,840,000	4090
Moderator Resin (Shielded)	1	3060	2.1

The rate of oxygen supplied for burning (i.e., O_{2s}) is calculated by multiplying the ventilation rate by the fraction of O_2 in air (0.21). For complete combustion of one kg of polystyrene, 95.5 moles of O_2 are required or 2.33 m^3 O_2 per kg of polystyrene. Therefore, O_{2r} is estimated by multiplying the product of m" and S by 2.33.

The exposed burning surface area was based on the surface area of the affected waste containers. For a confined fire, the surface area was assumed to be the top surface area of the packages involved. For an unconfined fire, the surface area was assumed to be all exposed surfaces of the stacked package configuration. See Table 7-29 for the exposed surface area.

For example, for a fire involving one unshielded moderator resin container, the maximum fire duration is estimated to be 3.9 hours based on 1530 kg of combustible materials (dry resins), the top surface area of 2.1 m^2, and DR of 0.5. The heat release rate is 2.3 MW.

The estimated fire sizes range in magnitude from around 2 MW for a single package fire, to about 70 MW for a room fire (oxygen limited). This roughly corresponds to the heat release range from a small car fire to a large truck fire in a tunnel (NILSEN09).

Radionuclide Concentrations in Waste

The radionuclide concentrations for the DGR accident assessment in Table 7-30 are conservatively defined as follows:

- LLW (Bottom Ash, Box Compacted, Non-Processible) – The inventories are the "as-received at WWMF" radionuclide concentrations from the Reference L&ILW Inventory report (OPG10a). This is conservative since these as-received LLW concentrations correspond to about the 80th percentile of as-received LLW package dose rates. Furthermore, this assumes that the wastes have arrived at DGR from stations with limited decay time, and does not consider that over 70% of the LLW will already have been in storage at WWMF for many years before they are transferred to the DGR.

- Moderator Resins (Shielded) – The radionuclide concentrations corresponds to as-received resins direct from stations that would be in a shielded resin package.

- Moderator Resins (Unshielded) - For assessing accidents involving unshielded resin liners, the above as-received concentrations were decayed by 10 years to account for some storage at WWMF so the external dose rate would be reduced for handling in an unshielded resin liner. While the key gamma dose contributors like Co-60 decay during this period, the key accident inhalation dose contributors like C-14 do not significantly decay.

- Retube Waste (End Fittings) – These wastes would generally be allowed to decay prior to receipt at DGR in order to reduce their heat load, and possibly to meet dose rate requiremenst for transportation from the nuclear stations. The reference concentrations in Table 7-30 include 15 years total decay after removal from the reactor.

Finally, while these radionuclide concentrations represent conservative estimates, values might be higher in some packages due to variability between packages and uncertainties in the inventory estimates. Therefore, for accidents involving a small number of packages, these inventories were further increased by a factor of 10 to represent a maximum package radionuclide inventory.

Table 7-30: Concentrations (Bq/m^3) of Key Radionuclides in Selected Waste Categories

Radionuclide	Bottom Ash (Old)	Box Compacted	Non-Processible Boxed	Non-Processible Drummed	Moderator Resin [a]	Retube- End Fittings [b]
Am-241	1.7E+05	3.0E+05	4.4E+05	2.4E+01	-	-
C-14	8.1E+06	6.7E+06	7.3E+05	1.5E+07	2.7E+12	-
Ce-141	-	-	-	-	1.2E+09	-
Ce-144	-	-	-	-	5.3E+08	-
Cm-244	4.9E+04	1.3E+05	7.9E+03	3.5E+00	-	-
Co-60	1.6E+08	8.3E+07	2.0E+07	4.4E+07	5.1E+10	1.3E+13
Cs-134	7.6E+06	6.8E+06	3.0E+05	2.8E+06	4.1E+08	8.5E+06
Cs-137+ Ba-137m	1.2E+08	7.6E+07	7.8E+08	8.8E+07	3.6E+08	3.4E+02
Eu-152	-	-	-	4.4E+04	1.2E+09	6.5E-05
Eu-154	2.8E+06	2.3E+06	-	4.8E+04	6.4E+08	1.2E-02
Fe-55	1.0E+09	3.2E+05	9.6E+07	8.4E+07	1.4E+10	4.6E+13
Fe-59	-	-	-	-	-	2.5E-24
Gd-153	-	-	-	-	5.8E+11	-
H-3	2.5E+07	2.8E+11	3.0E+10	6.1E+11	1.4E+11	6.2E+09
La-140	2.6E+08	-	-	-	-	-
Mn-54	-	-	-	-	-	1.6E+08
Nb-94	1.4E+06	1.0E+06	-	1.1E+05	-	3.6E+08
Nb-95	-	-	-	-	8.5E+08	3.3E-31
Pb-210	-	-	3.0E+06	-	-	-
Pu-238	6.4E+04	6.1E+04	5.4E+03	3.8E+00	9.5E+02	-
Pu-239	1.0E+05	1.3E+05	1.0E+04	1.2E+01	1.3E+03	-
Pu-240	1.5E+05	1.8E+05	1.5E+04	1.7E+01	1.9E+03	-
Pu-241	2.6E+06	4.9E+06	5.4E+05	5.2E+02	3.7E+03	-
Ra-226	-	-	8.2E+04	-	-	-
Ru-106	3.7E+07	6.1E+07	1.0E+06	7.0E+05	1.4E+09	2.8E-12
Sb-124	-	-	-	-	3.1E+08	1.3E-15
Sb-125	1.6E+07	1.4E+07	7.8E+05	1.3E+06	9.9E+08	5.0E+09
Sn-119m	-	-	-	-	-	2.1E+06
Sr-90+Y-90	8.2E+07	3.6E+06	1.5E+07	1.7E+07	2.8E+07	4.9E+02
Te-125m	-	-	-	-	-	2.3E-09
Zr-95	-	-	-	-	-	1.4E-16
Net Container Volume (m^3)	3.4	2.3	2.5	1.2	3	2.7

Notes:
From Reference L&ILW Inventory report (OPG10a).
a. Resins will be either unshielded or shielded. The values given above are for shielded resins. Concentrations in the unshielded resins were obtained by decaying these values for 10 years.
b. Retube end fittings inventory was decayed a further 10 years from the already 5-year decayed values presented in Table B-3 of the Reference L&ILW Inventory report.

Non-Radiological Elements and Species Concentrations in Waste

The concentration of non-radiological species in the waste (Table 7-31) was calculated by dividing the total amount in the Reference L&ILW Inventory report (OPG10a) by the total waste volume in each waste category. No additional conservatism was included for the non-radiological elements, considering (1) the general applicability of the source term parameters (Tables 7-33 and Table 7-34) to surface contaminated materials rather than to bulk materials containing non-radiological elements, (2) the direct data on chemical composition for most waste types, (3) the conservatism in the criteria (worker IDLH values are based on 30-minute exposure, while workers are likely exposed for less than 5 minutes), and (4) that in fire accidents the smoke (e.g., CO and volatile organics) would likely be a greater hazard. The results provide information on the relative risks of the various package related accidents in terms of non-radiological species.

Table 7-31: Concentrations of Non-Radiological Species in Selected Waste Categories (kg/m^3)

Non-Radiological Parameters	Bottom Ash (Old)	Box Compacted	Non-Processible	Moderator Resin	Retube-End Fittings
Antimony	6.8E-01	7.3E-02	7.2E-03	3.7E-04	8.7E-03
Arsenic	8.2E-02	9.2E-03	9.0E-04	2.4E-04	5.4E-02
Barium	2.0E+00	2.3E-01	2.3E-02	5.0E-03	3.9E-06
Beryllium	-	-	1.7E-03	4.3E-04	1.9E-06
Cadmium	4.2E-03	4.9E-01	4.8E-02	6.2E-03	9.7E-05
Chromium	4.7E+00	5.2E-01	5.9E+00	1.1E-02	7.0E+01
Cobalt	8.2E-02	9.2E-03	8.7E-04	5.2E-04	1.2E-01
Copper	1.6E+01	1.7E+00	2.9E+01	3.8E-01	1.3E+00
Lead	8.8E+00	9.8E-01	2.3E+01	4.8E-02	9.7E-04
Manganese	2.7E+00	2.7E+00	3.8E-01	1.3E-02	4.6E+00
Mercury	1.4E-02	2.1E-03	3.3E-04	3.2E-05	2.9E-05
Nickel	1.4E+00	1.5E-01	3.8E-01	2.4E+00	1.6E+00
Selenium	-	3.5E-03	3.5E-04	4.3E-04	6.8E-05
Strontium	6.8E-01	7.3E-02	9.0E-03	2.8E-04	5.8E-02
Uranium	-	-	5.0E-03	4.8E-05	2.9E-06
Zinc	2.7E+01	3.1E+00	6.6E-01	2.1E-01	6.8E-03
Zirconium	1.9E-01	2.0E-02	2.1E-03	1.2E-04	1.3E-02
Asbestos	-	-	4.5E+00	0.0E+00	0.0E+00
Dioxins & Furans	5.2E-05	0.0E+00	0.0E+00	0.0E+00	0.0E+00

Note:
From Reference L&ILW Inventory report (OPG10a).

Fractions and Factors

The following sections discuss the selection of values for Damage Ratio (DR), Airborne Release Fraction (ARF), Respirable Fraction (RF), and Leakpath Factor (LPF) that were used for the calculation of source terms.

The basis for the assigned values is the U.S. DOE Handbook (USDOE94) and the subsequent U.S. DOE standard (Section 4.4 for DR and Section 4.5 for ARF/RF, USDOE07). Values used in the WIPP (USDOE05) and Texas LLW facility (WCS07) assessments were also reviewed.

DR

DR is that fraction of material that is actually impacted by the accident conditions. DRs are based on container/package robustness, as well as waste form and accident type. Table 7-32 provides the DR values used in the source terms.

Table 7-32: Selection of DRs

Waste Category		Fire [a]	Breach	
			Low Energy	High Energy
LLW	Bottom Ash	Not combustible	0.25	1
	Box Compacted	0.5	0.1	1
	Non-Processible Boxed	0.5	0.1	1
	Non-Processible Drummed	0.5	0.05	1
ILW	Moderator Resin (Unshielded)	0.5	0.1	1
	Moderator Resin (Shielded)	0.1	0.05	0.5 [b]
	Retube- End Fittings	Not combustible	0.05	0.5 [b]

Notes:
a. Value for confined burning. For unconfined burning, DR=1.
b. Value for general high energy breach. For breach due to drop in shaft (i.e., cage fall), DR=1.

For fires, the DR is 1.0 for unconfined waste burning, 0.5 for confined burning in single-walled packages, and 0.1 for burning in overpacked double-walled containers. In unconfined burning, lid loss occurs and the contents are exposed. In confined burning, the lid remains in place and gases escape around the lid. In the DGR, the LLW disposal-ready packages are steel-walled and the lid is not gas-tight, so they are expected to burn confined. The non-processible drummed waste is double-contained in both drums and the drum rack overpacks. However, the DR for non-processible drummed waste is conservatively based on 0.5.

Unshielded moderator resins are in a single-walled container; shielded moderator resins are in overpacked concrete-shield double-walled packages.

For low-energy breaches, such as low-speed collisions, low height drops, and puncture, single-walled packages are assigned a DR of 0.1. For overpacked, double-walled packages, the damage is less and DR = 0.05. The exception is bottom ash, where due to the more mobile nature of the waste form, the DR is 0.25, taking partial credit for the overpacking of ash bins.

For high-energy breaches, the DR is set to unity for the single-walled containers, and 0.5 for shielded resins due to the robust containers. Full drop of package in shaft is a special case, in which DR = 1 for all packages.

ARF

The ARF is based on the waste form and accident type. Table 7-33 provides the ARF values used in the source terms.

Table 7-33: Selection of ARFs

Waste Category		Fire [a]	Breach	
			Low Energy	High Energy
LLW	Bottom Ash	Not combustible	0.002	0.01
	Box Compacted	0.001	0.001	0.01
	Non-Processible Boxed	0.001	0.001	0.01
	Non-Processible Drummed	0.001	0.001	0.01
ILW	Moderator Resin (Unshielded)	0.001	0.001	0.01
	Moderator Resin (Shielded)	0.0005	0.001	0.01
	Retube- End Fittings	Not combustible	0.0001	0.001

Note:
a. Value for confined burning and non-volatiles; for volatiles, ARF = 1; ARF = 0.01 under unconfined burning, except for release of bulk metals (copper, lead, chromium) and asbestos (non-processible waste) where ARF = 0.001.

For fires, the bounding values for ARF for contaminant release from confined burning of combustible waste is 0.0005, and for unconfined burning the ARF = 0.01, based on U.S. DOE models (consistent with WIPP, Chapter 3.4.1.2 of USDOE05; Table 4.5-1, USDOE07). Although the DGR waste packages are not expected to lose their lids since they are not gas-tight and therefore gas pressures will not generally build up to high pressure, it is assumed that some fraction of lids are lost for single-walled packages, and therefore ARF = 0.001 is applied to the LLW and unshielded ILW resins, while ARF = 0.0005 is applied for double-walled packages such as shielded resins.

However, during full room fire underground, an ARF of 0.01 for an unconfined fire is assumed, since this is an extreme fire and it is assumed that there may also be some collapse of stacked packages.

The above values apply in particular to release of surface contamination (USDOE07). This is generally applicable for radionuclides in combustible DGR wastes. However for evaluation of potential chemical hazards it may not be applicable, because in some wastes, the material is in a bulk form (e.g., lead blocks, asbestos blankets, copper piping, and steel parts) and the above ARFs would overestimate the amount that would be released from these materials in a fire. The release of particulates formed by oxidation in fires of metals and of powder from non-metallic solids are reported to have ARF ~ $3x10^{-5}$ to 0.001 in unconfined burning situations (Ch. 4.1, USDOE94; MISHIMA08), at least an order of magnitude less than above. Therefore, for bulk non-combustible metals (lead, copper, chromium) and asbestos in non-processible waste, an ARF of 0.001 is applied.

The ARF for volatile elements, notably H-3, C-14, mercury, and selenium, are assumed to be 1 for fire scenarios.

For low-energy breach accidents (drop, puncture, crush), the ARF is 0.001 for surface-contaminated materials (USDOE05, consistent with WIPP). The exception are ash for which the ARF is assumed to be 0.002 due to the finer nature of this material, and retube waste for which the ARF is assumed to be 0.0001 due to bulk activation material.

For high-energy breach accidents, U.S. DOE models (USDOE07) are based on suspension of bulk powders from shock impacts due to falling debris from structural collapse or external energy. A conservative ARF value of 0.01 is assumed for most wastes, except for retube waste where the contamination is bulk activation and 0.001 is selected.

RF

The RF of particulates is used only in the calculation of impacts of radionuclides and non-radiological species through the inhalation pathway (i.e., they are not credited during the calculation of immersion dose rates from radionuclides). The RF is based on the waste category characteristics and accident type. Table 7-34 gives the RF values used in the source terms.

For fire accidents it is conservatively assumed that fine particles are released and RF=1.

For low-energy breach accidents, the RF is generally 0.1 for most accidents. This is based on values from suspension of debris impacting powder in cans. For high-energy breach accidents, a higher bounding RF value of 0.2 is assessed. The bounding RF value for free fall of cohesionless powders is 0.3. This value is adopted for bottom ash.

H-3, C-14, mercury, and selenium are potentially volatile elements, and released amounts are assumed to be completely respirable, irrespective of accident scenario, i.e., RF = 1.

LPF

The LPF is the fraction of the release not attenuated along the leak path, e.g., by deposition as the contaminants move from within the container to the outside of the container (USDOE05).

A LPF of 1 is used for all unmitigated accidents except for cage fall and roof collapse. The assumption of LPF = 1 is conservative. In particular, the leakage of any species from an underground accident would involve passing through several hundred meters of tunnels, with potential for particles to impact/deposit on surfaces and then become fixed in place. The assumption of LPF = 0.1 for the specific roof collapse and cage-fall scenarios is justified, because the debris from these accident scenarios is expected to attenuate the source term.

Table 7-34: Selection of RFs

Waste Category		Fire [a]	Breach [a]	
			Low Energy	High Energy
LLW	Bottom Ash	Not combustible	0.3	0.3
	Box Compacted	1	0.1	0.2
	Non-Processible Boxed	1	0.1	0.2
	Non-Processible Drummed	1	0.1	0.2
ILW	Moderator Resin (Unshielded)	1	0.1	0.2
	Moderator Resin (Shielded)	1	0.1	0.2
	Retube- End Fittings	Not combustible	0.1	0.2

Note:
a. All released volatiles are assumed to be respirable (i.e., RF = 1), irrespective of accident type.

7.5.3.2 Exposure Duration

Worker exposure under accidents assumes that the worker remains in the vicinity of the accident for 5 minutes. At surface, there are multiple exits. Underground, there are multiple emergency refuge stations, with a portable station always kept close to the main work areas, so workers do not have to travel far to reach safety or obtain protective equipment. In addition, there are multiple detection systems that would alarm (smoke, radioactivity), and multiple communication systems to notify all workers, including a stench gas system in which a smelling agent is released into the ventilation system. Furthermore, worker training on accident response, availability of Personal Protective Equipment (PPE) for specific tasks, and the possibility of simply moving upstream from the release point, would all help limit the worker exposure duration.

Public exposure under accidents assumes that the person is in the direct plume path at the closest Bruce nuclear site boundary, and is exposed there for one hour. It is assumed that at longer times, the release will have been terminated (e.g., fire put out or isolated by firedoors or temporary walls, or ventilation turned off, or in case of breach the bulk of the cloud will have dissipated). Impacts of longer fire exposures are specifically discussed later.

7.5.3.3 Dispersion Modelling for Releases

The following sections summarize the methodology for dispersion calculations for outdoor and indoor accident scenarios.

Outdoor Dispersion for Short Distances for Worker Exposure

Breach Accidents

For an outdoor breach accident, the dispersion of released contaminants depends on the air conditions. The worker exposure is affected by their location relative to source. Also, they are assumed to be moving during the exposure period, and will be exposed to some averaging of the dispersing cloud. The worker is exposed to an average concentration estimated by:

$$C_{WO} = Q / V_{AIR} \text{ (breach, outdoors)} \quad (7\text{-}10)$$

where:

C_{WO} = Worker outdoor air concentration (Bq/m^3 or µg/m^3)

V_{AIR} = Effective volume of air (m^3)

The effective volume of air over which the worker is exposed can be estimated by several methods.

- Classic Gaussian puff dispersion models provide good estimates at larger distances. Extrapolating to distances within a few metres of a point source leads to airborne dispersion factors (ADF) ~ 0.5 s/m^3 at low wind speeds along the plume centerline. For workers exposed directly to the plume near the source for worker exposure time (T_{EXP_W}) = 5 minutes, the effective dispersion volume is simply $V_{AIR} \sim T_{EXP\_W}$ / ADF ~ (300 s) / (0.5 s/m^3) ~ 600 m^3.

- The IAEA Q system approach for estimating exposure to transportation accidents suggests uptake fractions of around 10^{-3} to 10^{-4}, which is equivalent to dispersing the release into V_{AIR} of ~600 to 6000 m^3 for 1.2 m^3/hr breathing rates and half-hour exposure times (IAEA08b).

- Assuming that the puff is initially released over an area of about 2 m laterally and 2 m high (roughly the scale of a single package), and that the air moves through this at about 2 m/s (less than the average wind speed of 3.5 m/s at the Bruce nuclear site), then the puff is dispersed into V_{AIR} of about (4 m^2) x (2 m/s) x (300 s) ~ 2400 m^3 over 5 minutes.

- Assuming an eddy diffusion model without advection, with a ground level puff release, and a worker standing 3 m away at 1.6 m height, then the time-weighted average exposure is equivalent to dispersion within V_{AIR} ~ 17,000 m^3 for 0.1 m^2/min turbulent diffusivity, 400 m^3 for 1 m^2/min, and 1400 m^3 for 10 m^2/min (Chapter 7, KEIL09).

In the present model, the effective air volume was estimated as V_{AIR} ~ 1000 m^3.

If the wind speeds are significant, or if the worker is not directly downwind from the plume, or if there is turbulence due to flow around vehicle or package, or if the worker moves several metres away, then the average concentration at the worker location would be much smaller, or equivalently the effective air volume would be larger. Also, accidents involving several packages would likely be dispersed over a larger volume simply because the packages take up more volume.

Fire Accidents

For a fire accident, assuming that the steady burn conditions are quickly reached, then the worker exposure to releases from an outdoor fire is estimated as:

$$C_{WO} = QR \times T_{EXP\_W} / V_{AIR} \quad \text{(fire, outdoors)} \qquad (7\text{-}11)$$

where:

QR = Source term release rate (Bq/s or µg/s) during a fire, given by Equation 7-6

T_{EXP_W} = Worker exposure time (s)

The heat of the fire would cause turbulence and buoyancy, which would disperse the contaminants away from the worker at ground level. The worker would also tend to move out of the plume path, since this would be clearly marked by the smoke. Therefore, the effective volume over which the contaminants are dispersed would be larger than for breach accidents. However, for the present analysis, the volume assumed for worker exposure in a fire accident is conservatively assumed to be similar to that for breach, V_{AIR} ~ 1000 m^3.

Indoor Dispersion for Worker Exposure

The model used for indoor non-fire scenarios for the calculation of source term dispersion inside a building (e.g., WPRB or underground room) was based on the source term being instantly released throughout some local volume, with the area ventilated at a constant rate.

Breach Accidents

For a large building with relatively low air movement, such as the WPRB, the calculation of average indoor air concentration during non-fire scenarios, over the worker exposure time is modelled using the same approach as the above outdoors dispersion model, Equation 7-10, with V_{AIR} ~ 1000 m^3 for low air flow conditions.

For underground locations, where there is a well defined and significant air flow rate, the average air concentration downstream from the breach is described by:

$$C_{WI} = Q / [F \times T_{EXP\_W}] \qquad \text{(breach, underground)} \qquad (7\text{-}12)$$

where:

C_{WI} = Worker indoor air concentration during non-fire scenarios (Bq/m^3 or µg/m^3)

F = Ventilation rate (m^3/s) relevant to DGR location (See Table 7-35)

Equation 7-12 applies downstream from the release point and represents a well-mixed pulse of contaminated air; the air concentration upstream from the accident location would be much less.

Table 7-35: DGR Ventilation Rates for Accident Assessment

Location	**Ventilation rate (m^3/s)**
WPRB	2.8 [a]
Main Shaft	100 [b]
Main Shaft Station	18 [c]
Access Tunnel	60 [b]
Underground Rooms	18 [c]

Notes:
a. Approximately 0.5 air changes/hr.
b. Variable, nominal value during daytime operations.
c. Variable, nominal value in an area.

Fire Accidents

During indoor fire scenarios, the source term is assumed to be released at a constant rate. In a large building such as the WPRB with low air flow due to ventilation, the smoke would tend to rise due to buoyancy, and mix within the room volume. Therefore, the average indoor air concentration during the exposure time is given by the equation for a mixed, ventilated room:

$$C_{WI} = QR / F \times [\,1 - V_{Room} / (F \times T_{EXP\_W}) \times (1 - \exp(-F \times T_{EXP\_W} / V_{Room}))\,] \quad \text{(fire, WPRB)} \tag{7-13}$$

where:

V_{Room} = Room volume (m^3)

Table 7-35 gives DGR ventilation rates used in the accident assessments. The volume of the waste package loading area in the WPRB is 20,000 m^3.

For underground fire scenarios, the air flow is significant and directional, and the average air concentration is defined similar to Equation 7-12. Similarly, this conservatively assumes the worker remains in downstream airflow from the accident for the exposure duration.

$$C_{WI} = QR / F \quad \text{(fire, underground)} \tag{7-14}$$

<u>Outdoor Dispersion for Long Distances for Public Exposure</u>

For dispersion outside a building, the released source term (ventilated out of the building or emitted at certain rate) is multiplied by ADFs to estimate air concentrations at the location of public receptors:

$$C_P = ER \times ADF \qquad (7\text{-}15)$$

where:

C_P = Average air concentration near public receptor (Bq/m^3 or µg/m^3)

ER = Average emission rate (Bq/s or µg/s)

ADF = Atmospheric dilution factor (s/m^3) (See Table 7-36 for relevant ADF)

<u>Emission Rate</u>

Breach Accidents

The average rate of emission, ER_O of material from an outdoor non-fire accident scenario depends on the source term, Q, released over the public exposure time:

$$ER_O = Q / T_{EXP\_P} \quad \text{(breach, outdoors)} \qquad (7\text{-}16)$$

where:

ER_O = Average emission rate for exposure period (Bq/s or µg/s) from outdoor release of material

T_{EXP_P} = Public exposure time (s)

The average rate of emission, ER_I of material ventilated out of the WPRB during a non-fire accident scenario is based on the assumption that the source term will be well-mixed within the building over the public exposure time, and the relevant ventilation rate, F:

$$ER_I = Q \times [\, 1 - \exp(-F \times T_{EXP\_P} / V_{Room}\,)\,] / T_{EXP\_P} \quad \text{(breach, WPRB)} \qquad (7\text{-}17)$$

where:

ER_I = Average emission rate over exposure period during non-fire scenarios (Bq/s or µg/s) from indoor release of source term through ventilation

The average rate of emission, ER_I of material from an underground non-fire accident scenario, depends on the source term, Q released over the public exposure time:

$$ER_I = Q / T_{EXP\_P} \quad \text{(breach, underground)} \tag{7-18}$$

Fire Accidents

The average rate of emission, ER_O of material from an outdoors fire accident scenario depends on the source term, QR, assumed to be constant over this time:

$$ER_O = QR \quad \text{(fire, outdoors)} \tag{7-19}$$

where:

QR = Source term release rate (Bq/s or µg/s)

Similarly, the rate of emission, ER_I due to a fire accident underground also occurs at the rate QR as per Equation 7-19, as it is assumed to be directly vented for the duration of the exposure period.

The average rate of emission, ER_I of material ventilated out of the WPRB during a fire scenario is calculated based on the assumption that the source would tend to be well-mixed in the WPRB before being ventilated out of the building due to the lower ventilation rate, and consistent with Equation 7-13:

$$ER_I = QR \times [\, 1 - V_{Room} / (\, F \times T_{EXP\_P}\,) \times (\, 1 - \exp(\, -F \times T_{EXP\_P} / V_{Room}\,))\,] \tag{7-20}$$

(fire, WPRB)

<u>ADFs</u>

ADFs are used to provide estimates of the amount of dispersion or dilution experienced by a contaminant released into the atmosphere, between the point of emission and the public receptor location.

The public is assumed to be exposed for 1 hour at site boundary to the DGR accident releases. Beyond that time, it is assumed that the accident releases will have passed or will have been controlled; the implications of an unmitigated underground room fire is also considered as a “what if” case.

The ADFs used for the present accident assessment are based on WWMF values (OPG06a) for a public receptor at the nearest site boundary 750 m from the WWMF. These are conservative estimates for the DGR buildings, which are slightly further from the site boundary. The ADFs used for the various accident scenarios are summarized below and in Table 7-36.

- For all non-fire scenarios, dispersion of the released source term to the public was calculated using the non-fire ground-release ADF of 1.6 x 10^{-4} s/m^3.
- For an above ground fire, a short-term ADF of 4.3 x 10^{-6} s/m^3 was assumed as a conservative estimate for public receptors located at 1.1 km from the point of emission, due to a buoyant fire plume rise.
- For underground fires, the smoke plume from the fire is expected to have been cooled during its transit to surface, and, consequently, there is no significant thermal plume effect. Therefore, for underground fires, the ADF for a ground-release (non-fire) was assumed from Table 7-36.

Table 7-36: ADFs for the Public (OPG06a)

Time Frame	Above Ground Fire ADF (s/m^3)	Underground Fire and Non-Fire ADF (s/m^3)
First hour	4.3E-06	1.6E-04
Prolonged	1.2E-06 (6 hrs)	1.5E-05 (24 hrs)
Long-term	-	4.4E-06

7.5.3.4 Consequence for Radiological and Non-Radiological Releases

Potential pathways assessed in different bounding scenarios are shown in Table 7-37.

Table 7-37: Accident Exposure Pathways

Pathway	Radionuclides		Non-Radiological Species	
	Worker	Public	Worker	Public
Inhalation	Included	Included	Included	Included
Immersion	Included	Included		
External radiation	Included[a]	N/S	N/A	N/A

Notes:
N/A Not Applicable
N/S Not a Significant Pathway
a. External radiation considered insignificant during fire accidents

The methodology for calculation of the dose received through different pathways is explained in this section.

Inhalation Pathway

Workers and members of the public can be exposed to radionuclides and non-radiological chemicals through inhalation.

Radionuclides

Worker Exposure in a Ventilated Building and Outdoors:

Inhalation dose from exposure to each radionuclide is calculated as the product of the concentration in air of the radionuclide which is respirable, air uptake at the worker inhalation rate during the assumed exposure time, and a worker inhalation dose coefficient (DC_{WINH}).

$$WD_{INH} = C_W \times INH_W \times T_{EXP\_W} \times DC_{WINH} \quad (7\text{-}21)$$

where:

WD_{IN} = Dose to workers through inhalation (mSv)

C_W = Air concentration near workers (Bq/m^3)

INH_W = Worker inhalation rate – 1.6 m^3/hr (adult, moderate activity, USEPA97)

T_{EXP_W} = Time of exposure (h) (Workers are assumed to be exposed to accident for 5 minutes or 0.08 h prior to evacuation)

DC_{WINH} = Inhalation dose coefficient for workers (mSv/Bq) (See Table 7-38)

It should be noted that in the breach scenario where C-14 is released into the air, 25% is considered as particulate, while 75% is considered as CO_2. In the fire scenario, 100% of C-14 is assumed to be released, and it is all considered as CO_2. Inhalation dose coefficient for C-14 varies according to particulate or CO_2 as given in Table 7-38.

Table 7-38: Dose Coefficients for Adult

Radionuclide	Air Inhalation (mSv/Bq)		Air Immersion (mSv/year)/(Bq/m^3)
	Worker [1]	Public [2]	Worker and Public [2]
Am-241	3.9E-02	4.2E-02	2.1E-05
C-14 (as CO_2)	1.2E-08 [a]	1.2E-08	8.2E-08
C-14 (as particulate)	2.0E-06 [a]	2.0E-06	8.2E-08
Ce-141	3.6E-06	3.2E-06	9.8E-05
Ce-144	4.9E-05	3.6E-05	1.1E-04
Cm-244	2.5E-02	2.7E-02	1.1E-07
Co-60	2.9E-05	1.0E-05	3.8E-03
Cs-134	9.6E-06	6.6E-06	2.2E-03
Cs-137	6.7E-06	4.6E-06	8.1E-04
Eu-152	3.9E-05	4.2E-05	1.7E-03
Eu-154	5.0E-05	5.3E-05	1.8E-03
Fe-55	9.2E-07	3.8E-07	0.0E+00
Fe-59	3.5E-06	3.7E-06	1.8E-03
Gd-153	2.5E-06	2.1E-06	9.8E-05
H-3	3.0E-08 [b]	3.0E-08 [b]	0.0E+00
La-140	1.5E-06	1.1E-06	3.5E-03
Mn-54	1.5E-06	1.5E-06	1.2E-03
Nb-94	4.5E-05	1.1E-05	2.3E-03
Nb-95	1.6E-06	1.5E-06	1.1E-03
Pb-210	1.1E-03	1.1E-03 [c]	1.4E-06 [d]
Pu-238	4.3E-02	4.6E-02	1.1E-07
Pu-239	4.7E-02	5.0E-02	1.1E-07
Pu-240	4.7E-02	5.0E-02	1.1E-07
Pu-241	8.5E-04	9.0E-04	2.0E-09
Ra-226	1.6E-02	3.5E-03	9.0E-06
Ru-106	6.2E-05	2.8E-05	3.3E-04
Sb-124	6.1E-06	6.4E-06	2.7E-03
Sb-125	4.5E-06	4.8E-06	5.9E-04

Radionuclide	Air Inhalation (mSv/Bq)		Air Immersion (mSv/year)/(Bq/m^3)
	Worker [1]	Public [2]	Worker and Public [2]
Sn-119m	2.0E-06	2.0E-06 [c]	2.2E-06 [d]
Sr-90	1.5E-04	3.6E-05	2.8E-05
Te-125m	3.3E-06	3.3E-06 [c]	1.1E-05
Zr-95	5.5E-06	4.8E-06	1.1E-03

Notes:
1 Most conservative value from ICRP 68 (ICRP95).
2 CSA N288.1 (CSA08b).
a. Public dose coefficient used as a more conservative value.
b. Based on HTO, which is most conservative among all gaseous forms of H-3 (CSA08b).
c. Dose coefficient for air inhalation to public not available, therefore DC for worker is used.
d. See (ECKERMANN96).

Public Exposure:

Public dose exposure through inhalation is similarly calculated:

$$PD_{INH} = C_P \times INH_P \times T_{EXP\_P} \times DC_{PINH} \quad (7\text{-}22)$$

where:

PD_{INH} = Dose to the public through inhalation (mSv)

C_P = Air concentration near public receptor (Bq/m^3) (See Equation 7-15)

INH_P = Public inhalation rate (m^3/hr) – 0.96 m^3/hr

T_{EXP_P} = Time of exposure (h) (Public exposure time assumed to be 1 hour)

DC_{PINH} = Inhalation dose coefficient for public (mSv/Bq) (See Table 7-38))

Immersion Pathway

The immersion dose from exposure to each radionuclide was calculated as the product of the total concentration (inclusive of respirable and non-RFs) in air of the radionuclide, exposure time, and an immersion dose coefficient:

Worker Exposure

$$WD_{IMM} = (C_W / RF) \times T_{EXP\_W} \times DC_{IMM} \tag{7-23}$$

where:

WD_{IMM} = Dose to workers through immersion (mSv)

C_W = Air concentration near workers (Bq/m^3)

RF = Respirable fraction (-) (See Table 7-34)

T_{EXP_W} = Worker time of exposure (year) based on a worker exposure time of 5 minutes

DC_{IMM} = Immersion dose coefficient for workers ($(mSv/year)/(Bq/m^3)$) (See Table 7-38)

Public Exposure

$$PD_{IMM} = (C_P / RF) \times T_{EXP\_P} \times DC_{IMM} \tag{7-24}$$

where:

PD_{IMM} = Dose to public through immersion (mSv)

C_P = Air concentration near public receptor (Bq/m^3)

T_{EXP_P} = Public time of exposure (year) based on a public exposure time of 1 hour

DC_{IMM} = Immersion dose coefficient ($(mSv/year)/(Bq/m^3)$) (See Table 7-38)

External Radiation

MicroShield Version 8.02 (GROVE09) was used to estimate external radiation dose to workers in above ground and underground non-fire scenarios. External dose to workers during fire scenarios was considered negligible, since the waste is in the packages (at least during the early stages when the workers are present), and since the immersion pathway is included for radionuclides released into the air. External radiation to members of the public was considered negligible in all scenarios due to the relatively large distance (1.1 km) between waste packages and the public.

Input parameter values such as waste matrix and container specifications are common to both normal operations and accident assessment. See Section 7.4.4.1.

MicroShield results show that the highest effective dose equivalent rate is for the anterior/posterior geometry, which assumes that the person is standing facing the source. This exposure geometry was conservatively assumed here. MicroShield neglects backscattering from the floors and walls as well as skyshine. However, as illustrated in the normal operations assessment, these are small contributors to the external dose for conditions close to the packages, as would apply to workers.

In scenarios where more than one package are involved, depending on the actual geometry and the gap between containers/packages, the consequences for external radiation can be conservatively estimated by scaling a basic bounding consequence for a single container/package by the number of containers/packages involved, or by assuming one large equivalent package which represents all container/packages involved. This is conservative because, for example, it neglects internal container walls and shielding.

Non-Radiological Species

Impacts of exposure to non-radiological species were assessed as follows.

Worker Exposure

Air concentrations of non-radiological species near workers were compared to the IDLH values shown in Table 7-5.

Public Exposure

Impacts of short-term exposure to non-radiological chemicals on members of the general public were assessed through comparing estimated concentrations with the PAC 1 inhalation criteria in Table 7-4.

7.5.3.5 Ventilation System Failure Modelling

Loss of ventilation in an active emplacement room may occur due to hypothetical short-term loss of power or failure of fans. If the ventilation fails, the air flow through the room will decrease, and H-3 and C-14 containing gases will build up in the room.

Assuming an approximately uniform outgassing rate from packages along the room length, the formula for the average room air concentration during normal operations in the room is given below:

$$C(t) = \frac{QR}{2\,F}\left[1 - \exp\left(-t\left(F/V\right)\right)\right] \qquad (7\text{-}25)$$

where:

F = Ventilation rate through the room (m^3/s)

QR = Total room contaminant release rate (Bq/s)

V = Void volume of the room (m^3)

Under steady-state conditions, the average room concentration is, therefore, given below:

$$C_{avg} = \frac{QR}{2\,F} \qquad (7\text{-}26)$$

The ventilation rate through the room during day operations, F, is 18 m^3/s (Table 7-35).

After failure of the ventilation system (F = 0 m^3/s), and assuming that the outgassing from the packages continues without change, the average room concentration then increases linearly as:

$$C(t) = C_{avg} + \frac{QR}{V}t \qquad (7\text{-}27)$$

For worker exposure, the air concentration for H-3 and C-14 is then compared to the corresponding DAC in Table 7-2.

7.5.4 Accident Consequence Assessment Results

The results of the accident consequence assessment are summarized in Table 7-39 and Table 7-40. Details of the results are given in Appendix A. These tables list the bounding scenarios and their estimated consequences. The estimated consequences are expressed as a dose value for radiological exposure, and as a maximum ratio of the estimated non-radiological species air concentration to a corresponding health and safety criterion (Table 7-4 and Table 7-5).

Impacts on Public

Radiological dose over a 1 hour exposure period to the public at the nearest Bruce nuclear site boundary are much less than the 1 mSv limit for any accident scenario.

Air concentrations of non-radiological species released during any accident scenario are less than the PAC 1 criteria for the public.

Although unlikely that a member of the public would be exposed at the Bruce site boundary for more than one hour, longer exposures would not exceed the criteria. Specifically, assuming complete burn of an underground room over a few hundred hours, the public radiological dose at the nearest site boundary is less than 1 mSv for a room containing LLW and for a room containing unshielded ILW moderator resin.

Impacts on Workers

Radiological doses to workers over a 5 minute exposure time are much less than the 50 mSv limit for any accident scenario. In addition, in the case of a ventilation system failure, workers exposed to H-3 and C-14 would be subjected to air concentrations much less than the DACs.

Air concentrations of non-radiological species released during the accident scenarios are less than the IDLH criteria for workers.

7.5.5 Assumptions and Uncertainty in Accident Assessment

Table 7-41 summarizes the main uncertainties in assessing the impact of all accidents, and how these have been addressed using conservative models and assumptions.

7.5.6 Preventative and Mitigation Measures

The impacts on public from potential accidents at the DGR were found to be generally small, and always within criteria.

The assessment qualitatively considers the likelihood of these potential bounding accidents in terms of identifying them as possible, unlikely or not credible. Measures to reduce their likelihood have already been considered within the design and will be further emphasized during detailed design and later during operations. These measures include:

- Minimization of combustible materials and ignition sources, especially near waste packages;
- Use of overpacking and shielding on higher activity packages;
- Limited number of packages handled in any transfer;
- limited equipment speeds;
- Fire detection and suppression equipment, such as automatic fire suppression systems on diesel transfer equipment;
- Contamination and dose rate monitoring;
- Access to refuge stations and safety equipment;
- Appropriate worker training and operating procedures; and
- Emergency communication systems.

Table 7-39: Summary of Impacts from Above Ground Accidents

Accident Stress Type	Above Ground Bounding Scenario	Selected Waste Category	Source Term Parameters						Radionuclide Doses (mSv)		Non-Radiological Species Maximum Ratio of Concentration to Criteria	
			# of Pkgs	DR	ARF	RF	LPF	$1/T_{FD}$ (1/h)	Workers	Public	Workers	Public
Fire [d]	Outdoor Unshielded Waste Package Fire	Box Compacted	8	0.5	0.001	1	1	0.20	0.2	<0.001	0.5 (Se)	<0.001
		Non-Processible Boxed	8	0.5	0.001	1	1	1.43	0.2	<0.001	0.4 (Se)	0.004 (Cr)
		Non-Processible Drummed	8	0.5	0.001	1	1	1.43	1.3	<0.001	0.2 (Se)	0.002 (Cr)
		Moderator Resin (Unshielded)	1	0.5	0.001	1	1	0.25	1.5	<0.001	0.01 (Se)	<0.001
	Indoor Unshielded Waste Package Fire	Box Compacted	24	0.5	0.001	1	1	0.10	0.006	<0.001	0.02 (Se)	<0.001
		Non-Processible Boxed	24	0.5	0.001	1	1	0.71	0.006	<0.001	0.02 (Se)	0.001 (Cr)
		Non-Processible Drummed	24	0.5	0.001	1	1	0.67	0.05	<0.001	0.007(Se)	<0.001
		Moderator Resin (Unshielded)	1	0.5	0.001	1	1	0.36	0.04	<0.001	<0.001	<0.001
		Combined LLW and ILW (Non-Processible Drummed and Unshielded Moderator Resin)	24 [a] 2 [b]	0.5 0.5	0.001 0.001	1 1	1 1	0.67 0.26	0.12	<0.001	0.008 (Se)	<0.001
	Shielded ILW Package Steam Release	Moderator Resin (Outdoors)	1	0.1	1 [c]	1	1	0.62	1.6	<0.001	0.01 (Se)	<0.001
		Moderator Resin (Indoors)	1	0.1	1 [c]	1	1	0.62	0.04	<0.001	<0.001	<0.001

Accident Stress Type	Above Ground Bounding Scenario	Selected Waste Category	Source Term Parameters						Radionuclide Doses (mSv)		Non-Radiological Species Maximum Ratio of Concentration to Criteria	
			# of Pkgs	DR	ARF	RF	LPF	$1/T_{FD}$ (1/h)	Workers	Public	Workers	Public
Low Energy Container Breach	Outdoor Waste Package Breach	Bottom Ash	8	0.25	0.002	0.3	1	N/A	0.03	<0.001	0.8 (Cr)	0.03 (Cr)
		Box Compacted	8	0.1	0.001	0.1	1	N/A	0.004	<0.001	0.01 (Cd)	<0.001
		Non-Processible Boxed	8	0.1	0.001	0.1	1	N/A	0.004	<0.001	0.06 (Cu)	0.002 (Cr)
		Non-Processible Drummed	8	0.05	0.001	0.1	1	N/A	0.003	<0.001	0.01 (Cu)	<0.001
		Moderator Resin (Unshielded)	1	0.1	0.001	0.1	1	N/A	0.6	<0.001	0.007 (Ni)	<0.001
		Moderator Resin (Shielded)	1	0.05	0.001	0.1	1	N/A	0.4	<0.001	0.007 (Ni)	<0.001
	Indoor Waste Package Breach	Bottom Ash	24	1	0.01	0.3	0.1	N/A	N/C	<0.001	N/C	0.08 (Cr)
		Box Compacted	24	1	0.01	0.2	0.1	N/A	N/C	<0.001	N/C	0.004 (Cr)
		Non-Processible Boxed	24	1	0.01	0.2	0.1	N/A	N/C	<0.001	N/C	0.05 (Cr)
		Non-Processible Drummed	24	1	0.01	0.2	0.1	N/A	N/C	<0.001	N/C	0.02 (Cr)
		Bottom Ash	1	0.25	0.002	0.3	1	N/A	0.07	<0.001	0.1 (Cr)	0.002 (Cr)
		Box Compacted	1	0.1	0.001	0.1	1	N/A	0.008	<0.001	0.001 (Cd)	<0.001
		Non-Processible Boxed	1	0.1	0.001	0.1	1	N/A	0.01	<0.001	0.007 (Cu)	<0.001
		Non-Processible Drummed	1	0.05	0.001	0.1	1	N/A	0.006	<0.001	0.002 (Cu)	<0.001
		Moderator Resin (unshielded)	1	0.1	0.001	0.1	1	N/A	0.6	<0.001	0.007 (Ni)	<0.001
		Moderator Resin (shielded)	1	0.05	0.001	0.1	1	N/A	0.5	<0.001	0.007 (Ni)	<0.001

Accident Stress Type	Above Ground Bounding Scenario	Selected Waste Category	Source Term Parameters						Radionuclide Doses (mSv)		Non-Radiological Species Maximum Ratio of Concentration to Criteria	
			# of Pkgs	DR	ARF	RF	LPF	$1/T_{FD}$ (1/h)	Workers	Public	Workers	Public
(cont'd)	(cont'd)	Combined LLW and ILW (Non-Processible Drummed and Unshielded Moderator Resin)	24 [a] 2 [b]	1 1	0.01 0.01	0.2 0.2	0.1 0.1	N/A N/A	N/C	<0.001	N/C	0.02 (Cr)
Inadequate Shielding	-	Moderator Resin	1	N/A	N/A	N/A	N/A	N/A	N/E	N/A	N/A	N/A

Notes:
N/A: not applicable for the scenario.
N/C: not considered. For the "what if" roof collapse scenario, the impact to workers is more likely to be incurred through injury due to conventional hazard rather than release of harmful radionuclides or non-radiological species. Therefore, the impact to workers was not considered.
N/E: not estimated, but the worker dose is expected to be < 10 mSv/year.
a. Non-processible drummed waste.
b. Unshielded moderator resin.
c. Volatiles.
d. Fire exposures are based on 5 minute exposure for workers and 1 hour exposure for public. $1/T_{FD}$ is the source burn rate fraction per hour.

Table 7-40: Summary of Impacts from Underground Accidents

Accident Stress Type	Underground Bounding Scenario	Selected Waste Category	Source Term Parameters						Radionuclide Doses (mSv)		Non-Radiological Species Maximum Ratio of Concentration to Criteria	
			# of Pkgs	DR	ARF	RF	LPF	$1/T_{FD}$ (1/h)	Workers	Public	Workers	Public
Fire [c]	Unshielded Waste Package Fire During Transfer	Box Compacted	1	0.5	0.001	1	1	0.20	0.04	<0.001	0.01 (Se)	<0.001
		Non-Processible Boxed	1	0.5	0.001	1	1	1.43	0.03	<0.001	0.01 (Se)	0.02 (Cr)
		Non-Processible Drummed	1	0.5	0.001	1	1	1.43	0.3	0.006	0.004 (Se)	0.009 (Cr)
		Moderator Resin (Unshielded)	1	0.5	0.001	1	1	0.26	0.3	0.006	0.003 (Se)	<0.001
	In Room Unshielded Waste Package Fire	Box Compacted	2400	1	0.01	1	1	0.0013	N/C	0.003[d]	N/C	0.07 (Cr)
		Non-Processible Boxed	2400	1	0.001/0.01 [a]	1	1	0.0078	N/C	0.002	N/C	0.5 (Cr)
		Non-Processible Drummed	2400	1	0.001/0.01 [a]	1	1	0.0073	N/C	0.02	N/C	0.2 (Cr)
		Moderator Resin (Unshielded)	1200	1	0.01	1	1	0.0034	N/C	0.02[e]	N/C	0.02 (Ni)
	Shielded ILW Package Steam Release	Moderator Resin (Transfer)	1	0.1	1 [b]	1	1	0.62	0.3	0.006	0.003 (Se)	<0.001
		Moderator Resin (In-Room)	1	0.1	1 [b]	1	1	0.62	0.3	0.006	0.003 (Se)	<0.001
Low Energy Container Breach	Waste Package Breach During Transfer	Bottom Ash	1	0.25	0.002	0.3	1	N/A	0.04	<0.001	0.02 (Cr)	0.004 (Cr)
		Box Compacted	1	0.1	0.001	0.1	1	N/A	0.005	<0.001	<0.001	<0.001
		Non-Processible Boxed	1	0.1	0.001	0.1	1	N/A	0.01	<0.001	0.001 (Cu)	<0.001
		Non-Processible Drummed	1	0.05	0.001	0.1	1	N/A	0.004	<0.001	<0.001	<0.001
		Moderator Resin (Unshielded)	1	0.1	0.001	0.1	1	N/A	0.6	<0.001	0.001 (Ni)	<0.001
		Moderator Resin (Shielded)	1	0.05	0.001	0.1	1	N/A	0.4	<0.001	0.001 (Ni)	<0.001
Low Energy Container Breach	In Room Waste Package Breach	Bottom Ash	3	0.25	0.002	0.3	1	N/A	0.2	<0.001	0.05 (Cr)	0.01(Cr)
		Box Compacted	4	0.1	0.001	0.1	1	N/A	0.03	<0.001	<0.001	<0.001
		Non-Processible Boxed	5	0.1	0.001	0.1	1	N/A	0.08	<0.001	0.007 (Cu)	0.001 (Cr)
		Non-Processible Drummed	5	0.05	0.001	0.1	1	N/A	0.04	<0.001	0.002 (Cu)	<0.001
		Moderator Resin (Unshielded)	4	0.1	0.001	0.1	1	N/A	2.8	<0.001	0.005 (Ni)	<0.001
		Moderator Resin (Shielded)	3	0.05	0.001	0.1	1	N/A	0.8	<0.001	0.004 (Ni)	<0.001

Accident Stress Type	Underground Bounding Scenario	Selected Waste Category	Source Term Parameters						Radionuclide Doses (mSv)		Non-Radiological Species Maximum Ratio of Concentration to Criteria	
			# of Pkgs	DR	ARF	RF	LPF	$1/T_{FD}$ (1/h)	Workers	Public	Workers	Public
High Energy Container Breach	Cage Fall	Bottom Ash	2	1	0.01	0.3	0.1	N/A	0.02	<0.001	0.07 (Cr)	0.02 (Cr)
		Box Compacted	2	1	0.01	0.2	0.1	N/A	0.02	<0.001	0.009 (Cd)	<0.001
		Non-Processible Boxed	3	1	0.01	0.2	0.1	N/A	0.01	<0.001	0.08 (Cu)	0.02 (Cr)
		Non-Processible Drummed	3	1	0.01	0.2	0.1	N/A	0.02	<0.001	0.04 (Cu)	0.008 (Cr)
		Moderator Resin (Unshielded)	2	1	0.01	0.2	0.1	N/A	0.6	<0.001	0.05 (Ni)	<0.001
		Moderator Resin (Shielded)	1	1	0.01	0.2	0.1	N/A	1.4	0.002	0.05 (Ni)	<0.001
		Retube- End Fittings	1	1	0.001	0.2	0.1	N/A	6.	0.004	0.03 (Cr)	0.007 (Cr)
Loss of Ventilation	Ventilation System Failure	All Waste	N/A	N/A	N/A	N/A	N/A	N/A	-	N/A	N/A	N/A

Notes:
N/A: not applicable for the scenario.
N/C: not considered. In-room unshielded waste package fire would take many minutes to develop from an initial small fire. During this period, the worker risk is addressed by the single waste package fire scenario. By the time the fire has developed into a room fire, any workers would have left the area and gone to refuge stations or safe locations. Therefore, the impact to worker during room fire was not considered.
- Worker doses were not calculated. Instead, the air concentrations for H-3 and C-14 were estimated and found to be much less than the corresponding DACs for workers.
a. ARF=0.001 for bulk metals and asbestos, and 0.01 otherwise.
b. Volatiles.
c. Fire exposures are based on 5 minute exposure for workers and 1 hour exposure for public. $1/T_{FD}$ is the source burn rate fraction per hour.
d. Public dose is 0.06 mSv at site boundary for full 740-hr fire duration.
e. Public dose is 0.2 mSv at site boundary for full 290-hr fire duration.

Table 7-41: Approach to Addressing Important Uncertainties

Parameter	Related Accident Scenario	Uncertainty	Approach
Source Terms	Fire scenarios	Release rate during fire.	• Fires are conservatively assumed to instantly reach a steady-state burning rate; the burning rate is more likely to be less in the initial stages of the fire. • Fire sizes are generally large compared with real fires. • Assumes that fire lasts long enough to expose workers and public before suppression. • Assumes that ventilation / fire doors are not shut down to contain smoke before the 1 hour public exposure period. • Release fractions do not specifically consider the combustibility of the radioactive or non-radioactive material. • All releases are assumed completely respirable.
	Breach scenarios	The airborne fractional release rates (ARF, RF and DR) were estimated from DOE handbooks, which were developed for different containers and waste forms.	• DOE values are conservative values from their experiments. • Similar values are used in WIPP. • Values are believed reasonable or conservative for DGR conditions; in particular, they are assumed applicable to bulk non-radioactive materials.
	Shielded ILW steam release scenarios	The source term assumed that a shielded ILW package would be affected by an external fire.	• Assumes that shield thermal inertia is sufficient to prevent fire, but that waste does heat up. • Assumes that waste is heated enough to release all volatiles.

Parameter	Related Accident Scenario	Uncertainty	Approach
Dispersion to Workers	Above ground outdoor and WPRB fire and breach scenarios	Effective air volume for estimating concentration for worker exposure.	• Little credit for wind or air circulation over this exposure period. • No accounting for thermal plume or buoyancy, nor for the number of packages involved which would have more source term dispersion than a single package.
	All indoor fire scenarios	Assumption regarding source term dispersion near worker.	• Worker was conservatively assumed to be downwind from the fire. • Steady-state burn rate and air concentrations were assumed.
Dispersion to the Public	All scenarios	ADFs.	• The ADF was based on the 90th percentile short-term factor derived from a Gaussian plume model. • No credit for puff release characteristics, deposition of particulate, and/or higher wind speeds associated with daytime atmospheric conditions. • Underground fire scenario releases modelled as non-thermal plume. • Public assumed to be downwind and in the plume path, at the closest Bruce nuclear site boundary point, and remain there for 1 hour exposure as reference. • Public dose at site boundary also assessed for complete burn over a few hundred hours ofa room of LLW or unshielded ILW resins.

Parameter	Related Accident Scenario	Uncertainty	Approach
Criteria for Receptor Exposure	Non-fire scenarios	Use of IDLH criteria for short-term worker exposure to non-radiological species.	• The IDLH criteria are intended for a nominal exposure period of half-hour. However, workers are unlikely to remain in the vicinity of a significant accident for this long without taking protective measures.
	All scenarios	Use of PAC 1 criteria for public exposure to non-radiological species.	• The PAC criteria are public exposure guidelines for short-term exposure, i.e.up to approximately 1 hour time scales. • PAC 1 values are adopted given the conservative public exposure model (i.e., at Bruce nuclear site fence line, in plume pathway). • The selected PAC 1 criteria are the lowest of the elemental or oxide forms of each non-radiological species where available.
Shielding (External Radiation)	All breach scenarios	MicroShield is not able to fully represent geometrical details, and provides approximate values for dose rates. MicroShield does not include backscattering or skyshine. There is uncertainty associated with the assumed crack size and spill geometry.	• MicroShield results are more accurate in the near-field, with simple geometry, which is the primary application here. • A comparison of Co-60 dose estimated by MicroShield and MCNP for a DGR underground emplacement room showed that the use of MicroShield was conservative. • Reasonably conservative parameters were assumed – crack in container assumed proportional to DR, no shielding, and the worker was positioned directly in line with crack for duration of exposure.

7.6 Contingency Planning

For situations in which consequences of accident assessment are not negligible, mitigation will be achieved through one or more of the following:

- Design mitigation;
- Preventive measures to reduce further the likelihood of such accidents;
- Controls installed on equipment to restrain their movement (e.g., limit switches);
- Administrative controls (mainly through procedures); and
- Worker training.

For accidents assessed to have larger consequences, contingency plans will be in place, and emergency response, including mine rescue, will be available to protect the workers. Emergency response is addressed in Chapter 10. Contingency plans will be developed in support of the Operating Licence application.

7.7 Summary and Conclusions

The results of this preliminary Preclosure Safety Assessment provide a quantitative estimate of hazards and impacts. The key results are listed below.

7.7.1 Normal Operations Assessment

- The maximum dose rate to the public due to normal DGR operations from airborne and waterborne releases was estimated to be far below the CNSC regulatory limit of 1 mSv/year, and is similar to the impacts that would be expected from WWMF for similar LLW and ILW radionuclide inventories.
- Inhalation doses estimated for DGR workers are all much less than OPG's occupational dose target of 10 mSv/year. This indicates that there are no concerns with respect to inhalation due to normal DGR operations, and the need for mitigation can be addressed in the context of ALARA. The highest dose rates occur within the WPRB staging area if there are many packages staged there, and the ventilation shaft and ventilation drifts which will not be normally occupied.
- External (gamma) dose calculations were carried out to the nearest DGR fence line (80 m from the WPRB) and to the nearest Bruce nuclear site fence line (about 1.1 km from the WPRB). The estimated dose rate is well below the non-NEW compliance dose limit of 0.5 µSv/hr at the DGR fence line, and well below the OPG

site boundary dose target for members of the public of 10 µSv/year (and in both cases much less than the regulatory limit).

- The external dose calculations for workers show that high dose rates are possible in specific locations, especially near the face of an array of higher dose rate LLW or ILW packages in emplacement rooms. Generally, workers would not need to spend much time in these locations, nor are most packages at high dose rates. However, it will be planned to monitor the radiation fields in these locations, and if necessary to limit the worker exposure, use shielded forklifts and/or use greater stand-off distances. This will be considered further within the context of ALARA.

- The external dose calculations also show that if multiple waste packages are routinely staged within the WPRB, then shielding or thicker walls (similar to LLSBs) will need to be incorporated around the staging area in the final detailed design to ensure that the external dose rate outside of the WPRB is below the OPG 25 µSv/hr building exterior radiation protection requirement and that dose rates in the office/control room are below the dose target of the 10 mSv/year.

7.7.2 Accident Assessment

For the accident assessment, a variety of bounding accidents were considered. The accidents were quantitatively assessed for several waste categories that represent the range of wastes to be handled. The assessment focused on the potentially hazardous material within the wastes.

The results of the accident assessment are that:

- Major DGR accidents are unlikely to occur;

- Credible DGR accidents do not exceed radiological dose criteria for workers or public;

- Credible DGR accidents do not exceed the relevant non-radiological species criteria for workers or public; and

- In most cases, the public safety criteria are met by large margins.

The following accident scenarios were identified as having the highest impacts. These are noted here to provide guidance in development of the detailed design and the operating procedures.

- Breach accidents involving ash containers (worker- hazardous elements in the dust);

- Fire accidents involving box compacted and non-processible wastes (worker radiological and hazardous elements in smoke); and

- Fire accidents involving multiple packages in an emplacement room (public-hazardous elements in smoke).

The conventional safety hazards of these accidents are also important. A preliminary assessment of these hazards and mitigations was considered in the Preliminary Conventional Safety Assessment report (NWMO11ac), and will be considered further as part of the detailed design.

Overall, both WWMF experience as well as the DGR analyses summarized here indicate that the wastes can be handled and emplaced without undue risk to workers or the general public.

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8. POSTCLOSURE SAFETY ASSESSMENT

This chapter provides a summary of the DGR radiological and non-radiological safety assessment for human and non-human biota for the period following repository closure. Details of the assessment methodology and results are given in the Postclosure Safety Assessment report (NWMO11af) and its supporting references.

The postclosure safety is quantitatively assessed through considering a range of potential future scenarios. Specifically, it considers the expected evolution of the DGR system with time (i.e., Normal Evolution Scenario) and the potential impacts of low-probability events leading to penetration of barriers and abnormal loss of containment (i.e., Disruptive Scenarios or “What if” Scenarios).

8.1 Acceptance Criteria

The following sections define acceptance criteria for the postclosure safety assessment. These criteria are consistent with the CNSC Regulatory Guide G-320 (Section 6.1 of CNSC06a), and have been accepted by the CNSC (CNSC08, CNSC09a, CNSC10).

8.1.1 Radiological Criteria for the Normal Evolution Scenario

The underlying principle for protection of humans is that future generations are protected to the same level as current generations (CNSC04a). Therefore, the CNSC limit on annual effective dose for members of the public of 1 mSv/year, is also applicable in the postclosure period.

CNSC Regulatory Guide G-320 (CNSC06a) states that to account for the possibility of exposure to multiple sources and to help ensure that doses resulting from the facility are ALARA, an acceptance criterion that is less than the regulatory limit should be used. Therefore, in the normal evolution scenario, a more restrictive dose criterion of 0.3 mSv/year was defined. This dose criterion is consistent with ICRP (ICRP07) and IAEA guidance (IAEA06b), allows for potential exposure from multiple sources in the future, and is approximately an order of magnitude below the individual dose rate received from natural background radiation in Canada (GRASTY04).

Specifically, the criteria for public radiological exposure are (OPG08b, CNSC08):

- Dose constraint of 0.3 mSv/year to critical group;
- Optimization below dose constraint;
- Doses are calculated for average adult member of the critical group(s); and
- Assessment encompasses the time of maximum calculated impact.

8.1.2 Radiological Criteria for the Disruptive Scenarios

A tiered approach is adopted for disruptive scenarios, recognizing that some scenarios are very unlikely or "what if" scenarios intended to test the robustness of the repository system. First, a dose criterion of 1 mSv/year is used for radiological exposure of humans under credible disruptive scenarios.

Second, if calculated doses exceed 1 mSv/year for a scenario, the acceptability of results from that scenario is examined on a case-by-case basis, taking into account the likelihood and nature of the exposure, uncertainty in the assessment, and conservatism in the dose criterion. Where the probability of exposure can be quantified without excessive uncertainty, a measure of risk can be calculated based on the probability of exposure and the health effects if the exposure occurs. This can be compared with a reference health risk value of 10^{-5}/year (OPG08b, IAEA06b).

Human intrusion is a special case. According to G-320 (CNSC06a), *"human intrusion scenarios are to be assessed separately, and the intrusion scenario probability should be considered in interpreting dose results. Reasonable efforts should be made to limit the dose from a high-consequence intrusion scenario and to reduce the probability of the intrusion occurring."* The fundamental concept of the DGR – the wastes are isolated at approximately 680 m depth - is specifically intended to reduce the probability of intrusion.

8.1.3 Radiological Criteria for Non-Human Biota

Potential radiological impacts on non-human biota are assessed for both normal evolution and disruptive scenarios. These potential impacts are compared with screening-level criteria expressed as No-Effect Concentrations (NECs). The criteria are listed in Table 8-1 for radionuclides of interest (NWMO09b, CNSC09a).

They are derived from Estimated No Effect Values (ENEVs) for indicator species relevant to current conditions at the DGR location. The ENEVs used are the most conservative values provided by Environment Canada and Health Canada (EC03) and the United Nations Scientific Committee on Effects of Atomic Radiation (UNSCEAR96). The radionuclide concentration corresponding to the ENEV is calculated for each indicator species in each applicable medium (e.g., surface water), assuming nil concentration in the other media. The NEC is then defined as the lowest concentration in each medium for all indicator species.

If any radionuclide concentrations exceed the NECs under the Normal Evolution Scenario, an Ecological Risk Assessment (ERA) will be carried out for the radionuclides that exceed criteria. The ERA will take into account uncertainties and the potential need for the effect of several radionuclides to be summed.

If any concentrations exceed these NECs under disruptive scenarios, then the acceptability would be judged on a case-by-case basis taking into account the likelihood and nature of the exposure, uncertainty in the assessment, and conservatism in the dose criterion.

Table 8-1: Acceptance Criteria (NECs) for Protection of Non-Human Biota from Potential Radiological Impacts

Radionuclide	Media			
	Groundwater (Bq/L)	Soil (Bq/kg)	Surface Water (Bq/L)	Sediment (Bq/kg)
C-14	1.6E+6	3.5E+2	2.4E-1	2.8E+5
Cl-36	3.0E+5	5.0E+0	3.1E+0	4.1E+4
Zr-93	5.9E+6	2.8E+5	1.8E+0	5.0E+6
Nb-94	3.6E+4	1.3E+2	1.6E-2	2.6E+4
Tc-99	8.1E+5	6.0E+1	8.0E-1	3.0E+6
I-129	9.0E+5	1.9E+4	3.2E+0	1.2E+6
Ra-226	5.9E+2	2.8E+2	5.9E-4	9.3E+2
Np-237	5.8E+2	5.0E+1	5.8E-2	1.1E+3
U-238	5.6E+2	4.9E+1	2.3E-2	6.6E+4
Pb-210	1.8E+5	3.7E+3	5.0E+0	6.3E+3
Po-210	5.4E+2	3.0E+1	7.0E-3	1.1E+5

8.1.4 Criteria for Non-Radioactive Contaminants

Potential impacts from non-radioactive elements or chemical species in the waste or packaging are assessed for both normal evolution and disruptive scenarios in environmental media relevant to human health and environmental protection.

The calculated environmental concentrations are compared with benchmark concentration limits in surface water, groundwater, soil and sediment. Consistent with the recommendations of the CNSC Regulatory Guide G-320 (CNSC06a), the benchmark concentrations are taken from federal and provincial environmental objectives and guidelines, in particular the Environmental Quality Guidelines published by the Canadian Council of Ministers of the Environment (CCME).

The criteria are listed in Table 8-2 (NWMO09b, CNSC09a, NWMO10c, CNSC10). These are based on the most conservative guideline concentrations for surface water, groundwater, soil and sediment from CCME and MOE guidelines (CCME07, MOE09b, MOEE94). For several elements of potential interest (Br, Gd, Hf, Nb, Sc, Sn, Sr, and Te), no criteria were available from CCME or MOE. In these cases, the exposure was evaluated based on surface water criteria from other sources (NWMO10c).

The impacts from non-radioactive contaminants released from the DGR are assessed in a tiered approach. Elements are screened first based on a comparison of estimated environmental concentrations with the criteria given in Table 8-2.

If any concentrations exceed these criteria under normal evolution scenarios, these species will be assessed further in a tiered approach with decreased conservatism in models. If any concentrations exceed these criteria under disruptive scenarios, then the acceptability would be judged on a case-by-case basis taking into account the likelihood and nature of the exposure, uncertainty in the assessment, and conservatism in the criteria.

Table 8-2: Acceptance Criteria for Protection of Human and Non-Human Biota from Non-Radioactive Contaminants

Species	Media			
	Groundwater (µg/L)	Soil (µg/g)	Surface Water (µg/L)	Sediment (µg/g)
Ag	0.3	0.5	0.1	0.5
As	13	11	5	6
B	1700	36	200	-
Ba	610	210	-	-
Be	0.5	2.5	11	-
Br	-	-	1700	-
Cd	0.5	1	0.017	0.6
Chlorobenzene	0.01	0.01	0.0065	0.02
Chlorophenol	0.2	0.1	0.2	-
Co	3.8	19	0.9	50
Cr	11	67	1	26
Cu	5	62	1	16
Dioxins/Furans	1.5E-5	7E-6	0.3	-
Gd	-	-	7.1	-
Hf	-	-	4	-
Hg	0.1	0.16	0.004	0.2
I	-	-	100	-
Li	-	-	2500	-
Mn	-	-	200	-
Mo	23	2	40	-
Nb	-	-	600	-
Ni	14	37	25	16
PAH	0.1	0.05	0.0008	0.22
Pb	1.9	45	1	31
PCB	0.2	0.3	0.001	0.07
Sb	1.5	1	20	-
Sc	-	-	1.8	-
Se	5	1.2	1	-
Sn	-	-	73	-
Sr	-	-	1500	-
Te	-	-	20	-
Tl	0.5	1.0	0.3	-
U	8.9	1.9	5	-
V	3.9	86	6	-
W	-	-	30	-
Zn	160	290	20	120
Zr	-	-	4	-

8.2 Assessment Approach

The safety assessment has been undertaken in a systematic manner, consistent with CNSC guidance (Chapter 7.0 of CNSC06a) and with international best practice, as embodied in the IAEA draft safety guide DS-355 on the safety case and safety assessment for radioactive waste disposal (IAEA08a) and the recommendations of the IAEA program for ISAM (IAEA04b).

Figure 8-1 provides a schematic of the process used to develop the safety assessment. The steps are described below.

1. The assessment context is defined, documenting the high-level assumptions and the constraints, notably regulatory requirements and assessment timeframe (Section 8.3).

2. The system is described, with current information on the waste, repository, geological setting and surface environment pertinent to postclosure safety (Section 8.4).

3. A range of potential future evolutions (scenarios) is systematically identified, ranging from likely ("expected") to very unlikely ("what if") (Section 8.5).

4. Conceptual and mathematical models are developed for these scenarios (Section 8.6, Normal Evolution Scenario; Section 8.7, Disruptive Scenarios).

5. The scenarios are analyzed and the results are assessed with respect to the performance of the system, its overall robustness, and the nature and role of key uncertainties (Sections 8.6, Normal Evolution Scenario; Section 8.7, Disruptive Scenarios; Section 8.8, Uncertainties).

The safety assessment is conducted as part of an iterative process in conjunction with site characterization, waste characterization and facility design. Quality management, including software and data control, are described in Section 8.6.2.8.

Uncertainties are addressed primarily through deterministic calculation cases. These cases include a reference case that provides the most accurate representation of the DGR system, and then a series of cases with generally more conservative assumptions with respect to processes or parameter values. See Section 8.6.2.7 for the set of calculation cases, and Section 8.8 for evaluation of the assessment uncertainties.

As CNSC G-320 (CNSC06a) and IAEA draft safety guide DS-355 (IAEA08a) note, the safety assessment is part of a larger safety case. This overall safety case, including in particular the integration of safety arguments, is addressed in Chapter 14.

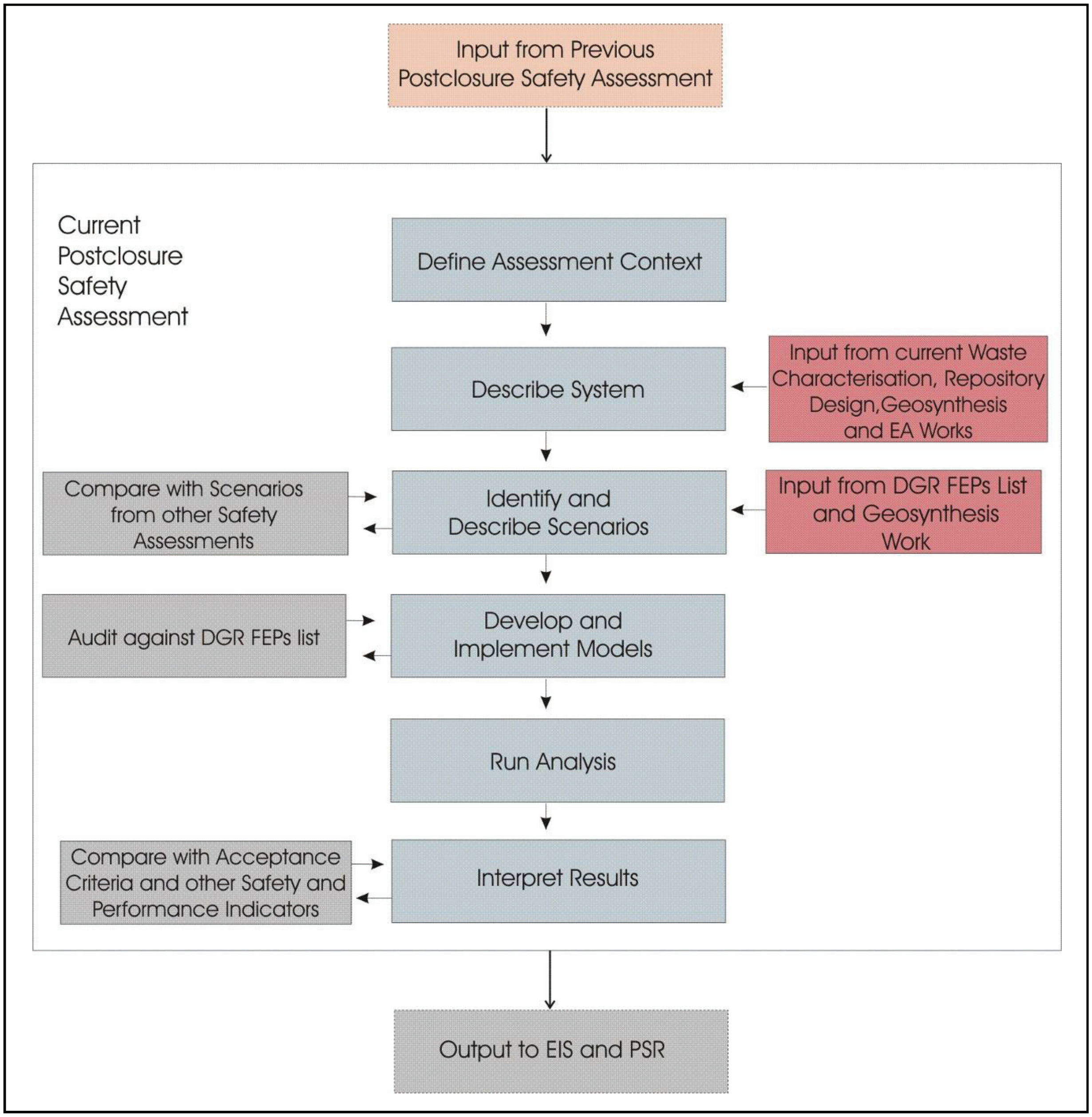


Figure 8-1: Approach Used in the Postclosure Safety Assessment

8.3 Assessment Context

Key components of the assessment context are summarized below:

Purpose:	Quantitatively assess the postclosure radiological and non-radiological safety of the proposed DGR.
Background:	Regulatory framework: Section 1.4.1 Geosphere description: Chapter 4 Waste inventory: Chapter 5 Facility description: Chapter 6 Shaft seal description: Chapter 13
Endpoints:	Radiation dose to humans. Environmental concentrations of radionuclides and non-radioactive species. Contaminant amounts or fluxes within various spatial domains.
Acceptance Criteria:	Radiological criteria to human (Section 8.1.1 and 8.1.2) and to non-human biota (Section 8.1.3). Non-radiological criteria to human and non-human biota (Section 8.1.4).
Treatment of Uncertainties:	Consideration of a range of scenarios, from expected (likely) to "what if" (very unlikely) scenarios. Use of conservatism in scenarios, models and data. Use of a stylized approach for the representation of future human actions and biosphere evolution. Deterministic calculation cases to explore uncertainties in models and data. Probabilistic assessment for a reference case condition.
Timeframe:	one million years (1 Ma) baseline. Encompasses the period over which most radioactivity in the waste has decayed and the maximum risk is expected to occur. Some analyses extended beyond 1 Ma to estimate the maximum impacts from some scenarios.

8.4 System Description

A high-level description of the DGR system is provided below.

Waste:	The total emplaced volume of L&ILW is approximately 200,000 m^3 (Table 5-7). The wastes are emplaced in a range of steel and concrete waste containers and overpacks. See Table 5-8 for radionuclide inventory, and Table 5-10 for chemical and element inventory. The total activity at closure is about 16,000 TBq. Key radionuclides in terms of total activity include H-3, C-14, Ni-63, Nb-94 and Zr-93. The waste generates about 2 kW of decay heat at time of closure. See Chapter 5 for further details.
Repository:	The repository is at a depth of around 680 m and comprises two shafts, a shaft and services area, access and return ventilation tunnels, and 31 waste emplacement rooms in two panels (Figure 6-7). The repository is not backfilled. At closure, a concrete monolith is placed at the base of the shafts (Figure 13-1), and then the shafts are backfilled with a sequence of materials (bentonite/sand, asphalt, concrete and engineered fill) (Figure 13-2). See Chapters 6 and 13 for further details.
Geological Setting:	The DGR is located in low permeability Ordovician argillaceous limestones, with 230 m of shales above and 160 m of limestones below. Significant underpressures exist in the Ordovician rocks, whereas overpressures exist in the Cambrian below the DGR. Above the Ordovician shales, there are 325 m of Silurian shales, dolostones and evaporites. The porewater in the Silurian and Ordovician sediments is highly saline (TDSs of 150 to 350 g/L) and reducing, with pH buffered by carbonate minerals. Above the Silurian sediments, there are 105 m of Devonian dolostones, the upper portions of which contain fresh, oxidizing groundwater that discharges to Lake Huron. Site investigations at the Bruce nuclear site have not found commercially viable mineral or hydrocarbon resources. See Chapter 4 for further details.
Surface Environment:	The present-day topography is relatively flat and includes streams, a wetland, and, at a distance of approximately 1 km, Lake Huron. The annual average temperature is about 8 °C with an average precipitation rate of around 1.1 m/year. The region around the Bruce nuclear site is mainly used for agriculture, recreation and some residential development. Groundwater is used for municipal and domestic water in this region, while the lake provides water for larger communities. The lake is used for recreation and commercial fishing. A significant aboriginal traditional activity in the region is fishing in Lake Huron. See Chapter 2 for further details.

The deep geologic repository provides the high-level safety functions of isolation and containment of the L&ILW. The site and design support these safety functions through a variety of safety relevant features or attributes, as summarized below:

Site Geology	- Multiple low-permeability bedrock formations enclose the DGR. - Predictable, horizontal geology with large lateral extent. - Stable deep diffusion-dominated groundwater system, even under glaciation. - Seismically quiet. - Geomechanically stable rock. - Low natural resource potential. - Low rock permeability limits the rate of repository resaturation. - Ordovician underpressures provide a convergent flow system. - Guelph Formation and Salina A1 Unit upper carbonate are permeable formations that can divert gas or solutes migrating upwards from repository via geosphere or shaft. - Chemical conditions limit contaminant mobility.
Layout	- DGR located at 680 m depth in thick limestone formation. - Shafts are placed in an islanded arrangement separate from waste panels. - Waste emplacement rooms are not backfilled, providing space for gas. - Waste emplacement rooms are aligned with rock principal stress and have thick room pillars for mechanical robustness.
Shaft	- Concrete monolith at base of shafts provides long-term structural support of the shaft seals; it also helps delay water and gas flow. - The bentonite/sand mix in the shafts is the primary seal; it is a durable low-permeable material that can swell under DGR saline conditions. - The asphalt mix is a secondary shaft seal that provides an independent self-sealing barrier to transport. - The concrete bulkheads at the Guelph and Salina A1 levels isolate the bentonite from flow in these units, and provide structural support for the overlying seals. - The shaft concrete liner and HDZ are removed before the shaft seals are installed. - Engineered fill is used in the shaft in the shallow groundwater zone, and topped with a concrete cap. - Site characterization boreholes are sealed when no longer needed.
Waste and packages	- Waste packages are not designed for long-term integrity. - Corrosion resistant Zircaloy delays release of the long-lived radionuclides Nb-94 and Zr-93. - 80% of the waste volume is LLW. - The most important radionuclides at closure are tritium and C-14 due to their early release as gas. Tritium decays within a few hundred years; C-14 decays in about 60,000 years, likely before the onset of glaciation at site.

The preliminary design of the repository is described in Chapter 6. The underground layout of the rooms and access tunnels is shown in Figure 8-2. It is based on flow-through ventilation without ducting.

However, the postclosure safety assessment was initiated using the original preliminary design, also shown in Figure 8-2, which used single-ended emplacement rooms with return ventilation ducting. The handling and emplacement basis for the T-H-E liner wastes was also changed, allowing for simpler handling within the DGR. These design changes were made for operational safety and reliability reasons. In the final preliminary design, the final overall excavated volume was a little larger, while the size of the tunnels and shaft stations were reduced. The shaft seal concept was unchanged.

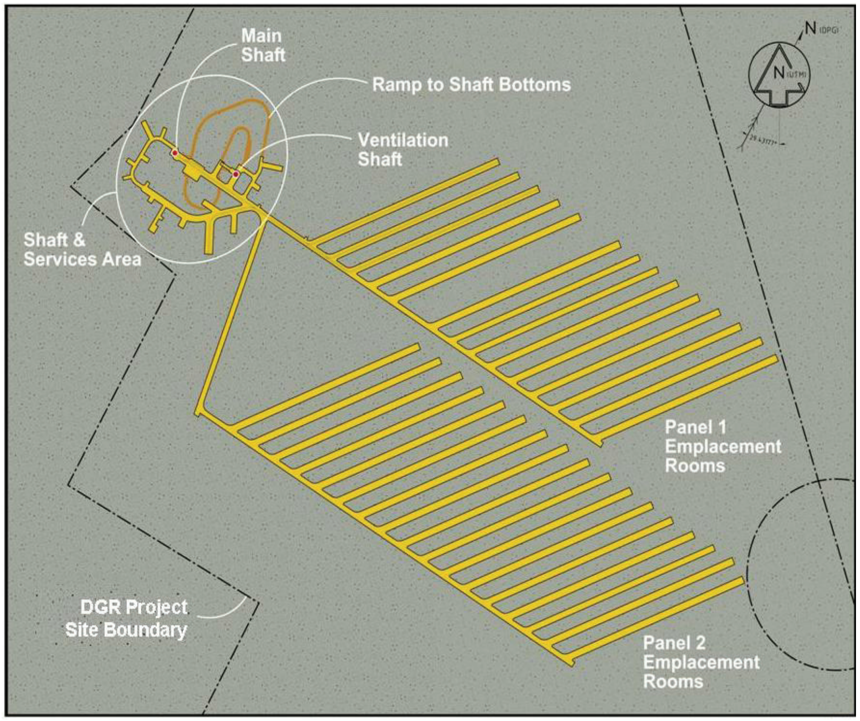
The change from the original to the final preliminary design was made after the present assessment was largely complete. To demonstrate that the postclosure assessment conclusions were not changed, additional calculation cases were undertaken based on the final preliminary design.

8.5 Scenario Identification

The postclosure safety of the DGR is assessed through considering a range of potential future scenarios. These are not intended to predict the future, but rather to identify a range of possible future evolutions of the DGR against which the performance of the system can be assessed.

In order to define these scenarios, the analysis considers the various external, internal and contaminant factors that could affect the DGR system and its evolution. These factors may be further categorized as FEPs.

A structured list has been prepared to organize the potential factors (Figure 8-3). This list is based on experience in various international repository programs, in which similar processes were followed to identify scenarios (NEA99).



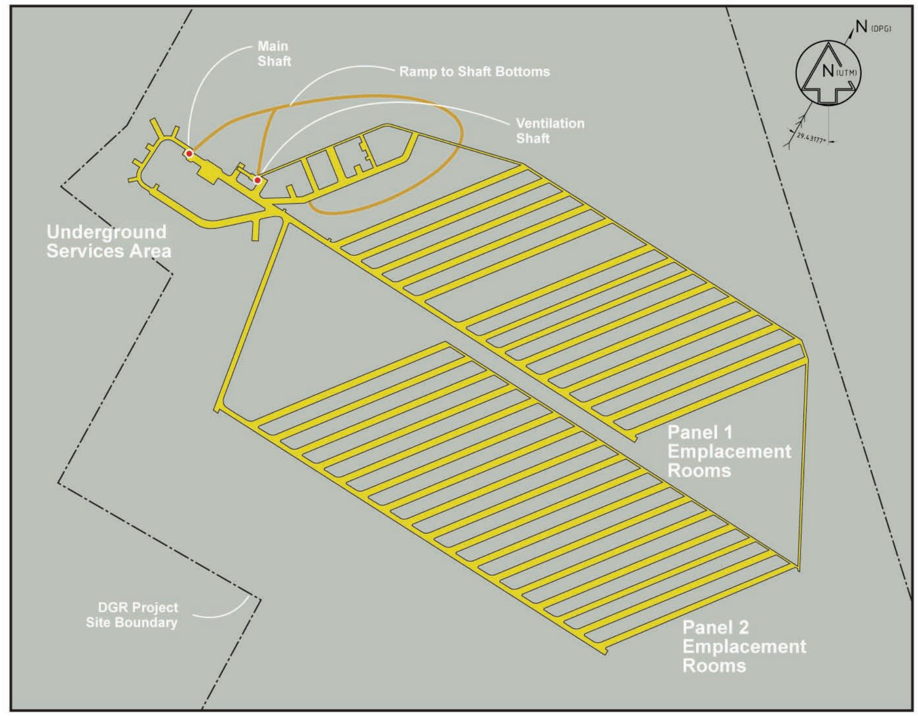


Figure 8-2: Original (Left) and Final (Right) Preliminary Design

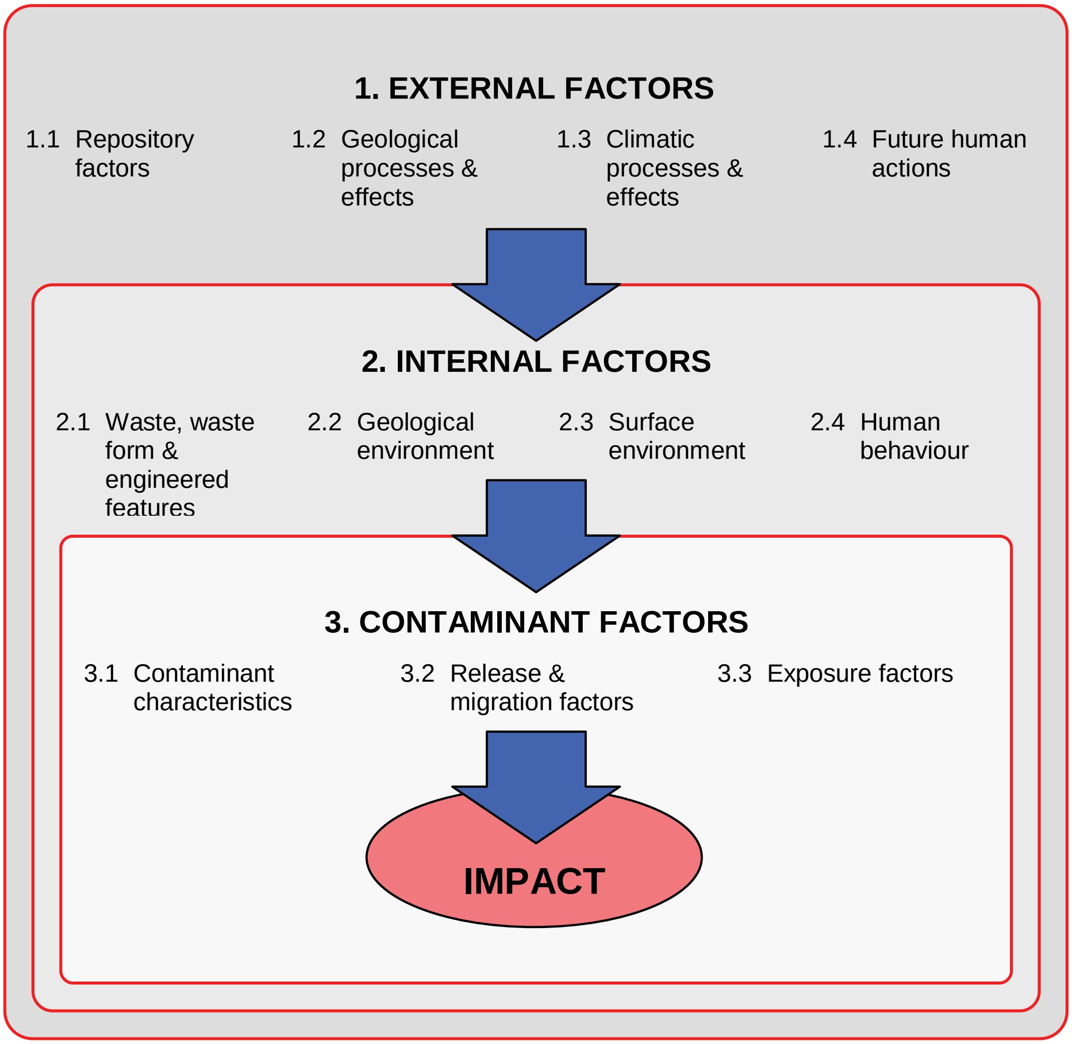


Figure 8-3: External, Internal and Contaminant Factors

The internal and contaminant factors (Internal FEPs) occur within the spatial and temporal boundaries of the DGR system, whereas the external factors (External FEPs) originate outside these boundaries. The External FEPs provide the system with its boundary conditions, and in particular include factors originating outside the DGR that might cause change in the system. Included in this group are decisions related to repository design, operation and closure since these are outside the temporal boundary of the postclosure behaviour of the DGR.

If these External FEPs can significantly affect the evolution of the system and/or its safety functions (i.e., isolation and containment) within the assessment timescale (one million years), they can be considered to be scenario-generating FEPs in the sense that if they occur they could define a particular future scenario that should be considered within the postclosure safety assessment.

The External FEPs were based on the reference list prepared by the NEA. Table 8-3 (NEA99) was based on experience in several national repository programs. These External FEPs were then analyzed to determine whether they were likely to affect the DGR system and its evolution over the next one million years. Table 8-3 indicates which External FEPs were considered relevant to the likely future evolution of the DGR. This assessment is documented in the Postclosure Safety Assessment report (NWMO11af) and the FEPs report (NWMO11ag).

Potential unlikely or alternative states for these External FEPs were then considered in order to identify additional scenarios that could compromise the long-term safety. As a further check, the potential for Internal FEPs to compromise the system's long-term safety was also evaluated. As the long-term safety of the DGR is based on the strength of the geosphere barrier and the shaft seals, the unlikely states considered focus on those scenarios in which these can be bypassed.

The range of potential factors were considered, but ruled out on various grounds as described in the Postclosure Safety Assessment report (NWMO11af) and the FEPs report (NWMO11ag). For example, small earthquakes are considered within the main scenario. Large earthquakes are unlikely at the site, but may occur in connection with glacial cycles. However, the effects of a large earthquake are bounded by the other selected scenarios so this is not evaluated as an explicit separate scenario. Similarly, repository gas pressures do not rise high enough to cause fracturing of the rock, so this scenario is not evaluated.

Consistent with G-320 (CNSC06a) and the CEAA/CNSC EA guidelines (CEAA09), the resulting scenarios are classified into those that consider the expected evolution of the DGR system with time (i.e., the Normal Evolution Scenario) and those that examine the potential impacts of low-probability events leading to penetration of barriers and abnormal loss of containment (i.e., Disruptive Scenarios or “What if” scenarios). The scenarios are listed in Table 8-4.

Table 8-3: External FEPs Considered

1.1	Repository Factors	
	1.1.01	Site investigations *
	1.1.02	Design of repository *
	1.1.03	Schedule and planning *
	1.1.04	Construction *
	1.1.05	Operation *
	1.1.06	Waste allocation *
	1.1.07	Repository closure *
	1.1.08	Quality assurance *
	1.1.09	Repository administrative control *
	1.1.10	Accidents and unplanned events
	1.1.11	Retrieval
	1.1.12	Repository records and markers *
	1.1.13	Monitoring
1.2	Geological Processes and Effects	
	1.2.01	Tectonic movement
	1.2.02	Orogeny
	1.2.03	Seismicity *
	1.2.04	Volcanic and magmatic activity
	1.2.05	Metamorphism
	1.2.06	Hydrothermal activity
	1.2.07	Denudation and deposition (large-scale) *
	1.2.08	Diagenesis
	1.2.09	Pedogenesis *
	1.2.10	Salt diapirism and dissolution
	1.2.11	Hydrological response to geological changes *
	1.2.12	Geomorphologic response to geological changes
	1.2.13	Deformation (elastic, plastic or brittle) *
1.3	Climate Processes and Effects	
	1.3.01	Global climate change *
	1.3.02	Regional and local climate change *
	1.3.03	Sea-level change
	1.3.04	Periglacial effects *
	1.3.05	Local glacial and ice-sheet effects *
	1.3.06	Warm climate effects (tropical and desert)
	1.3.07	Hydrological response to climate changes *
	1.3.08	Ecological response to climate changes *
	1.3.09	Human behavioural response to climate changes *
	1.3.10	Geomorphologic response to climate changes *
1.4	Future Human Actions (Active)	
	1.4.01	Human influences on climate *
	1.4.02	Social and institutional developments *
	1.4.03	Knowledge and motivational issues (repository)
	1.4.04	Drilling activities *
	1.4.05	Mining and other underground activities
	1.4.06	Un-intrusive site investigations
	1.4.07	Surface excavations
	1.4.08	Site development *
	1.4.09	Archaeology
	1.4.10	Water management (groundwater and surface water) *
	1.4.11	Explosions and crashes
	1.4.12	Pollution
	1.4.13	Remedial actions
	1.4.14	Technological developments
	1.4.15	Deliberate human intrusion
1.5	Other External Factors	
	1.5.01	Impact of meteorites and human space debris
	1.5.02	Evolution of biota
Notes: * Considered relevant to likely future evolution of DGR system over 1 Ma timescale. See Postclosure Safety Assessment report (NWMO11af) or the FEPs report (NWMO11ag).		

Table 8-4: Scenarios Evaluated in the Postclosure Safety Assessment

Normal Evolution Scenario	The Normal Evolution Scenario is the expected long-term evolution of the repository and site following closure. Over the 1 Ma assessment timescale, the scenario includes waste and packaging degradation, gas generation and build up, rockfall, earthquakes and, eventually, glacial cycles.	
Disruptive ("What if") Scenarios	Human Intrusion	Inadvertent intrusion into the DGR via an exploration borehole.
	Severe Shaft Seal Failure	Poorly constructed or substantially degraded shaft seal.
	Poorly Sealed Borehole	Poorly sealed or substantially degraded seals in site investigation/monitoring borehole.
	Vertical Fault	Transmissive vertical fault in the vicinity of the DGR.

The Normal Evolution Scenario is described and analyzed in Section 8.6, while the Disruptive Scenarios are described and analyzed in Section 8.7.

8.6 Normal Evolution Scenario

8.6.1 Scenario Description

The Normal Evolution Scenario is the expected long-term evolution of the repository and site following closure. The following high-level narrative of this system evolution is based on the more detailed discussion presented in the System and its Evolution report (NWMO11ah) and the FEPs report (NWMO11ag). This description is then used to develop the quantitative assessment model in Section 8.6.2.

The heat generated by radioactive decay within the repository is small – about 2 kW at the time of closure and decaying. This is low relative to the steady natural geothermal flux through the repository panel footprint of 10 kW. The repository will remain near its natural ambient temperature condition of around 20^{o}C.

During the years following closure, there will be corrosion of the carbon steel containers and degradation of organic materials in the wastes. The atmosphere in the repository will become anaerobic as oxygen is consumed by corrosion. Subsequent degradation of the wastes and packaging materials in the DGR will proceed by slower anaerobic processes. These anaerobic processes will generate various decomposition products, but in particular will generate gases – especially H_2 from the corrosion of metals, and CO_2 and CH_4 from the microbial decomposition of organics. The radioactivity in the waste may locally enhance corrosion in some packages, but overall it is too low to generate appreciable radiolytic gases.

These corrosion/degradation reactions usually require water. There is a small amount of water initially present in the wastes, but continued corrosion will depend on water seeping into the DGR from the host rock and/or shafts. Since the surrounding host rock and the shaft seals have low permeability, the rate of water supply may limit the corrosion/degradation rate. Furthermore, the water that starts to collect in the repository will have high salinity, as well as potentially high dissolved metals that discourage microbial activity and delay degradation.

As the wastes and packaging corrode and degrade, producing gases, the gas pressure inside the repository will increase (Figure 8-4). Due to the low permeability of the host rock and shaft seals, most of the gases will be retained within the repository void space and hence the gas pressure in the repository can rise to levels at or slightly above the steady-state hydraulic pressure at the repository depth (Chapter 5 of NWMO11aj). At higher gas pressures, both repository water and gas would be pushed into the surrounding rock and shaft, such that the system tends to equilibrate back towards the normal hydrostatic pressure.

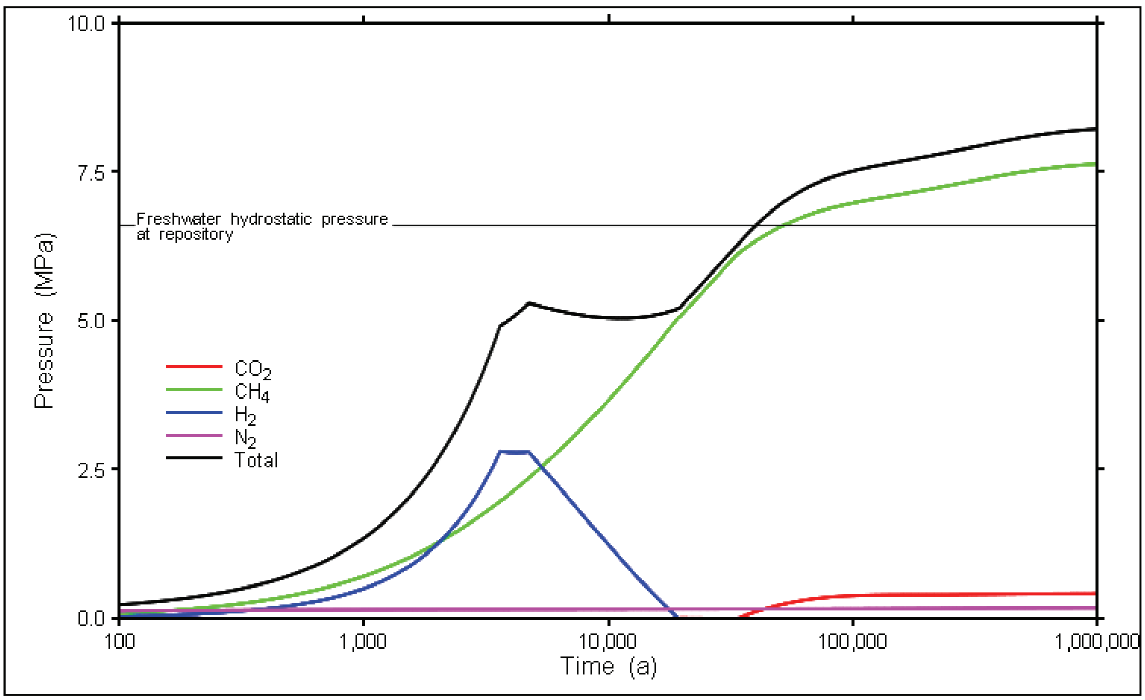
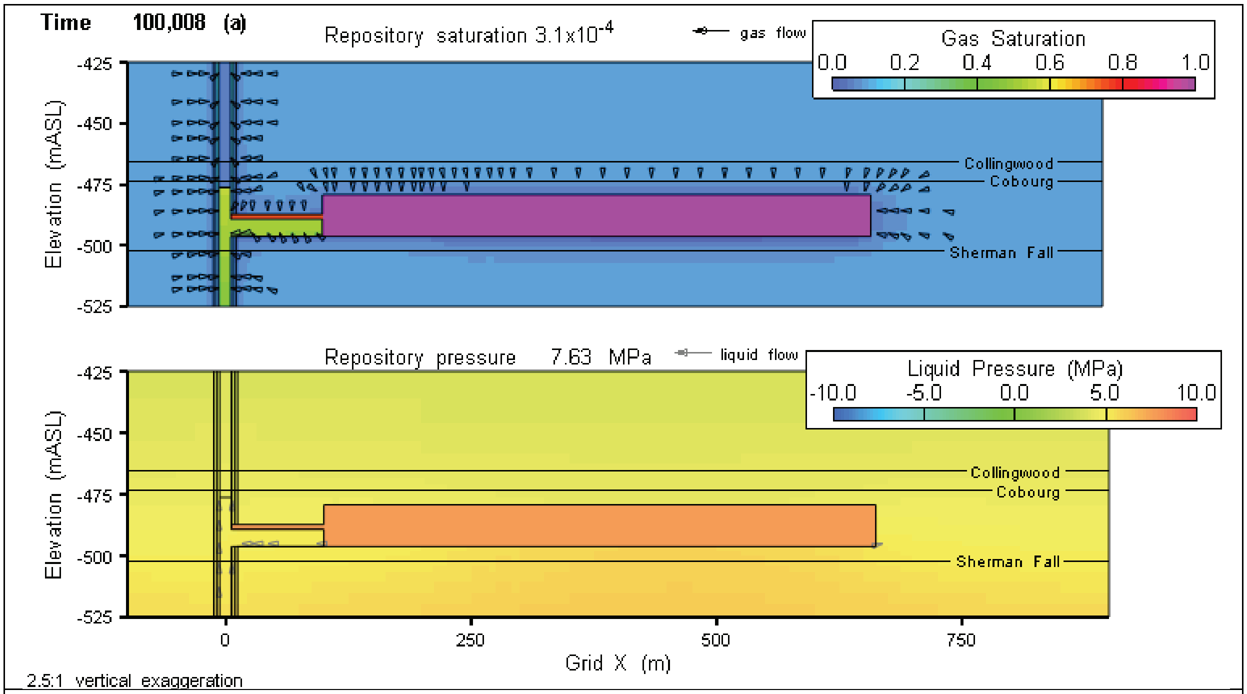


Figure 8-4: Repository Gas Pressure and Composition for the Normal Evolution Scenario Reference Case

The low permeability of the host rock around the repository, plus the gas pressure in the repository and the water consumption by corrosion reactions, limit the rate of water saturation of the repository. Calculations for reference conditions show repository water saturation will remain low. The shafts will resaturate more rapidly than the DGR, because they are backfilled (smaller volume), are exposed to more permeable rock formations, and do not have gas generation.

Figure 8-5 shows the saturation profile and pressures in the repository and adjacent rock at about 100,000 years after most of the gas generation has occurred (Chapter 5 of NWMO11aj). At this time the repository is virtually 100% gas, while the shaft and surrounding rock are at around 10% gas saturation (within the rock porosity of 1-10%), the initial estimated gas content of these rocks. The concrete monolith at the shaft base and a small region of rock above the monolith are largely unsaturated. There is slow gas movement from the surrounding rock into the repository and eventually through the monolith area and into the shaft.



Notes:
The host rock has an initial hydraulic pressure of about 5 MPa and a 10% gas saturation. The repository is represented as a porous volume that includes the maximum extent of rockfall.

Figure 8-5: Saturation, Flows and Pressures around the Repository for the Normal Evolution Reference Case after about 100,000 Years

The primary gases generated in the repository will be H_2, CO_2 and CH_4 from corrosion and organic degradation, with small amounts of N_2 left from the initial air in the DGR. CO_2 will equilibrate with the limestone (carbonate) minerals and with carbonate in any repository water; some will react with repository cement to form calcite or with iron to form siderite. H_2 and CO_2 contain energy that can be used by microbes, especially methanogens producing CH_4. In the long-term, the repository will contain mostly methane gas, consistent with natural gas reservoirs in sedimentary rocks.

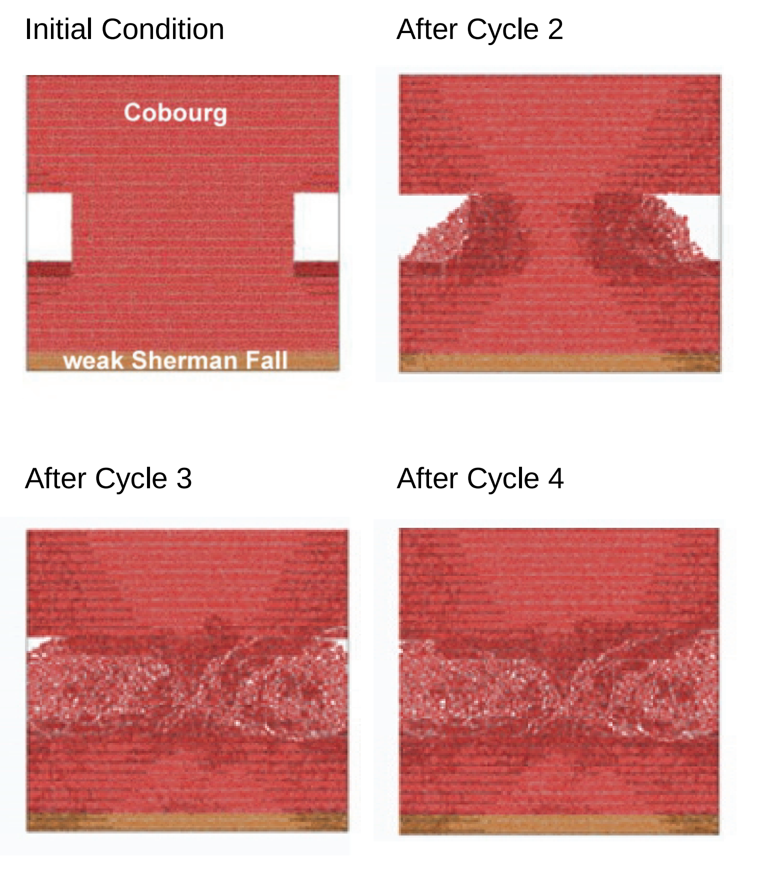
Figure 8-4 illustrates the calculated gas pressure and composition as a function of time for the Normal Evolution Scenario Reference Case (NWMO11aj). The figure shows the dominance of methane. Very little free CO_2 is present because it has reacted – in this case predominantly by methanogenesis with hydrogen.

The large amount of carbonate rock around the repository will act as a chemical buffer, tending to bring the water chemistry to neutral pH. The carbonate will balance the tendency to high pH from the cement present in the repository (in the floor and waste packages) and the tendency to low pH from CO_2 gas. Calculations indicate that only a small amount of carbonate rock will dissolve under these conditions (Section 4.5.1 of NWMO11ah).

The host rock around the repository has good rock mechanical quality, the rooms will be aligned with the in-situ stress conditions for maximum stability, and there is a thick pillar between rooms. Furthermore, the region has low seismic activity; large earthquakes are very unlikely. Consequently, the rooms are expected to remain open for thousands of years.

However, since the rooms are not completely filled by waste or backfill, it is expected that rockfall from the roof and walls will occur, due to eventual degradation of engineered rock support and, in the longer term, due to seismic and/or glacial events. This process will continue intermittently over a period of a few hundred thousand years, until the collapsed rock fills the available space and is able to support the roof and prevent further failure (the fallen rock takes up more volume than intact rock, so the room initial void space is redistributed over a larger volume but the total amount is unchanged). Modelling results are shown in Figure 8-6. It is calculated that the rockfall will propagate about 10 m into the repository roof before it stabilizes, and therefore will not affect the overlying geological formations (Section 6.4.4 of NWMO11c).

The peak gas pressures are in the range of 7 to 9 MPa, similar to or above the equilibrium hydraulic pressure at the repository level (about 7.4 MPa hydrostatic head; about 7.8 MPa in steady-state with the Cambrian overpressure). This pressure is well below the 17 MPa lithostatic pressure, and the 20-30 MPa horizontal rock stresses, so this gas pressure is retained with the repository and does not cause fracturing. This calculated peak repository gas pressure is consistent with the gas pressure in natural gas fields. Geomechanical modelling of the DGR showed no fracturing at peak gas pressures of 7 MPa; and formation of several metre long horizontal fractures at 15 MPa (Section 6.4.4 of NWMO11c).



Note: Adapted from Section 4.5.2.5.

Figure 8-6: Rockfall Within and Around the Emplacement Rooms After Four Glacial Cycles

Most of the waste packaging is not designed to be long-lived after closure. As the packages corrode, repository water will contact the wastes. This is particularly true for the large volume of LLW present in simple carbon steel containers. The higher activity ILW containers are more robust, and will take longer to degrade. With rockfall, all packages eventually fail. However, the failed packages may continue to provide some physical or chemistry control (e.g., alkalinity in concrete containers) inhibiting the release of contaminants, especially the ILW retube and resin containers.

Contaminants will be released from the waste due to the generation of gases and due to contact with water. The rate of release of contaminants to gas and water will vary with the type of wastes. Release of H-3 will be fast. Release of C-14 will also occur relatively quickly, as most of this is in surface sorption on IX resins. It will be released as gas from unsaturated wastes, or into water as the repository saturates. Within the water, carbon will be primarily as carbonate/bicarbonate, at levels controlled by equilibrium with the surrounding limestone rock. C-14 will equilibrate between the carbon in gas and carbon in water within the repository. Some will also react with cement and iron and precipitate as calcite or siderite, and some will exchange with stable carbon in the surrounding limestone rock. Release of the long-lived Nb-94 and Zr-93, however, will be slow, since these are embedded within the corrosion-resistant Zircaloy pressure tubes (which are placed in robust steel and concrete containers).

The contaminants will be contained by the low-permeability host rock and shaft seals. The radioactivity of the waste will drop to less than the natural activity of the overlying rock after about 100,000 years. However, migration of some dissolved or gaseous contaminants will occur via the geosphere and the shafts.

The shafts are sealed primarily with a bentonite/sand mixture that will swell and self-seal within the shaft. An asphalt seal may be emplaced in the Ordovician shales to provide an independent seal material, and the concrete monolith and bulkheads will provide mechanical support as well as an initial low-permeability barrier. The shaft seal concept is similar to that of the WIPP facility (HANSEN00).

In the longer term the concrete is expected to degrade due to mechanical stresses and chemical reactions. The bentonite/sand and asphalt will likely interact at their interfaces with other materials, especially the concrete. However the use of low-heat sulphate-resistant cement, the low permeability of the seals and the rocks, and the low temperature in the shaft, will limit the extent of interaction (Section 4.5 of NWMO11ah). Some reactions will increase porosity, while other reactions will reduce porosity (e.g., salt precipitation, cement carbonation).

The shaft seal system also includes an EDZ. This is a more permeable ring of rock at the interface between the rock and the shaft seals. Geomechanical modelling indicates that the shaft EDZ forms early after excavation, and then remains stable with time, even under glaciation loading (Section 4.5.2.4). Self-sealing of the EDZ in the long-term is not considered in the present scenario, although this can occur in other sedimentary rocks (NEA10). Also, the safety assessment assumes that the EDZ is not interrupted by shaft bulkheads and seals.

Upward contaminant migration from the repository would be delayed by the underpressures in the Ordovician shales, which tend to pull flow into these rocks including from above (i.e., downward water flow). The permeable Guelph Formation and Salina A1 Unit upper carbonate rock formations above the Ordovician shales also act as a transport barrier to vertical transport – it is easier for gas to move horizontally in these porous rocks than vertically upwards, and any advective groundwater flow in these units would entrain dissolved contaminants. Finally, chemical sorption, precipitation and exchange reactions will also slow down the movement of contaminants. (Chapter 5 of NWMO11aj).

Some contaminants may eventually discharge to the shallow groundwater zone, and then into the surface environment. People living on or near the site could be exposed to these contaminants through the use of groundwater drawn from a well, through fishing and swimming in the lake, and through the use of local land for farming, hunting, recreation, and dwelling. Natural dilution and dispersion in the shallow groundwater zone and lake would limit the potential impacts, especially for anyone not living directly above of the repository.

Important radionuclides in the repository are H-3 (tritium) and C-14, because they are a significant fraction of the total radioactivity at repository closure; they can be relatively easily released from the wastes; and they are mobile in both gas and water. However, the repository will ensure containment for the few hundred years required for tritium to decay to negligible levels. Therefore, the 60,000 year period after closure (around ten half-lives) during which C-14 significantly decays, is of particular interest. Other residual radionuclides are primarily only mobile within the groundwater pathway.

Beyond the 60,000 year time frame, climate cooling and resulting glaciation may be relevant. There have been nine major glacial cycles in the past million years. Key factors contributing to these cycles – variations in solar insolation to the northern hemisphere and the arrangement of the continents – will not change appreciably over the next million years. Although global warming is likely to delay the onset of the next cycle to beyond 60,000 years, it is assumed that glacial cycling will resume in the long-term, with a periodicity of approximately 100,000 years (Section 4.5.2.1).

Although there could be changes due to global warming in the near term (i.e., over the next thousand years), and these could be important to the ecosystems, the region around the DGR is expected to remain in a broadly temperate climate state until the onset of the next glacial cycle. As climatic conditions eventually cool, the ecosystem around the site will change from temperate to tundra. Human habits would also change, with agriculture becoming less likely for example.

The site will be eventually covered by an ice-sheet, with a few ice-sheet advances and retreats over a typical glacial cycle. The subsequent warming of the climate at the end of the glacial cycle would cause the ice-sheet to retreat, and the re-establishment of tundra and eventually temperate ecosystems around the site. An illustrative climate change sequence for the next 120,000 years is shown in Figure 8-7.

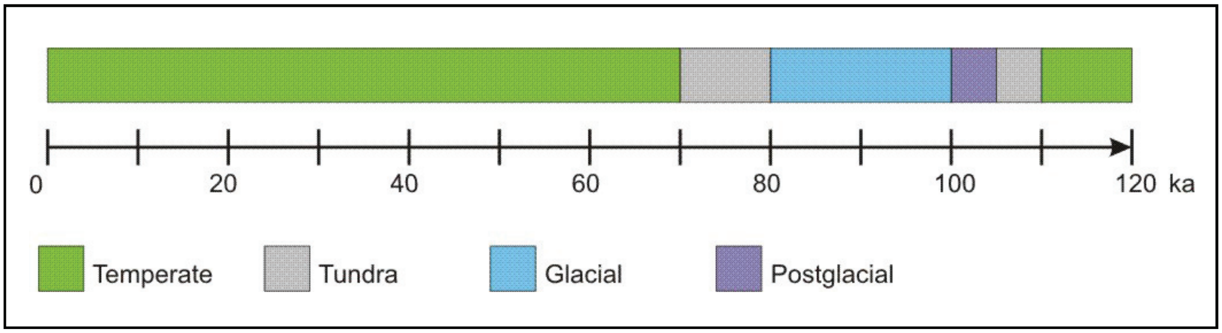
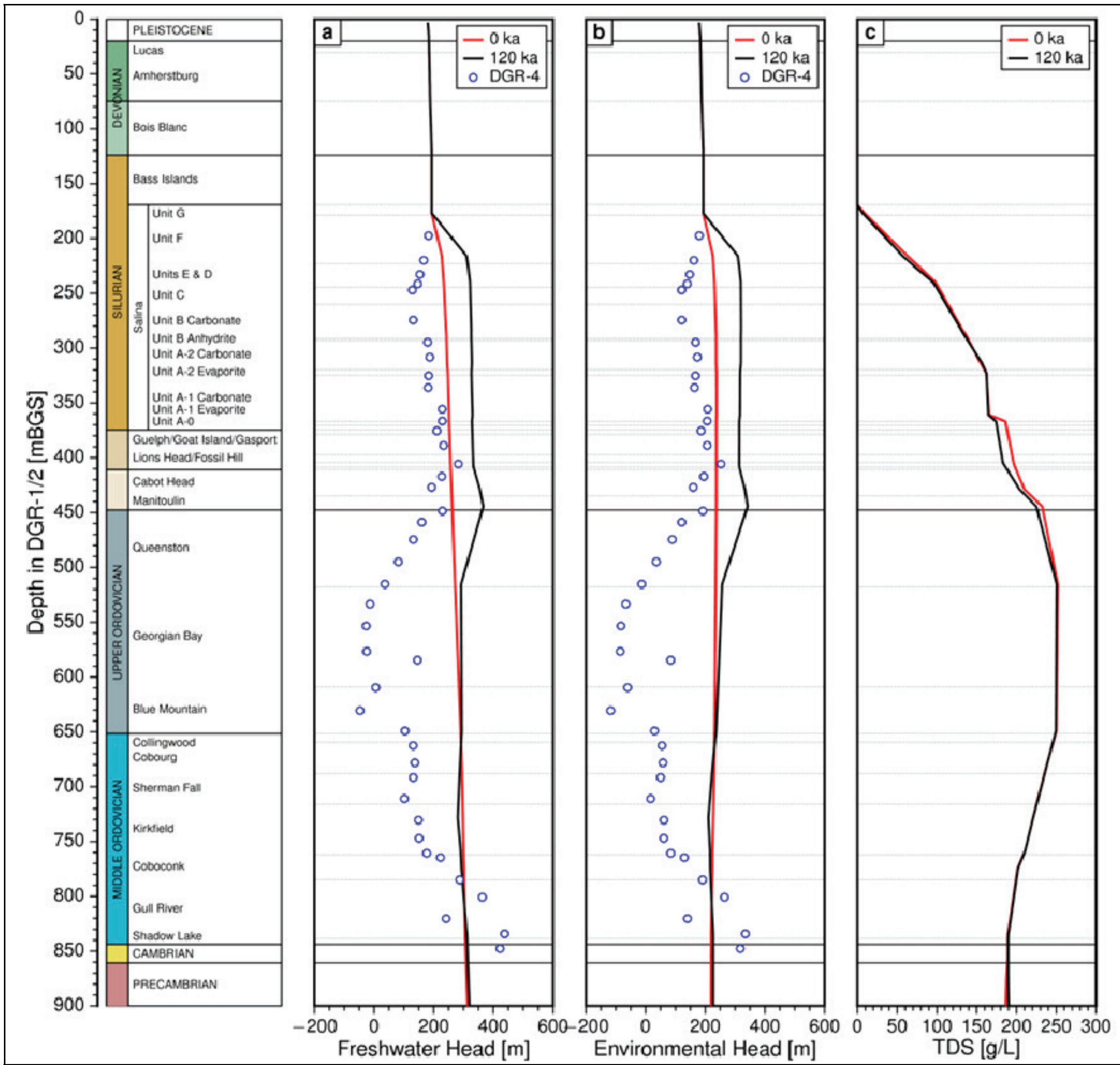


Figure 8-7: Assumed Sequence of Climate States for the Next 120,000 Years

The ice-sheet will cause major changes in the surface and shallow groundwater zone, including permafrost, hydraulic pressures and flow rates, and in the infiltration of glacial waters. Gradients within the permeable formations of the intermediate groundwater zone – Guelph Formation, Salina A1 Unit upper carbonate – will vary in direction and magnitude as the ice-sheets advance and retreat.

However, the impacts of glacial cycles on the deep groundwater zone are expected to be primarily transient changes in the stress and hydraulic pressures resulting from ice-sheet loading and unloading. This is supported by evidence from the site itself, where the deep groundwaters do not show signs of impact from past glaciations, nor are there signs of faulting or fracturing due to glaciation stresses. This is also supported by modelling of the behaviour of the groundwater and geomechanical environment around the repository, presented in Section 4.5.2.5. For example, Figure 8-8 shows model results indicating little change in hydraulic head profiles and salinity profiles over a 120,000 year glacial cycle, especially in the deep Ordovician rocks. The overall rock will remain intact, and contaminant transport will remain diffusion-dominated, as in previous glacial cycles.

In the very long-term, the repository will primarily consist of limestone rock, iron corrosion products and other minerals, methane gas and brine.



Notes:
a) freshwater head, (b) environmental head, and (c) total dissolved solids concentration versus depth at beginning (0 a) and end (120,000 a) of paleoclimate simulation. Freshwater and environmental heads for site characterization borehole DGR-4 are shown. Figure adapted from Figures 5.30 and 5.32 of the Geosynthesis (NWMO11c).

Figure 8-8: Modelling of Glacial Effects at the DGR Site

8.6.2 Models, Implementation and Data

8.6.2.1 Conceptual Model – General

Figure 8-9 and Box 1 summarize the main aspects of the conceptual model for the Normal Evolution Scenario.

Relative to the likely evolution described in Section 8.6.1, the conceptual model analyzed for the safety assessment includes the following simplifications:

- Containers do not provide any barrier to contaminant release;
- Tritium and C-14 are released as gases;
- Solubility limits and sorption are either neglected or conservative values assumed;
- Rockfall occurs quickly after repository closure, damaging packages and increasing the vertical extent of repository;
- Microbial reactions occur as long as moisture is present;
- Water consumption by anaerobic corrosion and gas generating reactions within the repository is not included in the repository water balance;
- Waste organics degrade fully to gas;
- The concrete monolith and bulkheads in the shafts are degraded from the time of closure;
- No horizontal groundwater flow in the Guelph Formation and Salina A1 Unit upper carbonate; and
- Constant simplified biosphere model with family living on top of repository.

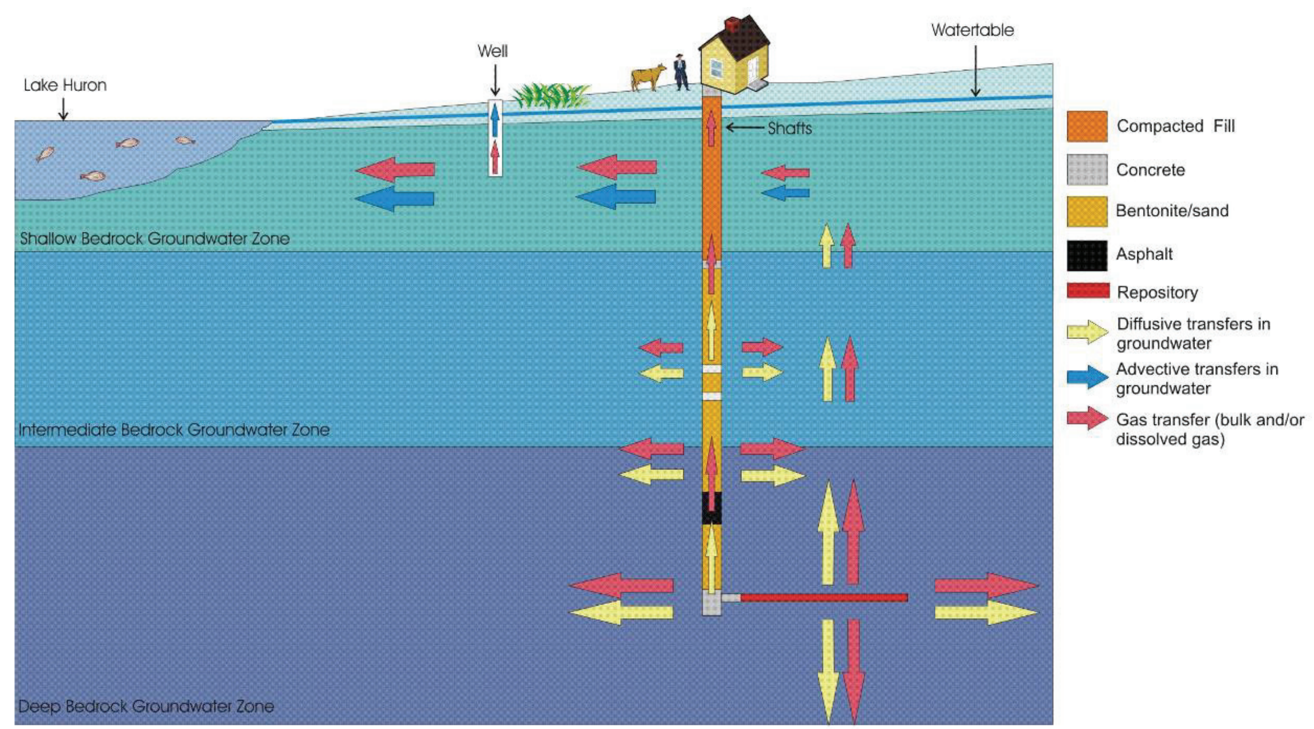


Figure 8-9: Normal Evolution Scenario: Schematic Representation of Potential Transport Pathways

Box 1: Key Aspects of the Conceptual Model for the Normal Evolution Scenario

Waste and Repository

- Reference waste inventory of about 200,000 m^3 (emplaced volume) and 16,000 TBq.
- Reference repository design with no backfill, except concrete monolith at shaft base.
- Rockfall occurs at closure, reaching a stable equilibrium.
- Metals degrade anaerobically to release H_2; organics degrade microbially to release CH_4 and CO_2.
- Resaturation of repository is determined by water inflow/outflow, gas generation, gas inflow/outflow and gas pressure.
- Contaminants are released into water via instantaneous and congruent release processes; no credit is given to waste packaging as a chemical or physical barrier.
- H-3 and C-14 are also released as gas as a result of waste degradation.
- Once released from waste, H-3, C-14, Cl-36, Se-79, I-129 partition between water and gas in the repository.
- No sorption of contaminants in repository, and solubility limitation only for Carbon.
- Contaminants may migrate into the host rock and shafts by diffusion and/or advection.

Geosphere and Shafts

- Very low permeability host rock with no significant fracturing or joints, some anisotropy in diffusion and permeability along versus across bedding planes.
- Underpressures in the Ordovician rocks are present initially but may equilibrate over time.
- Overpressure in the Cambrian sandstone remains constant over assessment timeframe.
- Ordovician rocks are partially unsaturated, with some methane gas.
- No significant groundwater flow in permeable Guelph Formation and Salina A1 Unit upper carbonate.
- EDZs exist around all excavations, including the shafts; no self-sealing due to creep or precipitation processes.
- Some degradation of concrete structures, but no further significant change in bulk properties of shaft seal materials or damage zones occurs over assessment timescale.
- Relative permeability of gas phase is described by van Genuchten models for capillary pressure.
- Contaminants may migrate through the host rock by diffusion.
- Contaminants may migrate up the shafts by diffusion and/or advection in groundwater or gas through the shaft seals and/or damaged zones.
- Zr, Nb, Cd, Pb, U, Np and Pu may sorb in the shafts and geosphere.

Biosphere

- Constant temperate climate conditions.
- Shallow groundwater flow discharges into the near-shore lake bed.
- Groundwater is pumped from a well located downstream from the repository, for domestic and farming use.
- Potential impacts are estimated based on assuming a self-sufficient family farm located on the repository site and using groundwater from well.

8.6.2.2 Conceptual Model – Release of Contaminants from the Waste

Contaminants can be released from the waste to water and air (Figure 8-10).

Releases to **water** occur once repository water has contacted the waste. The majority of the contaminants associated with LLW are expected to be released quickly on contact with water (i.e., instant release). This is because the wastes are in 'light' packaging that is likely to degrade relatively rapidly postclosure and because contamination is generally present on the surfaces of the wastes.

Many of the ILW wastes are packaged more heavily for operational reasons (i.e., with additional containment and shielding). For these wastes, the packaging could form a barrier to water-waste interaction and contaminant release. However, the potential effect of ILW packaging is conservatively ignored.

For some of the ILW wastes, the contamination is present in the matrix of the materials (i.e., the irradiated core components and retube wastes). For these wastes, contaminants only become available for release as the material itself corrodes. Such a process is represented with a congruent release model driven by the corrosion rate.

Solubility limits have not been applied to contaminant releases, except for C-14 where carbonate equilibria control is assumed due to the surrounding limestone rock.

Radioactive trace **gases** are also generated in the form of C-14 labelled CH_4 and CO_2; H-3 released as tritiated water vapour and tritiated hydrogen gas; and I-129, Cl-36 and Se-79 which may be volatilized, particularly if they become methylated by microbial processes in the DGR. Releases can occur under saturated and unsaturated conditions, and none of the waste packages are taken to be gas tight. Therefore, gaseous releases can occur immediately on repository closure.

8.6.2.3 Conceptual Model – Migration of Contaminants from Repository

Once contaminants have been released from the waste into the repository **water**, they can migrate from the emplacement rooms through diffusion into the surrounding EDZ and geosphere, and via advection/diffusion through the concrete monolith and its associated damaged zone at the base of the shafts (Figure 13-1 and Figure 8-11). When the repository is partially saturated, diffusion of contaminants in water can only occur from the base and part of the sides of the repository to the geosphere. During periods of desaturation of the repository due to increasing gas pressure, contaminants in water are forced from the repository by the enhanced gas pressure.

Contaminants dissolved in the water may be retained by sorption and precipitation within the repository. However, the current assessment conservatively neglects sorption in the repository for all elements. It is also assumed that no precipitation of elements occurs, once they have been released into repository water. Precipitation or exchange of C-14 with the carbonate host rock is not considered.

The majority of the **gas** contaminants are retained in the repository due to the low permeability of the host rock. However, some can be released from the repository through dissolution into repository water or porewater within the adjacent host rock and by subsequent migration away from the repository through the host rock or along the access tunnel to the shaft.

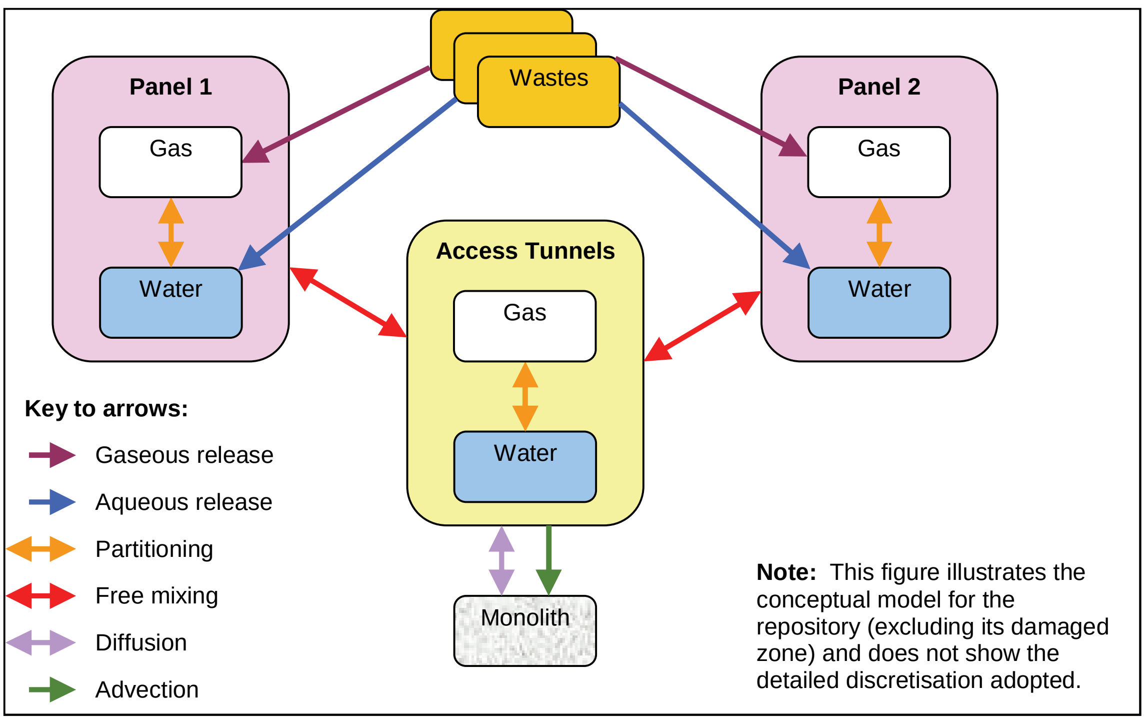


Figure 8-10: Conceptual Model for Repository Contaminant Release and Migration Processes

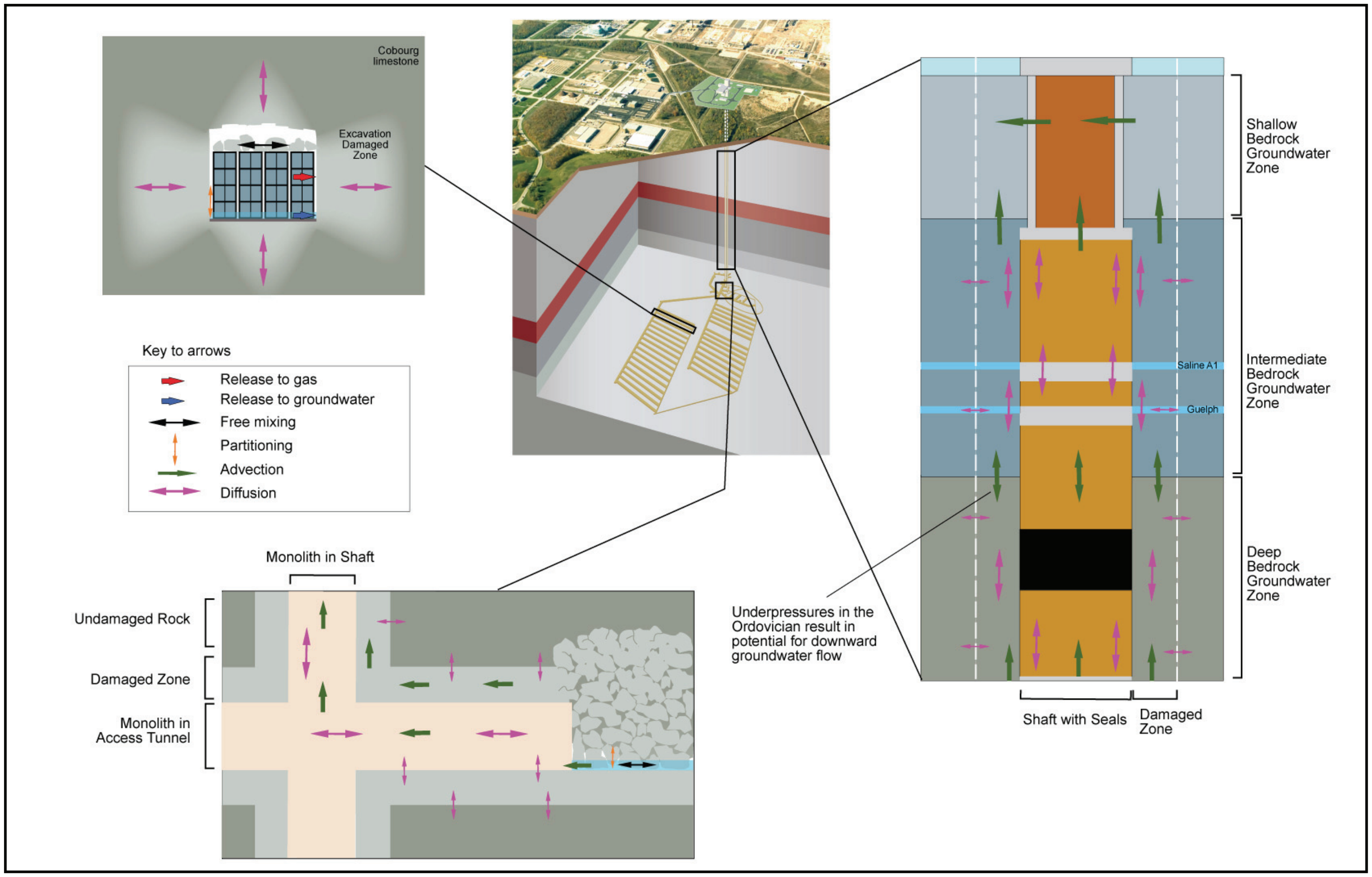


Figure 8-11: Normal Evolution Scenario: Conceptual Model for Repository and Geosphere Contaminant Release and Migration Paths

8.6.2.4 Conceptual Model – Migration of Contaminants Via Geosphere and Shafts

The shafts will be backfilled using a combination of low permeability materials (Figure 13-2). The concrete monolith and bulkheads are partially degraded over time due to chemical reactions and physical stresses. This degradation is assumed to occur from closure in the conceptual model.

During construction of the repository and its shafts, a damaged zone will develop due to the mechanical disturbance and relaxation of the rock into the excavation. This damaged zone is a more permeable pathway than the surrounding rock. Geomechanical modelling (Section 6.4.3 of NWMO11c) indicates that most of the damaged zone develops early after excavation. During repository closure, the shaft liner and part of the damaged zone will be removed in the intermediate and deep groundwater zones (Section 13.6.3.1). Once backfilled with the shaft seal, the damaged zone is supported and does not develop appreciably further. The effect of glacial loads on damaged zone evolution around the shafts in the deep and intermediate groundwater zone is small and incorporated into its parameterization (i.e., the maximum extent of damaged zone is assumed from the start). In the model, processes that could reduce the damaged zone are neglected (e.g., self-sealing due to creep of clay components within the shale or shaft seal, or precipitation of minerals).

Detailed modelling considers both diffusion and advection pathways for contaminant transport in **groundwater** in the host rock. Similarly both processes are considered in detailed modelling of transport in the shafts and their associated damaged zones. Conservative estimates for sorption of certain elements are considered in the geosphere and shafts. Once in the shallow groundwater zone, contaminant transport is advective towards Lake Huron with discharge to the biosphere in the near-shore region (Figure 8-9).

The Guelph Formation and Salina A1 Unit upper carbonate are relatively more permeable than the surrounding formations. Slow topographically driven flow occurs within these formations, but it is limited by the low hydraulic gradients. Groundwater flow in these formations would divert contaminant transport from the shafts and reduce the amount of contamination migrating upwards. Horizontal groundwater flow in the Guelph Formation and Salina A1 Unit upper carbonate is, therefore, ignored in the Reference Case, and contaminants therefore only move upwards within the site area, maximizing the potential local exposure. However, even without flow, these formations provide a more porous and permeable layer into which some of the contaminants that reach this level can diffuse (horizontally), especially free gas.

The low hydraulic gradient in the Cambrian will also limit migration of any contaminants that might have diffused down from the repository. Migration in the Cambrian will be further limited by the long distance to outcrop discharge points (in excess of 100 km).

Certain contaminants (i.e., H-3, C-14, Cl-36, Se-79 and I-129) will be present in the **gas** phase in the repository and have the potential to migrate from the DGR via gas permeation in addition to transport in groundwater. Free gas tends to migrate vertically upwards from the repository, while dissolved gas migration follows the groundwater pathways for both advection and diffusion. The rate of gas permeation through the rock and shaft is a function of the gas pressure, the seal or rock threshold capillary pressure, and the permeability of the media under two-phase flow conditions. At the DGR site, the gas movement is impeded by the very low permeability limestone and shale horizons, and their underpressures.

Depending on the case, gas reaching the shallow system dissolved in groundwater may be released as free gas due to the lower pressures in the shallow system; correspondingly, free gas reaching upper formations may dissolve into groundwater, and some may be swept away in the flowing groundwater in the upper aquifer.

8.6.2.5 Conceptual Model – Migration of Contaminants in the Biosphere

The conceptual model considers stylized, constant temperate conditions which are comparable with those found at present at the site. Consequently, the types of biosphere pathways are similar to those that would occur now.

The main migration path into the biosphere are through the groundwater discharging into the near shore of Lake Huron, through the extraction of groundwater from a well and used for irrigation, and through free gas release. For gas released as methane, some fraction can be converted into CO_2 by methanogenic processes in the soils, where it becomes more biologically accessible. After release through these pathways, the contaminants can transfer into plants or animals or fish, as well as between compartments, such as from surface waters into the lake (Figure 8-12).

The biosphere model includes compartments representing surface water as a stream discharging into the lake (Figure 8-13). The lake (Lake Huron) is represented by a multicompartment model for the several main basins, and includes a site near-shore region.

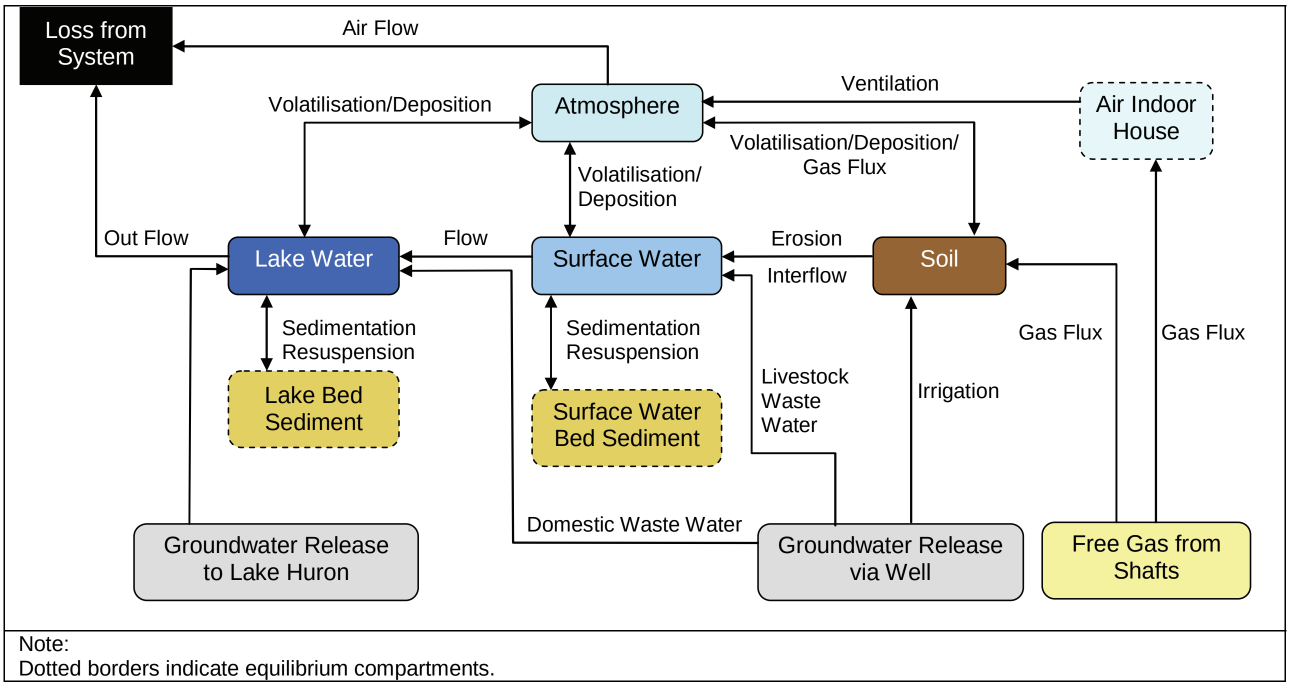


Figure 8-12: Biosphere Conceptual Model

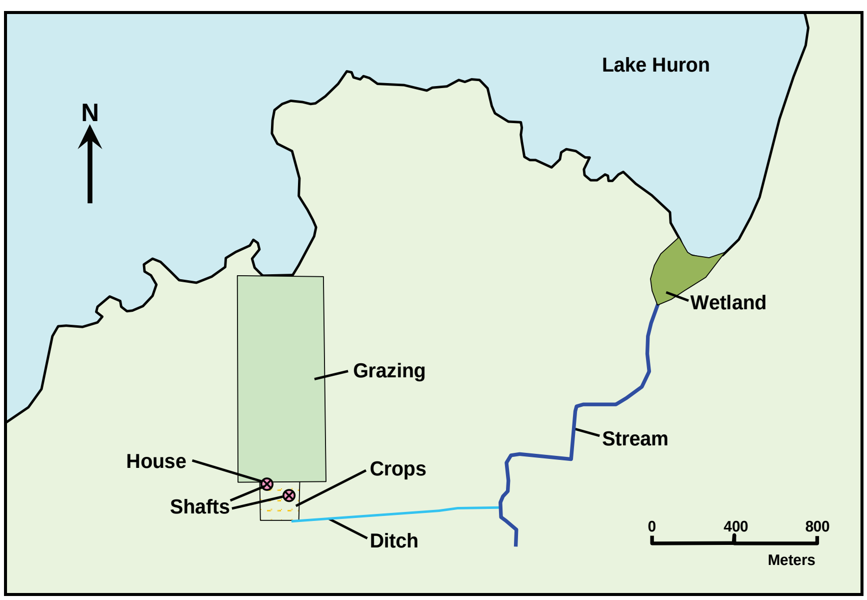


Figure 8-13: Conceptual Layout of the Biosphere System

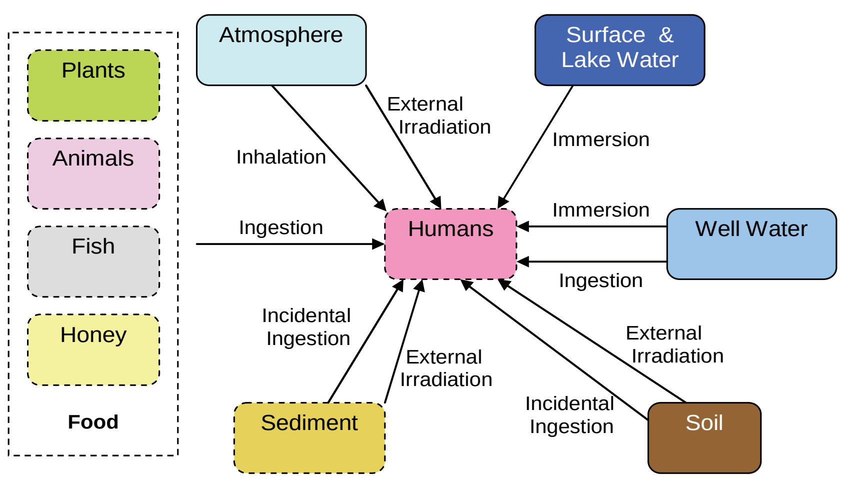
8.6.2.6 Conceptual Model – Human Lifestyles

It is assumed that eventually land use at the site will become unrestricted. Human exposure to any contaminants in the biosphere could occur by a variety of pathways as illustrated in Figure 8-14. Contaminants in soil and water would expose humans by external irradiation, and can be assimilated by plants and animals that may in turn be eaten by humans. Inhalation exposure and external air irradiation could occur if contaminants are volatilized and released from soil and water.

In order to assess potential impacts, a future "Site Resident" critical group is defined. This conservatively-defined hypothetical family lives on a farm on the repository site (Figure 8-13). Their house is over the main shaft. They grow their own grain, fruit and vegetables from fields that are located above the repository, and in particular on the ventilation shaft. They pump water from a well drilled into the shallow groundwater zone at a location that maximizes capture of any release from the shaft, for drinking, domestic use, watering animals, and irrigating garden and feed crops.

The family comprises two adults, a child and an infant. The livestock comprise dairy and beef cattle, pigs, lambs, goats and chickens. They hunt locally for deer and rabbits, catch fish from a nearby stream and from Lake Huron, and consume local honey. They swim recreationally in the lake.

The specific pathways included are based on the CSA N-288.1 (CSA08b) biosphere model.



Note:
Dotted borders indicate equilibrium compartments.

Figure 8-14: Human Exposure Pathways Conceptual Model

8.6.2.7 Calculation Cases

The treatment of uncertainty is central to any assessment to establish the safety of a radioactive waste repository. Many organizations use the following three broad categories to structure their analysis of uncertainties in postclosure safety assessments[1] (MARIVOET08, CNSC06a):

- **Future or scenario uncertainty** – uncertainty in the evolution of the repository system and human behaviour over the timescales of interest;
- **Model uncertainty** – uncertainty in the conceptual, mathematical and computer models used to simulate the behaviour of the repository system (e.g., due to approximations used to represent the system); and
- **Data uncertainty** – uncertainty in the data and parameters used as inputs in the modelling (e.g., due to incomplete site-specific data, and parameter estimation errors from interpretation of test results).

Scenario uncertainty is addressed through considering a range of scenarios. So in addition to the Normal Evolution Scenario, the impacts of other scenarios are assessed in Section 8.7.

Model uncertainties (conceptual and mathematical) and data uncertainties associated with the Normal Evolution Scenario are addressed through the evaluation of a set of calculation cases that are designed to bound the effects of these uncertainties with the Normal Evolution Scenario Reference Case (**NE-RC**). These cases are summarized in Figure 8-15 and Table 8-5.

Most of the uncertainties are addressed through deterministic calculations. This provides very clear information on the influence of the varied process or parameter. The disadvantage is that these are not able to provide as complete coverage of the parameter space. Some probabilistic modelling is therefore included, but is focussed on contaminant transport parameters around the NE-RC reference case. Cases with significantly different groundwater flow or gas flow are only considered within the deterministic set.

[1] The boundaries between these categories can overlap in that, depending upon how models are formulated, an uncertainty may be classed as a model or a data uncertainty.

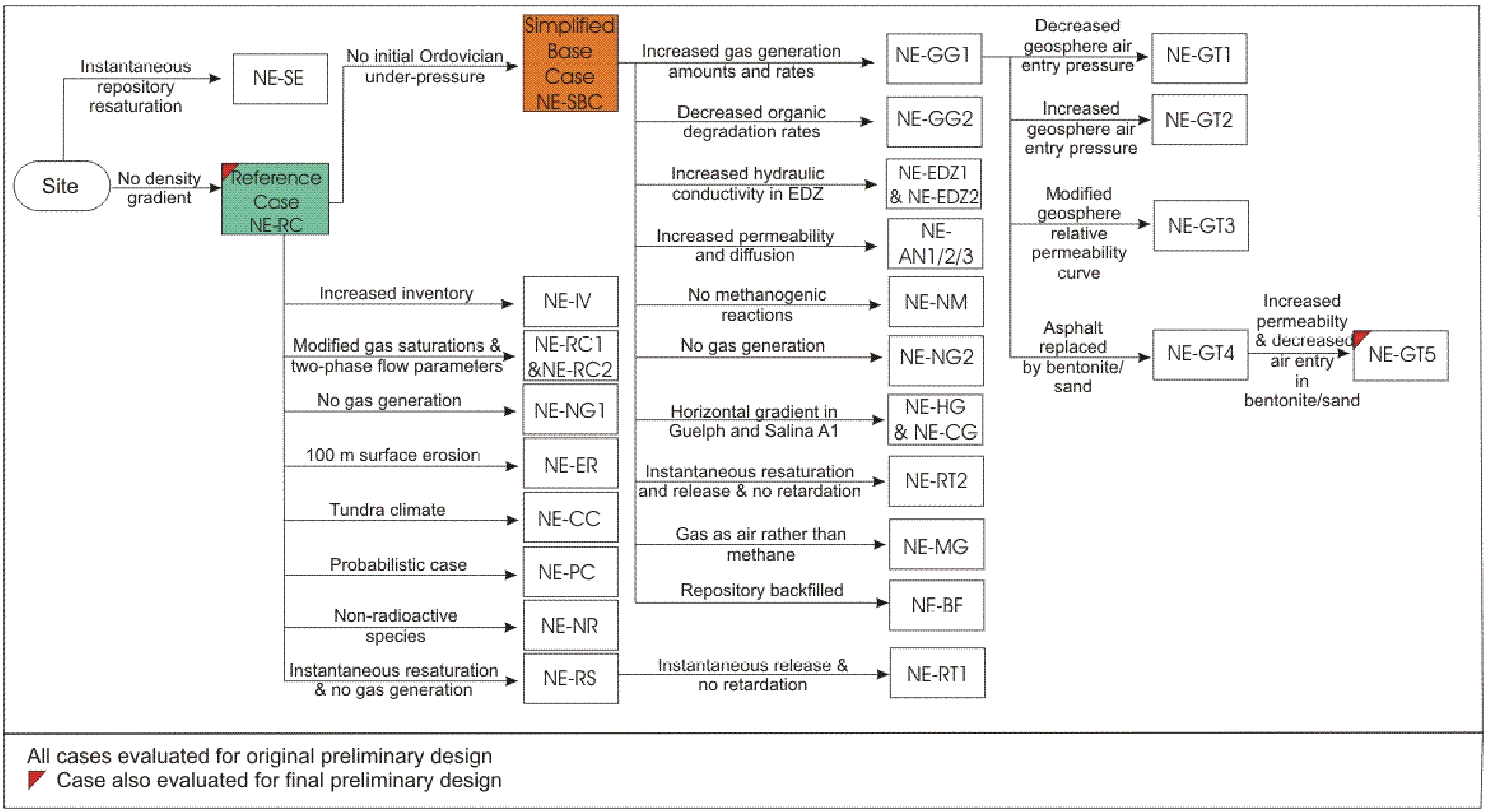


Figure 8-15: Normal Evolution Scenario: Summary of Calculations Cases

Table 8-5: Calculation Cases for the Normal Evolution Scenario

Case ID	Case Description
NE-RC	Reference case parameters based on reference inventory, original preliminary design and site characterization data. Assume steady-state Cambrian overpressure (+165m), 10 m rockfall from closure, initial Ordovician underpressures allowed to equilibrate, no salinity gradient, constant temperate climate and no horizontal flow in the Cambrian, Guelph Formation and Salina A1 Unit upper carbonate. The gas modelling included initial gas saturations of 10% in the Ordovician rocks. The groundwater modelling assumed instant resaturation and no gas generation.
NE-SBC	As NE-RC but with: • No underpressures in the Ordovician; and • No partial gas saturation in the Ordovician.
NE-RS	As NE-RC but with: • Immediate water resaturation of repository; and • No gas generation in repository.
NE-EDZ1	As NE-SBC but with EDZ hydraulic conductivities increased to maximum values in Data report (Table 5-7 and 5-8 of NWMO11am), i.e.: • Shaft inner EDZ increased by two orders of magnitude; • Shaft outer EDZ increased by an order of magnitude; and • Repository EDZ increased by an order of magnitude.
NE-EDZ2	As NE-EDZ1, but with a 9 m wide zone around monolith at repository level in which the HDZ and EDZ around the monolith are milled out and replaced by concrete to interrupt the HDZ.
NE-HG	As NE-SBC but with: • Horizontal groundwater flow in the Guelph Formation (gradient of 0.0026) and Salina A1 Unit upper carbonate (gradient of 0.0077) (Section 5.4.1.1 of NWMO11am); and • 1.25 km travel path along Guelph Formation and Salina A1 Unit upper carbonate to lake.
NE-AN1	As NE-SBC but with changes in horizontal to vertical anisotropy of hydraulic conductivity. Anisotropies of 10:1 and 1000:1 are replaced by 2:1 and 20:1, respectively, with horizontal hydraulic conductivity fixed as in NE-SBC.
NE-AN2	As NE-SBC but with changes in horizontal to vertical anisotropy of effective diffusion coefficient. Anisotropies of 2:1 are replaced by 10:1, with a vertical effective diffusion coefficient fixed as in NE-SBC.
NE-AN3	As NE-SBC but with increased vertical permeability resulting in no anisotropy except for 10:1 (horizontal to vertical) in Coboconk and Gull River formations.
NE-SE	As NE-RC but with a saline fluid density profile based on the measured profile with depth. A linear increase in density between 1000 and 1185 kg m^{-3} is adopted between the top of the model (Salina F) and the Guelph Formation. Below the Guelph, a constant density of 1185 kg m^{-3} is adopted.
NE-NG1	As NE-RC but with no gas generation.
NE-NG2	As NE-SBC but with no gas generation.
NE-MG	As NE-SBC except that gas used is air rather than methane. Case recognizes that the different gases generated in the DGR will have different characteristics than the "bulk" gas (methane) considered in NE-SBC.
NE-RC1	As NE-RC but with gas saturations in Ordovician equal to residual gas saturation of 5%.
NE-RC2	As NE-RC but with gas saturations and two-phase flow parameters on a formation basis as given in the DGSM (NWMO11k).

Case ID	Case Description
NE-GG1	As NE-SBC but with: • Increased metal inventory (~ 25% increase); and • Metal corrosion and organic degradation rates increased to maximum rates in Data report (Tables 3-20 and 3-21 of NWMO11am) (up to an order of magnitude increase).
NE-GG2	As NE-SBC but with organic degradation rates decreased to minimum rates in Data report (Table 3-21 of NWMO11am) (up to an order of magnitude decrease).
NE-NM	As NE-SBC but with no microbial reactions generating methane.
NE-GT1	As NE-GG1 but with decreased van Genuchten air-entry pressure and less steep air-entry curve. NE-GG1 is used as basis because it generates overpressures in the repository which are more suitable for testing gas transport in the intact rock near the repository.
NE-GT2	As NE-GG1 but with increased van Genuchten air-entry pressure and steeper air entry curve.
NE-GT3	As NE-GG1 but with relative permeability curve modified with residual liquid saturation and residual gas saturation set to zero.
NE-GT4	As NE-GG1 but with asphalt layer in shaft replaced by bentonite-sand seal.
NE-GT5	As NE-GG1 but with: • Asphalt seal in shaft replaced by bentonite/sand; • Gas entry pressures for shaft materials reduced by factor of two to 5 x 10^6 Pa; and • Bentonite/sand hydraulic conductivity reduced by an order of magnitude to 10^{-10} m/s.
NE-BF	As NE-SBC but with repository backfilled with coarse aggregate material with a porosity of 0.3.
NE-RT1	As NE-RS but with: • Instantaneous release of radionuclides to groundwater; and • No radionuclides sorbed or solubility limited in repository or geosphere.
NE-RT2	As NE-SBC but with: • Instantaneous release of radionuclides to groundwater; and • No radionuclides sorbed or solubility limited in repository or geosphere.
NE-IV	As NE-RC but with radionuclide inventory increased by a factor of ten.
NE-ER	As NE-RC but with surface erosion of 100 m in 1,000,000 years.
NE-CC	As NE-RC but with alternative constant state biosphere (i.e., tundra rather than temperate).
NE-CG	As NE-HG, but with dose to "Site Shore Resident" and "Downstream Resident" evaluated. These people are exposed via consumption of lake fish and water from the near shore and the South Basin of Lake Huron, respectively.
NE-PC	As NE-RC but with probabilistic treatment of some parameters.
NE-NR	As NE-RC but with the inventory of non-radioactive species emplaced in the repository.

8.6.2.8 Mathematical Models and Software Implementation

The mathematical modelling approach used in the assessment is based on the use of an assessment-level (system) model incorporating all key processes relevant to contaminant release, transport and impact, supported by detailed models for the groundwater flow and transport, and gas generation and transport processes.

Assessment-level models are implemented in AMBER Version 5.3 (QUINTESSA09). This computer code represents contaminant transport within a compartment model approach, with water and gas flows being provided as input from separate detailed modelling codes. Two detailed codes have been used in the current assessment – FRAC3DVS-OPG and T2GGM (Figure 8-16).

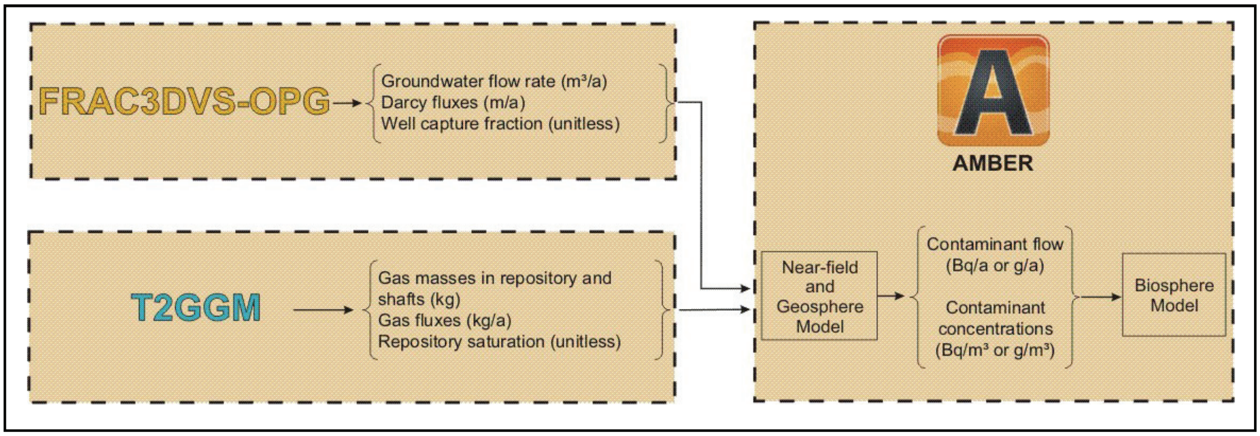


Figure 8-16: Information Flow Between the Detailed Groundwater (FRAC3DVS-OPG) and Gas (T2GGM) Codes and the Assessment Model (AMBER)

FRAC3DVS-OPG (Version 1.3.0) is a three-dimensional finite-element/finite-difference groundwater flow and contaminant transport code (THERRIEN10). FRAC3DVS-OPG was used in finite-element and equivalent-porous-medium representations of the saturated host rock and repository. A three-dimensional simplified model of the deep and intermediate groundwater zones was implemented to evaluate groundwater flow and transport. A separate three-dimensional model of the shallow groundwater zone was also implemented (the 3DSU model) to evaluate flow and transport from the shafts to the well and lake (see Section 4.2 of NWMO11an).

T2GGM (Version 2.1) is a code that couples the Gas Generation Model (GGM) and the widely used TOUGH2 (NWMO11ap). GGM, a project-specific code, models the detailed generation of gas within the repository due to corrosion and microbial degradation of the metals and organics present, and TOUGH2 models the subsequent two-phase transport of the gas through the repository and geosphere. The coupling of

GGM and TOUGH2 allows the interactions between gas generation/pressure and water saturation in the repository to be represented explicitly. Four different but complimentary models of the DGR system were implemented (see Section 4.3 of NWMO11aj).

- A detailed three-dimensional geometry of the repository, the shafts and the surrounding geosphere (the 3DD model).
- A simplified three-dimensional representation of the repository and the surrounding geosphere that includes the shafts and associated EDZ (the 3DSRS model).
- A simplified three-dimensional representation of the repository and the surrounding geosphere that does not include the shafts (the 3DSR model).
- A two-dimensional vertical and radial representation of the shaft systems that connect the repository to the shallow groundwater zone (the 2DRS model).

The process of mathematical model development and implementation has been undertaken under the postclosure safety assessment's quality plan (QUINTESSA10) and Quintessa's ISO 9001:2008 quality management system. The AMBER code is managed and developed under Quintessa's quality management system that incorporates the requirements of TickIT software quality system (www.tickit.org). The code is documented in the Normal Evolution Scenario report (NWMO11ak). Both the FRAC3DVS-OPG and T2GGM code have been qualified to NWMO software quality requirements (NWMO10d) as documented in FRAC3DVS-OPG User's Guide (THERRIEN10) and T2GGM (NWMO11ap), respectively.

The quality of the analysis was ensured through:

- Use of suitably qualified staff;
- Use of peer-reviewed and published literature;
- An iterative process, building on previous safety assessments as well as improvements in the facility design and site knowledge;
- Formal data freeze and data clearance processes to ensure that a consistent set of parameters for the facility design and site characterization;
- Use of quality-assured software, with verification of calculation input and results; and
- Peer review of results.

8.6.2.9 Data

Most of the data are specific to the DGR system and have been taken from its waste and site characterization programs. The overall DGR program has been structured such that the safety assessment has been produced in multiple iterations, with data freezes in synchronization with the inventory, design and geoscience programs.

Data required for safety assessment was either obtained from published literature or referenceable documents, or was released for use within the DGR project using a data clearance process. In the latter case, approved data have been documented using a data clearance form that records the persons providing and approving the dataset, together with the purpose and nature of the dataset, its status/history, and any limitations/restrictions on its use/application.

Table 8-6 summarizes the reference values used for the key parameters for the Normal Evolution Scenario's Reference Case. The bases for many of the important parameters are described in Chapter 4 (Geoscience), Chapter 5 (Waste Inventory) and Chapter 6 (Facility Description). Further details on model parameters used in this safety assessment are provided in the Data report (NWMO11am), or the other detailed modelling supporting reports.

One set of parameters of particular interest are the hydraulic conductivities of the host rock and shaft seal materials. The host rock hydraulic conductivity is very low, as demonstrated by various analyses in Chapter 4. The EDZ around the shaft is modelled as one-shaft-radius-thick, based on the maximum extent calculated in geomechanical modelling, see Figure 4-81, but assumed to apply to the entire shaft. The properties will vary across this thickness; they are modelled as an inner and outer EDZ region. In the reference case, the effective vertical hydraulic conductivity of the inner EDZ is set to 100 times the host rock hydraulic conductivity across the shaft height, and the outer EDZ is set to 10 times the host rock hydraulic conductivity. This is based on experience with EDZ in underground laboratories in other sedimentary rocks, and considering the rock properties, horizontal bedding plane direction and stress conditions at the DGR site.

The reference shaft seal concept is based on a combination of low-permeable bentonite/sand, concrete and asphalt. The primary seal is a 70/30 wt% bentonite/sand mixture. For compacted in-situ material, a reasonable target is a dry density of 1600 kg/m^3. This corresponds to an equivalent montmorillonite dry density of around 1215 kg/m^3. At groundwater salinities of 100 and 350 g/L, which bracket the range of conditions around the shaft, the hydraulic conductivity of bentonite/sand ranges from 4×10^{-12} and 1×10^{-11} m/s, see Figure 8-17 (NWMO11am). Similarly, swelling pressures of 0.4 to 1 MPa would be expected; see Figure 8-18. Additional characteristics and experience with bentonite/sand seals is summarized in Box 2.

Table 8-6: Reference Values for Key Parameters for Normal Evolution Scenario

Parameter	Value(s)
Repository	
Repository depth	680 m
Number of emplacement rooms	Panel 1: 14; Panel 2: 17
Volume of emplacement rooms	Panel 1: 1.7 x 10^5 m^3; Panel 2: 2.5 x 10^5 m^3
Average width of emplacement rooms	Panel 1: 8.25 m; Panel 2: 8.5 m
Average repository height	7 m (used to represent the initial height throughout the repository)
Distance between Panel 1 access tunnel and Panel 2 emplacement rooms	20 m
Panel 1 access tunnels dimensions	L 537 m, W 5.4 m, H 7.0 m
Panel 2 access tunnels dimensions	L 787 m, W 5.9 m, H 7.0 m
Monolith dimensions (within repository)	L 85 m, W 11.8 m, H 7.0 m (only modelled from open access tunnels to base of a combined shaft)
Monolith dimensions (within shafts)	Radius 5.9 m; H 13 m (from repository ceiling level upwards)
Panel footprint	2.4 x 10^5 m^2
Excavated volume	Excavated: 5.3 x 10^5 m^3; Void: 4.2 x 10^5 m^3.
Waste volume (as disposed)	Panel 1: 6.8 x 10^4 m^3 ; Panel 2, 1.3 x 10^5 m^3
Waste inventory	8.8 x 10^2 TBq LLW, 1.6 x 10^4 TBq ILW at 2062 See Tables 5-8 and 5-10 for inventories.
Mass of organics (waste, packages & engineering)	2.2 x 10^7 kg
Mass of concrete (waste, packages & engineering)	2.1 x 10^8 kg (includes monolith)
Mass of metals (waste, packages & engineering)	6.6 x 10^7 kg
Backfilling of rooms and tunnels	None except monolith in immediate vicinity of shafts
Monolith properties	K_h and K_v 1 x 10^{-10} m/s; porosity 0.1; effective diffusion coefficient 1.25 x 10^{-10} m^2/s (degraded from closure)
Repository HDZ	K_h 1 x 10^{-6} m/s, K_v = K_h; porosity 4 x rock mass Emplacement rooms and tunnels: 0.5 m thick above/below and sides Supported tunnels: 2 m thick above/below, 0.5 m thick sides
Repository EDZ	K_h 10^3 x rock mass, K_v = K_h; porosity 2 x rock mass Emplacement rooms and tunnels: 8 m thick above/below and sides Supported tunnels: 3 m thick above/below and sides
Rockfall	Rockfall affects all rooms and tunnels,10 m into ceiling immediately after closure
Corrosion rates	Carbon steel and galvanized steel: 1 µm/year (unsaturated), 2 µm/year (saturated), Passivated carbon steel, stainless steel and Ni-alloys: 0.1 µm/year Zr-alloys: 0.01 µm/year
Degradation rates	Cellulose: 5 x 10^{-4} /year Ion exchange resins, plastics and rubber: 5 x 10^{-5} /year
Solubility and sorption in repository	Solubility limitation only considered for aqueous C releases (0.6 mol/m^3). No sorption considered
Shaft	
Internal diameter (lower section)	Main: 9.15 m; Ventilation: 7.45 m; Combined: 11.8 m (concrete lining and HDZ removed)
Length (lower section)	483.5 m (top of monolith to top of bulkhead at top of intermediate groundwater zone)
Internal diameter (upper section)	Main: 6.5 m; Ventilation: 5.0 m
Length (upper section)	178.6 m (top of upper bulkhead to ground surface)
Backfill and seals	Sequence of bentonite-sand, asphalt, LHHPC and engineered fill. LHHPC bulkheads (degraded from closure) keyed across the inner EDZ
Vertical and horizontal hydraulic conductivity	Bentonite-sand: 1 x 10^{-11} m/s; Asphalt: 1 x 10^{-12} m/s; LHHPC: 1 x 10^{-10} m/s; Engineered fill: 1 x 10^{-4} m/s
Diffusion and transport porosity	Bentonite-sand: 0.3; Asphalt: 0.02; LHHPC: 0.1; Engineered fill: 0.3

Parameter	Value(s)
Effective diffusion coefficient	Bentonite-sand: 3 x 10^{-10} m^2/s; Asphalt: 1 x 10^{-13} m^2/s; LHHPC: 1.25 x 10^{-10} m^2/s; Engineered fill: 2.5 x 10^{-10} m^2/s
EDZ	Inner EDZ, 0.5 x shaft radius thick, K_v x 100 rock mass, K_h = K_v; porosity 2 x rock mass Outer EDZ, 0.5 x shaft radius thick, K_v x 10 rock mass, K_h = K_v; porosity = rock mass
Sorption in shaft and EDZ	Conservative estimates for Zr, Nb, Cd, Pb, U, Np and Pu. See Section 4.6.3 of Data report (NWMO11am).
Geosphere	
Host rock type	Low permeability argillaceous limestone (Cobourg Formation)
Temperature at repository depth	22 °C
Groundwater composition at depth	Na-Ca-Cl dominated brine; TDS: 131-375 g l^{-1}; pH: 6.5 to 7.3; Eh: reducing
Hydraulic heads	+165 m at top of the Cambrian sandstone Observed variable head profile with underpressures in the Ordovician (up to -290 m) 0 m at the top of the Lucas Formation (top of the shallow groundwater zone)
Deep groundwater zone:	8 x 10^{-15} to 4 x 10^{-12} m/s (1 x 10^{-9} in the Shadow Lake Formation and 3.0 x 10^{-6} in the Cambrian sandstone)
horizontal hydraulic conductivity	10% of horizontal hydraulic conductivity for all, but Coboconk and Gull River formations (0.1%) and Cambrian which is isotropic
vertical hydraulic conductivity	0.009 to 0.097
transport porosity	2.2 x 10^{-13} to 2.4 x 10^{-11} m^2/s (some anisotropy)
effective diffusion coefficient	0
horizontal hydraulic gradient	0
Intermediate groundwater zone:	
horizontal hydraulic conductivity	5 x 10^{-14} to 2 x 10^{-7} m/s
vertical hydraulic conductivity	10% of horizontal hydraulic conductivity for all formations other than Guelph Formation and Salina A1 Unit upper carbonate which are isotropic
transport porosity	0.007 to 0.2
effective diffusion coefficient	3 x 10^{-14} to 6.4 x 10^{-11} m^2/s (some anisotropy)
horizontal hydraulic gradient	0
Shallow groundwater zone:	
horizontal hydraulic conductivity	1 x 10^{-7} to 1 x 10^{-4} m/s
vertical hydraulic conductivity	10% of horizontal hydraulic conductivity for all formations
transport porosity	0.057 to 0.077
effective diffusion coefficient	6 x 10^{-12} to 2.6 x 10^{-11} m^2/s
horizontal hydraulic gradient	0.003
Sorption in geosphere	Conservative estimates for Zr, Nb, Cd, Pb, U, Np and Pu. See Section 5.5.1.3 of Data report (NWMO11am).
Biosphere	
Average annual surface temperature	8.2 ºC
Average total precipitation	1.07 m/year
Ecosystem	Temperate
Groundwater release paths	1) 80 m deep well located 500 m down gradient of combined shaft. Well demand of 6388 m^3/year for self-sufficient farm with crop irrigation. 2) near-shore lake bed (for discharge from shallow groundwater zone)
Gas release paths	Soil and House located above repository
Sorption in biosphere	For all elements except for B, Li, Tl and W
Land use	Agriculture, recreation, forestry
Receptor (Critical Group)	Site resident, living on repository site and farming. Habit data provided in Section 7.1 of the Data report, NWMO11am, based on CSA N288.1 (CSA08b)
Human dose coefficients	See Section 7.2 of Data report (NWMO11am).
Abbreviations used in the table:	K_v: vertical hydraulic conductivity K_h: horizontal hydraulic conductivity LHHPC: Low Heat High Performance Cement TDS: Total Dissolved Solids L: Length W: Width H: Height

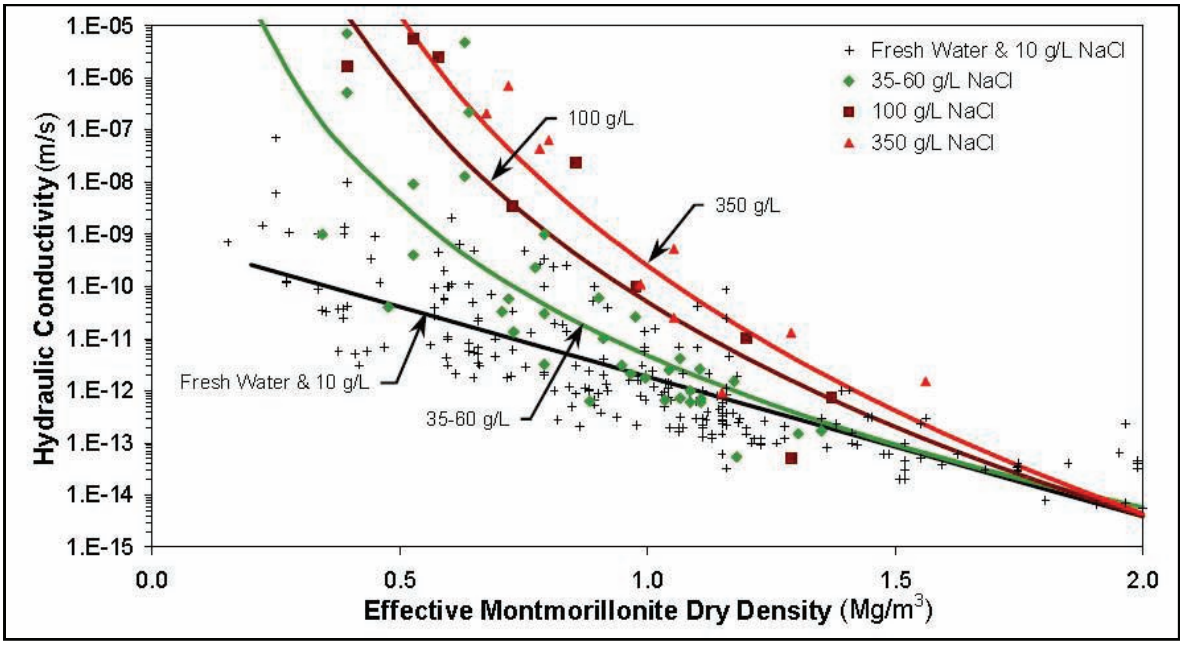


Figure 8-17: Hydraulic Conductivity of Bentonite/Sand

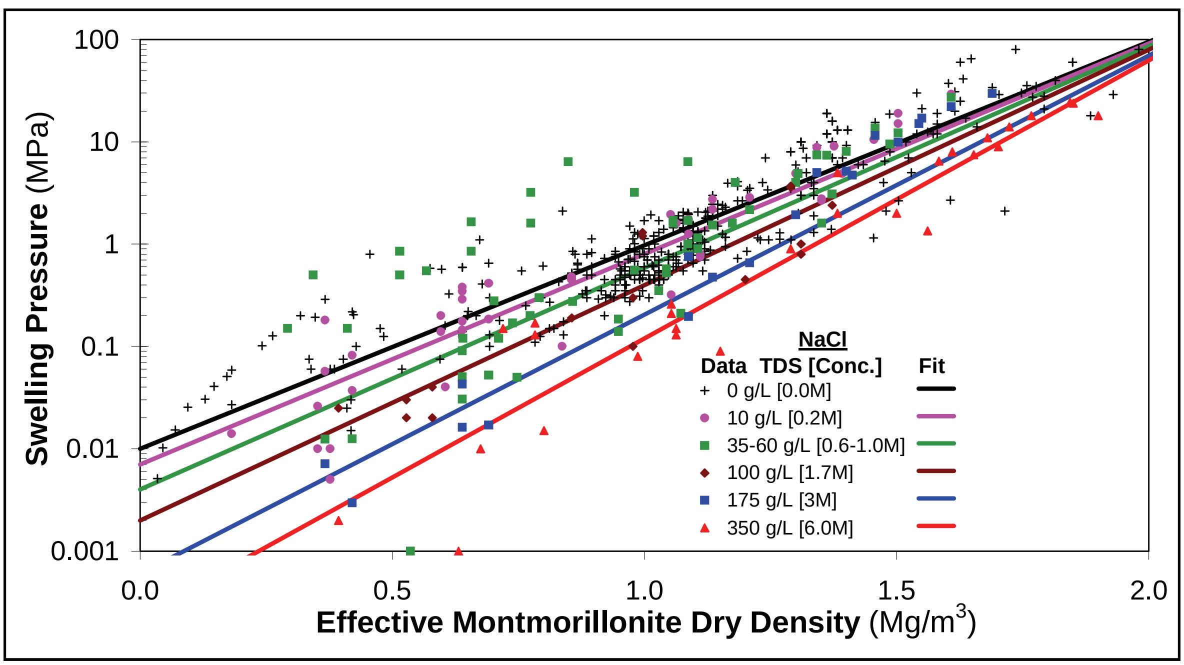


Figure 8-18: Swelling Pressure of Bentonite/Sand

BOX 2: Bentonite/Sand Seals

- The reference bentonite/sand mixture is primarily clay. The 70/30 bentonite/sand ratio was selected since it retains a clay-dominated composition, while being easier to handle than 100% clay and having improved mechanical properties. The density was chosen so that it would swell with water under the DGR saline conditions.

- There is experience in Canada with using bentonite-sand mixtures as repository seals, including the AECL Underground Research Laboratory Buffer-Container Experiment Test (50:50), ITT Test (50:50), Tunnel Sealing Experiment (70:30) and the Enhanced Seal Project (60:40 and 70:30) (DIXON02, MARTINO07).

- Achieving the desired properties of the seal requires appropriate quality control during the emplacement process. This includes the use of a graded grain size distribution for the sand component, as well as water control during placement. The seal is expected to be placed in layers and compacted in-situ.

- Bentonite is known to be a durable material, with natural deposits that are many millions of years old that still contain montmorillonite (LAINE10).

- Higher temperatures (> 100^{o}C) and alkaline conditions encourage mineralogical transformations, but the DGR shaft will be at low temperatures (< 25^{o}C), with only localized alkaline conditions near the concrete monolith and bulkheads.

- The effects of water salinity and groundwater chemical species are more complex. There is some evidence of reduced stability under certain high salinity conditions, but also evidence of no reaction other than cation exchange. Although there is no direct data on bentonite stability under the highly saline Na-Ca-Cl site groundwater conditions at the DGR site, there are some natural analogs, notably some Spanish bentonites, that have been exposed to saline Na-Cl (sea) water over millions of years, and show no significant mineral alteration (LAINE10, SAVAGE05).

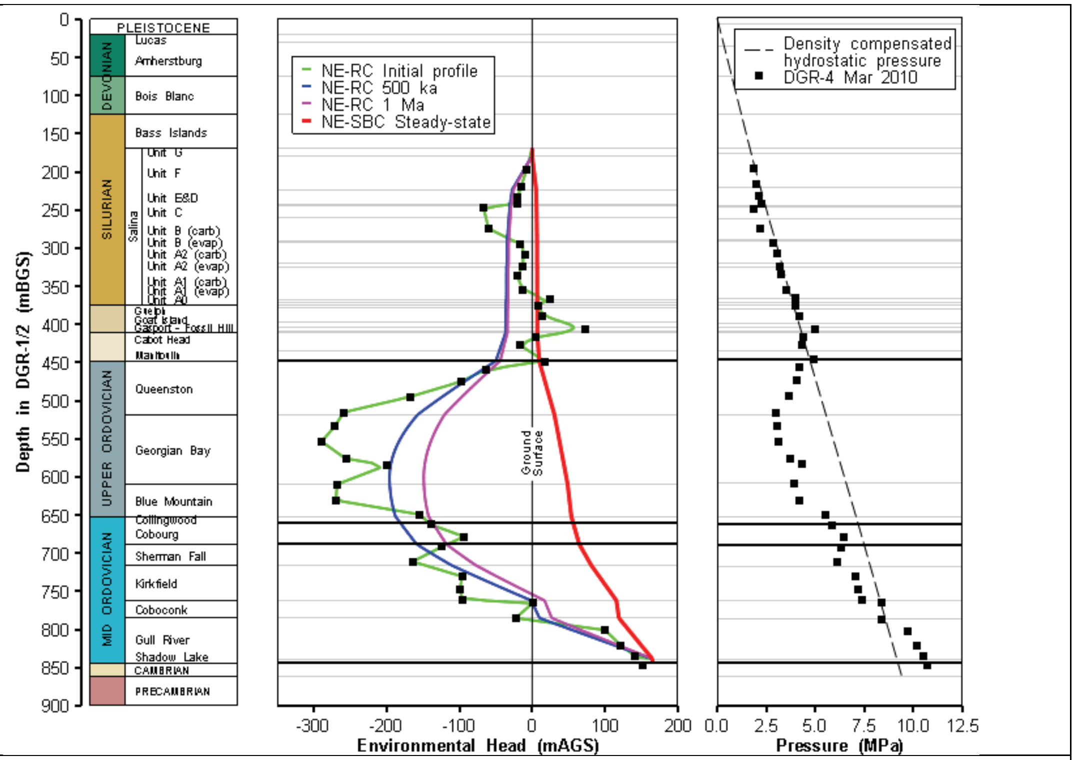
- Simple estimates indicate that the bentonite degradation processes such as illitization will be slow at the DGR (Appendix E.3 of NWMO11ah). There will be a reaction zone adjacent to concrete surfaces, and against the shaft wall, but these are expected to be limited in extent. They will be limited in part due to the low temperatures, which limits the rate of reaction, and due to the low permeabilities of the shaft seal and rock, which limits the rate of supply of reacting species. Also, the groundwater at the DGR site is near neutral pH, and the concrete bulkheads will be fabricated from low-pH cement.

8.6.3 Key Modelling Assumption for the Normal Evolution Scenario

The postclosure safety assessment modelling results are based on a simplified representation of the repository system. Key modelling assumptions for the Normal Evolution Scenario are summarized in Table 8-7.

The Normal Evolution Scenario considers two principal cases, see Figure 8-19. The **Reference Case (NE-RC)** is based upon transient groundwater flow starting with the underpressures observed in the Ordovician sediments and overpressures observed in the Cambrian sandstone. The Reference Case builds directly on the results of the geosphere characterization program (NWMO11k) and the associated Geosynthesis (NWMO11c) which is summarized in Chapter 4.

A more conservative **Simplified Base Case (NE-SBC)** is also evaluated, in which there is no underpressure, but a steady vertical head gradient towards the shallow groundwater zone is maintained as a consequence of the overpressured Cambrian.



Note: Adapted from Figure 5-3 of the Groundwater Modelling report (NWMO11an). Detailed groundwater and gas models focussed on the low-permeability intermediate and deep geosphere.

Figure 8-19: Hydraulic Head and Pressure Profiles for the Reference Case (NE-RC) and Simplified Base Case (NE-SBC)

Table 8-7: Key Modelling Assumptions for the Normal Evolution Scenario's Reference and Simplified Base Case

Key Assumption	Motivation/Reason for Assumption	Impact of Assumption
Conceptual Model		
Microbial reactions occur in DGR if there are energy sources and water present.	Water chemistry in repository could be unfavourable for microbes, but they tend to be ubiquitous when food and water are available.	Maximizes the generation of gases, which increases the pressure in the DGR. This delays resaturation but enhances the potential for gas migration to the surface. Variant cases explore different gas generation rates (NE-GG1, NE-GG2), no gas generation (NE-NG1, NE-NG2), no methane-generating microbial activity (NE-NM), and instant resaturation (NE-RS).
"Non-water-limited" repository model.	Conservatively allows for some unidentified water sources.	Water consumed in anaerobic corrosion and gas generating reactions is not included in the repository water balance. This overstates the amount of water available to support repository reactions. The importance of this assumption is tested in "water-limited" WL cases.
Degradation reactions proceed to completion with respect to generation of gases.	Conservative assumption to avoid the complexities of microbial reactions.	Maximizes the generation of gases, which increases the pressure in the DGR and also delays resaturation. See also variant cases described above.
Concrete monolith and bulkheads in the shafts are degraded from closure.	Timing of degradation is uncertain. Simplifies modeling of concrete.	Degraded concrete is more permeable than undegraded concrete by around two orders of magnitude, so this assumption maximizes the potential for contaminants to migrate from the DGR.
Bentonite/sand seal does not significantly degrade for relevant timescales.	Bentonite and sand are durable natural materials. Degradation is expected to be slow under the low-flow, low-temperature DGR conditions.	Bentonite/sand seal properties consistent with exposure to brine are used as reference. There will likely be interaction zones at the interface between bentonite/sand and concrete bulkheads, but these will be localized. More permeable seals are considered; see NE-GT5 and SF-BC.
Asphalt seal does not significantly degrade for relevant timescales.	The organic component of the asphalt seal is a degradation-resistant bitumen. In the absence of UV radiation and conditions favorable to microbes, degradation is expected to be slow.	The case NE-GT4 shows that the asphalt could be replaced with bentonite/sand without significantly affecting the shaft performance. Asphalt provides a redundant seal material. If instead the asphalt were to degrade, the shaft seal would still be provided by the bentonite/sand seals.

Key Assumption	Motivation/Reason for Assumption	Impact of Assumption
Damaged rock zones around repository and shafts remain connected, and do not seal due to creep or precipitation processes.	Shales contain clay and some self-sealing would be possible, but the long-term behavior of damaged zones in these rocks has not been studied.	Provides a more permeable pathway from the repository than the geosphere, and so represents a conservative assumption that maximizes the potential for contaminants to migrate from the DGR via the shafts.
The Reference Case represents transient development of the observed hydraulic pressure gradients, while the Simplified Base Case assumes that the gradients have reached a steady-state equilibrium from closure.	Addresses uncertainty associated with the future development of hydrogeological pressure gradients.	The Reference Case includes the initial underpressures observed in Ordovician formations, which results in the potential for downward groundwater flow in the shafts. The Simplified Base Case represents a constant steady-state gradient upward from the DGR to the shallow groundwater zone and therefore has potential for constant upward groundwater flow from the DGR. The Simplified Base Case is therefore conservative and maximizes potential contaminant transport from the DGR to the shallow groundwater zone.
There is no significant effect of glacial cycles on contaminant transport in the deep groundwater zone.	Consistent with site evidence, regional hydrogeological modelling, and shaft geomechanical modelling (Section 4.4.4, Section 4.5.4).	Impacts on stress may result in rockfall in the repository, which is represented but assumed to occur at closure. Ice-sheet loading and unloading will cause changes in the vertical groundwater pressure gradients (initially increasing the downward gradient, and then reversing to an upward gradient). These could potentially affect contaminant transport. However, the impacts on groundwater pressure gradients are not sufficient to change the deep geosphere from diffusion dominated transport.
Horizontal groundwater flow in the Guelph Formation and Salina A1 Unit upper carbonate is not represented.	Flow is very slow, and discharge point is not certain but likely distant. In long-term, flow and discharge point are likely to change with the passage of ice-sheets.	Horizontal groundwater flow in the Guelph Formation and Salina A1 Unit upper carbonate would divert contaminant transport away from the shallow groundwater zone. The formations would provide longer transport pathways to the biosphere, with associated greater dispersion and decay. Ignoring these pathways therefore maximizes contaminant transport to the local biosphere above the repository. The impact of this assumption is explored via the NE-HG case that includes horizontal groundwater flow in the Guelph and Salina A1.
Groundwater well located downstream from repository.	Future location of a groundwater well is uncertain.	The well is placed a short distance downstream from the repository in the plume path to capture contaminants from repository.

Key Assumption	Motivation/Reason for Assumption	Impact of Assumption
Constant present-day temperate biosphere.	Provides a readily understandable estimate of potential impact on humans. Use of a constant biosphere provides clarity on changes due to repository/geosphere evolution. Simplifies modelling.	The range of biosphere conditions relevant to the DGR site cover temperate, tundra, glacial and post-glacial conditions. Human uses of the site will be more limited under non-temperate conditions. Representing contaminant releases to a constant temperate biosphere state is therefore a useful indicator of potential exposures to humans, as it maximizes the range of exposure pathways and the use of local resources. The NE-CC case considers release to a tundra biosphere system.
Potential receptor is a self-sufficient farming family that maximizes its use of local resources and lives in a house located on top of the main shaft.	Addresses uncertainty surrounding future human behaviour.	Maximizing use of local resources and living in a house located on top of the main shaft are conservative assumptions in respect of potential exposures to humans. Alternative potential receptors (critical groups) who are exposed via high consumption of lake fish and water from the near shore, or from the South Basin of Lake Huron are considered in the NE-CG calculation case.
Mathematical Model		
Single combined shaft and access tunnel pathway.	Simplifies the representation of the DGR system in the models.	The properties and key geometric aspects of these features are preserved In particular, the cross-sectional areas of shaft and shaft EDZ are the same as for the two shafts.
Instantaneous collapse of waste package stacks at closure.	Addresses uncertainty concerning the timing of collapse of waste stacks.	Minimizing the height of waste in the DGR maximizes the amount of waste that can come into contact with groundwater in a partially resaturated repository. This maximizes potential contaminant releases to groundwater and is therefore conservative for this pathway
Packaging does not limit access of water to the wastes or releases in liquid or gaseous phases.	The packages are not designed for long-term containment. Duration of their integrity is uncertain.	The exclusion of the packaging from the contaminant release models is conservative as it allows earlier release. (Note that the amount of metal and concrete in packaging is taken into account in determining the associated gas generation, non-radioactive species and repository chemical evolution.)

Key Assumption	Motivation/Reason for Assumption	Impact of Assumption
Limited solubility and sorption in repository and geosphere	Limited data available to support the solubilities and sorption of elements under repository conditions, notably the high salinity.	Conservative assumption. The limiting case of no solubility limitation and no sorption of any elements is assessed (NE-RT1 and NE-RT2).
Instantaneous rockfall at closure	Rockfall is likely to be gradual over thousands of years, but exact timing is uncertain. Simplifies the representation of the process in the gas and groundwater models.	Maximizes the extent of damage to the host rock and to waste packages from DGR closure. This reduces the thickness of low permeability rock above the repository and maximizes the interface area between the repository and the rock around the concrete monolith, close to the shafts and is therefore conservative for potential contaminant migration.
Salinity gradient in geosphere not represented	Simplifies the representation of the DGR system in the gas and groundwater models.	Ignoring the salinity gradient is generally conservative since it is expected to limit contaminant migration due to density effects. The effect of salinity gradients is partially included in transient calculation cases as initial head profiles are based on environmental heads which are compensated for fluid density. The NE-SE case explicitly represents saline fluid density effects.
The bulk gas transported through the geosphere as a single gas.	Simplifies the representation of the DGR system in the detailed gas model.	A gas mixture could interact differently with the shaft and host rock than a single gas. However, generally methane is the likely dominant gas in the host rock and in repository. Conservatively, no differentiation is made between uncontaminated and contaminated gas. Variant cases consider air (NE-MG) and H_2 (NE-NM).

8.6.4 Normal Evolution Scenario Results

8.6.4.1 Normal Evolution – Radioactive Decay

A fundamental process occurring in the DGR is the steady radioactive decay of the wastes. Figure 8-20 shows the total activity of the wastes as a function of time, based on three categories of wastes. Figure 8-21 shows the ingestion radiotoxicity of the wastes – a measure which specifically includes the relative hazard of the different radionuclides. The existing natural radioactivity of the rock above the repository is shown in Figure 8-20, and the corresponding natural radiotoxicity is shown in Figure 8-21. In Figure 8-20 the top of the grey band corresponds to the rock within the Bruce nuclear site, while the bottom corresponds to the rock directly above the repository. The main natural radionuclides are K-40, U-238 (and decay chain, especially Po-210), Rb-87 and Th-232.

These figures show that the 80% of the waste volume that is LLW will have largely decayed to low levels in a few hundred years. It is the 10% of the waste volume in the refurbishment (retube) ILW that contains most of the long-lived radioactivity – in particular Zr-93. Figure 8-20 shows that the total radioactivity of the wastes is less than that of the rock within about 100,000 years. Figure 8-21 shows that wastes remain more concentrated, with the radiotoxicity of the retube waste about 100 times that of the rock per cubic meter at longer times.

Due to the good containment provided by the DGR system, some peak impacts may not occur within one million years. Calculated results may therefore be presented beyond one million years to show that these impacts are small. Graphs use a grey background for the period beyond one million years to emphasize the illustrative nature of the results over such timescales. Furthermore, in many cases, the calculated impacts are less than 10^{-6} mSv/year, and while they are presented here to allow comparison of different calculation case results, such low doses should be considered as negligible.

The results and the associated commentary presented in this section are, of necessity, a summary of the more detailed results and commentary presented in the supporting reports. For a more detailed analysis, the reader should consult the relevant supporting report (NWMO11ak, NWMO11aq, NWMO11aj and NWMO11an).

Note in particular that all models include a steady Cambrian overpressure. The initial Ordovician underpressures are included in the Normal Evolution Reference Case based models, but assumed to have quickly dissipated in the Simplified Base Case models.

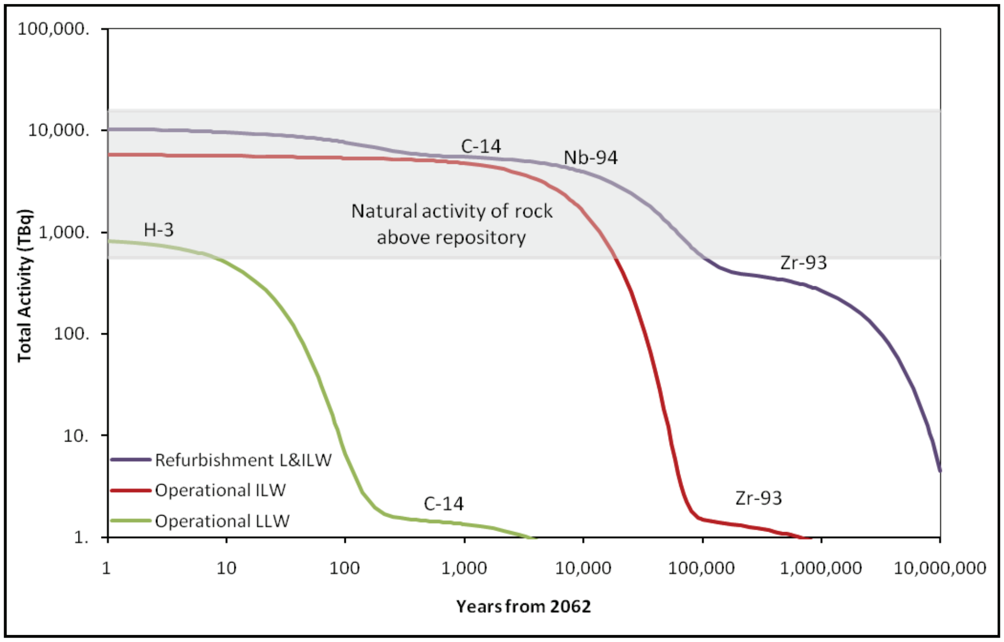


Figure 8-20: Total Radioactivity of the Waste as a Function of Time

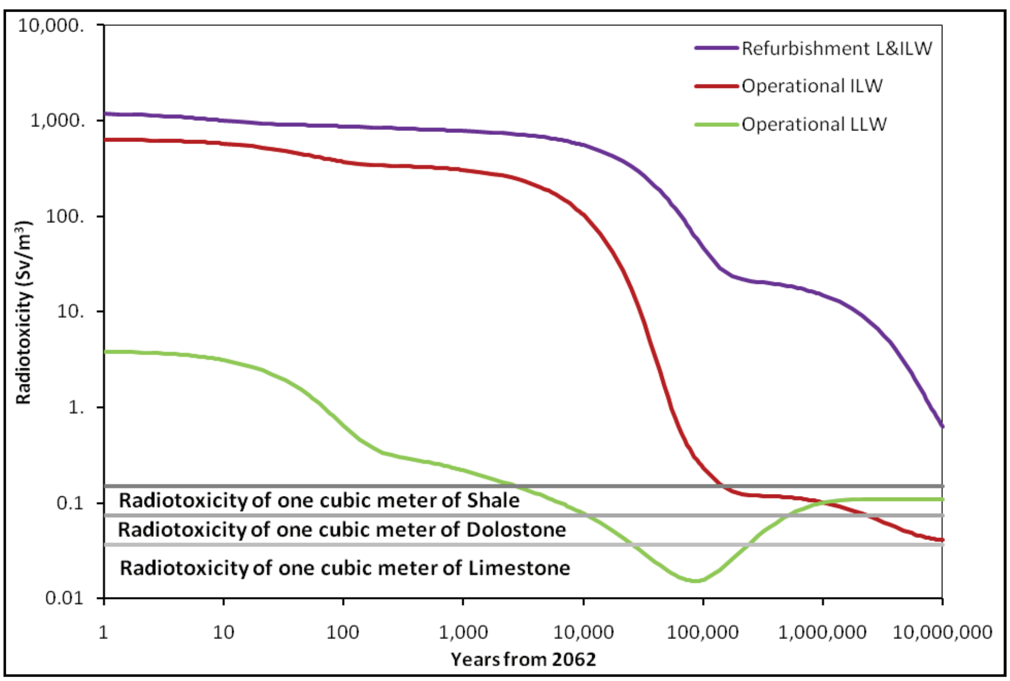


Figure 8-21: Radiotoxicity of 1 m^3 of Waste as a Function of Time

8.6.4.2 Normal Evolution – Containment in Repository

The important initial behaviour of the repository is the slow in-seepage of water from the rock, and the slow degradation of waste and containers leading to build up of gas (Figure 8-4). The balance of these processes leads to low amounts of water within the repository, Figure 8-22 (note that the emplacement rooms are about 7 m high).

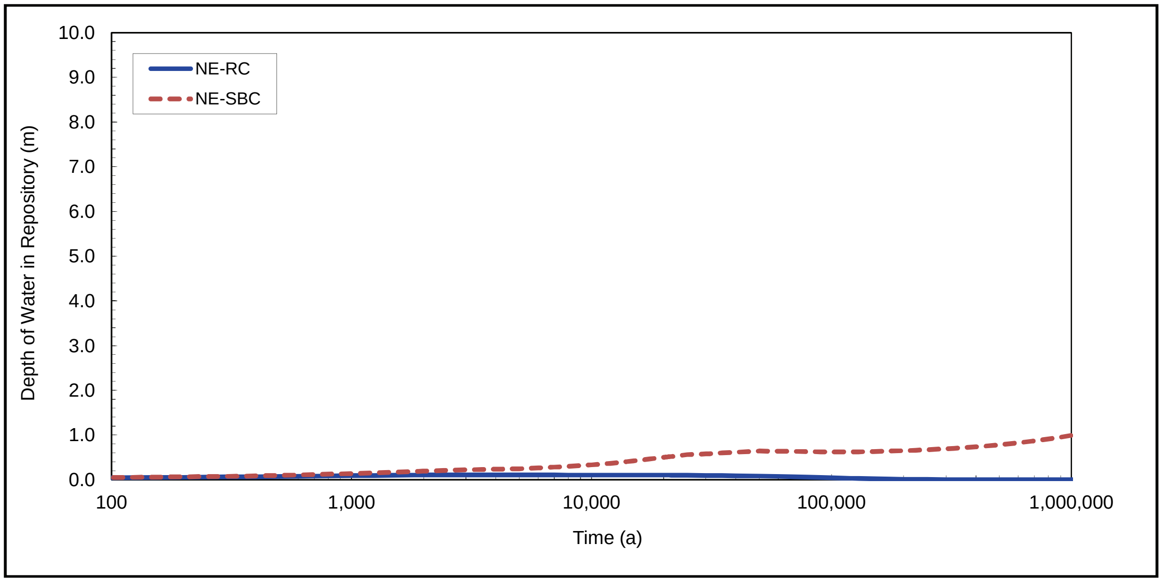


Figure 8-22: Depth of Water in the Repository for the Reference Case (NE-RC) and Simplified Base Case (NE-SBC)

There is enough water available from the host rock in this case to sustain corrosion and degradation reactions. Figure 8-23 and Figure 8-24 show how the form of iron and carbon in the repository changes from the initial source materials into primarily Fe_3O_4 and methane, respectively (Section 5.1.1 of NWMO11aj). The carbon figure also shows the influx of some methane from the host rock into the repository.

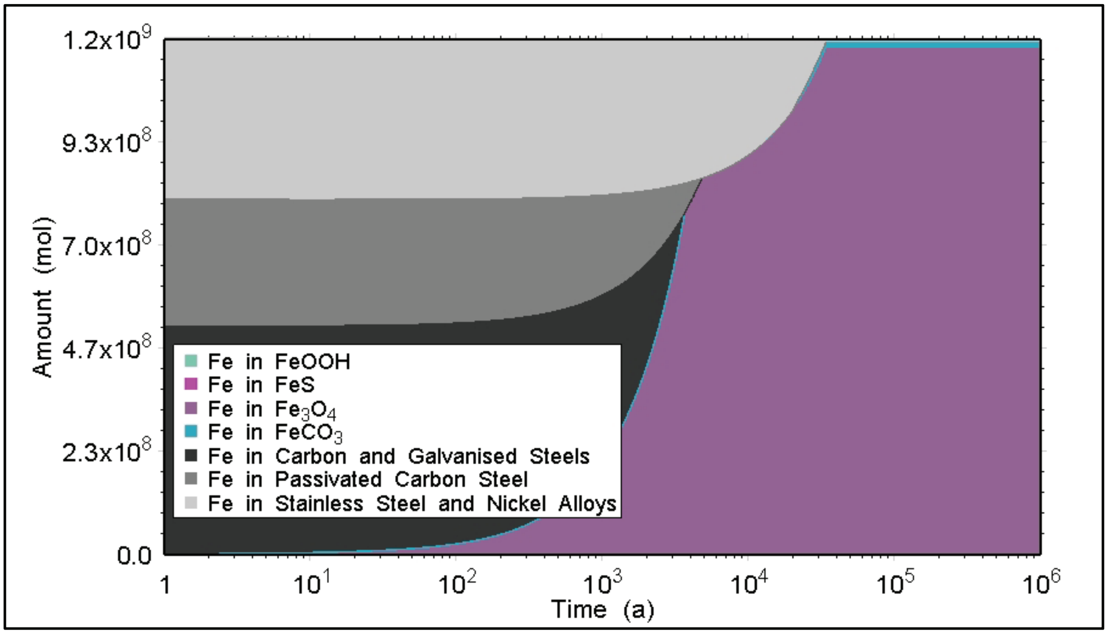


Figure 8-23: NE-RC: Iron Atom Stack Plot

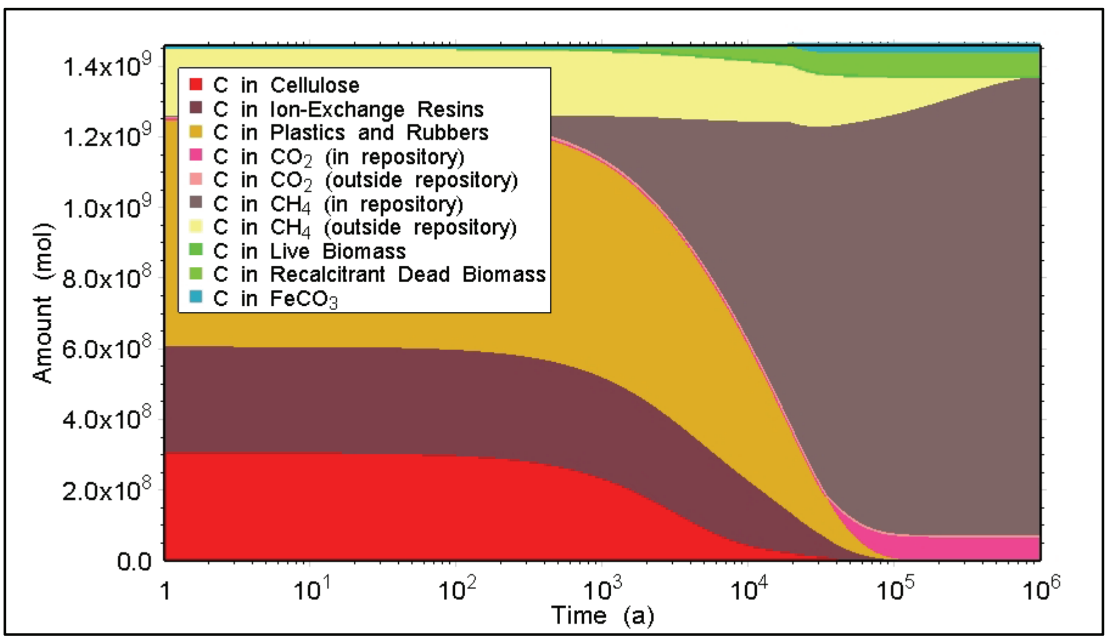
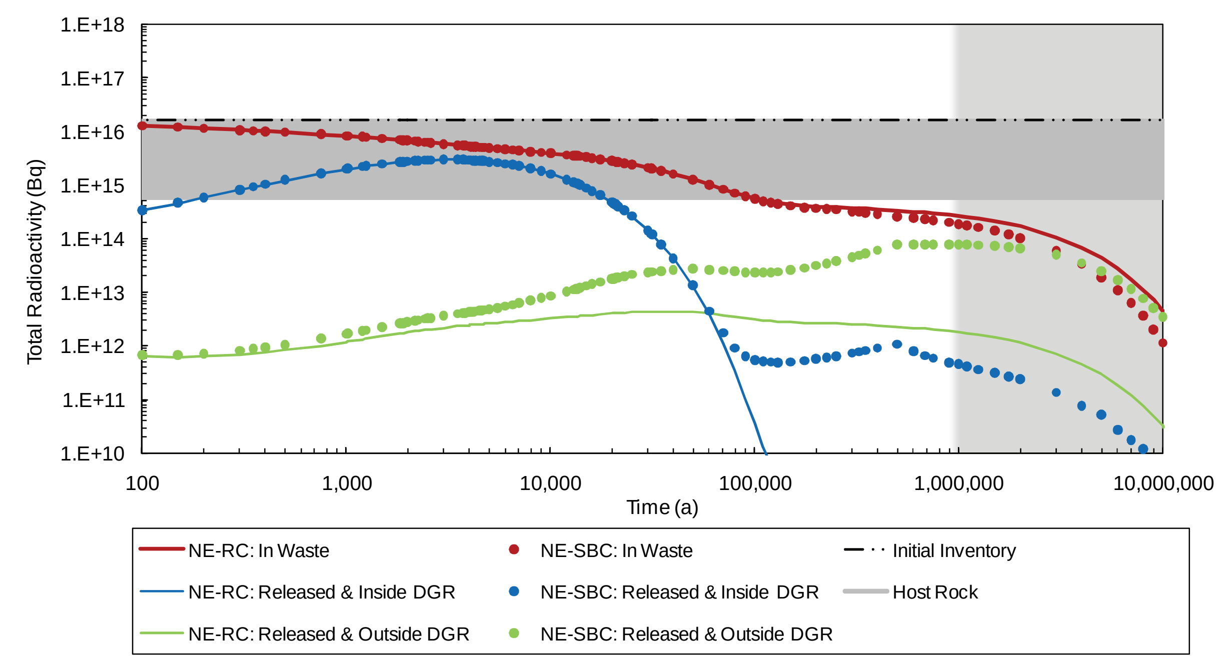


Figure 8-24: NE-RC: Carbon Atom Stack Plot

Although the waste containers themselves are not considered to be long-lasting, the combination of slow saturation of the repository, slow degradation of some waste forms, and slow diffusion from the repository into the surrounding rock provides effective containment for most of the radionuclides.

This is illustrated in Figure 8-25, which shows the amount of radioactivity in the waste, the amount released from the waste but remaining within the DGR, and the amount released from the DGR to the host rock and shafts. The figure shows the higher saturation in the Simplified Base Case results in a greater release from the wastes at long times in comparison to the Reference Case. For comparison, the figure also shows the natural radioactivity in the rock above the repository as a horizontal grey band. The upper part of this band corresponds to the Bruce nuclear site, the lower part of this band corresponds to the DGR footprint.

Figure 8-25 also shows that the amount of radioactivity outside the waste reaches a maximum of 18% of the initial inventory in both cases. This is due to the release of C-14 (from resins) as gas within the DGR. The amount of radioactivity outside the DGR reaches a maximum of 0.03% of the initial inventory for the Reference Case and 0.5% for the Simplified Base Case.



Notes: Horizontal grey band is the range of natural rock radioactivity above the repository. Lower level is repository footprint area, upper level is the Bruce nuclear site area.

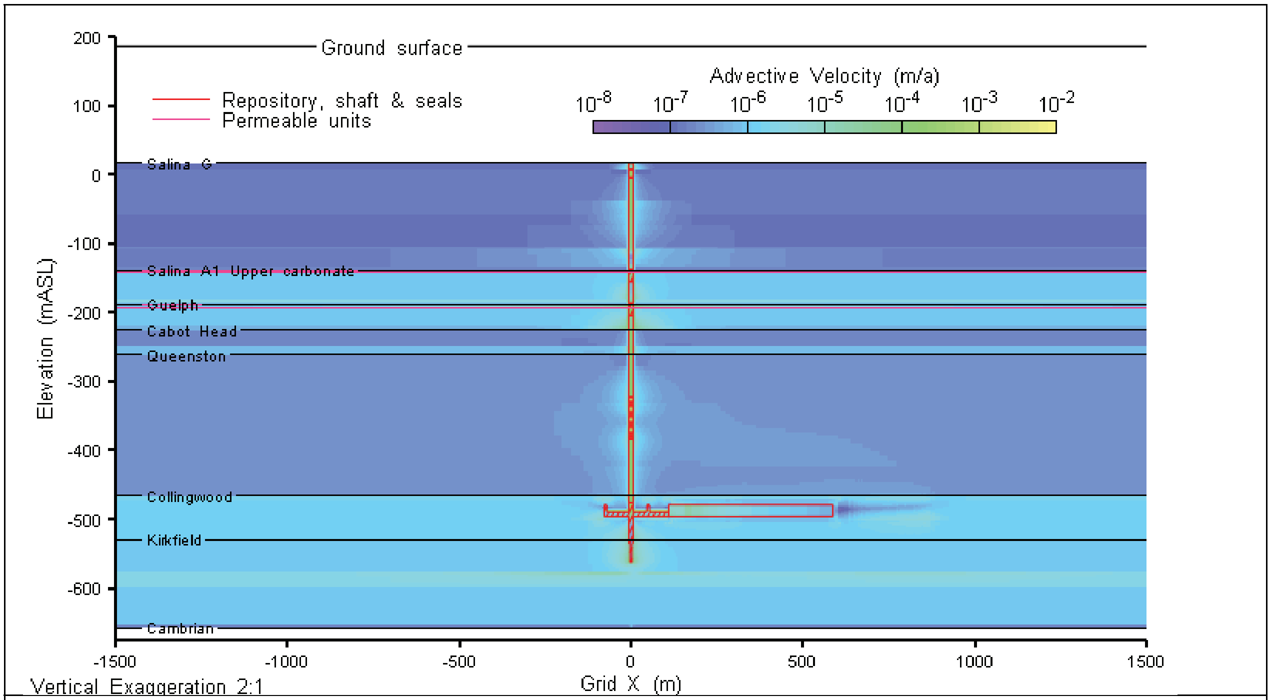
Figure 8-25: Total Radioactivity Within and Outside Repository in the NE-RC and the NE-SBC Cases

8.6.4.3 Normal Evolution – Containment in Geosphere and Shafts

The host rock surrounding the DGR has very low permeability, such that there is no vertical and horizontal advection of groundwater, and contaminants can only diffuse away from the repository.

Figure 8-26 shows the groundwater velocities in the intermediate and deep bedrock calculated for the Simplified Base Case, which is conservative in relation to the Reference Case with regards to groundwater flow (Figure 5-22 of NWMO11an). The figure shows that calculated groundwater velocities are effectively zero at about 0.001 mm/year. This is consistent with a diffusion-dominated groundwater regime.

The evolution of pressures and saturations in the shaft is due to both gas and water flowing into or out from the shaft from the adjacent geosphere and repository driven by pressure differences.



Notes: Detailed groundwater model focussed on the low-permeability intermediate and deep geosphere. Permeable formations above Salina G were modelled separately.

Figure 8-26: Groundwater Velocities in the Rock for the Steady-State Simplified Base Case NE-SBC

Figure 8-27 shows the gas saturation and liquid pressure conditions in the shaft at 60,000 years (Figure 5-20 of NWMO11aj). Note that the gas saturations in the rock mass are due to the initial saturation of 10% in the intact rock pore in the Reference Case. Liquid pressures in the rock reflect the environmental head profile, including the Ordovician underpressures. The asphalt seal is visible as a gas saturated volume, with the gas passing through it and groundwater passing around it through the EDZ. Gas is moving from the repository up the shaft, and also flowing into the shaft from the formations. Water is moving both upward from the repository, and downward from the top of the model, into the underpressured shales. (Note that advective flows are indicated as arrows only in those cases where the flow rate exceeds 10^{-4} m/year.)

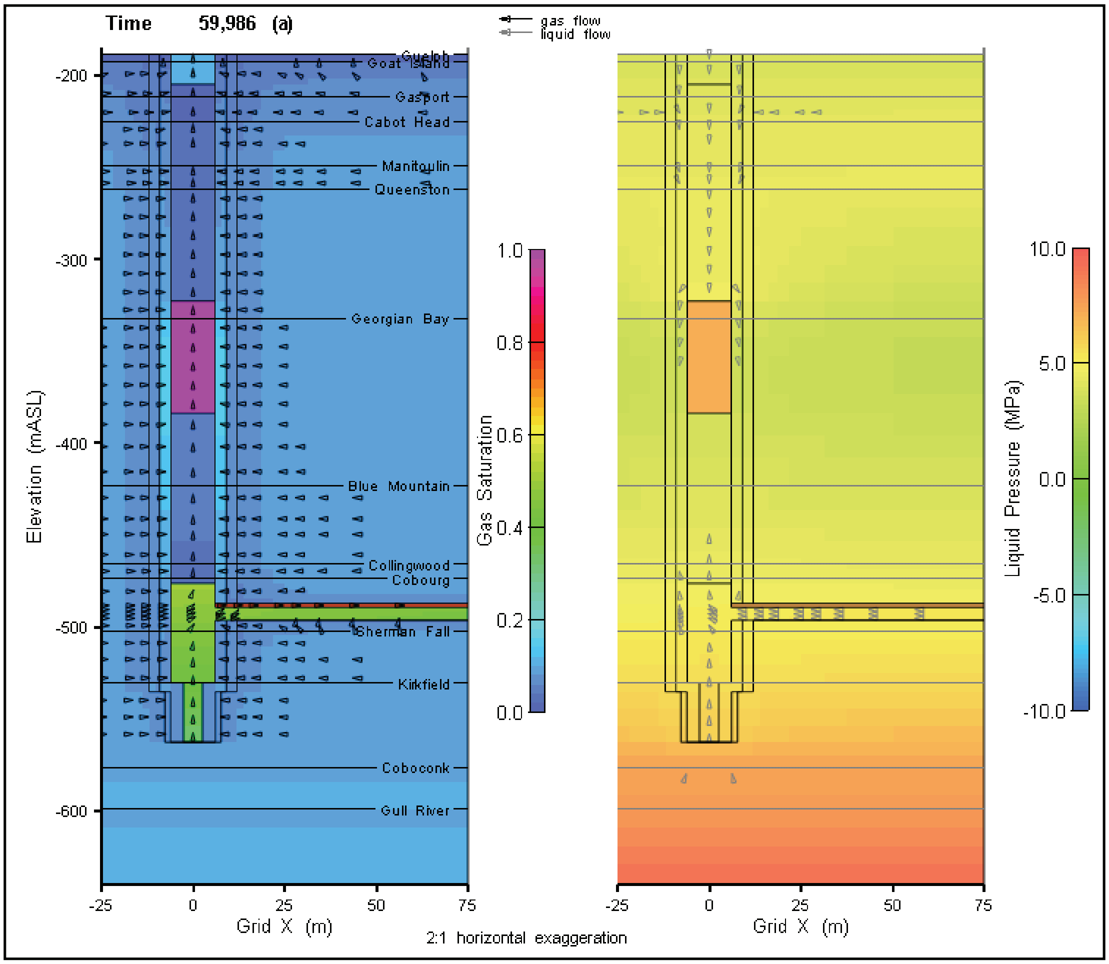


Figure 8-27: NE-RC: Shaft Saturations, Flows and Pressures at Year 60,000 for the Reference Case (NE-RC)

Figure 8-28 shows detailed transport modelling results, which illustrate the extent of transport through the groundwater over one million years, assuming resaturation of the DGR at closure and the instantaneous release of Cl-36 from the waste packages. Cl-36 is an important nuclide in groundwater transport since it has a relatively large inventory in the DGR, a long half-life, dissolves in water, and it is not sorbed and so is mobile. The results show slow diffusion of Cl-36 outwards from the repository panels in all directions and some diffusion up the shaft.

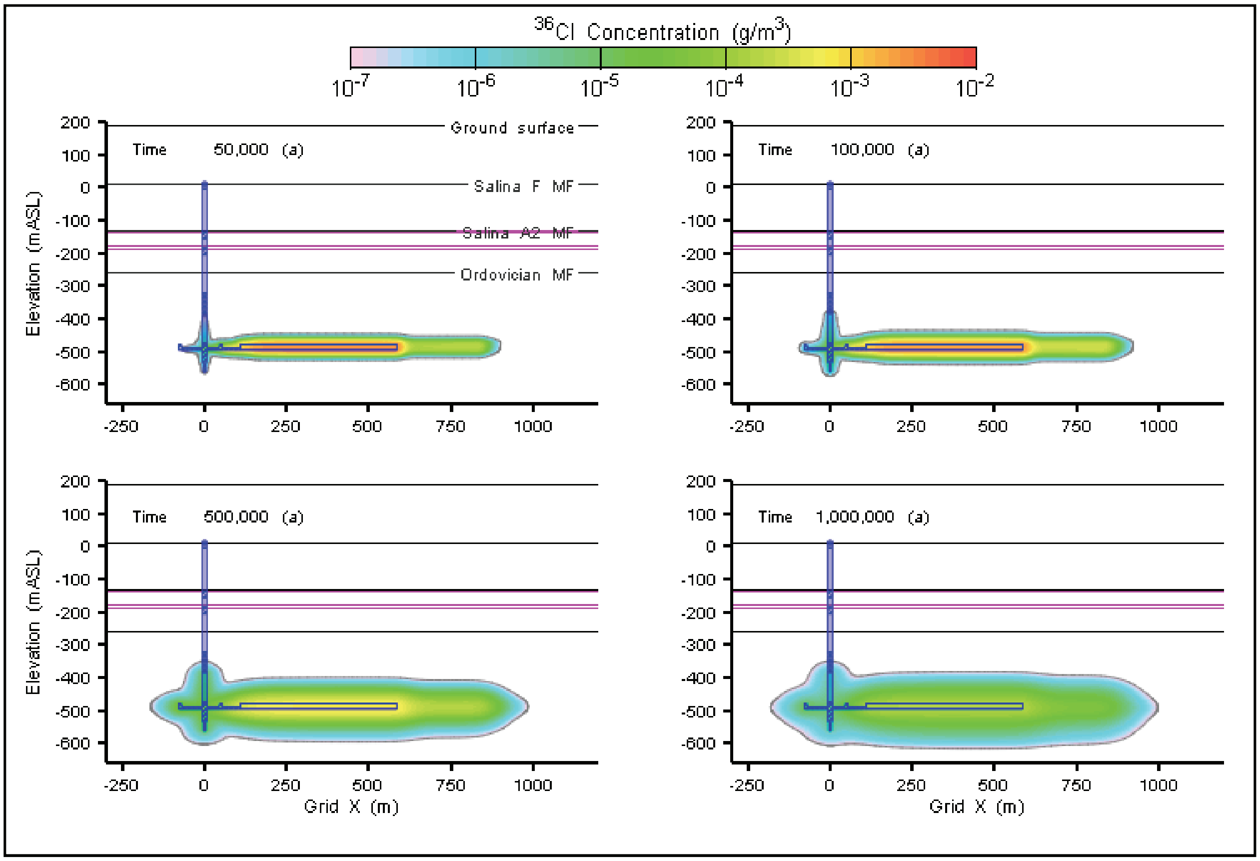


Figure 8-28: Transport of Cl-36 in Groundwater in the Geosphere for Instant Resaturation and Release (NE-RS)

Figure 8-29 shows the total calculated radionuclide concentrations in the formations above the DGR for the Reference and Simplified Base Cases. Nb-94 and Zr-93 (and its decay product Nb-93m) dominate the releases from the DGR beyond about 4,000 years. These radionuclides are sorbed onto shales (including the Collingwood and Blue Mountain Formations) but not on limestones (such as the Cobourg). Their greater retention on the shales means that concentrations in the Collingwood exceed those in the Cobourg, which is closer to the DGR, after about 100,000 years.

The concentrations decline with distance from the DGR, such that calculated peak concentrations in the rock are comparable to the natural background radioactivity in the Cobourg and Collingwood (mostly K-40 and U-238), and do not exceed 1 Bq/m^3 beyond the Queenston Formation. Diffusion of contaminants down into the Cambrian results in a peak concentration of around 3300 Bq/m^3 in the Cambrian for the Reference Case after about 1.5 million years[2].

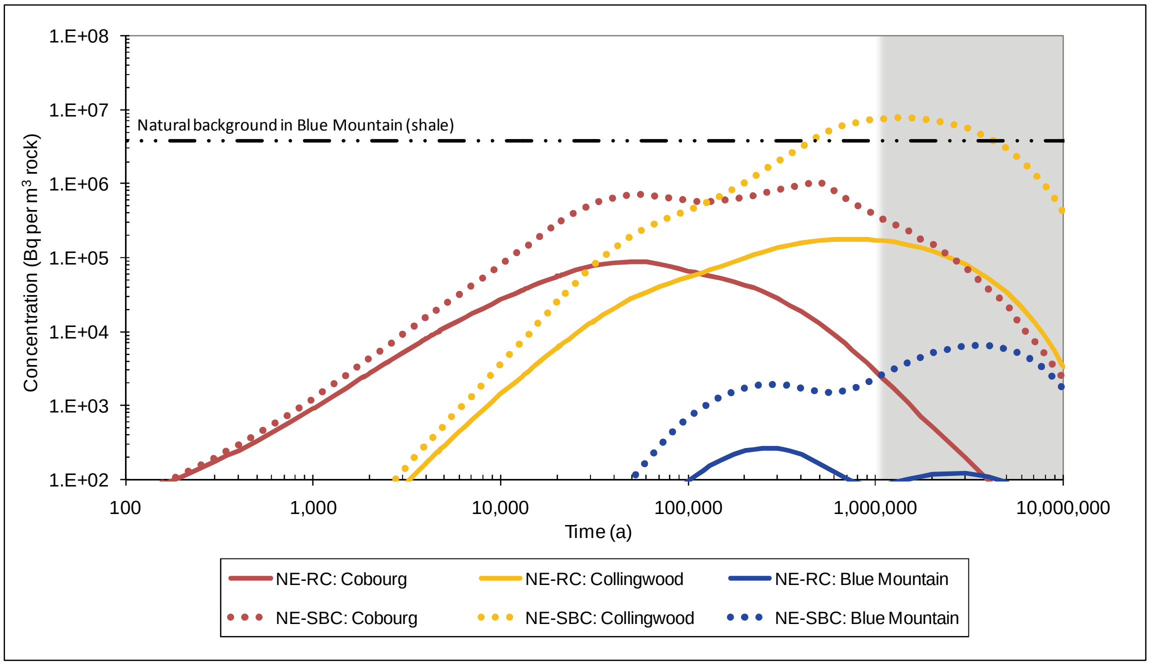


Figure 8-29: Radionuclide Concentration (Per m^3 of Rock) in the Deep Ground Waster Zone for the NE-RC and NE-SBC Case

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[2] Consumption of water with this concentration would result in a dose of around 0.002 mSv/year if it were assumed that water was pumped directly from the Cambrian and used without any treatment. This is not possible since the salinity of Cambrian water is around 200 g/L, a factor of 7 higher than seawater.

Figure 8-30 shows the calculated concentrations in the shaft sealing materials and demonstrates their effectiveness at minimizing contaminant transport. The figure shows that concentrations are reduced to very small levels. No concentrations greater than 1 Bq/m^3 are calculated for the seals above the top of the Manitoulin Formation in the Reference Case, and the Salina A1 Unit upper carbonate formation in the Simplified Base Case.

Figure 8-31 shows the calculated concentrations within the repository and shaft for nickel, which is a potentially hazardous non-radioactive element in the waste and waste packages. The transfer rate of other non-radioactive elements to the shaft was similar to or less than for nickel.

The low and slow level of repository resaturation, combined with the very low permeability of the host rock and the shaft seals means that effectively no contamination enters the shallow groundwater zone and then the biosphere (see Table 8-8). I-129 and Cl-36 dominate the small radionuclide flux, due to their long life and greater mobility. I-129 dominates in the Reference Case due to the slower transport in this case, and the longer half-life of I-129.

Table 8-8: Maximum Calculated Flux to the Shallow Groundwater Zone

Calculation Case	Max. Calculated Flux	Time of Max. Calculated Flux (Ma)	Main Contributor to the Max.
NE-RC: Reference Case	3×10^{-6} Bq/year	> 1	I-129
NE-SBC: Simplified Base Case	2×10^{-3} Bq/year	> 1	Cl-36
NE-NR: Non-radioactive Case	0.03 g/year	> 1	Ni

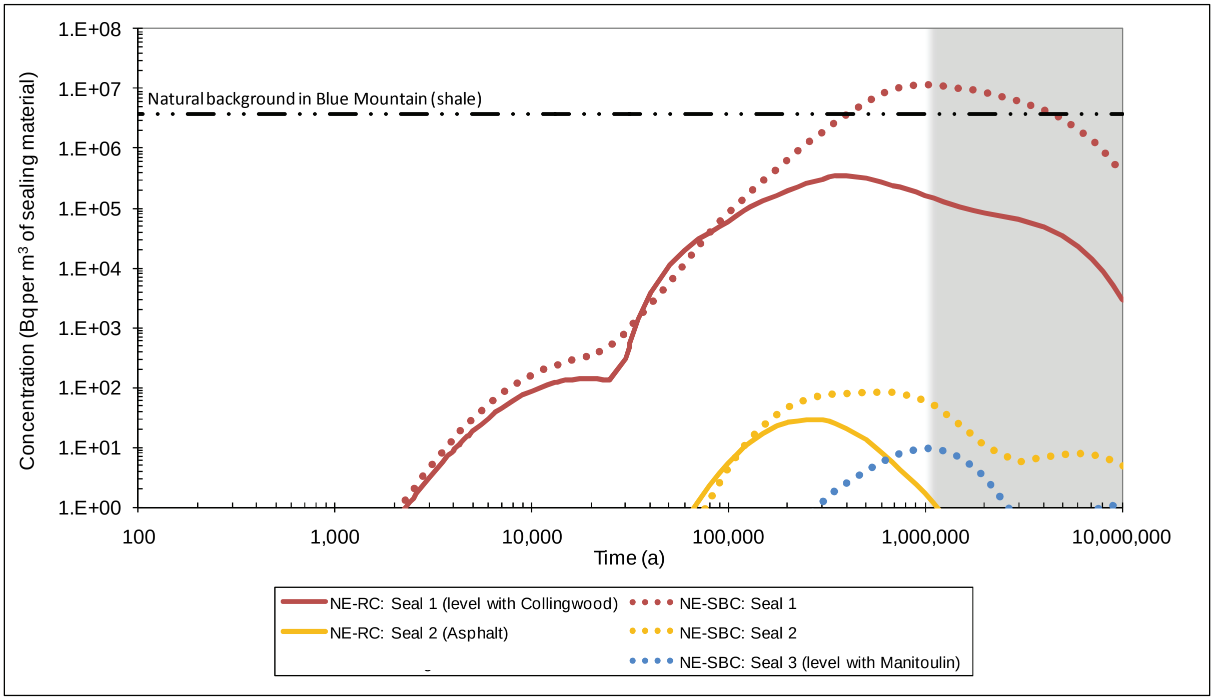

Figure 8-30: Radionuclide Concentration in Shaft for NE-RC and NE-SBC Cases

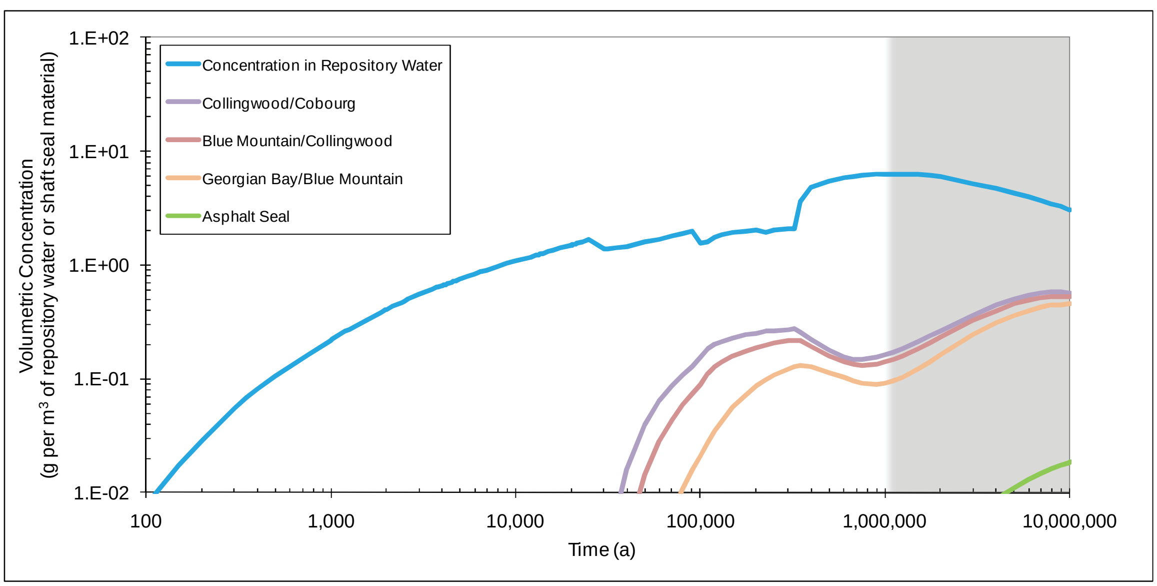

Figure 8-31: Nickel Concentration in the Repository and Various Shaft Horizons for NE-RC Case

8.6.4.4 Normal Evolution – Impacts

The very small release of contaminants to the biosphere results in very small concentrations in biosphere media as shown in Table 8-9.

For comparison, surface waters have a provincial background concentration of around 0.1 Bq/L gross-beta. Lake sediments from the regional study area have Cs-137 concentrations of around 0.2 Bq/kg, and naturally occurring K-40 of around 250 Bq/kg. Soils have concentrations of K-40 and Cs-137 of around 500 Bq/kg and 3 Bq/kg, respectively, at provincial background locations (Sections 2.4.3-2.4.6 of NWMO11ah).

The calculated doses to the site resident resulting from these small concentrations are negligible and are summarized in Table 8-10. The calculated doses are many orders of magnitude lower than the dose criterion of 0.3 mSv/year.

The calculated radionuclide concentrations in the biosphere for both the Reference and Simplified Base Cases are much smaller than the screening NECs for impacts on non-human biota in Table 8-1. The calculated concentrations of non-radioactive contaminants in biosphere media for the Reference Case are also much smaller than the environmental quality standards for groundwater, soils, surface water and sediments designed to protect human health and the environment (Table 8-2).

Table 8-9: Maximum Calculated Biosphere Concentrations

Calculation Case	Well Water (Bq/L)	Soil (Bq/kg)	Surface Water (Bq/L)	Sediment (Bq/kg)
NE-RC: Reference Case	6×10^{-15}	5×10^{-15}	1×10^{-17}	1×10^{-14}
NE-SBC: Simplified Base Case	3×10^{-12}	4×10^{-12}	6×10^{-15}	3×10^{-13}

Table 8-10: Maximum Calculated Dose to an Adult

Calculation Case	Max. Calculated Dose (mSv/year)	Time of Max. Calculated Dose (Ma)	Main Radionuclide Contributing to the Max.
NE-RC: Reference Case	2×10^{-15}	> 1	I-129
NE-SBC: Simplified Base Case	1×10^{-13}	> 1	I-129

8.7 Disruptive Scenarios

A key factor in the long-term safety of the DGR is the site geology, in particular the thick layers of low-permeability rock around and above the repository. The Disruptive Scenarios consider various unlikely cases in which the integrity of the geosphere barriers is breached: human intrusion, severe shaft seal failure, poorly sealed borehole, and vertical fault. (Earthquakes and glaciation were considered within the Normal Evolution Scenario.)

When analyzing the Disruptive Scenarios and their results, it is important to recognize that the likelihood of the events initiating the Disruptive Scenarios is expected to be very low. The likelihood of the as-modelled scenarios occurring is even lower as the scenarios make additional conservative assumptions, for example relating to human practices. Nevertheless, these scenarios provide insight into the robustness of the DGR system.

The selected Disruptive Scenarios are described and evaluated in the sections below. Figure 8-32 shows their locations assumed in the current assessment. Human intrusion occurs into Panel 1, which has the highest amount of ILW. The poorly sealed borehole is the closest existing borehole at repository depth. Two locations for the vertical fault are considered – one just outside the well-characterized site area at 500 m distant, and one within the area at 100 m from the waste panels.

The scenarios are evaluated separately rather than in combination since the individual scenarios have low probability and independent causes, and so their probabilities of occurring together are even lower.

Consistent with the Normal Evolution Scenario, a reference calculation is undertaken for each Disruptive Scenario. To avoid ambiguity with the Normal Evolution Scenario's Reference Case, the reference calculation for each Disruptive Scenario is termed the Base Case calculation. In addition to each Base Case calculation, a range of variant calculations has been undertaken for each Disruptive Scenario.

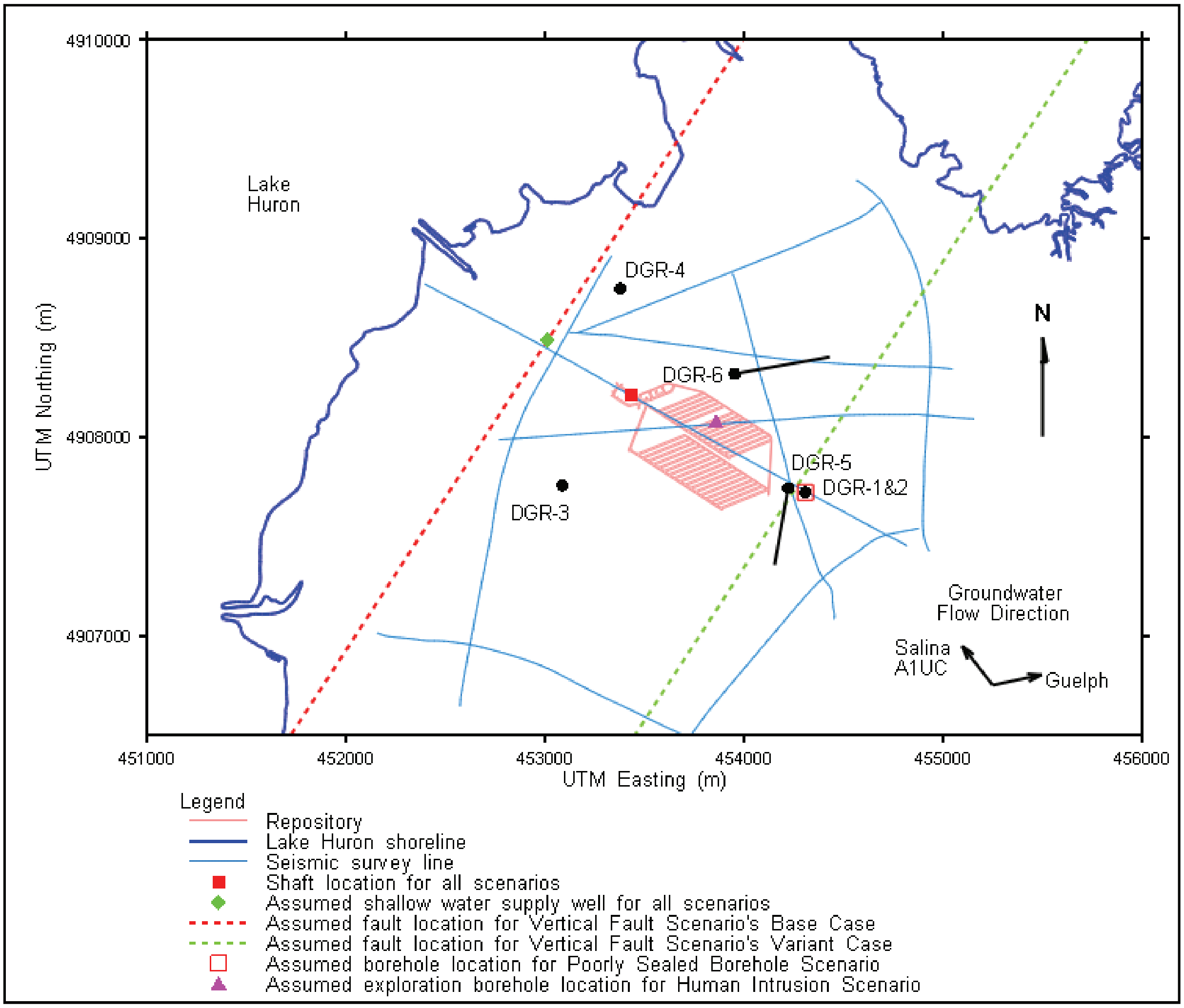

Figure 8-32: Location of Assumed Disruptive Scenarios Relative to the DGR Layout and the Site Characterization Deep Boreholes and Seismic Survey Area

8.7.1 Human Intrusion Scenario

8.7.1.1 Description

The depth of the repository, its limited footprint and the absence of resources in its vicinity, all mean that direct intrusion into the DGR is unlikely. Nonetheless, the Human Intrusion Scenario considers the possibility that at some time in the future after control of the site is no longer effective, people could conduct exploratory deep drilling in the vicinity of the site without realizing that it is present, and inadvertently intercept the repository[3]. Furthermore, it is conservatively assumed that standard deep drilling practices are not followed, i.e., there is no proper disposal of contaminated material nor appropriate closure of the borehole.

The intrusion results in the release of contaminants directly to the surface as contaminated gas or drill core and drilling debris (Figure 8-33). If the borehole is then abandoned, there are little further consequences since the repository remains unpressurized. Only if the borehole is continued down into the pressurized Cambrian and then poorly sealed, is there release of contaminated groundwater into the shallow groundwater via the borehole (Figure 8-34). The potential resulting exposure situations are summarized in Table 8-11.

Table 8-11: Exposure Situations for the Human Intrusion Scenario

Critical Group	Direct Release to Surface		Release to Shallow Groundwater System
	Release Mechanism		Release Mechanism
	Gas	Drill Core	Groundwater
Drill crew at wellhead	✓		
Resident near to drill site	✓		
Laboratory technician		✓	
Future site resident using contaminated soil		✓	
Future site resident using contaminated groundwater			✓

[3] Deliberate intrusion is not assessed since it is expected that the intruders would take appropriate precautions.

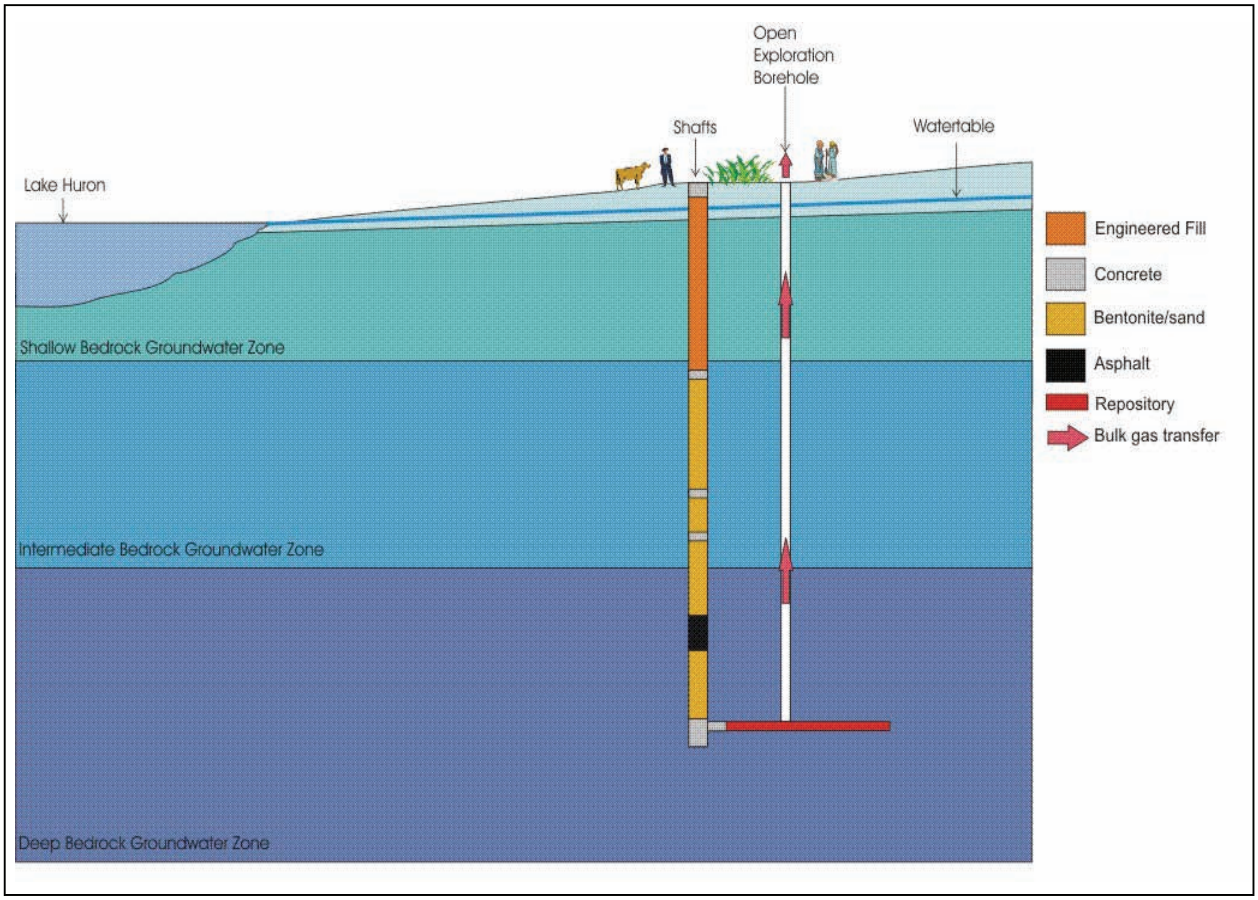

Figure 8-33: Human Intrusion Scenario: Schematic Representation of Short-Term Gas Release

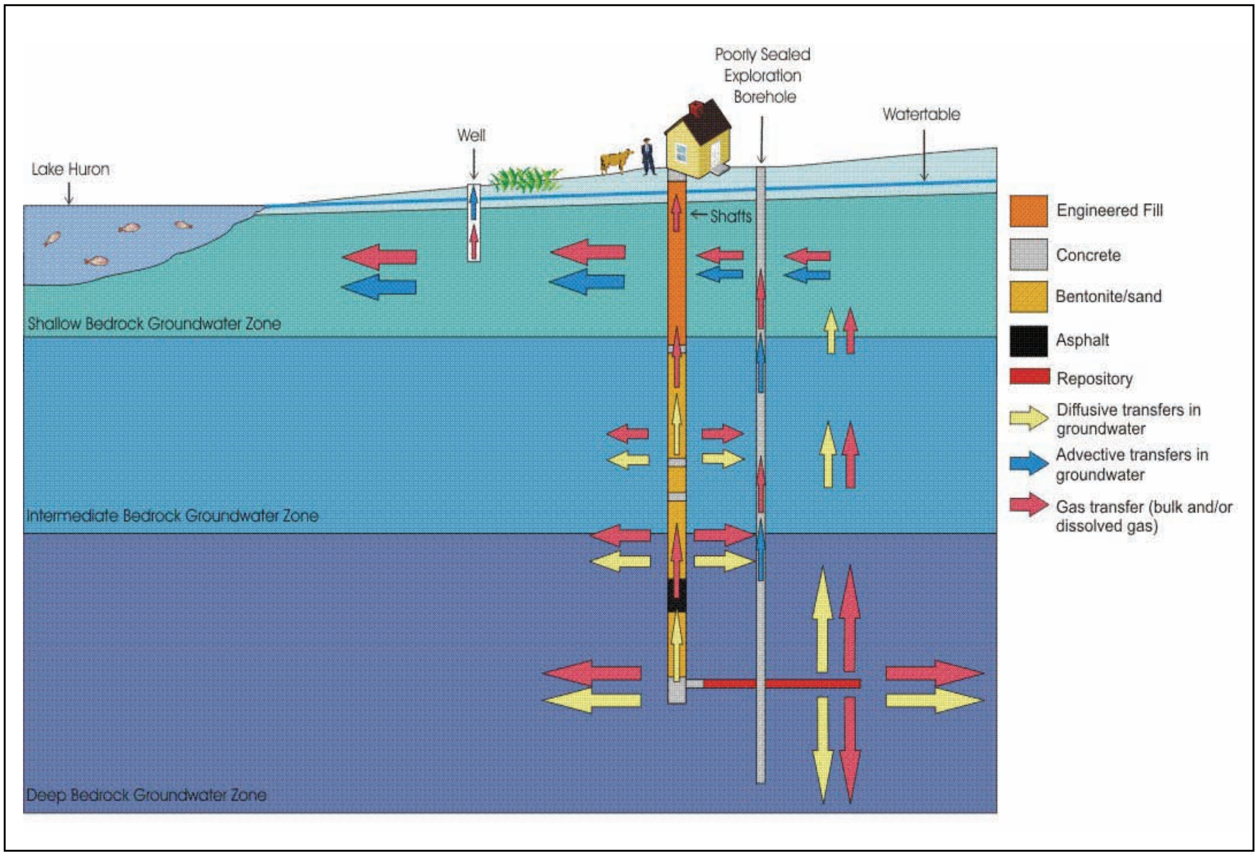

Figure 8-34: Human Intrusion Scenario: Schematic Representation of Long-Term Ground Water Release

8.7.1.2 Models, Implementation and Data

Conceptual and mathematical models and data have been developed for the scenario and they are described in Chapter 2 of the Human Intrusion and Other Disruptive Scenarios report (NWMO11aq). The key aspects of the conceptual model for the scenario are summarized in Box 3.

Table 8-12 and Table 8-13 summarize the calculation cases and the key modelling assumptions for the scenario, respectively.

Box 3: Key Aspects of the Conceptual Model for the Human Intrusion Scenario

Gas Release

- Intrusion via exploration borehole directly into an emplacement room in Panel 1 at some time after controls are no longer effective (i.e., after 300 years).
- Resaturation profile prior to borehole intrusion consistent with the Normal Evolution Scenario (Figure 8-22).
- Contaminants (H-3, C-14, Cl-36, Se-79, I-129 and Rn-222) released via borehole from the repository into surface environment as gas.
- Gas release via the borehole is limited by blow-out preventers as per normal practice in sedimentary rocks, but depressurization allowed to occur.
- Atmospheric dispersion of released gas.
- Direct impacts on drill crew and nearby residents considered.

Drill Core Release

- Intrusion via exploration borehole into an emplacement room in Panel 1 at some time after controls are no longer effective (i.e., after 300 years).
- Retrieval of waste in unconsolidated core and subsequent spreading over the surface soil resulting in direct impacts on drill crew and future site resident using the soil.
- Retrieval of an intact sample of waste in drill core and subsequent direct impacts on laboratory technician examining core considered.

Groundwater Release

Consistent with the Normal Evolution Scenario, other than:

- Intrusion via exploration borehole into an emplacement room in Panel 1 at some time after controls are no longer effective (i.e., after 300 years).
- Resaturation profile prior to borehole intrusion consistent with the Normal Evolution Scenario (Figure 8-22).
- The borehole is poorly sealed (seal has the properties of engineered fill) and the casing degrades allowing relatively rapid resaturation of the repository following intrusion and the migration of contaminants into the shallow groundwater system.
- Site resident living and farming on repository site is potentially exposed through contaminated groundwater pumped from the shallow groundwater system.

Table 8-12: Calculation Cases for the Human Intrusion Scenario

Case ID	Case Description
HI-BC HI-NR	As NE-RC but exploration borehole drilled from surface and terminated in Panel 1 of the repository. Considers the consequences of surface release of contaminated gas immediately following intrusion. Retrieval of contaminated drill core is also assessed. HI-NR considers non-radioactive species
HI-GR1	As NE-RC but considers an exploration borehole drilled at 300 years down into Panel 1 of the repository and terminated at repository depth, and then poorly sealed resulting in a hydraulic conductivity of 10^{-4} m/s and porosity of 0.25. Considers the long-term consequences of releases to repository water following repository resaturation.
HI-GR2	As HI-GR1 but borehole terminates in the pressurized Cambrian.

Table 8-13: Key Modelling Assumptions for the Human Intrusion Scenario

Assumption	Motivation/Reason for Assumption	Impact of Assumption
Surface Release Pathway (Gas)		
Gas is vented from a blowout preventer	Blowout preventers are standard practice in drilling in deep sedimentary rocks. But gas is assumed to be vented and not capped.	A release rate of 1 m^3/s (atmospheric) is assumed for the period that a drill crew is on site (see Section 2.4.3.1 of NWMO11aq). The calculated air concentration is directly proportional to the release rate.
Both drill crew and nearby resident are exposed	Considers potential impacts on both intruder and inadvertent public.	A drill crew would be exposed if the repository were intercepted. An offsite resident is less likely to be present, but has been conservatively assumed to be living 100 m from the drill site.
Surface Release Pathway (Drill Core)		
Waste is retrieved in drill core	The assumption is made to examine the possible consequences of direct exposure to waste by a laboratory technician.	The calculated dose to a laboratory technician is directly proportional to the activity concentration in the waste. The reference calculations consider the average concentration in Panel 1.
Core is examined in a laboratory, without precautions	The assumption is made to examine the possible consequences of direct exposure to waste.	The assumptions concerning exposure in the laboratory determine the dose and are conservative. However, the contaminated core may not be examined at all, or may rapidly be identified as requiring careful handling.

Assumption	Motivation/Reason for Assumption	Impact of Assumption
Drill crew do not wear personal protective equipment	Conservative assumption on future human behavior; ingestion and inhalation pathways are relevant.	In practice, external irradiation dominates the calculated exposures; therefore, this is not a key assumption.
Drill core debris is not disposed of properly	Conservative assumption on future human behavior.	Normal practice involves the collection of drill core and associated debris for storage/disposal under controlled conditions, therefore the assumed fate of the slurry in the model (dispersed in the soil used to grow food) would not occur. The calculated doses for a future resident of the drill site are directly proportional to the drill core concentration in soil.
Drill core debris is mixed with soil and not leached before being used for farming	Conservative assumption on future human behavior.	Leaching of soil will reduce the concentrations of contaminants in soil that is farmed, therefore the assumption is conservative. The assumption that the soil is used by a farmer soon after the completion of drilling, is also conservative.
Groundwater Release Pathway		
Intrusion borehole occurs after 300 years	This is the earliest credible time that all institutional control and memory might not be effective in preventing intrusion.	The impacts could be higher if the intrusion is early due to less radioactive decay.
Borehole extends to the Cambrian	Necessary for groundwater flow to occur up the borehole.	If the borehole were terminated at the repository, there would be no release of contaminants in water via the borehole as the repository would pressurize only very slowly; the increased pressure of the Cambrian is necessary to drive flows up to the shallow groundwater zone. It is, however, unlikely that a borehole would be continued after the repository had been encountered.
Borehole is poorly sealed	Necessary for flows to occur up the borehole.	The borehole would be expected to be sealed. The model assumes that this is not the case and the borehole has a hydraulic conductivity of 10^{-4} m/s. The hydraulic conductivity determines the rate of release of contaminants via the borehole.
No sorption occurs in the borehole	Conservatively permits the maximum rate of release of contaminants to the shallow groundwater zone.	Contaminants will sorb to the material filling the borehole, reducing the concentrations released to the shallow groundwater zone. The assumption that there is no sorption is conservative and maximizes the rate at which contaminants are released.
Notes: The Human Intrusion Scenario utilizes the same models for the near-field (surface release pathway), geosphere and biosphere as the Normal Evolution Scenario Reference Case, and the key assumptions in Table 8-7 therefore also apply.		

8.7.1.3 Human Intrusion Scenario Results

If an exploration borehole struck the DGR, contaminants could be released to the surface and result in human exposure. The calculations assume intrusion into Panel 1 where radionuclide concentrations are highest.

Under the reference conditions, the saturation of the repository is less than 1% throughout the calculations (Section 8.6.1). Under these conditions, liquid would not be released from the repository via an intruding borehole since the repository is largely unsaturated. However gas would be released since it would be at greater than atmospheric pressure (Figure 8-4). C-14 and Rn-222 are present in the gas at concentrations above 1 Bq/m^3 (see Figure 8-35). C-14 is present with the greatest activity, decreasing after 10,000 years due to radioactive decay. The concentration of Rn-222 decreases at first due to the decay of its Ra-226 parent but then shows ingrowth from longer-lived U-238/U-234.[4]

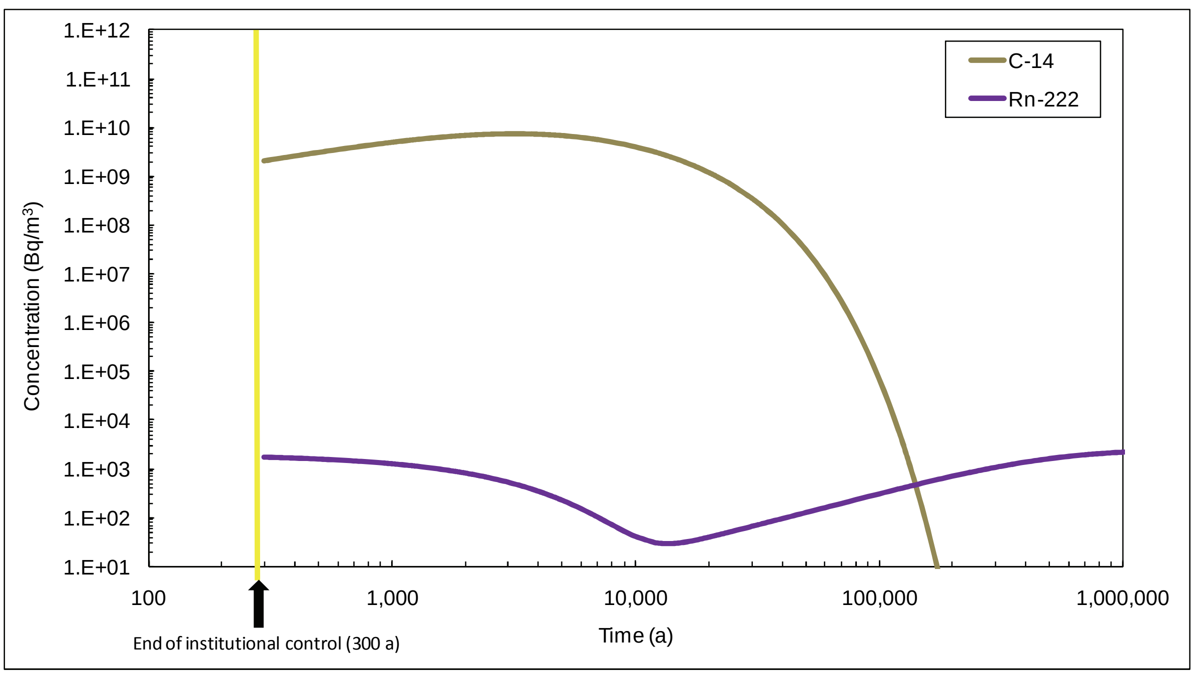

Figure 8-35: Calculated Concentrations of Radionuclides in Repository Gas at Repository Pressure, Human Intrusion, Base Case

[4] These concentrations do not include loss of C-14 by isotope exchange with stable carbon in the carbonate rock, and trapping and decay of Rn within its source material.

A wide variety of exposure pathways could occur for this scenario, so a range of potential receptors has been assessed – the drill crew and nearby residents (i.e., within 100 m of the drill site) exposed during the drilling, laboratory technicians exposed to the core sample, and future site residents exposed to soil contaminated with the extracted core. Calculated doses for these people are shown in Figure 8-36. The calculated dose to the drill crew peaks at about 1 mSv due to exposure to Nb-94 in the drill core debris. The calculated dose to the nearby resident peaks at about 0.1 mSv due to inhalation of C-14 released from the borehole. The dose to the future site resident is dominated by external irradiation from Nb-94 and peaks at about 1 mSv/year.

The human intrusion scenario has a low probability of occurrence. As an indication, an exploratory deep borehole drilling rate of around $10^{-10}/m^2$/year (equivalent to one deep borehole per 100 years per 10 km x 10 km area), and a panel plan area of approximately 0.1 km^2 correspond to a probability of occurrence of about 10^{-5}/year. This is a low probability per year. Over long time scales, it becomes likely – however, the potential dose impacts also decrease over long times, and in particular intrusion impacts fall below the dose criterion after about 10,000 years. Based on a probability of 10^{-5}/year, a peak dose of 1 mSv and a health risk of 0.057/Sv (ICRP07), the associated risk is around 6 x 10^{-10} serious health effects per year, well below the reference health risk value of 10^{-5}/year (Section 8.1.2).

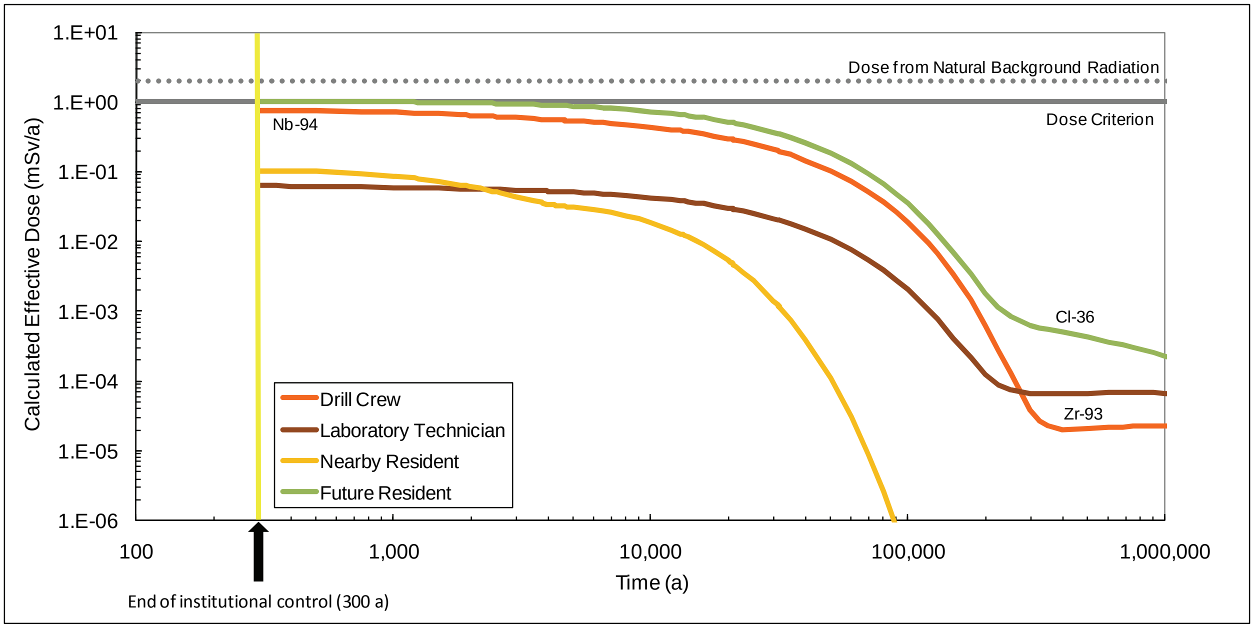

Figure 8-36: Calculated Doses from Surface Release of Gas and Core Resulting from Human Intrusion, as a Function of the Time of Intrusion for the Base Case

Calculations of the concentration of non-radioactive contaminants in soils contaminated by the drill core indicate that environmental quality standards given in Table 8-2 are not exceeded. If contaminated drill core is left on soil around the site (assumed to be an area of about 30 m x 40 m), then Pb, Ni, Cu, Mo and Cr concentrations are at about 10-30% of their environmental criteria, while all others are much lower.

Comparison of radionuclide concentrations in biosphere media against the screening NECs given in Table 8-1 for non-human biota show that C-14 and Nb-94 exceed the screening criterion by about a factor of 20 within the site assuming the contaminated drill core debris is left on site and mixed with soil, while all other radionuclides are below their criteria by at least a factor of 7. Since this intrusion is very unlikely, leaving drilling debris on site is against current regulations, and any exposure is localized around the drill site, the risk is low. Furthermore, less conservative ERA calculations show that the resulting doses to site-specific biota are around 3% of relevant dose criterion (Appendix G, NWMO11aq).

Standard practice requires that any site investigation borehole is sealed once investigations are complete. However, the scenario analysis also considered "what if" the borehole is poorly sealed, resulting in a continuing pathway for contaminants from the DGR to the shallow groundwater zone after an intrusion event. In this case, it is found that there are no further consequences, because the repository is not pressurized and there is little groundwater flow up the borehole and into the shallow groundwater system.

Detailed modelling has shown that contaminants could only be released from the repository through the borehole if the intruding borehole penetrated through the repository and continued into the pressurized Cambrian rocks and was not appropriately sealed. In this highly improbable case, the peak calculated dose to the adult site resident would be around 30 mSv/year for intrusion in 400 years, decreasing to 0.003 mSv/year after 60,000 years. The dose is dominated by exposure to C-14 via plant ingestion, due to the use of contaminated well water for irrigation.

In summary, this dose assumes inadvertent intrusion into the repository, plus further intrusion of the borehole down to the Cambrian, plus poor sealing of the borehole, plus a family farming on the site using a well that directly intercepts contaminated groundwater from the borehole. Assuming the same probability of occurrence as for intrusion into the repository (thereby assuming the probability of continuing into the Cambrian and poorly sealing the borehole is unity), the peak dose equates to a risk of around 2×10^{-8} of serious health effects per year, more than two orders of magnitude below the reference health risk value of 10^{-5}/year.

8.7.2 Severe Shaft Seal Failure Scenario

8.7.2.1 Description

The shafts represent a potentially important pathway for contaminant release, and therefore the repository design includes specific measures to provide good shaft seals, taking into account the characteristics of the geosphere. The Normal Evolution Scenario considers the likely behaviour of the shaft seals, which include some expected degree of degradation with time. The Severe Shaft Seal Failure Scenario considers a "what-if" case in which the shaft seals, and the shaft and repository damage zones, are assumed to have significantly degraded properties. For example, the seals may not be fabricated or installed appropriately, or the long-term performance of the seals may deteriorate due to unexpected physical, chemical and/or biological processes.

The key transport pathways for releases from the repository are summarized in Figure 8-37.

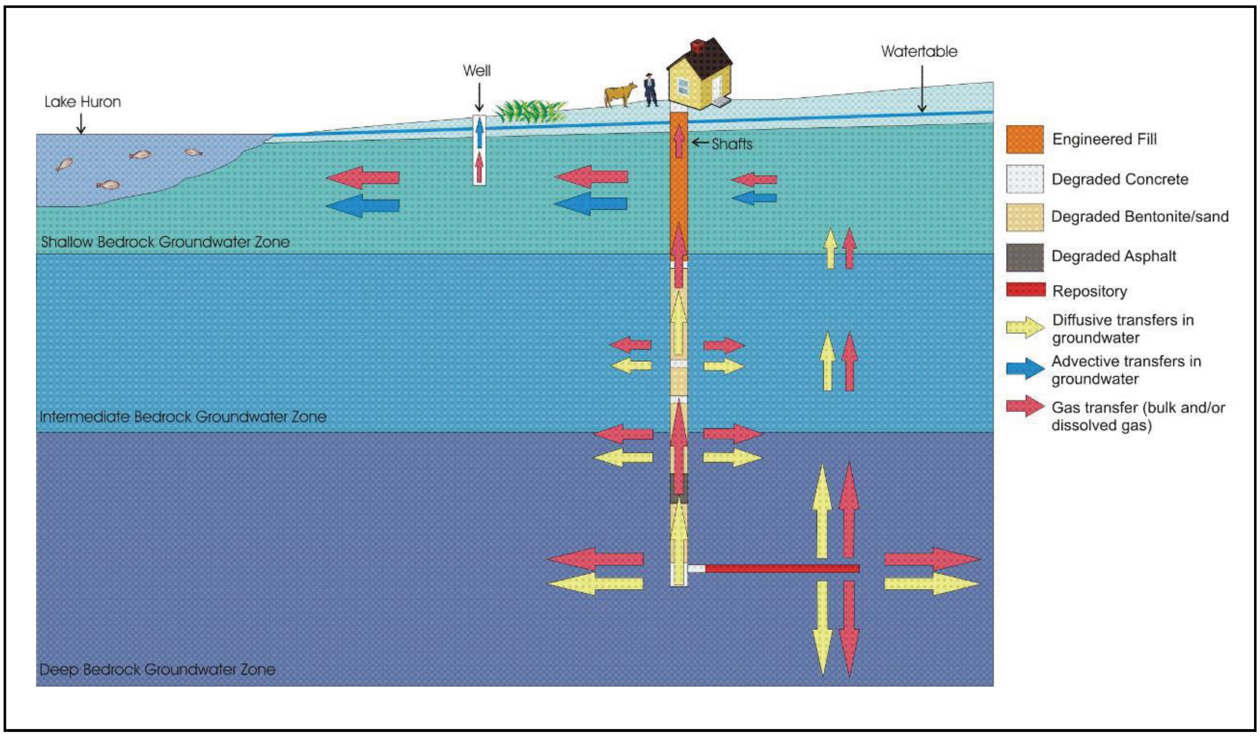

Figure 8-37: Schematic Representation of Potential Transport Pathways for the Severe Shaft Failure Scenario

8.7.2.2 Models, Implementation and Data

Conceptual and mathematical models and data have been developed for the scenario and they are described in Chapter 3 of the Human Intrusion and Other Disruptive Scenarios report (NWMO11aq). The key aspects of the conceptual model are summarized in Box 4. The model is the same as for the Normal Evolution Scenario (Section 8.6.2.1), except that changed parameter values represent the significantly degraded characteristics of the concrete monolith and shaft seals, and the increased permeability of the repository/shaft damage zones. These differences could result in increased flow of water down the shaft into the repository initially, and contaminated water and gas up the shafts from the repository later.

Table 8-14 and Table 8-15 summarize the calculation cases and the key modelling assumptions for the scenario, respectively.

Box 4: Key Aspects of the Conceptual Model for the Severe Shaft Seal Failure Scenario

Waste and Repository

- The repository EDZ permeability is increased by an order of magnitude compared with the Normal Evolution Scenario Reference Case.
- Resaturation of the repository is determined by detailed modelling which evaluates water inflow/outflow, gas generation, gas inflow/outflow, and gas pressure.
- Contaminants migrate into the host rock and shafts by diffusion and/or advection or by gas permeation (driven by repository gas pressure) or by gas dissolution into groundwater.

Geosphere and Shafts

- The shaft seals are degraded from the time of closure.
- The shaft EDZs have increased permeability (two orders of magnitude for inner EDZ and one order of magnitude for outer EDZ) compared with the Normal Evolution Scenario Reference Case.
- Sorption of Zr, Nb, Pb, U, Np and Pu on bentonite/sand is reduced by an order of magnitude compared with the Normal Evolution Scenario Reference Case.
- Initial underpressure in Ordovician formation, and steady overpressure in Cambrian formation, as in the Normal Evolution Scenario Reference Case.

Biosphere

- Model is the same as the Normal Evolution Scenario Reference Case.

Table 8-14: Calculation Cases for the Severe Shaft Seal Failure Scenario

Case ID	Case Description
SF-BC SF-NR	As NE-RC but properties of all seals and repository/shaft EDZs set to significantly degraded values from repository closure (e.g., seal hydraulic conductivity of 10^{-9} m/s, zero capillary pressure for shaft sealing materials, and reduced sorption). SF-NR considers non-radioactive species.
SF-ED	As SF-BC but increased shaft seal hydraulic conductivity (10^{-7} m/s) in order to understand the sensitivity of the system performance to shaft seal properties. This is in the range of a fine sand/silt material, about 4-5 orders of magnitude more permeable than the design-basis bentonite/sand and asphalt seals.

Table 8-15: Key Modelling Assumptions for the Severe Shaft Failure Scenario[a]

Assumption	Motivation/Reason for Assumption	Impact of Assumption
Instantaneous physical degradation of all shaft seals	One potential cause of broad degradation of shaft seals is poor installation practice, which would be effective from the time of closure.	Allows earlier ingress of water into repository through the shaft, which enhances contaminant release rates from the wastes. Also conservatively allows for earlier release from repository, before there is significant decay.
Reduced sorption in the shaft seals	There is no specific likely cause, although in principle a major change in the geochemical conditions could affect sorption.	Reduced capacity of shaft materials to retard radionuclides. In practice, the sorption assumptions do not have a very significant effect on releases, which are dominated by C-14 released through the shaft as free gas.
Further degradation of shaft/repository EDZs	The same failure mechanism that affects the shaft seals, affects the damaged zones around the shafts and repository.	Maximizes the flux of contaminants from the emplacement rooms to the base of the shafts and then up through the shaft. In practice, the shaft EDZs are not a significant pathway in this scenario due to the large volume of degraded shaft seal material.

Note:
a. The Severe Shaft Seal Failure Scenario utilizes the same models for the near-field, geosphere and biosphere as the Normal Evolution Scenario Reference Case, and the key assumptions in Table 8-7 therefore also apply.

8.7.2.3 Severe Shaft Seal Failure Results

The shaft seal includes multiple components utilizing a variety of materials that act individually and collectively as a barrier to contaminant transport. The Severe Shaft Seal Failure Scenario assesses a hypothetical situation in which there is a major breakdown in the performance of these barriers. Two situations are considered below.

- A Base Case, for which the hydraulic conductivity of all shaft seals are conservatively set at 10^{-9} m/s (i.e., at the top end of the range for bentonite-sand given in Section 4.5 of NWMO11am) with a porosity of 30% (SF-BC).
- An extra conservative case for which the hydraulic conductivity of all shaft seals are set at 10^{-7} m/s with a porosity of 30%, which is equivalent to fine silt and sand (SF-ED). This case is intended to test the parameter values at which shaft seals are not effective.

The degradation is assumed to be present at time of closure. The initial conditions at this time also include the underpressures observed in Ordovician formations.

The degraded shaft seals permit more rapid water inflow into the repository. Detailed modelling shows a greater degree of repository saturation in comparison to the Normal Evolution Scenario Reference Case (Figure 8-38). The resulting gas generation and reduced shaft seal capability allows the repository gas pressure to open a pathway that enables the repository gas to vent up the shafts (Figure 8-39). In the Base Case (SF-BC), this gas pathway is established after about 20,000 years.

For the Base Case, the bulk gas reaches the shallow geosphere zone at a peak rate of about 840 kg/year. About 5% of this gas flux would dissolve in the flowing shallow groundwater. The bulk gas carries C-14 labelled gases from the DGR, which can similarly dissolve in groundwater. Calculated concentrations in well water peak at about 3 Bq/L after about 23,000 years, consistent with the peak gas flux. Most of the peak gas flux to the shallow system does not dissolve in the groundwater and reaches the biosphere as free gas. Some of this bulk gas is assumed to enter a house that is conservatively taken to be positioned directly above the main shaft. The calculated radionuclide concentrations in the air inside the house peak at about 16,000 Bq/m^3 after about 23,000 years.

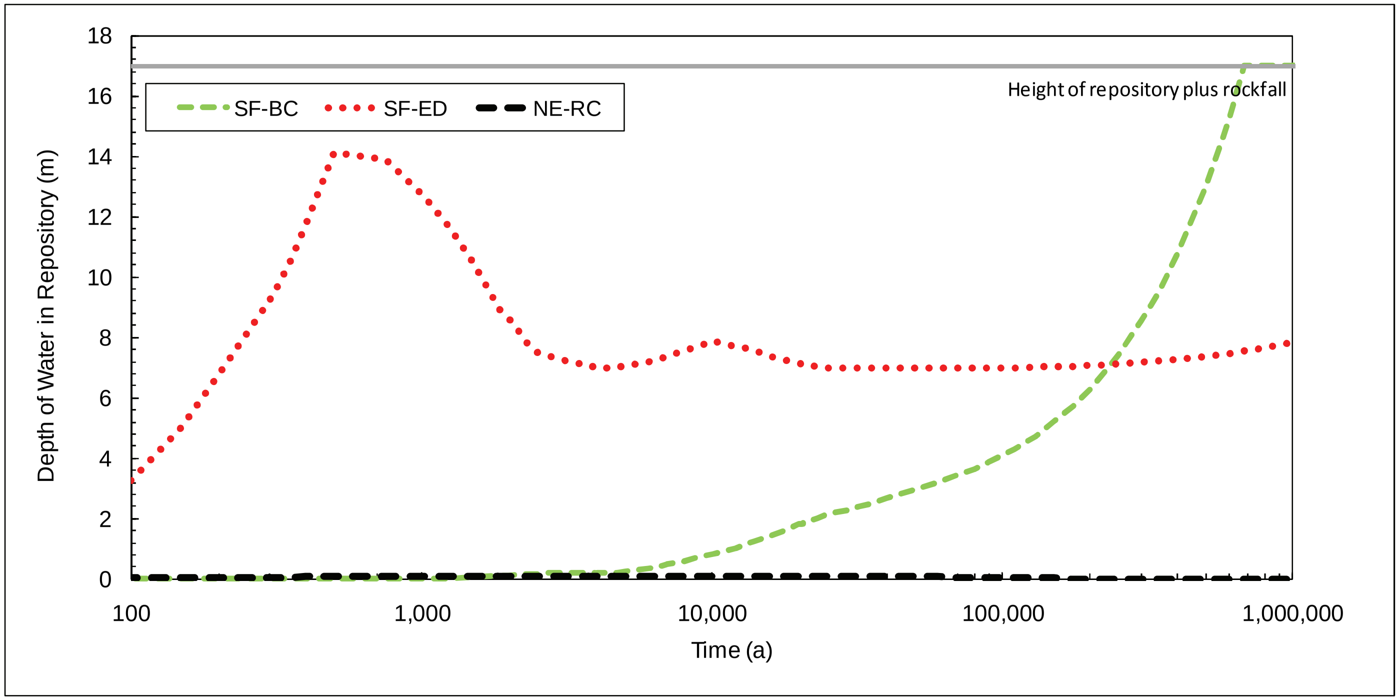

Figure 8-38: Depth of Water in the Repository for the Severe Shaft Seal Failure Cases

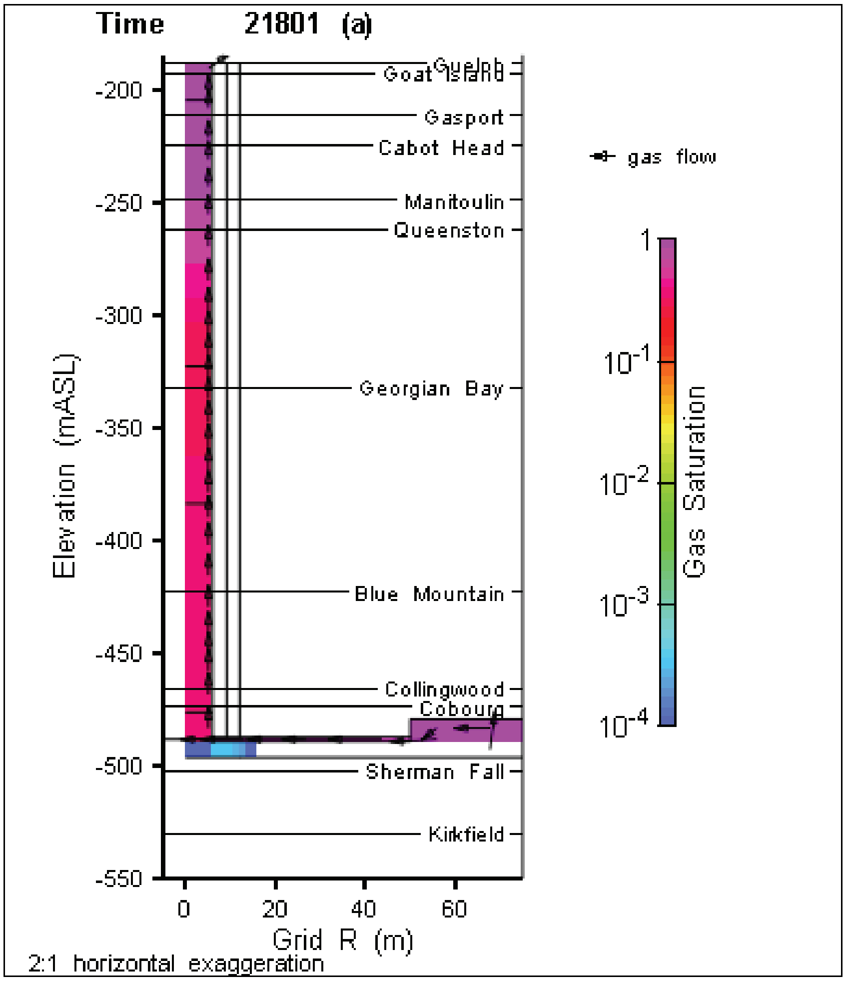

Figure 8-39: Shaft Gas Saturation and Flow for the SF-BC Case Showing Gas Venting via the Shafts

The Base Case results in calculated dose to the site resident that reaches a maximum of around 1.3 mSv/year after about 23,000 years (see Figure 8-40). This coincides with the peak release of C-14 labelled gases to groundwater in the shallow groundwater zone and directly to the biosphere. The dominant exposure pathways are inhalation within the house, positioned directly above the main shaft, and ingestion of plant produce, each of which contributes about 40% of the calculated peak dose. It is noted that a scenario likelihood of around 10^{-1} or less per year would result in the risk of serious health effects being less than the reference value of 10^{-5}/year. The probability of instant severe shaft seal degradation combined with a house positioned directly above one of the shafts can reasonably be considered to be significantly lower than this.

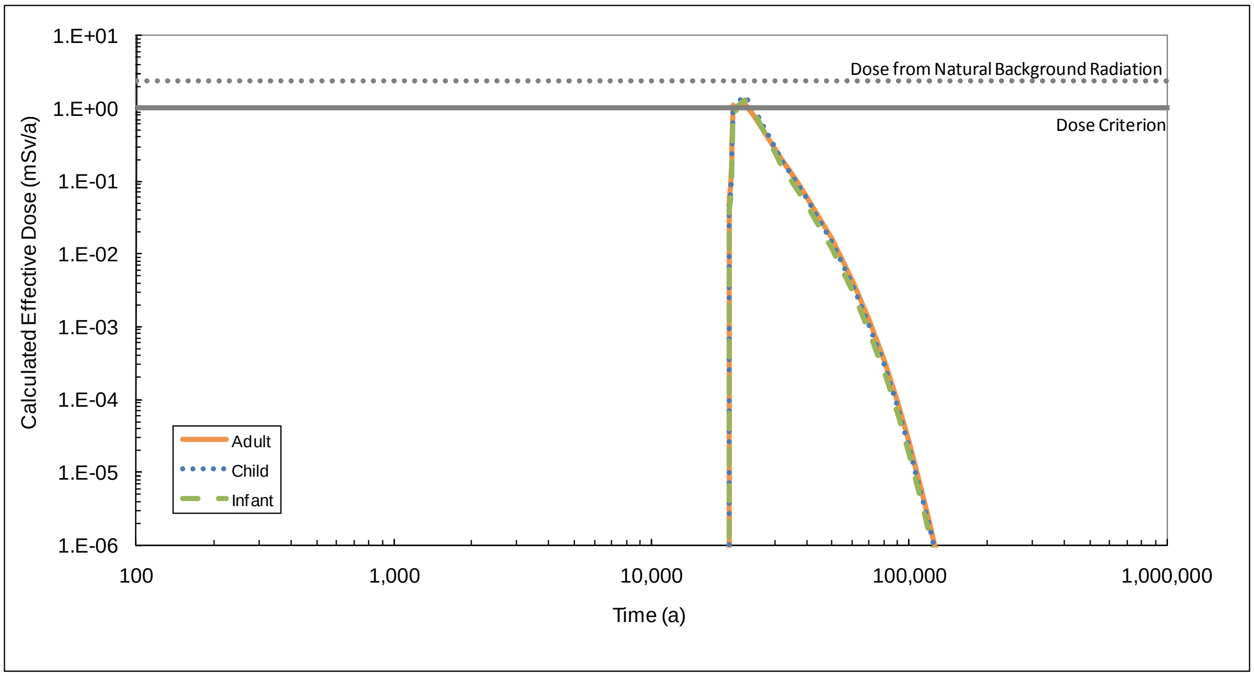

Figure 8-40: Calculated Effective Doses to Site Residents for the Severe Shaft Seal Failure Scenario, Base Case (SF-BC)

Calculated concentrations in biosphere media (soils, surface water, sediment) remain relatively low for the Base Case. The peak calculated concentrations for C-14 in soils and sediments remain below the NECs for protection of non-human biota given in Table 8-1. The peak calculated C-14 concentration in local surface water of 0.3 Bq/L is a factor of 1.4 above the associated screening NECs for the protection of non-human biota. Shaft seal failure is an unlikely scenario and these consequences would only apply if the failure is within about 50,000 years after DGR closure (due to C-14 decay). Also, the high concentration is in the local stream, and is slightly above the screening

NEC criterion. Based on these considerations, and the conservatism in the screening criterion, the actual risk to non-human biota is expected to be low. Calculated biosphere concentrations for all other radionuclides are more than seven orders of magnitude below their associated criteria.

There is a negligible release of non-radioactive contaminants via the groundwater pathway, and all calculated values are many orders of magnitude below the environmental quality standards given in Table 8-2.

In order to understand the performance of the DGR system, an extra degradation shaft seal failure case (SF-ED) is considered in which the shaft seal hydraulic conductivity is assumed to be a further factor of 100 higher. In this case, the calculated flux of contaminants to the shallow groundwater zone is again dominated by the transport of C-14 labelled gases with bulk gases via the shafts. The assumptions for the degradation of the shaft seals in case SF-ED results in calculated dose to the adult site resident living above the repository that reaches about 80 mSv/year after around 3800 years. The dominant radionuclide is C-14 and the dominant exposure pathway is the inhalation within the house position directly above the main shaft, which contributes about 75% of the peak calculated dose. It is emphasized that this calculation case is an extremely conservative case and was undertaken with the purpose of investigating the sensitivity of dose impacts to shaft seal properties.

These Severe Shaft Seal Failure cases would require that around 500 m of low-permeable shaft seal degrades to have an effective conductivity of 10^{-9} m/s or higher. This is very unlikely under the DGR conditions of low-flow, low-temperature, and use of multiple low-permeable seal materials. It is also noted that this scenario would have little consequence if the degradation occurred after about 60,000 years when C-14 would have significantly decayed. This is also the earliest time that ice-sheets from the next glacial cycle might be expected, so glacial cycles are not an important factor.

Finally, it is noted that the consequences decrease with distance from the site. The estimated total amount of C-14 in the DGR is $6x10^{15}$ Bq (Table 5-8). Even if this entire DGR inventory of C-14 were to be released as gas within one year, it would be roughly equivalent to the current allowed WWMF DRL for C-14 of $4.6x10^{15}$ Bq/year (Table 7-3) and the peak dose to anyone living around the Bruce nuclear site would be about 1 mSv or less.

8.7.3 Poorly Sealed Borehole Scenario

8.7.3.1 Description

Several site investigation/monitoring boreholes have been drilled in the vicinity of the DGR down to and beyond the depth of the repository during the site investigation phase. In all cases, the boreholes are outside the repository footprint by at least 100 m. Furthermore, they will be appropriately sealed on completion of site investigation/ monitoring activities. However, the Poorly Sealed Borehole Scenario considers the consequences of a deep borehole not being properly sealed. Such a situation would be prevented by normal quality control. However, the situation is one of a limited number of potential events that could result in an enhanced permeability pathway to surface environment, and is considered as a "what if" scenario, possibly also arising due to unexpected physical, chemical and/or biological processes resulting in poor seal performance.

The key transport pathways for releases from the repository are summarized in Figure 8-41.

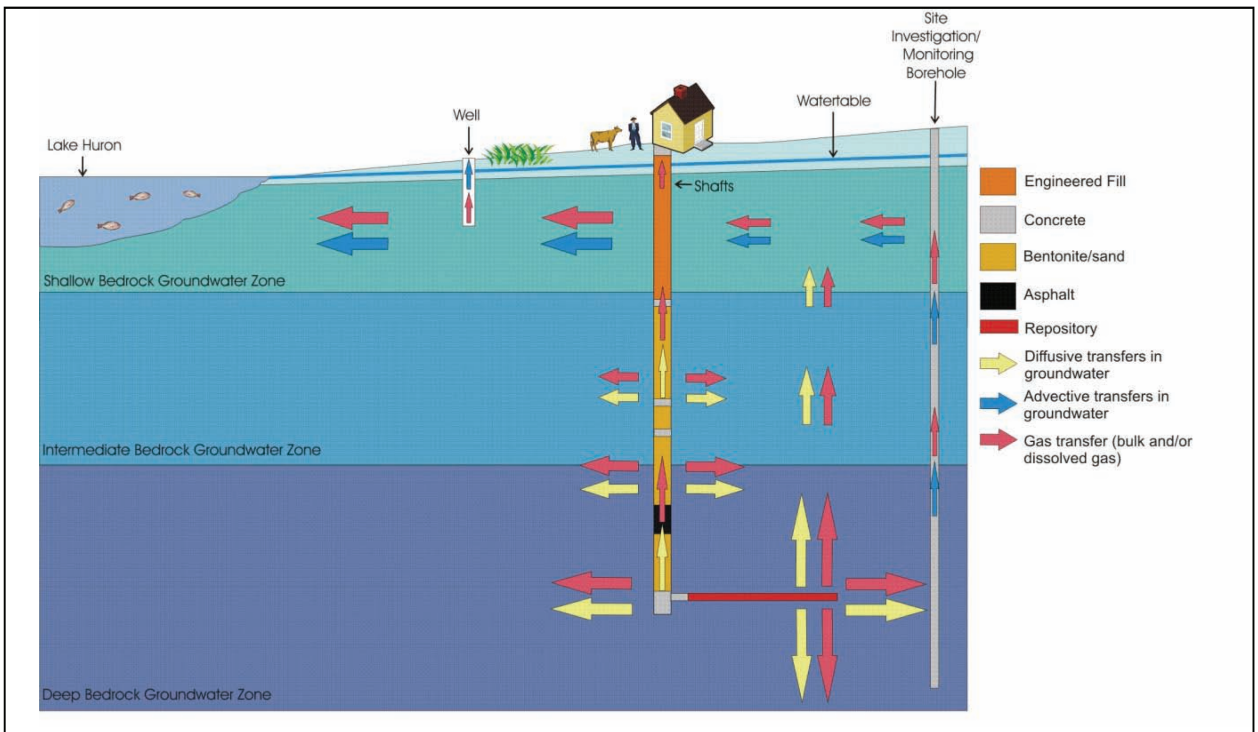

Figure 8-41: Schematic Representation of Potential Transport Pathways for the Poorly Sealed Borehole Scenario

8.7.3.2 Models, Implementation and Data

Conceptual and mathematical models and data have been developed for the scenario – these are described in Chapter 4 of the Human Intrusion and Other Disruptive Scenarios report (NWMO11aq). The key aspects of the conceptual model for releases from the repository are summarized in Box 5. The model is the same as for the Normal Evolution Scenario (Section 8.6.2.1). The only difference is that there is an additional pathway for contaminants to migrate from the repository to shallow groundwater zone - via the poorly sealed borehole. For quantitative estimate of potential impact, the DGR-2 site characterization borehole location is used, at 100 m southeast of Panel 2, as this is the closest borehole (Figure 8-32).

Box 5: Key Aspects of the Conceptual Model for the Poorly Sealed Borehole Scenario

Waste and Repository

- Instantaneous resaturation of the repository, which maximizes the release of contaminants into groundwater that may subsequently migrate via the borehole.

Geosphere and Shafts

- Poorly sealed site investigation/monitoring borehole located 100 m from southeast edge of Panel 2. Borehole extends from surface down to Precambrian.
- Contaminants may migrate along the poorly sealed borehole by advection, and no sorption is assumed to occur.

Biosphere

- Model is the same as the Normal Evolution Scenario.

Note: All other modelling assumptions are the same as for the Normal Evolution Scenario (Table 8-7)

Detailed modelling (Section 6.5.1 of NWMO11an) shows that the presence of the borehole does not perturb the regime in the vicinity of the repository to any notable degree. Horizontal flow rates from the repository towards the borehole are comparable to diffusion rates, and contaminants transported by the borehole have diffused through the rock prior to intercepting the conductive pathway. The conceptual model for contaminant transport therefore only considers a diffusive flux of contaminants from repository to the borehole.

Table 8-16 and Table 8-17 summarize the calculation cases and the key modelling assumptions for the scenario, respectively.

Table 8-16: Calculation Cases for the Poorly Sealed Boreholes Scenario

Case ID	Case Description
BH-BC BH-NR	As NE-RC but with a poorly sealed site characterization/monitoring borehole extending from the surface to the Precambrian and located 100 m to the southeast of Panel 2 (i.e., DGR 2) with a hydraulic conductivity of 10^{-4} m/s and porosity of 0.25. BH-NR considers non-radioactive species.

Table 8-17: Key Modelling Assumptions for the Poorly Sealed Borehole Scenario[a]

Assumption	Motivation/Reason for Assumption	Impact of Assumption
Borehole is not properly sealed	Necessary to investigate the possibility of an enhanced permeability pathway to the surface environment.	Established practice would ensure that the boreholes are sealed so as to prevent any residual flows through them once they were no longer used. The assumption that the borehole is not sealed is therefore very conservative. The borehole is conservatively assumed to have a fill similar to compacted sand. These conditions are necessary to permit a flow to occur upwards to the shallow groundwater zone
Repository resaturates at closure	Maximizes the potential release of contaminants from the repository that could subsequently be captured by the poorly sealed borehole.	Under the likely unsaturated conditions in the repository, there would be no significant release of contaminants in groundwater, and therefore the poorly sealed borehole would not provide a pathway for contaminant migration. Therefore, a conservative assumption is made that the repository rapidly resaturates and contaminants begin to be released in groundwater soon after closure.
No sorption occurs in the borehole	Conservative assumption. No specific seal material is identified.	The assumption that there is no sorption maximizes the rate at which contaminants can migrate through the borehole.
Note: a. The Poorly Sealed Borehole Scenario utilizes the same models for the near-field, geosphere and biosphere as the Normal Evolution Scenario Reference Case, and the key assumptions in Table 8-7 therefore also apply.		

8.7.3.3 Poorly Sealed Borehole Results

Detailed modelling indicates that the borehole has limited influence on the hydraulic conditions at the repository horizon because of the very low permeability host rock around the DGR.

The results also indicate that the flow of water up the poorly sealed borehole is up to 15 m^3/year into the shallow bedrock groundwater zone, which is flowing at a rate of about 60,000 m^3/year over the width of the repository.

The calculations are based on a repository that is resaturated at closure, which maximizes the release of contaminants to groundwater. Figure 8-42 shows the calculated radionuclide transfer flux to the shallow bedrock groundwater zone via the borehole. Calculated concentrations in biosphere media are very small, such that radionuclide concentrations are more than seven orders of magnitude lower than the NECs for non-human biota given in Table 8-1. Concentrations of non-radioactive contaminants are more than three orders of magnitude smaller than the associated environmental quality standards. The calculated dose to an adult site resident is very small, peaking at 4 x 10^{-8} mSv/year after about 900,000 years, much lower than the 1 mSv/year dose criterion[5].

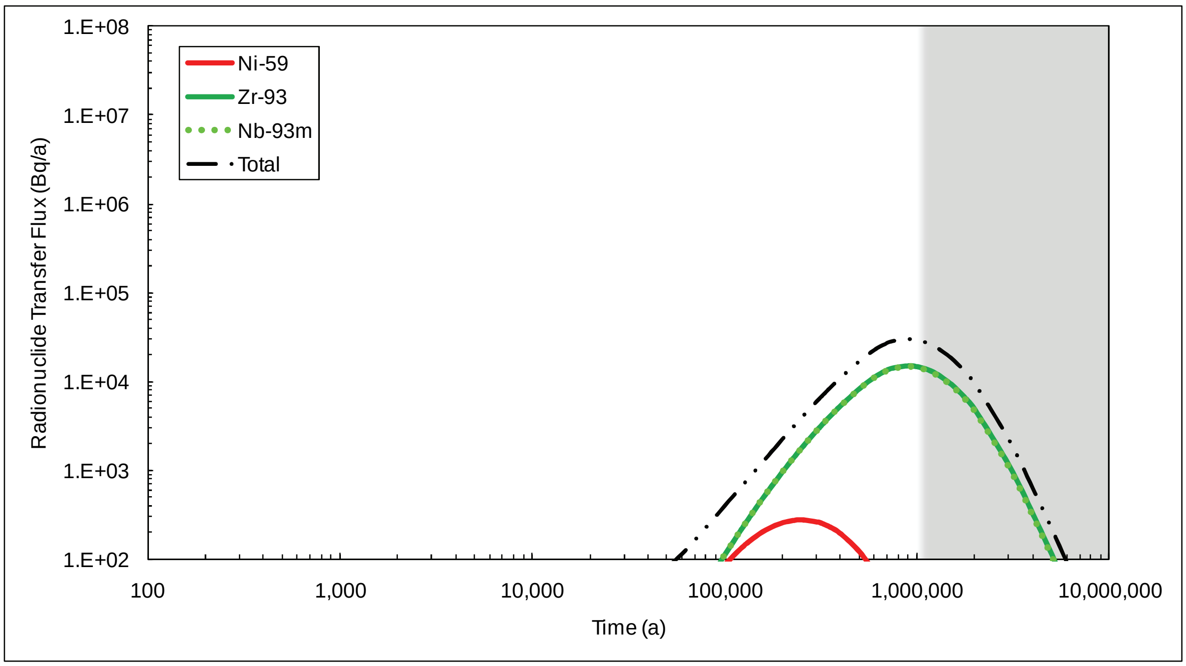

Figure 8-42: Calculated Radionuclide Transfer Flux to the Shallow Groundwater Zone via the Poorly Sealed Borehole

[5] Based on well capture rate for a self-sufficient farm well at 80 m depth in the permeable shallow aquifer. Even if 100% of the contaminant flux through the borehole were to be captured by a small single-family well of about 4x130=520 m^3/year (i.e., no dilution in the Shallow Bedrock Ground Zone), the peak drinking water dose would be about 3x10^{-5} mSv/year.

8.7.4 Vertical Fault Scenario

8.7.4.1 Description

There is strong geological, hydrogeological, and geochemical evidence that transmissive vertical faults/fracture zones do not exist within the footprint or vicinity of the DGR (Chapter 4 of this report, Section 2.3.9 of NWMO11c).

Despite this evidence, the Vertical Fault Scenario considers "what if" there was a transmissive vertical fault, either undetected or representing the displacement of an existing structural discontinuity, in close proximity to the repository. Such a fault could provide a pathway between the geosphere at the level of the repository and the overlying permeable Guelph Formation. Flow in the Guelph is assumed to discharge to the lake near shore by the site.

The fault extends from the Precambrian basement to the Guelph Formation, but not into the shallow groundwater zone, consistent with the regional evidence. Regionally, any such discontinuities are often associated with hydrothermal dolomitized carbonate and are found to originate in the Precambrian or Cambrian and extend up to the Ordovician shales where they terminate (ARMSTRONG10).

The key transport pathways for releases from the repository are shown in Figure 8-43.

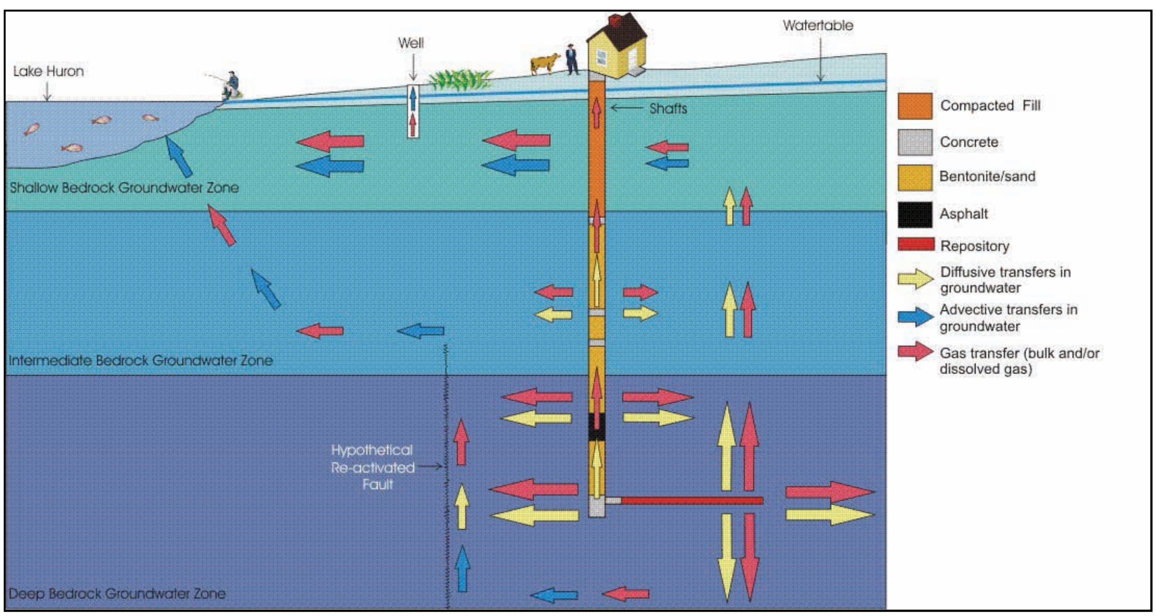

Figure 8-43: Schematic Representation of Potential Transport Pathways for the Vertical Fault Scenario

8.7.4.2 Models, Implementation and Data

Conceptual and mathematical models and data have been developed for the scenario – these are described in Chapter 5 of the Human Intrusion and Other Disruptive Scenarios report (NWMO11aq). The key aspects of the conceptual model for releases from the repository are summarized in Box 6. The only difference from the Normal Evolution Scenario conceptual model (Section 8.6.2.1) is that there is the additional pathway for contaminants to migrate vertically from the repository horizon into the overlying Guelph Formation and then into the lake. The fault is taken to be 500 m to the northwest of the repository - i.e., beyond the area considered in the site investigation program. An alternative fault location within the site characterization area is also considered at 100 m to the southeast of the repository (Figure 8-32). The overpressurized Cambrian is assumed to be unaffected, despite being connected by a permeable path to the lower head and permeable Guelph Formation.

Table 8-18 and Table 8-19 summarize the calculation cases and the key modelling assumptions for the scenario, respectively.

Box 6: Key Aspects of the Conceptual Model for the Vertical Fault Scenario

Waste and Repository

- Repository is assumed to be completely saturated from closure onwards. This is chosen conservatively to maximize the release of contaminants into groundwater that may subsequently migrate via the fault.

Geosphere and Shafts

- Hypothetical vertical fault connects the Precambrian to Guelph Formation.
- The overpressure in the Cambrian sandstone drives groundwater flow through the transmissive fault vertically upwards.
- No sorption of contaminants in the fault.
- Horizontal flow in Guelph Formation leading into lake near shore.

Biosphere

- Model is the same as the Normal Evolution Scenario.
- Considers a self-sufficient family farm located on the repository site and using groundwater from well.
- Also considers a group living on the site shore region where both the shallow groundwater zone and Guelph Formation are assumed to discharge, and that has a high fish diet.

Note: All other modelling assumptions are as for the Normal Evolution Scenario (Table 8-7)

Table 8-18: Calculation Cases for the Vertical Fault Scenario

Case ID	Case Description
VF-BC VF-NR	As NE-RC but with a single 1 m wide, high hydraulic conductivity (10^{-8} m/s) vertical fault located 500 m northwest of the repository, and between the Cambrian and the Guelph Formation. Horizontal flow in Guelph. VF-NR considers non-radioactive species.
VF-AL	As VF-BC, but with vertical fault located 100 m southeast of the repository.

Table 8-19: Key Modelling Assumptions for the Vertical Fault Scenario[a]

Assumption	Motivation/Reason for Assumption	Impact of Assumption
Vertical fault is present close to the repository	A vertical fault could provide a pathway connecting the repository horizon to overlying permeable rocks.	There is no evidence of the presence of transmissive vertical faults in the vicinity of the repository, although the presence of such a fault would likely have been noticed in the site characterization program. The impact is assessed at two locations; one just outside the detailed site characterization area, and one within the area.
Fault extends from Precambrian to Guelph formation	Consistent with regional fault structures, in which faults originate in basement structures and extend into the Ordovician sediments.	This provides a pathway to bypass the low permeability deep geosphere. Faults extending to surface are not credible since existing faults would likely have been visible, and creation of new faults to surface would require huge energy releases.
Repository resaturates at closure	Maximizes the potential release of contaminants from the repository that could subsequently move via groundwater to the vertical fault.	Assuming rapid resaturation of the repository maximizes the flux of contaminants through the hypothetical vertical fault.
No sorption occurs in the vertical fault	Minerals and other properties within the fault are not known.	The assumption that there is no sorption maximizes the rate at which contaminants can migrate through the fault.
Groundwater flow in Guelph and Salina A1 upper carbonate discharges to the lake near-shore over ~ 1 km pathlength	Discharge location is not known for certain.	This is a conservative assumption since the site evidence suggests that any flow from the Guelph Formation and Salina A1 Unit upper carbonate into the biosphere is likely to be over a significantly longer pathlength than approx. 1 km. Discharge into the near shore minimizes dilution of any contaminants released into the lake.

Note:
a. The Vertical Fault Scenario utilizes the same models for the near-field, geosphere and biosphere as the Normal Evolution Scenario Reference Case, and the key assumptions in Table 8-7 therefore also apply.

8.7.4.3 Vertical Fault Results

Two fault locations are considered, one at 500 m to the northwest of the repository (VF-BC) and an alternative case where the fault is located 100 m to the southeast of the repository (VF-AL).

The detailed groundwater modelling shows that the VF-BC case only has a minor impact on the hydraulic conditions in the repository. Since any vertical fault would connect to the pressurized Cambrian, a pressure gradient develops which directs groundwater movement away from the fault (Figure 8-44). Contaminants in the repository need to diffuse either directly to the fault (against the hydraulic gradient) or downwards to the Cambrian and then via groundwater flow to the fault, before they can be transported by groundwater advection up the fault to the Guelph Formation. The results indicate that the resulting radionuclide transfer flux to the Guelph peaks at about 3 MBq/year after more than one million years (as shown in Figure 8-45).

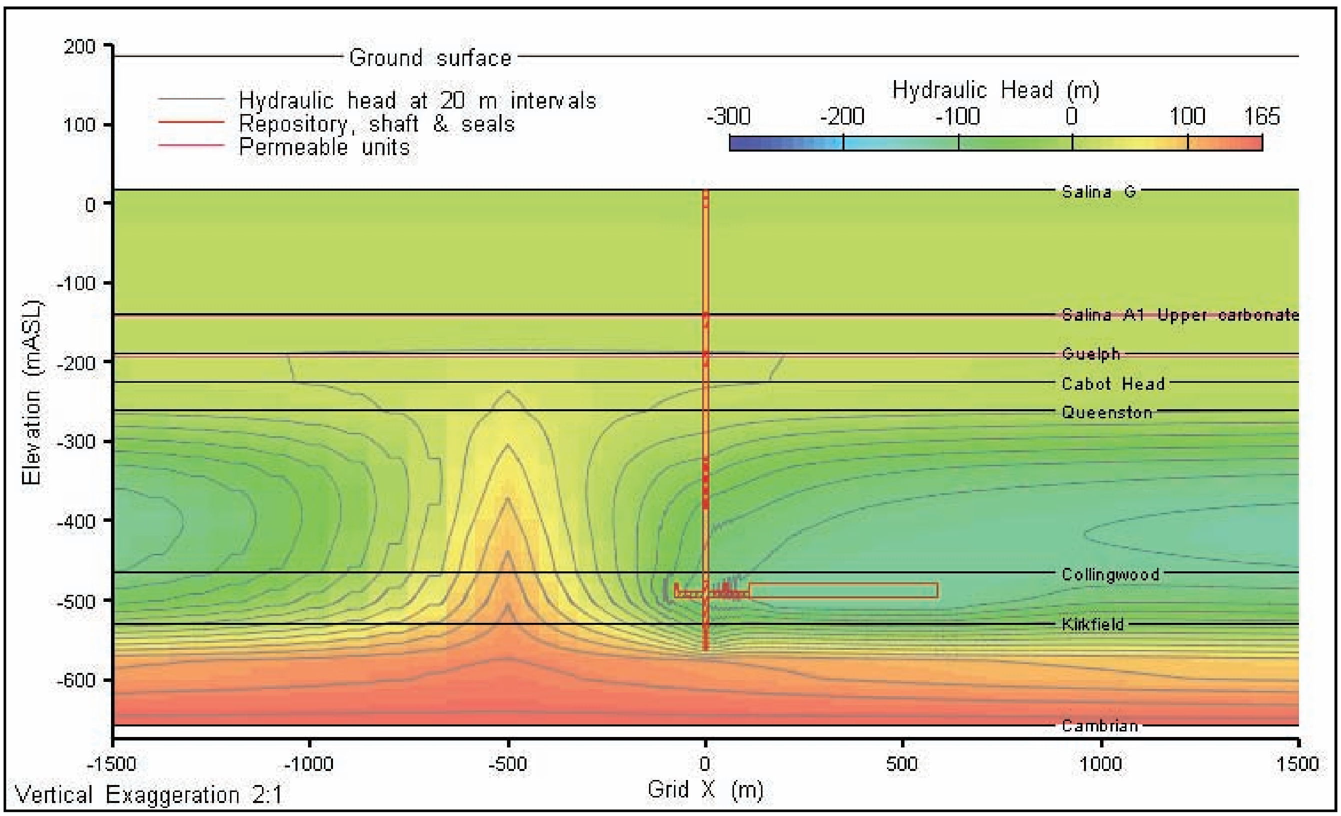

Figure 8-44: Hydraulic Heads for the VF-BC Case

Horizontal groundwater flow in the Guelph is assumed to discharge to the near-shore of the lake. The resulting dispersion means that calculated concentrations are at least seven orders of magnitude smaller than the NECs for non-human biota given in Table 8-1. Concentrations of non-radioactive contaminants remain at least four orders of magnitude below the associated environmental quality standards given in Table 8-2.

Calculated doses for the VF-BC case are similarly very small; the peak calculated dose to the maximally exposed group (the site shore resident) is 3 x 10^{-10} mSv/year after more than a million years, much smaller than the dose criterion.

Diffusion of contaminants over the entire repository footprint down to the Cambrian dominates over diffusion from the side of the DGR as a transport pathway to the fault. Therefore the closer proximity of the fault to the DGR for the variant fault location case (VF-AL) has relatively little impact on the calculated contaminant fluxes via the fault and the peak calculated dose to the maximally exposed group (the site shore resident) is the same, at 5 x 10^{-10} mSv/year[6].

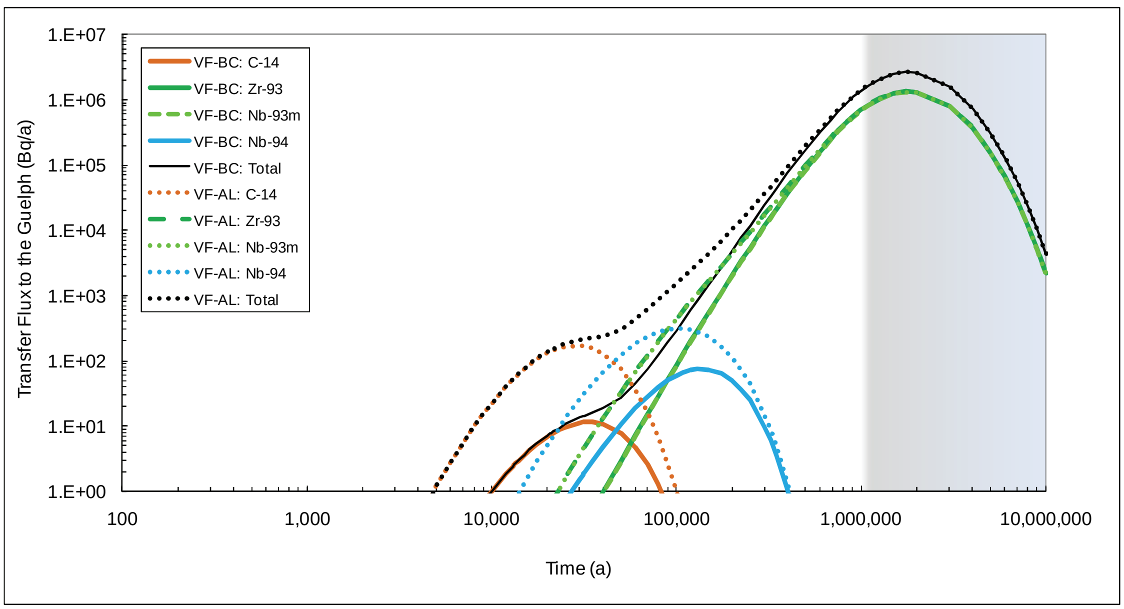

Figure 8-45: Calculated Fluxes of Contaminants in Groundwater from the Fault to the Guelph for a Vertical Fault Located 500 m (VF-BC) and 100 m (VF-AL) from the DGR

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[6] The peak concentration in the water entering the Guelph from the fault is about 500 Bq/L. Consumption of water at this concentration would result in a dose of around 0.3 mSv/year if it were assumed that water was pumped directly from the Guelph Formation without any treatment. Note also that the TDS content of Guelph water is around 375 g/L, a factor of 13 times higher than seawater.

8.8 Assessment Uncertainties

As noted in Section 8.6.2.7, uncertainties can be considered in three categories: scenario, model and data. The results from the reference/base calculation cases and the variant calculation cases provide information to assess the importance of the various sources of uncertainty. The results are summarized below; a more detailed analysis is provided in the supporting modelling reports (NWMO11ak, NWMO11aq, NWMO11aj and NWMO11an).

8.8.1 Scenario Uncertainty

The uncertainty in the future evolution of the site and repository is tested with the Normal Evolution Scenario and the four Disruptive Scenarios. Results for the reference/base cases for the Normal Evolution and Disruptive Scenarios are summarized in Table 8-20. Very low contaminant release to the shallow groundwater zone and negligible annual dose are calculated for the Normal Evolution Scenario (orders of magnitude below the dose criterion of 0.3 mSv/year).

For the Disruptive Scenarios, the calculated doses for the Human intrusion and Severe Shaft Seal Failure cases are at or just below the dose criterion of 1 mSv/year for times up to about 30 ka. However, when the low likelihood of such scenarios is taken into account, the risk benchmark of 10^{-5} health effects per year is not exceeded. The maximum calculated doses for the Poorly Sealed Borehole and Vertical Fault Scenarios remain well below the dose criterion.

Table 8-20: Calculated Maximum Doses and Fluxes for the Assessed Scenario

Scenario	Maximum Dose to an Adult (mSv/year)	Maximum Radionuclide Flux into Shallow Groundwater System	
		Groundwater (Bq/year)	Free Gas (Bq/year)
Normal Evolution			
Reference Case	2×10^{-15} [a]	3×10^{-6} [a]	0
Simplified Base Case	1×10^{-13} [a]	2×10^{-3}	0
Human Intrusion Base Case	1	n/a [b]	n/a [b]
Severe Shaft Seal Failure Base Case	1	5	2×10^{10}
Poorly Sealed Borehole Base Case	4×10^{-8}	3×10^{4}	n/a
Vertical Fault Base Case	5×10^{-10} [a]	n/a [c]	n/a [c]

Notes:
a. Occurs at the end of the calculation period (10 Ma).
b. Release only to surface in Human Intrusion base case.
c. Releases are intercepted by Guelph and discharged into lake, bypassing shallow system.

"What-if" calculations indicate that doses of tens of milliSieverts would require either:

1. That the intrusion borehole is continued on past the repository and down into the Cambrian formation, and that the borehole is not appropriately sealed, allowing for long-term flow of water from the Cambrian through the repository and then to the shallow groundwater system.

Or

2. That the entire shaft seal system (500 m of low-permeable material) would have to degrade to an effective conductivity of around 10^{-7} m/s, roughly equivalent to very fine sand and silt.

In both cases, the doses would apply to someone living directly on the repository site; impacts further afield (i.e., off the Bruce nuclear site) would be much lower.

In disruptive scenarios with potential for notable dose impacts, C-14 is the important radionuclide, along with Nb-94 for human intrusion. Therefore, these scenarios become unimportant on timescales of 60,000 years due to decay of C-14 (and Nb-94). This is also the earliest likely time for the onset of the next glacial cycle and advance of an ice-sheet at the site. Therefore, future glaciations are unlikely to have significant effects on impacts from these disruptive scenarios.

8.8.2 Model and Data Uncertainty

Model and data uncertainties associated with the Normal Evolution Scenario are addressed through the evaluation of a set of calculation cases that are designed to bound the effects of these uncertainties with the Normal Evolution Scenario Reference Case (**NE-RC**). These cases are summarized in Figure 8-15 and Table 8-5. The cases are discussed with respect to the following uncertainties:

- Repository resaturation;
- Waste Inventory;
- Contaminant release rate;
- Gas generation;
- Geosphere gas properties;
- Geosphere transport properties;
- Shaft seal performance;
- Geosphere over- and underpressures;
- Geosphere horizontal flow;
- Human receptors; and
- Glaciation.

8.8.2.1 Repository Resaturation

A significant feature of the Reference Case is that the repository is expected to remain mostly unsaturated for a very long time. This is important because it increases the volume available for gas and minimizes the potential for radionuclides to be released into groundwater and to migrate from the repository.

Figure 8-46 shows an overlay of the calculated saturation levels within the repository from all detailed gas modelling cases (Section 8.1 of NWMO11aj). The model included water seepage from the rock and from the shafts. In these particular "non-water-limited" cases, the consumption of water by reactions within the repository was not considered. (The water saturation levels are even lower if water consumption by these reactions is included; see Section 6.1 and Chapter 7 of NWMO11aj).

The results show that the repository is less than half saturated for all cases except those where the shaft is highly permeable and is able to supply water to the DGR level (SF-ED) or where there is no gas generation within the DGR (NE-NG).

In the limiting case where the repository is assumed to instantly resaturate at closure (NE-RS), the calculated doses increase by about a factor of 20. However, since the dose remains orders of magnitude below the dose criterion, the safety of the repository system is not sensitive to repository resaturation.

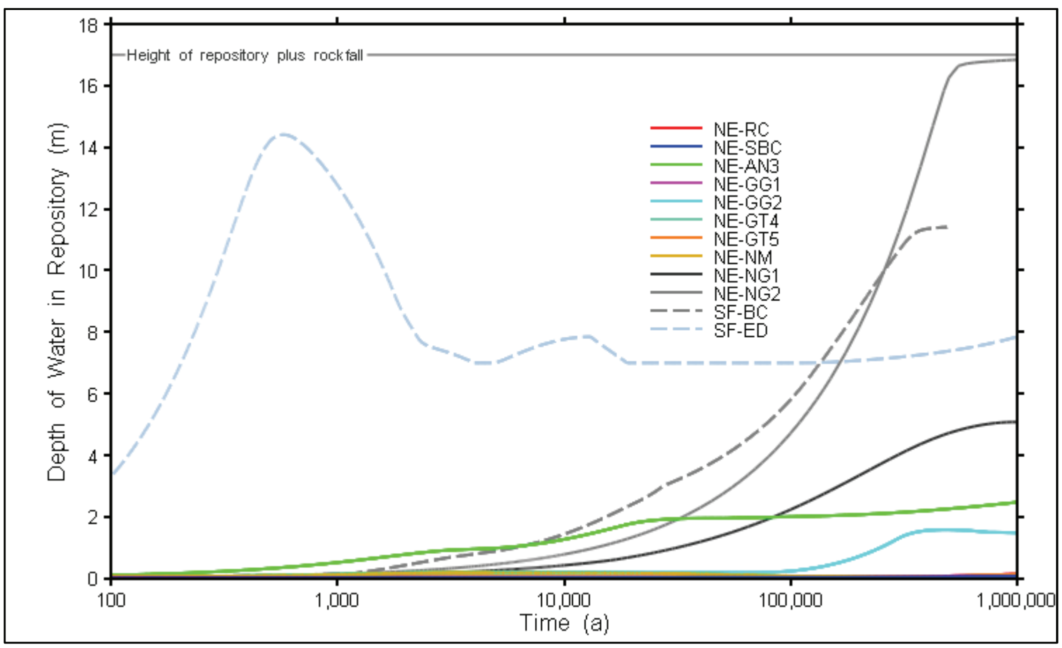


Figure 8-46: Depth of Water in Repository (Non-Water-Limited Cases)

8.8.2.2 Waste Inventory

The potential effect of uncertainties surrounding contaminant inventories in the wastes has been explored through a variant case in which the initial inventory is increased by an order of magnitude (NE-IV). The results indicate a linear response in the maximum calculated dose. However, since the peak dose results are orders of magnitude below the criterion, the safety of the repository is not sensitive to the inventory uncertainties.

8.8.2.3 Contaminant Release Rates

Contaminant release is conservatively represented as instant release on contact with repository water for most waste categories, with congruent release being used for wastes where contamination is bound within the waste itself. However, the actual release is dependent upon water entering the repository. (Tritium and C-14 are also released as gases.)

The effect of repository resaturation as a factor in controlling release is tested in a case with repository resaturation at closure (NE-RS), as well as cases assuming instant and complete release of the inventory to the repository water at closure together with no retardation (NE-RT1, RT2). The results are compared in Table 8-21 and Figure 8-47.

These results (see Table 8-21) show that when the contaminant release rates are maximized through resaturation of the repository at closure (NE-RS), the maximum calculated dose increases by a factor of 20. When the release models are also bypassed, with instant release to groundwater in a saturated repository (NE-RT1), the maximum calculated dose increases by more than six orders of magnitude (although this increase is also affected by the absence of sorption in the shafts and geosphere). This is due to the slow release of some important radionuclides bound within the Zircaloy metal. However, while relevant to the safety of the system, the maximum calculated doses remain well below the dose criterion, and overall safety is therefore not sensitive to realistic uncertainties in these processes.

Uncertainty concerning repository chemistry is treated through adopting conservative assumptions relating to the release of contaminants from the waste and its subsequent release from the repository. These include assuming no solubility limitation (except for C-14 releases) and no sorption on materials within the repository. While siderite formation is included as a (minor) process for precipitating carbon; other precipitation processes, such as calcite formation, are conservatively not represented.

Table 8-21: Maximum Doses to an Adult for Different Contaminant Release Assumptions

Case	Brief Description	Max. Calculated Dose (mSv/year)	Time of Max. Calculated Dose (Ma)
NE-RC NE-RS NE-RT1	Reference case (with underpressures) Resaturation at closure (with underpressures) Resaturation at closure, instant release to groundwater, no sorption (with underpressures)	2×10^{-15} 4×10^{-14} 4×10^{-9}	10 [a] 10 [a] 10 [a]
NE-RT2	Resaturation at closure, instant release to groundwater, no sorption (without underpressures)	5×10^{-9}	10 [a]

Note:
a. This represents the end of the calculation period.

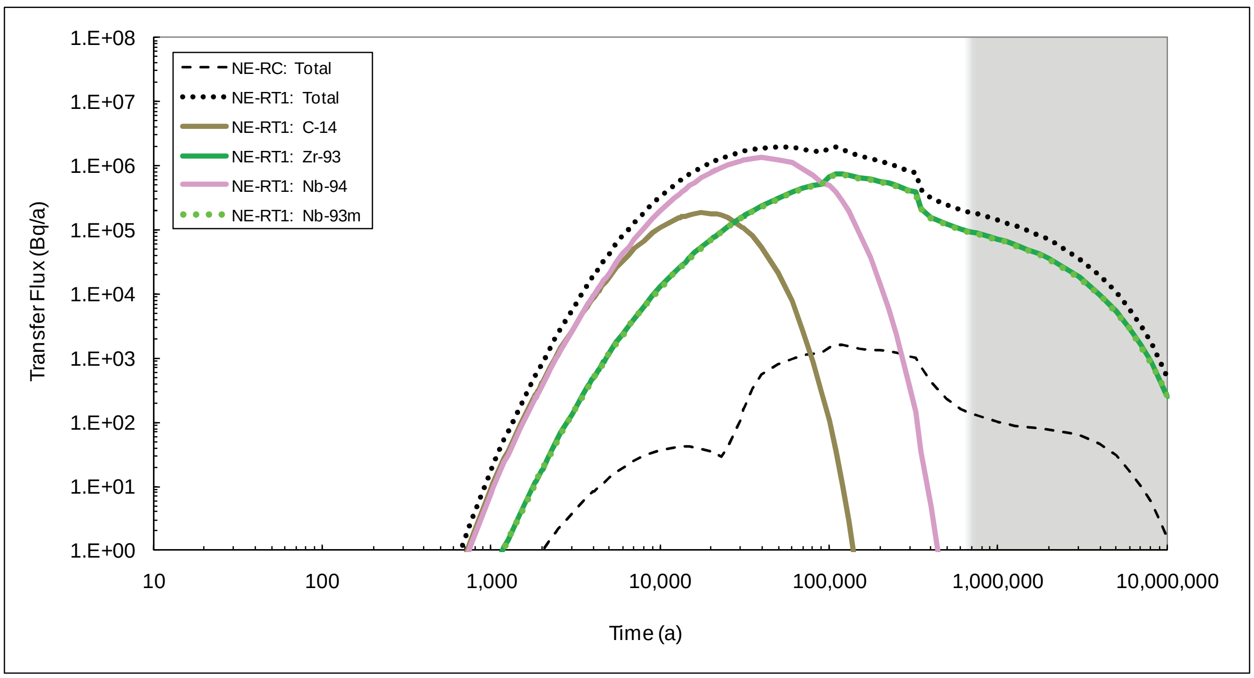


Figure 8-47: Radionuclide Flux to the Base of the Shafts with Instant Resaturation and Release and No Sorption (NE-RT1) Compared to Reference Case (NE-RC)

8.8.2.4 Gas Generation

The GGM within the repository draws on a number of assumptions about the corrosion behaviour of materials in the repository and the extent of microbial activity. The model is intended to maximize the amount of gas generation by assuming that corrosion processes and microbes are active, and by assuming that the organics are fully degraded into CO_2 and CH_4. It is possible however that conditions will be sufficiently dry or saline that there will be little corrosion or microbial activity. The effect of alternative assumptions for gas generation was, therefore, tested through several cases:

- NE-GG1 – Increased amount of metal and increased gas generation rates from corrosion and microbial reactions;
- NE-GG2 – Decreased organic degradation rates;
- NE-NM – No methanogenic reactions (e.g., $H_2 + CO_2 > CH_4$); and
- NE-NG1/NG2 - No gas generation.

The NE-GG1 case includes an increased inventory of metal in the DGR (e.g., reflecting a greater degree of packaging/overpacking), together with increased metal corrosion and organic degradation rates. The case results in increased gas generation which results in an earlier gas pressure peak, and the repository remaining almost completely unsaturated due to the gas pressures.

With the NE-GG2 case, the repository remains relatively unsaturated, and the peak pressure is similar to the NE-GG1 case but occurs later. Note that this case results in a different mix of H_2, CO_2 and CH_4 within the repository, and therefore affects the extent of methanogenesis, and the pressure evolution. Specifically the gas contains more H_2 and the peak pressure is similar to the high-gas-generation NE-GG1 case.

The NE-NM case assumes that methane generating microbes are not active; this includes organic degradation related reactions as well as gas phase reactions. The primary gas in the repository is therefore H_2 from metal corrosion. This results in a higher gas pressure within the repository.

At the other limit, the bounding case of zero gas generation was also evaluated (NE-NG).

The maximum calculated dose to a site resident is very low in all these cases, as shown in Table 8-22.

Table 8-22: Maximum Gas Pressures and Doses for Different Gas Generation Rates

Case	Brief Description	Peak Pressure (MPa)[a]	Max. Calculated Dose (mSv/year)
NE-RC	Reference case	8.2	2×10^{-15}
NE-SBC	Simplified base case	7.2	1×10^{-13}
NE-GG1	Increased gas generation rates	7.8	9×10^{-11}
NE-GG2	Decreased organic degradation rates	7.8	9×10^{-14}
NE-NM	No methanogenic reaction	9.2	5×10^{-14}

Figure 8-48 summarizes the repository pressures calculated for the above case plus other cases. The overall conclusion is that the gas pressure within the repository tends towards about 7-9 MPa, a range roughly corresponding to the natural hydrostatic pressure and the steady-state pressure due to the Cambrian overpressure. The balance reflects the tendency of the system to push gas into the rock and shaft at higher pressure, or for water and gas to seep into the repository at lower pressure.

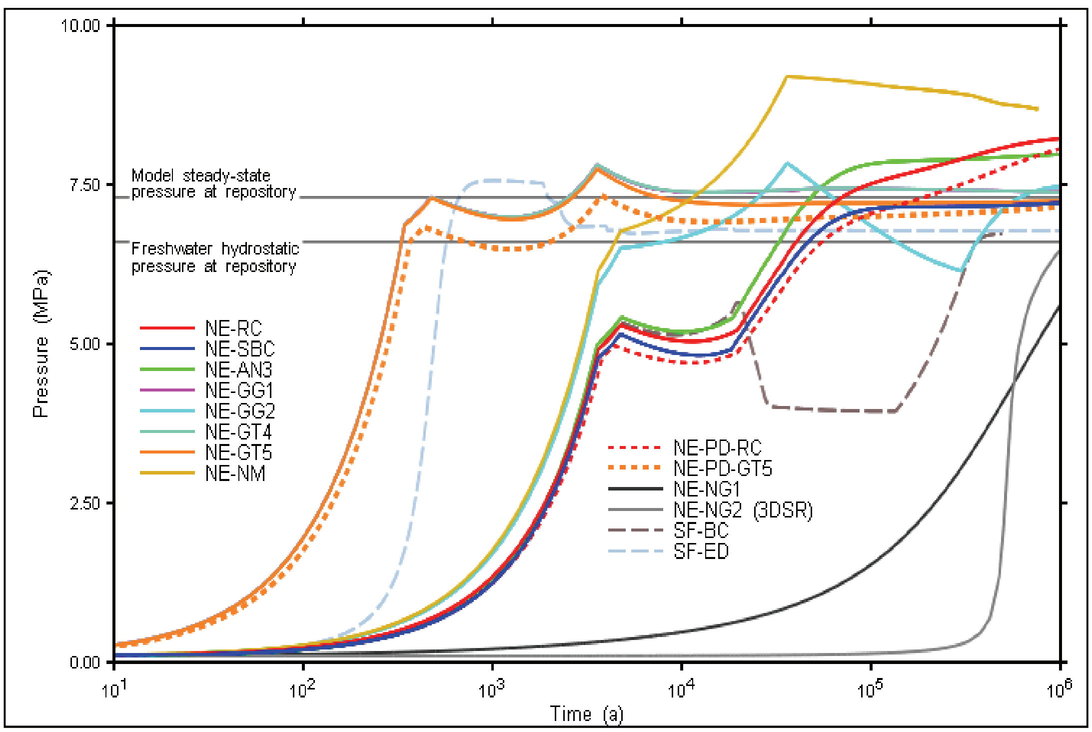


Figure 8-48: Calculated Pressure Profile in Repository for Various Cases (Non-Water-Limited Cases)

8.8.2.5 Geosphere Gas Properties

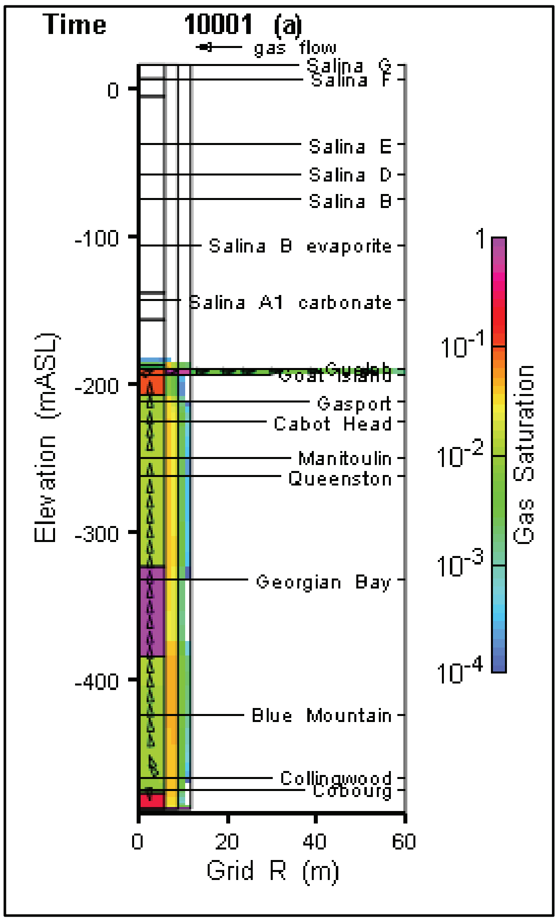
The rate of gas transport through the geosphere and shaft is dependent upon the gas pressure in the repository, as well as the initial gas saturation conditions and gas permeability properties of the shaft seals and host rock.

Various detailed gas modelling cases represent the Normal Evolution Scenario with different assumptions relating to initial gas saturations in the rock formations (NE-RC1 and NE-RC2). The results are also sensitive to assumptions on residual gas saturation within the host rock as this affects the gas permeability at low saturations.

The Reference Case uses a representative gas capillary pressure curve for most of the lower permeability intact rock units, rather than formation specific curves. The alternative cases NE-GT1 and NE-GT2 investigated the impact of bounding capillary pressure curves, while NE-GT3 used an alternative relative permeability curve.

Results for all these cases showed virtually no sensitivity to these gas-related parameters; in all cases there is essentially no transport of a separate gas phase in the rock when the rock is initially liquid saturated. The NE-RC2 case used formation specific two-phase flow parameters. These did not appreciably impact repository pressures, but did induce a higher level of transient behaviour in the geosphere.

The Guelph Formation and Salina A1 Unit upper carbonate are relatively porous and permeable. Detailed gas modelling indicates that if gas reaches these formations, they can divert some of the gases. This is illustrated in the NE-GG1 and NE-NM cases in which free gas was able to migrate from the DGR into the shafts (Figure 8-49). In these cases, the gas is captured by the permeable Guelph and does not extend beyond the Salina A2.



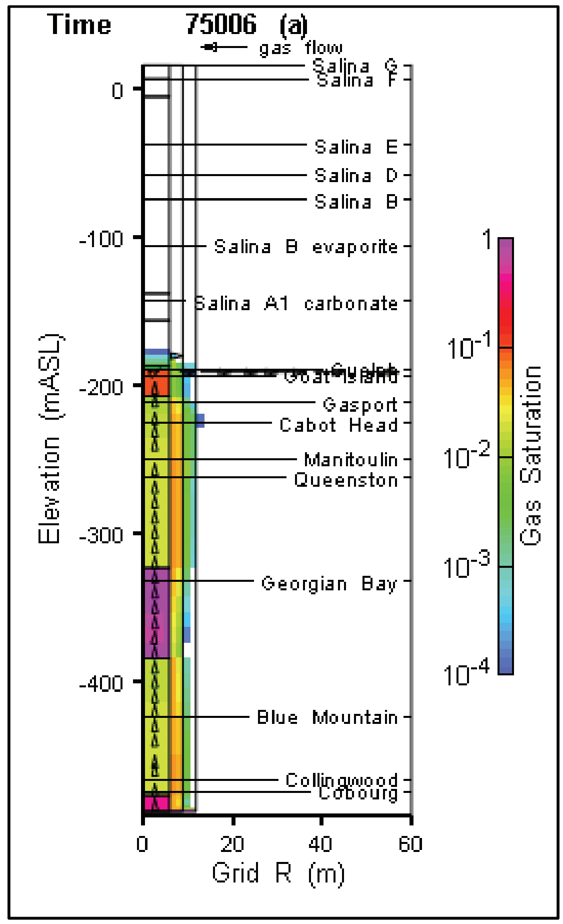


Figure 8-49: Gas Saturations and Flows for the NE-GG1 Case (Left) and the NE-NM Case (Right) Showing Diversion of Gas into the Porous Guelph Formation

8.8.2.6 Geosphere Transport Properties

The horizontal hydraulic conductivity of the rock is well established from site characterization. However, the vertical hydraulic conductivities have not been directly measured, but have been inferred from modelling and other factors as described in the Geosynthesis (NWMO11c). They are generally estimated to be about ten times less than the horizontal values. In the case NE-AN1, the horizontal:vertical anisotropy was reduced typically by a factor of five (i.e., increased vertical hydraulic conductivity). This had little impact on the transport results (compare Figure 8-50 and Figure 8-28) because diffusion is the dominant mechanism for mass transport from the repository.

Increasing the horizontal effective diffusion coefficients for the host rock (case NE-AN2) increases the spread of contamination at repository depth and results in less contaminants travelling up the shafts.

The Reference Case adopts conservative values for the sorption of contaminants within the host rock. The NE-RT1 and NE-RT2 cases entirely exclude sorption in the shafts and geosphere (and also assume instant resaturation and contaminant release from packages), resulting in an increase in the maximum calculated dose by more than four orders of magnitude. However, the dose remains well below the criterion.

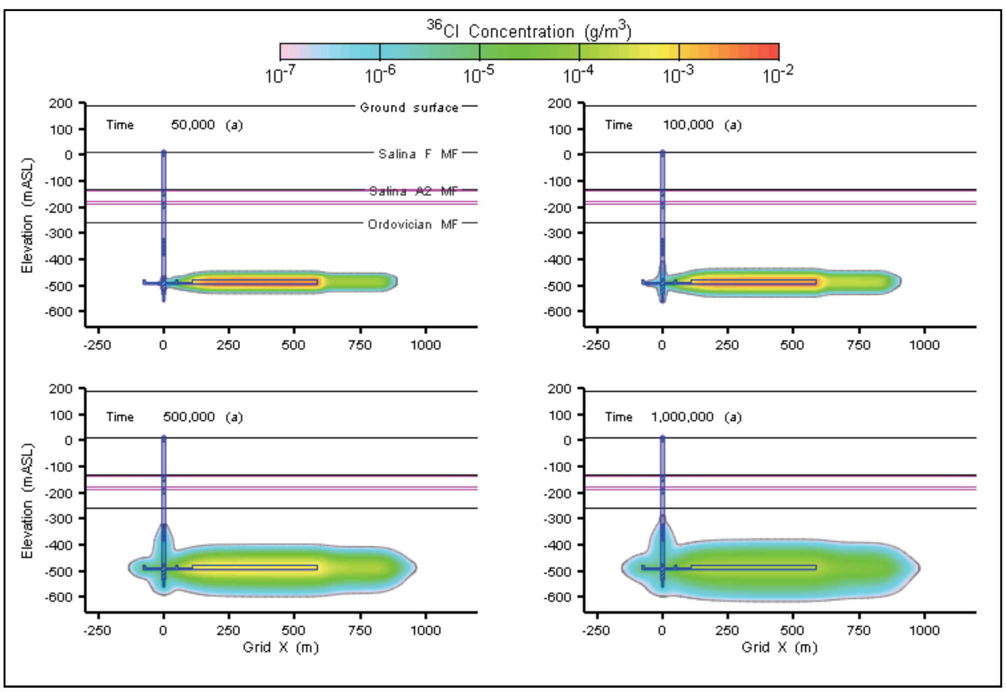


Figure 8-50: Transport of Cl-36 in the Geosphere for Instant Saturation and Release, and 10x higher Vertical Hydraulic Conductivity (NE-AN1)

8.8.2.7 Shaft Seal Performance

Although the shaft seal system is designed to have a low-permeability, its permeability is not as low as that of the host rock and therefore the shafts are the main pathway for any contaminant releases from the repository. Uncertainties in the properties of the seals and damaged rock zone around the shafts are therefore potentially important.

The reference case considers degraded concrete from the start, as well as a thick damaged rock zone. The uncertainties in the properties or degradation in the properties of the shaft seal were explored as follows:

- More permeable damaged zone in the rock around the shafts (NE-EDZ1);
- No asphalt layer (NE-GT4);
- More permeable (degraded) seal (NE-GT5); and
- Shaft Seal Failure Scenario (SF-BC and SF-ED).

The NE cases consider parameter uncertainties or variation within the design basis. The SF cases consider extreme parameter values well beyond the design basis.

Table 8-23 summarizes the hydraulic conductivities used in the Reference Case and in the various alternative NE and SF cases, for the main geosphere formations, the shaft damaged zone, and the shaft seal materials. This shows the range of degradation considered in the assessment. These cases are discussed below.

There is uncertainty in the extent and properties associated with damage to the host rock resulting from the excavations. The values adopted in the Reference Case reflect geomechanical modelling as well as relevant experience from other underground projects in sedimentary rocks (Section 6.4 of NWMO11c). In particular, the extent of the EDZ was based on the maximum extent calculated at any shaft position, and assumed to apply uniformly across the entire shaft column. It was divided into two regions to reflect the variation in hydraulic conductivity, with the inner EDZ assigned 100 times the host rock's vertical hydraulic conductivity and the outer EDZ assigned 10 times that of the host rock permeability.

This uncertainty will be further addressed through DGR site-specific information obtained during and after DGR construction. However, for this postclosure assessment, a variant case considers the potential effect of more severe damage to the host rock surrounding the shafts (NE-EDZ1). In this case, the inner EDZ is 10,000 times more permeable than the host rock, and the outer EDZ is 100 times. The variant results in an increase in the maximum calculated dose by about two orders of magnitude, but this remains well below the dose criterion.

Table 8-23: Vertical Hydraulic Conductivity (m/s) in Shaft Seal and Rock for Various Cases

Material	Base Value	Reference NE-RC / NE-SBC	Increased EDZ NE-EDZ1	Degraded Seal NE-GT5	Shaft Seal Failure SF-BC	Shaft Seal Failure SF-ED
Engineered fill (top 180 m)	10^{-4}	10^{-4}	10^{-4}	10^{-4}	10^{-4}	10^{-4}
Shallow aquifer zone	10^{-5} - 10^{-7}	10^{-5} - 10^{-7}	10^{-5} - 10^{-7}	10^{-5} - 10^{-7}	10^{-5} - 10^{-7}	10^{-5} - 10^{-7}
Guelph / Salina A1	10^{-7} - 10^{-8}	10^{-7} - 10^{-8}	10^{-7} - 10^{-8}	10^{-7} - 10^{-8}	10^{-7} - 10^{-8}	10^{-7} - 10^{-8}
Concrete monolith and shaft bulkheads	$2x10^{-12}$ fresh concrete	10^{-10} degraded	10^{-10}	10^{-10}	10^{-9}	10^{-7}
Bentonite / sand	10^{-12} freshwater	10^{-11} brine	10^{-11}	10^{-10}	10^{-9}	10^{-7}
Asphalt seal	10^{-12}	10^{-12}	10^{-12}			
Inner EDZ Silurian rocks	10^{-11} - 10^{-12}	10^{-11} - 10^{-12}	10^{-9} - 10^{-11}	10^{-11} - 10^{-12}	10^{-9} - 10^{-11}	10^{-9} - 10^{-11}
Inner EDZ Ordovician rocks	10^{-12} - 10^{-13}	10^{-12} - 10^{-13}	10^{-10} - 10^{-11}	10^{-12} - 10^{-13}	10^{-10} - 10^{-11}	10^{-10} - 10^{-11}
Silurian rocks	10^{-13} - 10^{-14}	10^{-13} - 10^{-14}	10^{-13} - 10^{-14}	10^{-13} - 10^{-14}	10^{-13} - 10^{-14}	10^{-13} - 10^{-14}
Ordovician rocks	10^{-14} - 10^{-15}	10^{-14} - 10^{-15}	10^{-14} - 10^{-15}	10^{-14} - 10^{-15}	10^{-14} - 10^{-15}	10^{-14} - 10^{-15}

Note: Shading indicates values in column changed.

Other cases assumed that the shaft seals were more permeable. Two cases were considered, both conservatively based on the NE-GG1 case with increased gas generation: one in which the asphalt layer was replaced with bentonite/sand (which has ten times higher permeability) (NE-GT4), and one in which the entire bentonite/sand column had degraded (NE-GT5). The latter assumed no asphalt seal, 10 times more permeable bentonite/sand seals, and two times lower gas entry properties for the bentonite/sand seal.

The results showed that in these cases, free gas could travel up the shafts, but was captured by the permeable Guelph Formation and Salina A1 Unit upper carbonate and did not travel beyond the Salina A2 Unit. That is, there was no free gas pathway to the shallow groundwater zone. However, gas reaching these formations from the repository would contain C-14. If the C-14 were dissolved in the groundwater at these formations, and then moved with the groundwater, the maximum calculated dose is significantly higher than in the Reference Case, but still more than five orders of magnitude below the dose criterion.

Uncertainty surrounding the performance of shaft seals is bounded by the unlikely Severe Shaft Seal Failure Scenario (see Section 8.7.2). In this case, with the shaft seals uniformly degraded to 10^{-9} m/s hydraulic conductivity, and the damaged zone also more permeable, the peak calculated doses reach 1 mSv/year due to the C-14 carried with free gases, which breaks through to the shallow groundwater zone and surface. If all the shaft seals are degraded to 10^{-7} m/s, then the potential dose impacts are tens of milliSieverts to someone living directly on the repository site.

8.8.2.8 Overpressures and Underpressures

Site characterization work has identified that the Cambrian sandstones are overpressured, while the Ordovician sediments are underpressured. There are several possible origins of these over/underpressures, and the likely cause(s), as well as their evolution are currently being investigated (Section 4.4.4.6).

The Reference Case includes the observed pattern of overpressure and underpressure. However, the Simplified Base Case assumes that the underpressures quickly dissipate after closure, whereas the high pressure in the Cambrian remains steady over the timescales of interest, resulting in a steady vertical upwards hydraulic head gradient. This is a conservative assumption, since mass flow from the repository will be significantly reduced as long as underpressures persist in the Ordovician units, as prevailing liquid gradients will be towards the underpressures, including the gradients within the shafts.

The maximum doses calculated for the Reference Case and Simplified Base Case are compared in Table 8-24, which shows that excluding the underpressures within Ordovician formations results in an increase by about a factor of 50, confirming that this assumption used in all SBC-based cases is conservative.

Another direct comparison is provided by the NE-RT1 and NE-RT2 cases, with instant resaturation and contaminant release and no sorption. NE-RT1 was based on the Reference Case geosphere, while NE-RT2 was based on the Simplified Base Case geosphere. As shown in Table 8-24, the SBC-based case has a higher peak dose, however it is only a factor of 1.2 higher in this case.

The calculated doses remain many orders of magnitude below the dose criterion of 0.3 mSv/year, irrespective of the overpressure/underpressure assumption. While the underpressures are favourable to repository performance, the overall safety of the repository is not highly sensitive to this factor due to the overall low permeability of the host rock and shafts.

Table 8-24: Summary of Maximum Doses to an Adult for Different Vertical Head Gradient Assumptions

Case	Brief Description	With Underpressure	Max. Calculated Dose (mSv/year)	Time of Max. Calculated Dose (Ma)
NE-RC	Reference case	Yes	2×10^{-15}	10 [a]
NE-SBC	Simplified base case	No	1×10^{-13}	10 [a]
NE-RT1	Instant Resaturation, RC	Yes	4×10^{-9}	10 [a]
NE-RT2	Instant Resaturation, SBC	No	5×10^{-9}	10 [a]

Note:
a. This represents the end of the calculation period

8.8.2.9 Geosphere Horizontal Flow

The potential for horizontal groundwater flow in the permeable Guelph Formation and Salina A1 Unit upper carbonate means any contaminants reaching these formations could be diverted away from the direct vertical pathway towards the shallow groundwater zone. Due to uncertainty about the future evolution of the gradients in these formations, flow in these formations is ignored in the Reference Case and Simplified Base Cases, so that transport is preferentially vertical.

A variant case (NE-HG) considers groundwater flow in the Guelph Formation and Salina A1 Unit upper carbonate, with both formations assumed to discharge a relatively short distance from the DGR (1.25 km) to the lake. The results (Table 8-25) demonstrate the conservative nature of discounting groundwater flow in these formations, through a reduction in the maximum calculated dose by more than two orders of magnitude for the NE-HG case.

Table 8-25: Maximum Doses to an Adult for Different Assumptions Groundwater Flow in the Intermediate Groundwater Zone

Case	Brief Description	Max. Calculated Dose (mSv/year)	Time of Max. Calculated Dose (Ma)
NE-SBC	Simplified base case (excluding underpressures), no horizontal flow	1×10^{-13}	10 [a]
NE-HG	Including horizontal groundwater flow in the Guelph and Salina A1 upper carbonate	5×10^{-16}	10 [a]

Note:
a. This represents the end of the calculation period.

8.8.2.10 Potential Human Receptors

The "site resident" critical group considered in the Reference Case and Simplified Base Case is defined on a conservative basis with the aim of maximizing potential exposures. For example, the family is assumed to drill a groundwater well into a contaminant plume in the shallow groundwater zone and maximize use of local resources through a self-sufficient farming lifestyle. The habits of the family are defined on a conservative basis, e.g., 95th percentile food consumption rates are used (CSA08b).

A variant case (NE-CG) considers potential exposures to alternative receptors, who maximize their use of the lake with a high fish diet. This case also includes flow in the Guelph that can capture contaminants and discharge them to the lake nearshore. The case shows a reduction in the calculated maximum dose by more than two orders of magnitude for a "site shore resident" that takes fish and water from the near-shore lake, and three orders of magnitude for a “downstream resident” that takes fish and water from the South Basin of Lake Huron and lives off-site. Given the low doses calculated to someone living directly on top of the repository, these lower "downstream" impacts are completely negligible. The repository will not affect other people living around the lake and using it for food and water.

8.8.2.11 Glaciation

Although glaciation will have a major impact on the surface and near-surface systems, its impact is not expected to be significant in the deep geosphere zones (Section 8.6.1). In particular, evidence from the site characterization and from detailed regional hydrogeologic modelling indicate that glacial cycles at the DGR site would have no significant effects on salinity/marker profiles with depth, indicating that solute transport at repository depth is not affected by glacial episodes (Section 5.4.10 of NWMO11c).

Therefore, the impact of glaciation on the repository’s overall safety is expected to be limited. Nevertheless, it is recognized that it could:

- Impact the performance of the shaft seals;
- Affect resaturation and rockfall in the repository;
- Impact the evolution of the over/underpressures; and
- Erode surface materials.

Results from several calculation cases can be used to provide an estimate of each of these potential impacts (Table 8-26), recognizing that these cases did not explicitly model transient glaciation. The results show that the impacts remain many orders of magnitude below the dose criterion.

Table 8-26: Potential Impacts of Glaciation

Case	Brief Description	Max. Calculated Dose (mSv/year)	Time of Max. Calculated Dose (Ma)
NE-RC	Reference case (including underpressures)	2×10^{-15}	10 [a]
NE-SBC	Simplified base case (excluding underpressures)	1×10^{-13}	10 [a]
NE-EDZ1	Degraded shaft EDZ performance (excluding underpressures)	2×10^{-11}	1.1
NE-RS	Alternative resaturation assumptions (including underpressures)	4×10^{-14}	10 [a]
NE-ER	Surface erosion of 100 m over 1 Ma (including underpressures)	1×10^{-13}	10 [a]

Note:
a. This represents the end of the calculation period.

8.8.3 Model Convergence

The postclosure safety assessment adopts a range of different codes/models. The different codes/models were developed to efficiently explore different FEPs. However, where the models overlap, the comparison of the results provides information on the uncertainty due to the numerical models themselves, such as gridding or convergence precision, or the importance of geometric effects. The comparison of the results obtained using the different codes shows good agreement and helps build confidence in the model results, as illustrated in the results described below.

Figure 8-51 shows a comparison of the vertical hydraulic head profile from independent modelling of the Reference Case and Simplified Base Case using the FRAC3DVS-OPG and T2GGM codes. The difference in the hydraulic head at the Guelph formation (the top of the figure) in the two NE-RC cases is due to differences in the location of the top boundary and the top boundary conditions (see also Figure 8-19). The agreement shows that the geosphere representation in both the groundwater and gas models is substantially equivalent in spite of the different modelling approaches and discretizations.

Figure 8-52 shows a comparison of the contaminant mass transport in groundwater between FRAC3DVS-OPG and AMBER for comparable cases with instant resaturation and instant release of Cl-36. The annual Cl-36 fluxes via both the host rock and shafts are compared at three depths. AMBER is a much simpler model that is intended to conservatively represent contaminant transport, which is confirmed with these results.

The detailed gas modelling with T2GGM considered four numerical models with emphasis on different aspects of the repository (3DD, 3DSRS, 3DSR and 2DRS). Figure 8-53 compares repository pressures and saturations for the NE-SBC case using three T2GGM repository models. Figure 8-54 presents shaft gas flow for the three T2GGM shaft models, for the NE-GG1 case (high gas generation). Results agree very well between the various 2D and 3D models, which is an indication that the numerical gridding and convergence are appropriate.

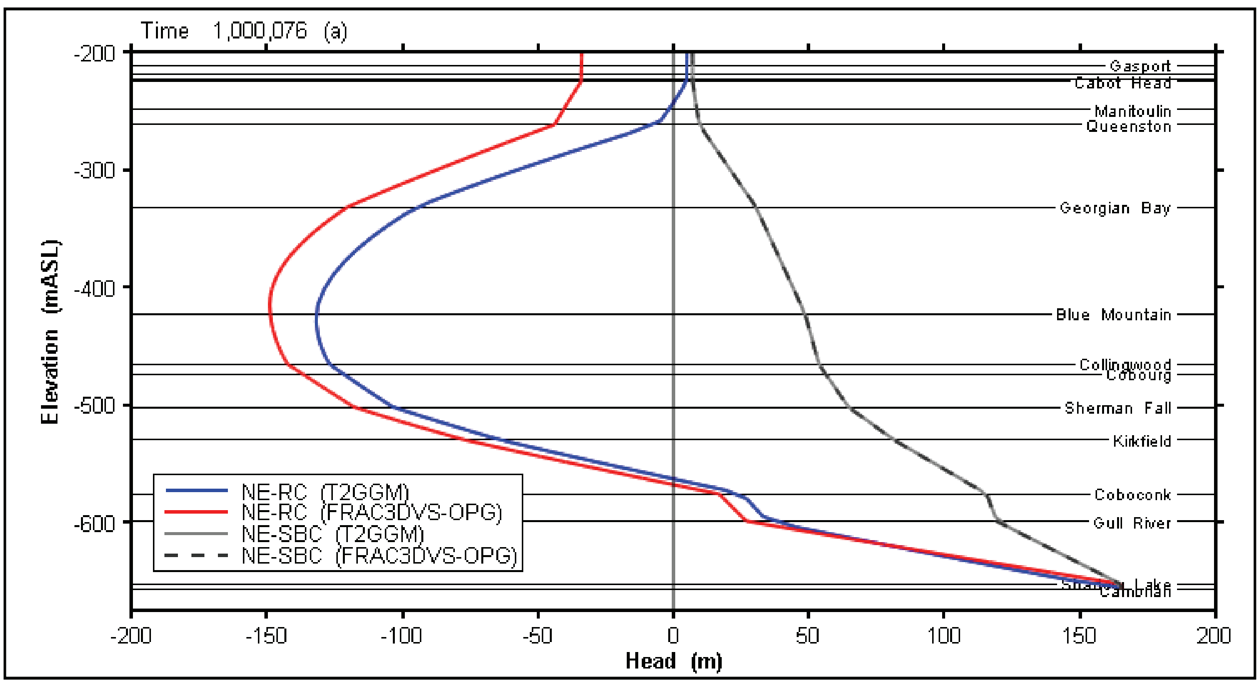


Figure 8-51: Comparison of Hydraulic Head Profiles at 1 Ma for the NE-RC and NE-SBC Cases Calculated by FRAC3DVS-OPG and T2GGM

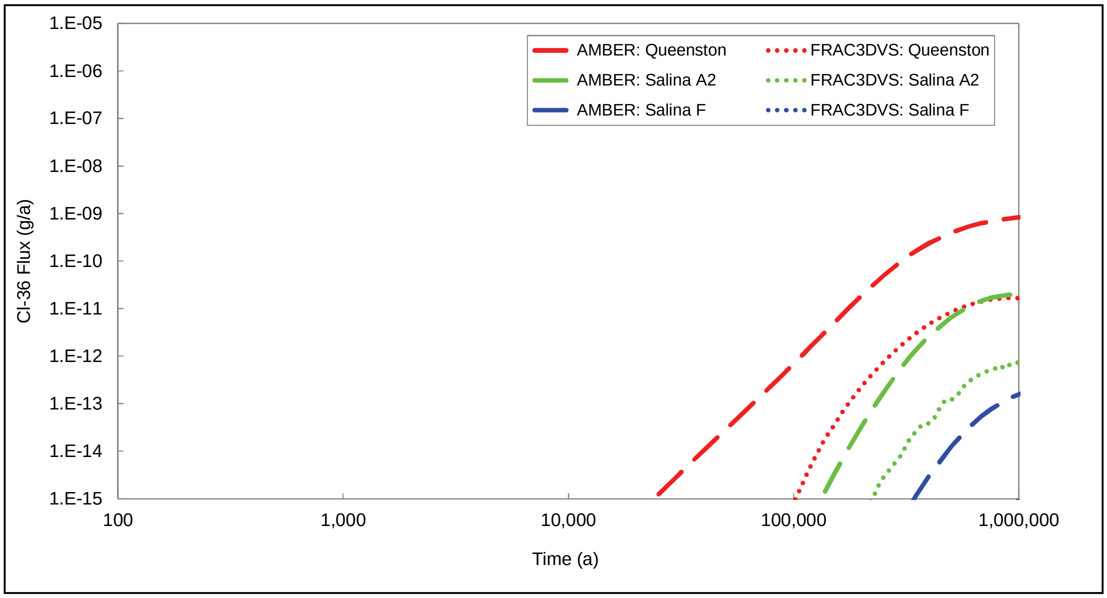


Figure 8-52: Comparison of Cl-36 Fluxes at Different Geosphere Levels Calculated by FRAC3DVS-OPG (NE-RC) and AMBER (NE-RT1)

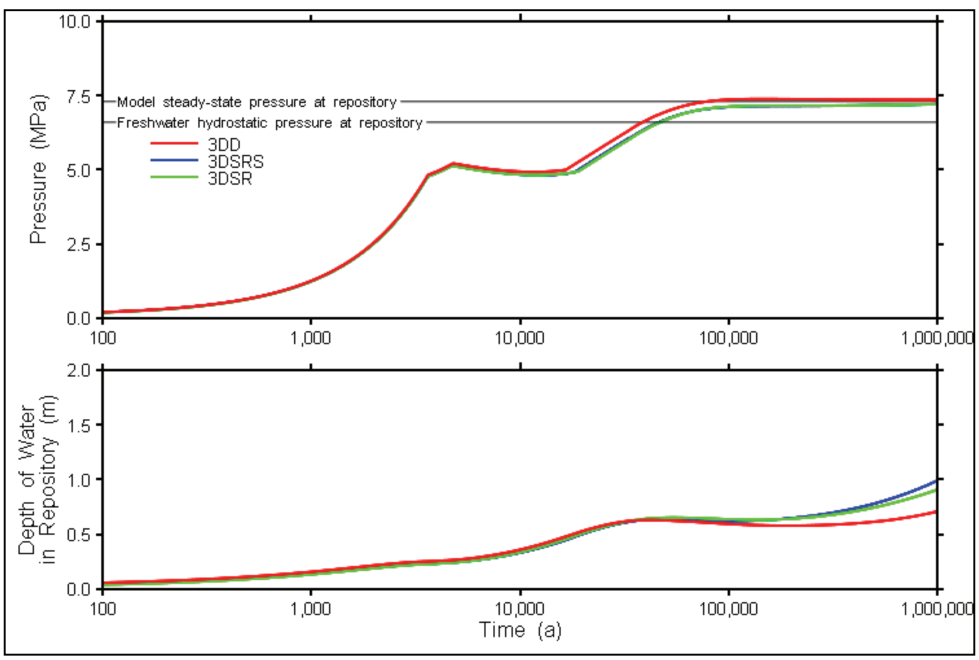


Figure 8-53: Comparison of Repository Pressures and Liquid Saturations for 3DD 3DSRS and 3DSR T2GGM Models of NE-SBC Case

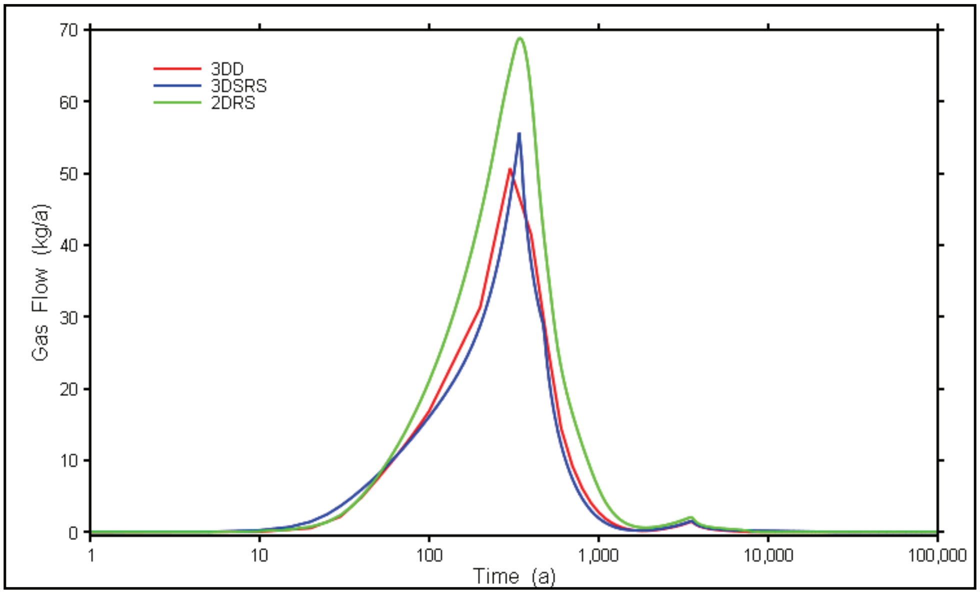


Figure 8-54: Comparison of Shaft Gas Flow Rates at the Collingwood Formation for 3DD, 3DSRS and 2DRS T2GGM Models of NE-GG1 Case

The detailed numerical model results have also been tested with simple analytic models in some limits.

Simple calculations have estimated the maximum gas pressure within the repository as the waste degrades. The simple calculations provide an estimated maximum gas pressure of 7.4 MPa, which compares well with the peak gas pressures calculated by T2GGM for the NE-RC and NE-SBC cases of 8.2 MPa and 7.2 MPa, respectively (see Appendix B of NWMO11aj).

Simple analytical calculations have also been undertaken for gas flow rates via the shafts based on the extra degraded variant to the Severe Shaft Seal Failure Scenario (SF-ED). A gas mass flow rate of 3.1 x 10^{-6} kg/s is calculated using the simple approach, which compares well to the value of 2.9 x 10^{-6} kg/s calculated by T2GGM (see Appendix B of NWMO11aj).

A simple contaminant transport model was developed for groundwater transport through the access tunnel and up the shaft, including radial transport into the adjacent rock through diffusion. The results were compared against FRAC3DVS-OPG results at the top of the Ordovician formations. The analytical model results confirm that only the very leading edge of the breakthrough curve reaches this location during the modelled timescale (see Appendix E of NWMO11an).

8.8.4 Probabilistic Calculation

Probabilistic calculations have been undertaken for leading radionuclides (C-14, Cl-36, Zr-93 and I-129) to investigate sensitivity of consequences to the release and transport parameters. Sampled parameters include the initial inventory, dimensions and corrosion rates for metal wastes, diffusion coefficients and sorption coefficients. (The ranges are described in NWMO11am) The general basis was the NE-RC reference case, specifically the gas and groundwater flows.

The results demonstrate that the concentration of leading radionuclides in well water may increase by up to about two orders of magnitude when the reference case parameters are varied over plausible ranges (Figure 8-55). The very small calculated doses indicate that the safety of the system is not sensitive to variations in these parameters.

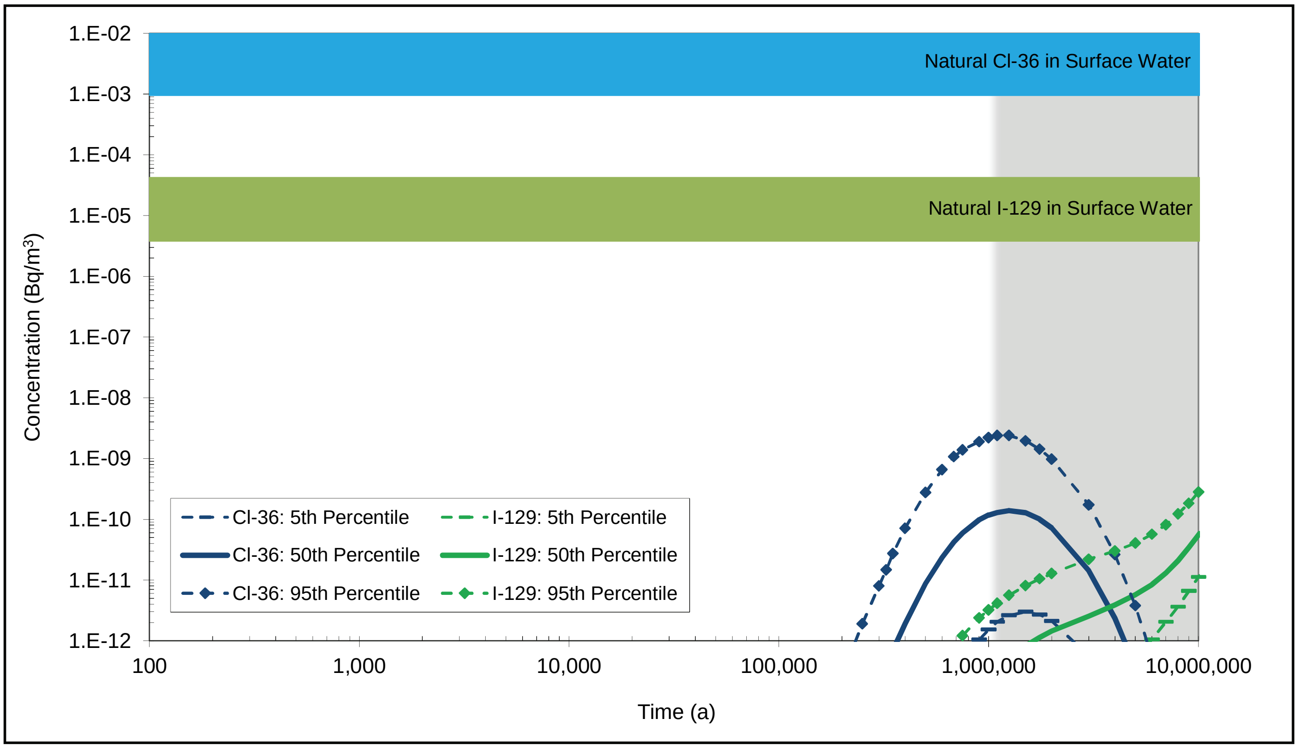


Figure 8-55: Calculated Well Water Concentrations for Cl-36 and I-129 from Probabilistic Sensitivity Calculations (NE-PC) Based on the Reference Case

8.8.5 Alternative Repository and Shaft Seal Designs

The DGR preliminary design incorporates postclosure safety assessment feedback regarding design options. Further input is planned during the detailed design.

8.8.5.1 Original and Final Preliminary Designs

As noted in Section 8.4, there were some design changes in the preliminary design during this assessment. Specific comparison cases were completed, which showed that the postclosure results for the final preliminary design were very similar to those calculated for the original preliminary design.

8.8.5.2 Backfilled Repository

The reference design is to emplace the packages in emplacement rooms but not to backfill these rooms. The advantages of not backfilling are reduced cost, reduced worker dose and greater retrievability during operations, and increased space for gas during postclosure. The option of backfilling the DGR has been investigated through the NE-BF case. In this case, the effective void space in the repository panels is reduced to 30% of the reference value.

The results indicate appreciably higher gas pressures within the repository initially. The gas pressure is sufficient to force free gas up the shafts, although this gas is captured by the relatively permeable Guelph Formation and does not reach the shallow groundwater zone or surface. The gas pressures may approach lithostatic pressure based on conservative, non-water-limited calculations; however lower pressures around hydrostatic pressure are predicted with water-limited calculations.

8.8.5.3 Asphalt Shaft Seal

The design considers an asphalt layer to provide an independent low-permeable seal material. However, the properties and durability of the asphalt seal are not as well established as those for bentonite/sand. The option of not using an asphalt seal was considered (NE-GT4 and NE-GT5, which are both based on the high gas generation case NE-GG1). The results show little effect on overall releases. That is, the asphalt seal layer is not required for shaft seal performance in the Normal Evolution Scenario. Its value is as an independent material that could provide confidence in the shaft performance under unexpected conditions where the bentonite/sand seal is degraded.

8.8.5.4 Monolith

Since the damaged zone around the monolith is an important pathway for contaminant transport, one case was analyzed in which the HDZ and EDZ around the monolith was blocked by a section of concrete (NE-EDZ2). The results were analyzed with respect to groundwater flow and an instant resaturated repository. In this case, blocking the HDZ and EDZ with concrete did not have much effect.

8.9 Summary and Conclusions

Consistent with the guidelines for the preparation of the EIS for the DGR (CEAA09) and with CNSC Regulatory Guide G-320 (CNSC06a), the postclosure safety assessment has evaluated the DGR's ability to perform in a manner that will protect human health and the environment.

The assessment considered potential impacts through consideration of a range of possible future scenarios. The most detailed analyses were carried out for an expected evolution scenario (the **Normal Evolution Scenario**).

Four disruptive ("what if") scenarios have also been evaluated that, although unlikely to occur, could disrupt or bypass the key geosphere barrier:

- Unintentional intrusion into the repository as a result of an exploration borehole (the **Human Intrusion Scenario**);
- The unexpected poor performance of the shaft seals (the **Severe Shaft Seal Failure Scenario**);
- Poor sealing of a site investigation/monitoring borehole in close proximity to the repository (the **Poorly Sealed Borehole Scenario**); and
- A hypothetical transmissive vertical fault in close proximity to the DGR footprint (the **Vertical Fault Scenario**).

8.9.1 Normal Evolution Scenario

The Normal Evolution Scenario Reference Case draws on the results of the site investigations and geosynthesis, and represents the site in the most detail. It includes the measured overpressure in the Cambrian sandstone below the DGR, and the measured underpressures and partial gas saturations in the Ordovician formations within which the DGR is located. Analyses included evaluation of water inflow from rock and shaft, gas generation and build up within the repository, corrosion and rockfall processes that would degrade waste packages, groundwater and gas flow through repository, host rock and shaft seals, and impacts on people living above and around the repository. Variant calculation cases were also assessed to explore uncertainties associated with the Normal Evolution Scenario.

The key results for these cases are described below.

- The full resaturation of the repository with water is gradual, taking more than one million years, due to the low permeability of the host rock and gas generation in the repository. The majority of the water seeps into the repository from the surrounding host rock rather than the shafts.

- Contaminants are contained within the repository and host rock, thereby limiting their release into the surface environment and their subsequent impacts. Reference Case calculations estimate that less than 0.1% of the initial waste activity is released into the geosphere around the repository, and much less is released into the shafts.
- Gases are contained within the repository and geosphere. The gas pressure is anticipated to equilibrate at 7 to 9 MPa, i.e., around the 7.4 MPa equilibrium hydrostatic pressure at the repository level, and well below the lithostatic pressure of about 17 MPa. The gas will be primarily methane in the long-term.
- The low-permeability geosphere and shaft attenuate the release of contaminants, providing time for radioactive decay to decrease the radioactivity in the repository.
- The maximum calculated dose for all calculated cases is more than five orders of magnitude below the 0.3 mSv/year public dose criterion (Figure 8-56). (In general, peak doses to children and infants are within a factor of three of the adult dose.)
- These results apply to a hypothetical family assumed to be living on the site in the future, and obtaining all of its food from the area. The potential dose would decrease rapidly with distance from the site. For example, calculated doses to a "downstream" group exposed via consumption of lake fish and water from Lake Huron are more than three orders of magnitude lower than the dose to the family living on the site.

8.9.2 Disruptive Scenarios

A tiered approach is adopted for disruptive scenarios, recognizing the speculative nature of some scenarios. First, a dose criterion of 1 mSv/year is used for radiological exposure of humans under credible scenarios. Second, if calculated doses exceed 1 mSv/year for a scenario, the acceptability of results from that scenario is examined on a case-by-case basis taking into account the likelihood and nature of the exposure, conservatism and uncertainty in the assessment, and conservatism in the dose criterion. Where feasible, they are compared to a reference health risk of 10^{-5}/year.

Consistent with the Normal Evolution Scenario, a reference calculation is undertaken for each Disruptive Scenario. To avoid ambiguity with the Normal Evolution Scenario Reference Case, the reference calculation for each Disruptive Scenario is termed the Base Case calculation. In addition to the Base Case calculations, some variant calculations were also undertaken. The key results are summarized below and in Figure 8-57.

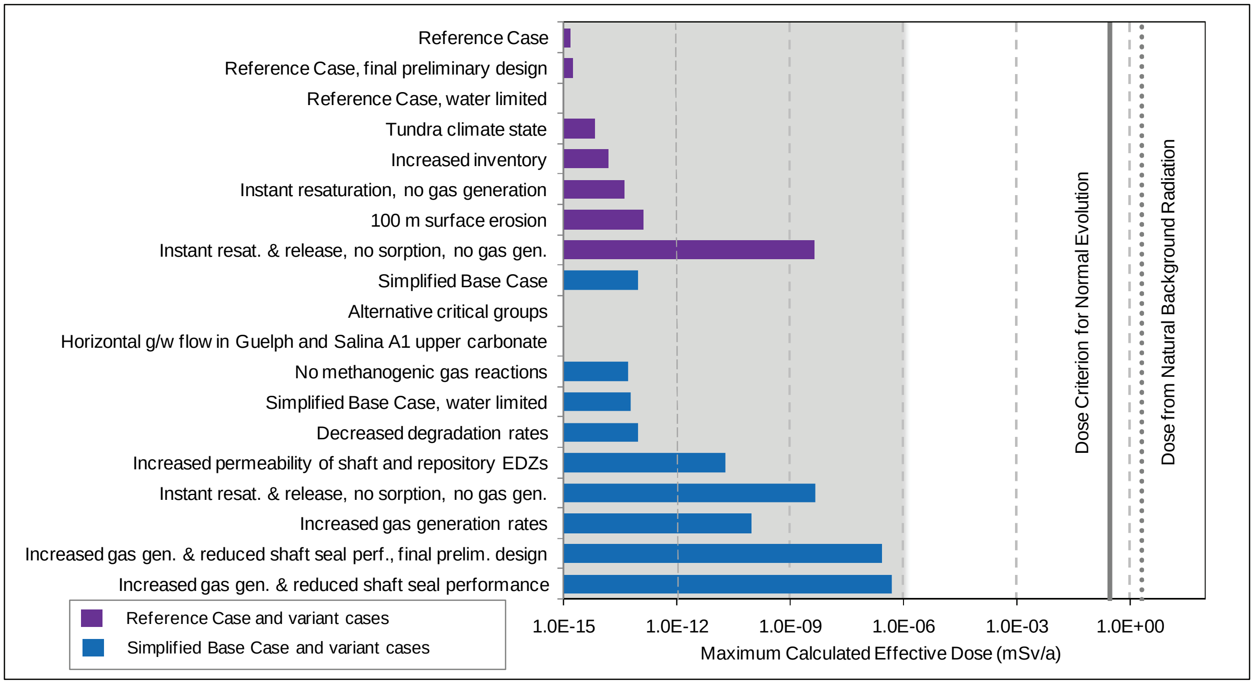


Figure 8-56: Normal Evolution Scenario: Maximum Calculated Doses for All Calculation Cases

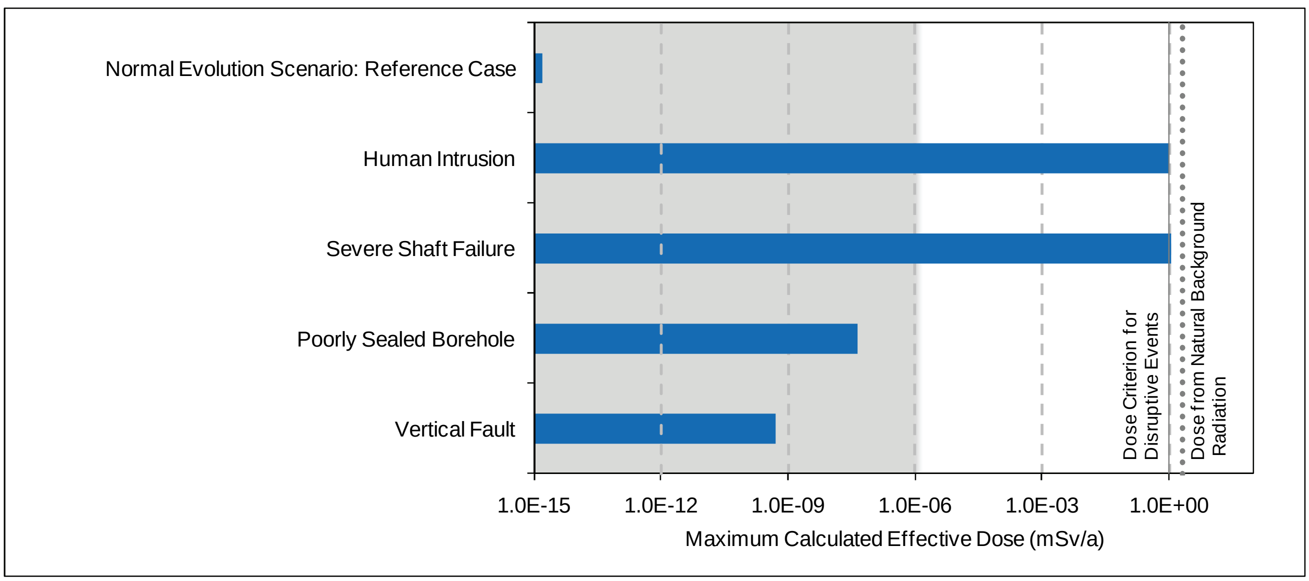


Figure 8-57: Disruptive Scenarios: Maximum Calculated Doses for Base Case Calculations

- For the *Human Intrusion Scenario*, if a borehole is drilled into the repository and gases and material from the repository are not appropriately contained, the calculated doses could be about 1 mSv for the drill crew and for a future person farming on the contaminated site. The likelihood of drilling into the repository in any given year is very low due to the lack of mineral resources and the repository's small footprint and depth, and high contaminant releases are unlikely when following standard deep drilling practices. Thus the peak risk of serious health effects is low, and much less than the reference health risk value of 10^{-5}/year.

- For the *Severe Shaft Seal Failure Scenario*, the maximum calculated doses are about 1 mSv/year, based on immediate failure of 500 m of low-permeability shaft seals (to 10^{-9} m/s hydraulic conductivity), reduced sorption in the shafts, increased degradation of shaft EDZs, and assuming a family is farming directly on top of the shafts (including a house located on the main shaft). The scenario is very unlikely. Therefore the risk from the severe shaft seal failure scenario is low.

- Calculated peak annual doses for the *Poorly Sealed Borehole Scenario* and the *Vertical Fault Scenario* are about several orders of magnitude less than the dose criterion.

- Additional cases were evaluated to determine what it would take to have a disruptive scenario with larger impacts. For the Human Intrusion Scenario, the borehole would have to be extended down to the Cambrian and then poorly sealed, so that there was water flowing up the borehole, through the repository and into the shallow groundwater system. For the Severe Shaft Seal Failure Scenario, the entire shaft would need to degrade by 4-5 orders of magnitude below design basis to a hydraulic conductivity of 10^{-7} m/s, about equivalent to fine silt and sand. In these cases, the peak dose to someone living on top of the repository site could be tens of milliSieverts.

- The primary risk in the disruptive scenarios is from release of C-14 containing gas from the repository. The potential impacts therefore decrease to well below the dose criterion after about 60,000 years due to C-14 decay. Since glaciation at the DGR site is not likely to occur prior to then, there is little risk from glaciation affecting these maximum peak doses from disruptive scenarios.

- Finally, it is noted that the impacts of the disruptive scenarios are local. The total content of C-14 in the repository on closure is approximately equal to the site annual DRL for air release of C-14 as CO_2. So even if the entire C-14 inventory were released as gas within a one year period, then dose impacts for people living outside the current Bruce nuclear site would be around or below the public dose criterion.

8.9.3 Key Radionuclides

- Most radionuclides are retained within the repository or geosphere.

- H-3, although a significant contributor to the waste radioactivity at closure, is fully retained within the repository and host rock where it decays.

- For scenarios that could result in releases of contaminants to the surface environment within about 60,000 years of closure, C-14 (mostly from ILW moderator resins) is the key radionuclide, together with Nb-94 (mostly from ILW pressure tubes) for human intrusion.

- For releases that occur at later times, Cl-36 (mostly from ILW pressure tubes), and I-129 (mostly from ILW PHT resins) become more important due to their longer half-life and their mobility.

- Nb-94 and Zr-93 are slowly released and mostly retained within the shaft and geosphere and so are not significant contributors to the calculated doses for groundwater releases.

8.9.4 Impacts on Non-Human Biota and Non-Radiological Impacts

- For the Normal Evolution Scenario, concentration of radionuclides and of non-radioactive contaminants in surface media are well below the relevant environmental protection criteria.

- For Disruptive Scenarios, impacts are also low. All non-radioactive contaminants and most radionuclides have calculated concentrations in surface media that are well below their screening concentration criteria for the base cases.

- There are some local exceedances of screening criteria for the Human Intrusion Scenario and the Severe Shaft Seal Failure Scenario. In particular, the concentration of C-14 and Nb-94 could locally exceed soil criteria by a factor of 20 if drilling debris from the repository were to be dumped on the surface at the site in the Human Intrusion Scenario. And C-14 could locally exceed the surface water screening criteria by a factor of 1.4 in the Severe Shaft Seal Failure scenario.

- Since these higher concentrations are local, the screening criteria are conservative, and the scenarios are very unlikely, the risk to biota from these scenarios is low.

8.9.5 Implications on Design

- The results indicate that there is no benefit to be gained from backfilling the repository due to the significant containment already provided by the host geology and the shaft seals. Backfilling may result in a higher gas pressure within the repository after closure.

- The calculations have emphasized the importance of the shaft seals in limiting contaminant fluxes in groundwater and gas from the repository. The damaged zone in the rock around the concrete monolith at the shaft base is a key pathway to the shafts.

- Some contaminants that do migrate up the shafts as gas or dissolved species can be laterally diverted into the higher permeability Silurian hydrostratigraphic units such as the Guelph Formation and Salina A1 Unit upper carbonate. The low-permeability shaft seals in the Silurian are effective in directing contaminant transport into these features.

8.9.6 Uncertainties

The long timescales under consideration mean that there are uncertainties about the way in which the system will evolve. These uncertainties have been treated in the current assessment through: the assessment of range of scenarios, models and data; and the adoption of conservative scenarios, models and data. The key uncertainties in terms of their importance to potential impacts are as follows.

1. **Gas pressure and repository saturation** are important in determining the potential release of radioactivity into repository water, and the potential for C-14 release through gas in the first 60,000 years. Therefore, the processes that control these parameters are important. They were approached in this safety assessment through use of a range of calculation cases to test the importance of uncertainties in those contributing processes.

2. **Shaft and EDZ properties** and their evolution with time. Variant calculation cases for the Normal Evolution Scenario and the Severe Shaft Seal Failure Scenario calculations emphasize the importance of the shaft seals, particularly in the first 60,000 years following closure.

3. **Glaciation effects** - Although geological evidence at the site indicates that the deep geosphere has not been affected by past glaciation events, glaciation is expected to have a major effect on the surface and near-surface environment, and is not entirely predictable. It should, however, be noted that ice-sheet coverage of the site is likely to occur only after 60,000 to 100,000 years, at which point the primary remaining hazard will be long-lived radionuclides in

groundwater rather than gaseous C-14. Calculations have shown that the deep groundwaters are stable and transport is diffusion-dominated, so dissolved radionuclides will be contained in the deep geosphere with large safety margins.

4. **Chemical reactions** - Under the highly saline conditions of the deep geosphere at the DGR site, several aspects of the chemistry are uncertain due to the limited database. In particular, this includes the sorption of contaminants on seal materials and host rocks, as well as mineral precipitation/dissolution reactions. Generally conservative values have been adopted in this assessment.

The geosphere is clearly key to the DGR safety. In general, the attributes of the geosphere are sufficiently well known to support the safety assessment (Section 4.6). However, some aspects are still uncertain, such as the cause of the over/underpressures. These geosphere uncertainties have been considered in this assessment through a range of scenarios, calculation cases and conservative parameter values. Although further resolution of these uncertainties is desirable to increase confidence, they have not been found to be important to the conclusions of this assessment.

The Geoscientific Verification Plan outlines plans to initiate tests of important processes and materials in the rock during DGR construction - for example EDZ measurements (NWMO11ar). Also, the shaft seal design will not be finalized until the decommissioning application several decades from now, and will take advantage of knowledge gained over the intervening period.

While these tests plus further safety and geoscience modelling work will help to improve confidence in the assessment, the results presented here show that the DGR meets the postclosure safety criteria, that it provides isolation and containment of the wastes, and that the system safety is robust, i.e., the system will maintain its integrity and reliability under a range of conditions. The uncertainties should be interpreted in the context of the low calculated impacts, for example calculated dose results for all Normal Evolution Scenario variant cases are all more than five orders of magnitude below the dose criterion.

8.9.7 Conclusions

The postclosure safety assessment has evaluated the DGR's ability to perform in a manner that will protect human health and the environment from the emplaced waste for an expected evolution scenario, as well as a number of disruptive ("what if") scenarios.

The assessment calculations for the Normal Evolution Scenario indicate that the DGR system provides effective containment of the emplaced contaminants. Most radionuclides decay within the repository or the deep geosphere (Figure 8-58). The amount of contaminants reaching the surface is very small, such that the maximum calculated impacts for the Normal Evolution Scenario are much less than the public dose criterion of 0.3 mSv/year for all calculation cases. In addition, potential impacts of radionuclides on biota and non-radioactive contaminants on humans and non-human biota are well below the relevant criteria.

The isolation afforded by the location and design of the DGR limits the likelihood of disruptive events potentially able to bypass the natural barriers to a small number of situations with very low probability. Even if these events were to occur, the analysis shows that the contaminants in the waste would continue to be contained effectively by the DGR system such that risk criterion is met.

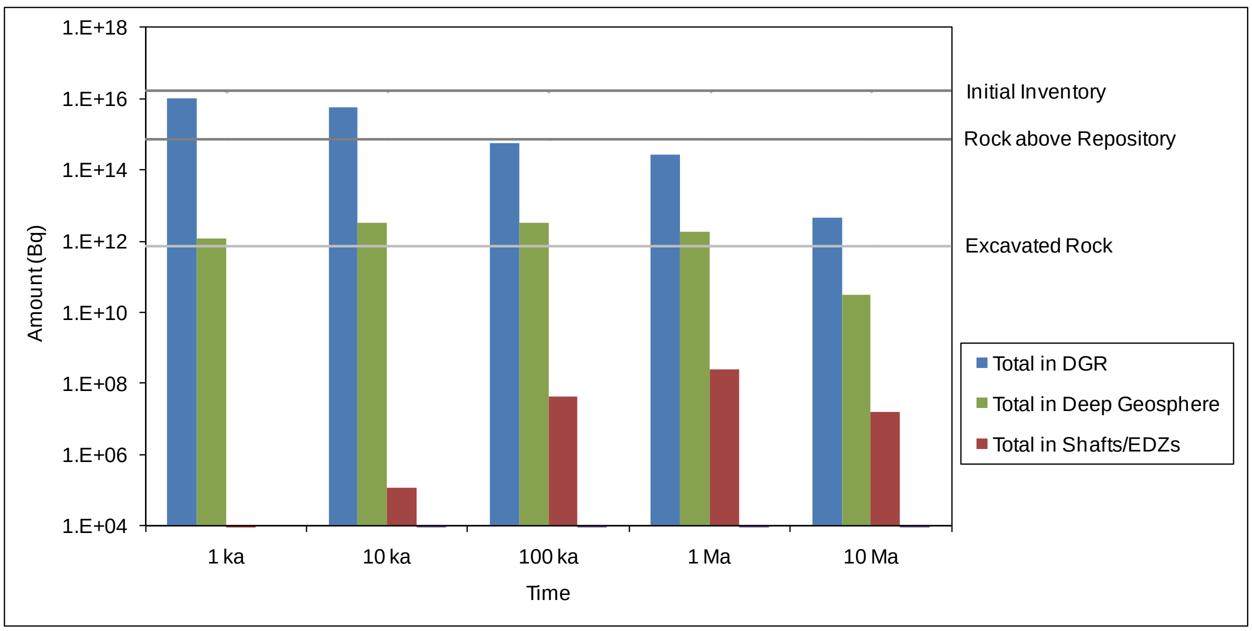


Figure 8-58: Distribution of Activity in System at Different Times for the Normal Evolution Scenario Reference Case

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9. SITE PREPARATION AND CONSTRUCTION

9.1 Introduction

Site preparation and construction of the DGR Facility will begin after receipt of a site preparation and construction licence and work will span a period of approximately five to six years. Following receipt of the licence, the site will be prepared for construction by clearing and grubbing, site grading, installing fencing, installing temporary construction services, and establishing the stormwater management system. The first major construction activities will be establishing the shaft collars followed by the erection of the main and ventilation shaft headframes, sinking of two shafts and construction of underground infrastructure, access tunnels and emplacement rooms at the repository level. At the same time various facilities and infrastructure will be constructed at surface. At the end of construction, the DGR Facility will be fully commissioned and all temporary construction facilities removed from the site in preparation for handover to DGR operations staff.

The management system, which describe organization, responsibilities and governance that will apply to site preparation, construction and commissioning activities, encompassing procedures specific to health, safety and environment, quality plans (verification and validation), training (construction and mining disciplines), human factors, etc., are outlined in the DGR Project Management System document (OPG11b) and the Design and Construction Phase Management System document (NWMO11a).

9.1.1 Construction Regulations

Although the DGR Facility falls under federal jurisdiction, according to Ontario Hydro Nuclear Facilities Exclusion from Part II of the Canada Labour Code Regulations - Occupational Health and Safety (SOR/98-180), the responsibility for workplace health and safety at all OPG nuclear facilities has been delegated to the Province of Ontario. Thus workplace health and safety during the construction of the DGR Facility will be regulated under the Ontario Occupational Health and Safety Act (OHSA90) and its associated regulations. Labour legislation in the Province of Ontario is enforced by the Ministry of Labour (MOL).

The construction of the DGR Facility will be regulated under Ontario's Occupational Health and Safety Act. Given the nature of the project, it is expected that the Ontario MOL will administer their regulatory supervision of the project primarily under the Mines and Mining Plants Regulation, R.R.O 1990 (Reg 854).

9.1.2 Construction Program and Schedule

Site preparation and construction of the DGR project will follow the construction program as referenced in the Design and Construction Phase Management System document (NWMO11a). The monitoring program during site preparation and construction is described in the DGR EA Follow-Up Monitoring Program (NWMO11at). A high-level schedule for the project, including the site preparation and key construction activities, is shown in Figure 9-1.

9.2 Site Preparation

The DGR project site is nominally 31 ha and a major portion of the site will be prepared for the establishment of construction facilities and services required for the construction of the DGR facilities. Site preparation will include:

- Fencing and security;
- Clearing and grubbing; and
- Site grading.

Approximately 22 ha of the DGR site is currently open industrial barren land and some of this land has been previously used to support construction activities elsewhere on the Bruce nuclear site (e.g., construction laydown). Currently a portion of this land is used for disposal of excess clean soil material from excavation activities at other OPG facilities on the Bruce nuclear site. The remaining 9 ha of the DGR site are mixed forest.

Currently a 44 kV single-pole power line crosses the southern end of the DGR site. This line will be relocated as part of the site preparation activities.

9.2.1 Fencing and Security

The DGR Facility, being located within the Bruce nuclear site, will be encompassed by the larger existing Bruce nuclear site security system. The Bruce nuclear site is entirely surrounded by a perimeter fence that restricts access to the site via land or water. Only authorized personnel and vehicles are allowed on the site. All construction personnel will be required to pass the Bruce Power site security clearance requirements. During DGR construction, the only access to the Bruce nuclear site will be via controlled checkpoints and it is anticipated that all construction traffic will be required to enter via the main Bruce nuclear site entrance.

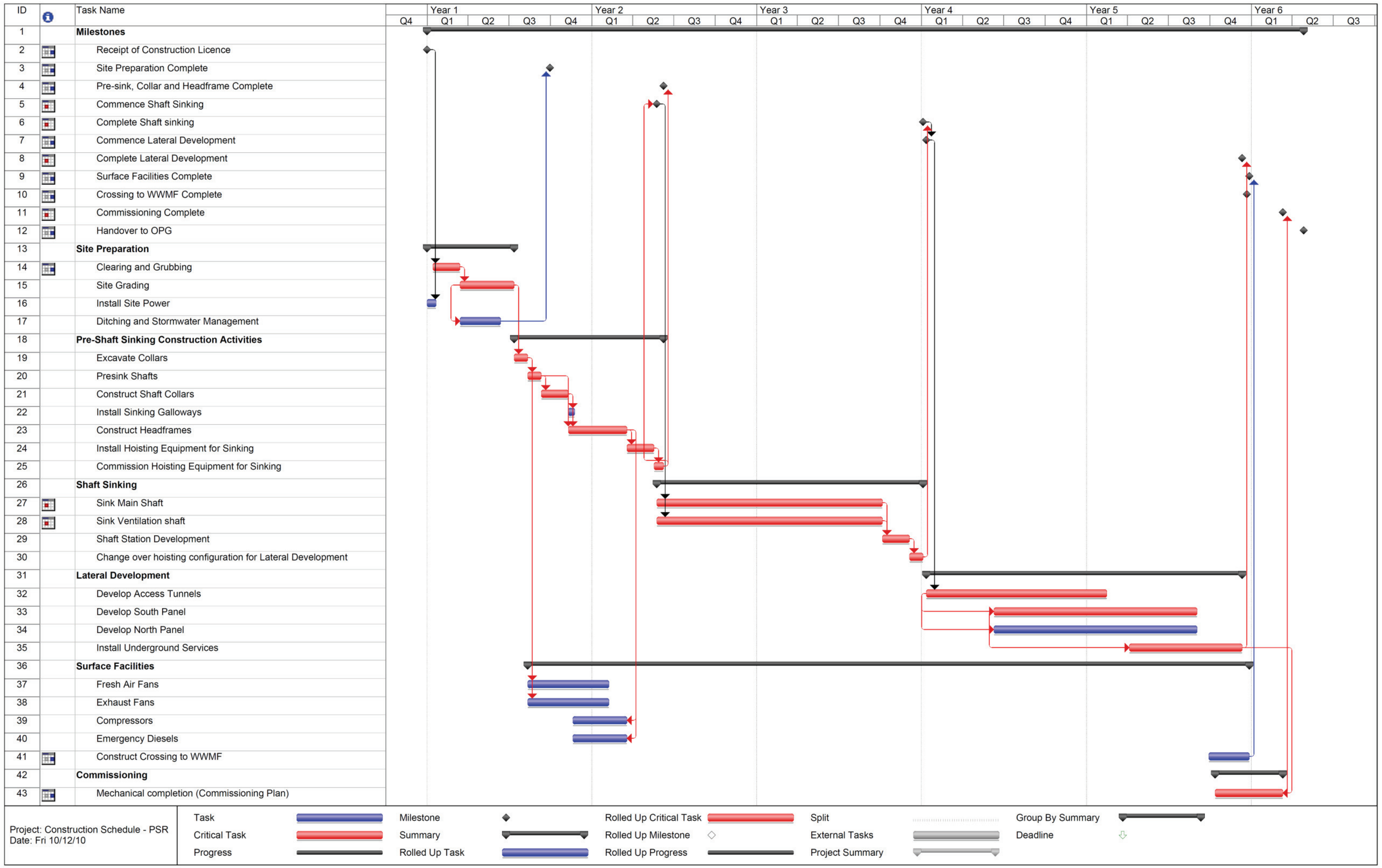

Figure 9-1: DGR Project Site Preparation and Construction Schedule

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Specific to the DGR, the entire construction island, including the WRMA, will be surrounded by a fence to isolate the DGR from other OPG and Bruce Power facilities. A separate gated entrance to the DGR site will lead directly off the Interconnecting Road and/or a new road along the abandoned rail bed to the east of the DGR site. During construction, there will be no access via the WWMF. Security on the DGR site during the site preparation and construction phase will be provided by the project at the main access point to the DGR site.

9.2.2 Clearing and Grubbing

Clearing and grubbing will be performed within the DGR site to remove existing vegetation. Clearing and grubbing will be timed to avoid environmentally sensitive periods (i.e., outside of the breeding bird season, generally mid-May through mid-July). Clearing and grubbing will be staged to ensure that the period of time in which soils are exposed is minimized.

Approximately 9 ha of forested area will be cleared in preparation for construction. Trees will be felled, skidded and piled in the cut area and may be disposed of by chipping or piling. Wood materials that are chipped will be reused on the DGR site or elsewhere on the Bruce nuclear site in landscaping activities. Some roots, stumps, embedded logs and debris will be grubbed to a depth of about 0.5 mBGS. Wood materials that cannot be chipped will be disposed at a suitable location either on the DGR site or elsewhere on the Bruce nuclear site in accordance with existing management practices.

9.2.3 Site Grading

Following clearing and grubbing, the ground surface on the DGR site will be graded to specified elevations and with grades that will ensure proper drainage of the site. The two major areas to be graded are shown in drawing H333000-WP404-10-042-0001 (see Chapter 17). The area in the location of all DGR buildings and ancillary facilities will be graded to an elevation of about 186 mASL. The area in the location of WRMA and associated stormwater management pond will be graded as necessary to receive waste rock. Both areas will have perimeter ditching to receive stormwater run-off which will ultimately discharge from the site via the stormwater management pond.

Prior to start of any site grading work, a comprehensive survey of the site will be performed to confirm that there are no buried services on the DGR site. In addition, an investigation will be performed to identify the location of any potentially contaminated materials on site. If found, these materials will be managed either on the DGR site or elsewhere, as required, in accordance with provincial regulations.

Top soil will be stripped from the site where excavation or grading is planned and stockpiled separately from other excavated material. The topsoil will be protected and kept in segregated piles until needed later for finished grading. Overburden material that would be unsuitable for use as topsoil will be stockpiled and used later for backfilling.

The site grading plan for the DGR site will take advantage of existing topography and soils. It is expected that it will not be necessary to import soils to site to achieve required grade elevations.

It is anticipated that site grading will be performed by using conventional earth moving equipment including bulldozers, dump trucks, blade scrapers and water trucks (for dust control).

9.3 Other Activities Required for Construction

9.3.1 Installation of Construction Services

Construction services will be installed within the DGR construction island (see Figure 9-2) for use by various contractors working on both surface and underground construction.

Services provided by the project will include:

- Construction road access to the fenced and gated construction island via the Interconnecting Road and/or a new road constructed along the abandoned rail bed;
- A levelled, graded and drained yard area with temporary construction roads in place for trailers and material storage/laydown areas;
- Electrical substation supplied by a 13.8 kV voltage transmission line (see Section 6.2.4.1 for description of the electrical system);
- Concrete batch plant with materials stockpiles to be located on the DGR site – common to all construction users;
- Service and fire water connection points to the existing Bruce Power network – common to all construction users;

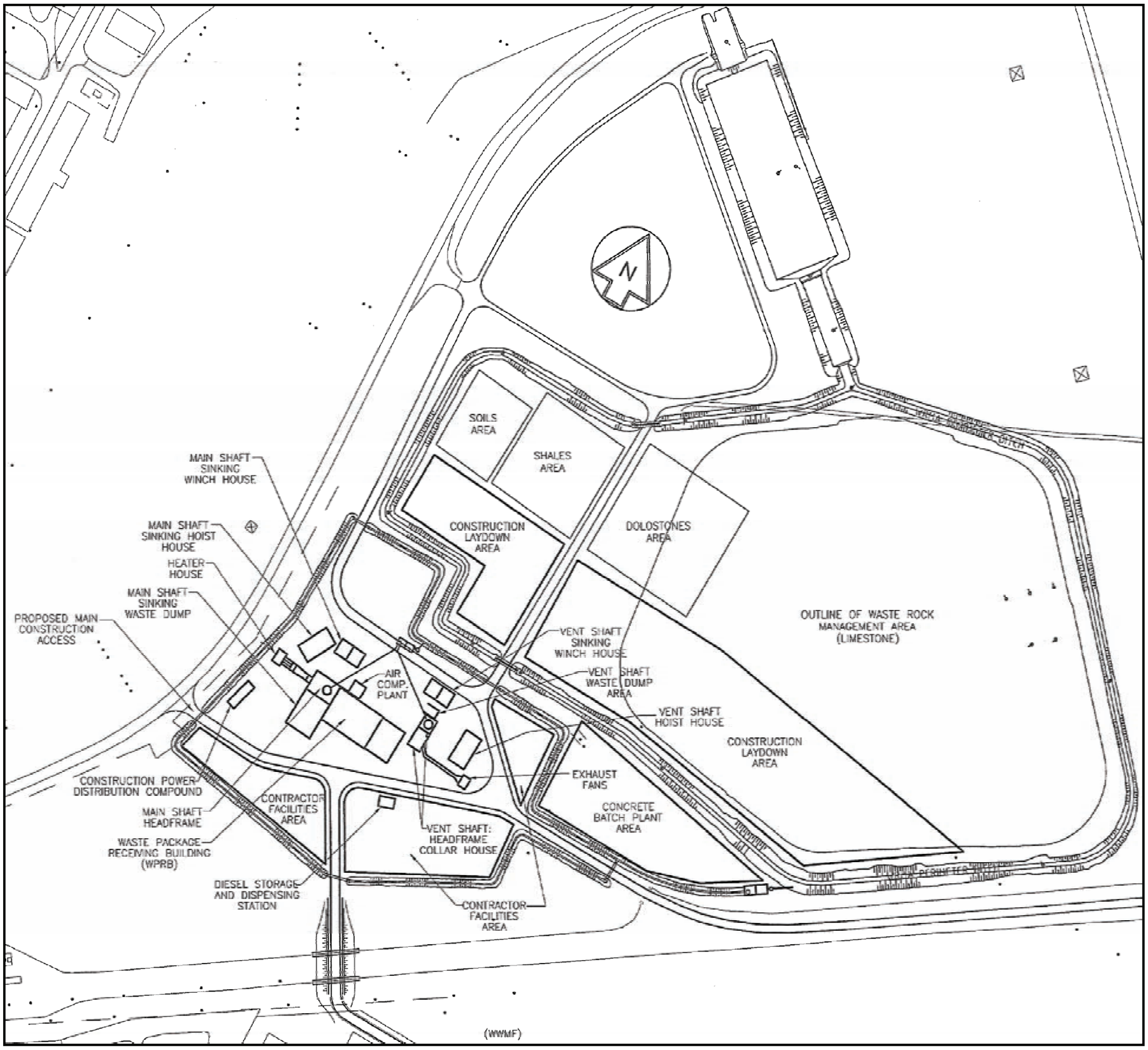


Figure 9-2: DGR Construction Layout

- Temporary diesel and unleaded fuel dispensing station – common to all construction users;
- Communication connection point to the existing fibre-optic line – common for all construction users;
- Security located at the main access point to the DGR site; and
- Emergency response and mine rescue (mine rescue supported by the mining contractor).

All other services will be provided by the various contractors as required (i.e., compressed air, temporary or back-up diesel generators, etc.). Prior to start of construction, each contractor will install temporary construction trailers for offices and storage, worker change rooms (e.g., mine dry facilities) and lunch rooms. Contractors will supply their own stand-alone washroom and sanitary facilities. A connection to existing sewer lines at the Bruce nuclear site will not be provided during construction.

9.3.2 Stormwater Management

The design of the stormwater management system is described in Section 6.2.4.8 and is comprised of a system of ditches and the stormwater management pond. The stormwater management system will be used to collect and treat all stormwater run-off inside the DGR construction island, as well as, any water pumped to surface during shaft sinking and underground development. Water pumped to surface from underground via the shafts during construction will be treated in a temporary water treatment plant prior to discharge into the existing ditch system to the retention pond.

The water treatment plant will be located in the vicinity of the main and ventilation shafts. The ultimate location of the plant will depend upon the final construction site layout for shaft sinking activities (i.e., location of temporary structures such as hoist and winch houses, etc.). Water pumped from the underground development activities will be directed to the water treatment plant, as required, to remove materials such as grit, oil and grease, prior to discharging into the site stormwater management system. Contaminants such as nitrogen, ammonia and saline groundwater will be treated, as necessary, in the stormwater management pond.

The location of stormwater drainage ditches and the pond are shown on drawing H333000-WP404-10-042-0001 (see Chapter 17). The ditches will be excavated at time of site grading by mechanical excavation equipment (see Section 9.2.3). The ditches will be vegetated, as necessary, to prevent erosion of soil from the bottom and sides of ditches. The stormwater management pond will be constructed in a manner similar to that used to construct the ditches with a low permeability base (natural or composite liner). A protective cover of granular material will be placed in the base and sides of the pond. The water flow control systems and the water sampling station will be constructed at the point of discharge from the pond. Discharge from the stormwater management pond will be monitored to confirm that it meets effluent limits and criteria established by the MOE for the site in accordance with the provincial regulations.

If necessary, improvements will be made to the drainage system downstream of the pond discharge location to ensure unobstructed flow of water to Lake Huron. These improvements may include replacement of existing culverts beneath the Interconnecting Road, and cleaning and/or enlarging the ditch between the Interconnecting Road and Lake Huron.

9.3.3 Waste Rock Handling

The waste rock produced from the development of the shafts will be transferred to surface as described in Section 9.4.5.1. At surface, the rock will be removed from the muck bay at each headframe by front-end loader and loaded into off-highway trucks. These trucks will transfer the materials to the appropriate temporary stockpiles at the WRMA (see Section 6.2.3). A bulldozer will be located at the waste rock piles to maintain appropriate grades of the stockpiles. Dust mitigation will be provided through the application of water (e.g., spraying piles or misting) as required.

If the temporary storage of shales from shaft development is required for a period of longer than one year, the shale stockpile will be capped to minimize the potential for erosion of these materials while also limiting infiltration into the pile. Overburden materials excavated from the site are likely to be suitable for use in capping.

Waste rock from the repository level development will be transferred from underground to surface via the ventilation shaft as described in Section 9.4.7.1. At surface, it will be loaded from the muck bay in the same manner as described above and transferred to the long-term waste rock pile.

A setback of 200 m from the Interconnecting Road is included in the design of the long-term waste rock pile (limestone). Visual screening will be provided by planting of trees.

9.3.4 Conventional and Hazardous Materials Management

During site preparation activities, conventional wastes will be limited as it is expected that trees and stumps from grubbing will be re-used on the site and soils will be used to establish site grading and berm requirements. Investigations of the site indicate that there are no pre-existing contaminated soils that require handling and disposal.

Conventional construction and domestic waste generated throughout the site preparation and construction phase will be collected at the site in industrial bins and disposed of or recycled off-site at a licensed waste management facility. Hazardous materials for disposal will primarily consist of oils and lubricants used to maintain and operate construction equipment. Waste oils, lubricants and solvents will be collected in totes on both surface and underground and transferred to an off-site licensed facility for disposal.

During construction, sanitary services will be supplied and managed by the contractor, with disposal off-site. This will include sanitary and mine dry facilities (change room and showers).

Conventional and hazardous waste volumes expected to be generated during the site preparation and construction activities are summarized in Table 9-1.

Table 9-1: Projected Range of Annual Output of Waste

Waste Material	Projected Range of Output
Brush and stumps	0
Contaminated soils	0
Oils and grease	35,000 – 45,000 L per year
Batteries	150 – 200 kg per year
Solvents, paints, etc.	1,500 – 2,500 L per year
Domestic waste	25,000 – 35,000 kg per year
Sanitary waste	8,000 – 12,000 kg per year
Underground sump discharge	100,000 – 120,000 m^3 per year
Waste rock (limestone)	832,000 m^3

Explosives will be required for excavation of the shafts and underground facilities. Handling of explosives on the project site will be in accordance with Part VI of the Mines and Mining Plants Regulations (Reg. 854).

9.4 Construction

After the DGR site has been prepared, and all contractor facilities and services are in place, construction of the DGR facilities will commence. Construction and repository development work will take approximately five years to complete and will be executed as described below.

The shaft pre-sink, would involve the following major construction activities:

- Ground improvement (e.g., grouting or freezing) around the two shafts;
- Preparation of the shaft collar areas (excavation to approximately 30 m depth) including installation of headframe foundations;
- Construction of main shaft and ventilation shaft headframes; and
- Installation of shaft sinking hoisting equipment.

Shaft sinking, would involve the following major construction activities:

- Sinking of the two shafts and construction of final concrete liners;
- Equipping each shaft (installation of shaft steel, permanent services, etc.) and installation of permanent hoisting equipment.

Final development would involve the following major construction activities:

- Construction of underground facilities at repository level; and
- Completion of surface facility construction.

The final activity is commissioning and would involve preparing the DGR Facility for handover to DGR operations staff.

The following sections provide a description of the aforementioned major construction activities. The layout of construction facilities is shown in Figure 9-2. The construction layout is arranged to form a “construction island”, in which all facilities will be grouped in relatively close proximity around the construction site. This arrangement will be fenced off from the rest of the Bruce nuclear site and allows for controlled access and security at the site.

9.4.1 Ground Improvement

Given the moderately high hydraulic conductivities in the upper portions of the dolostones (upper 180 m, excluding overburden), it is assumed that ground improvement will be required to control groundwater inflows and permit safe excavation under relatively dry conditions. The ground improvement program is intended to create a relatively impermeable annulus around the shafts to limit lateral groundwater flow from the permeable dolostones into the excavation. This can be accomplished from surface through either grouting or freezing. The following sections describe possible methods of ground improvement in the upper dolostones. One, or a combination, of these methods may be used to allow safe excavation of the two shafts. It is expected that ground improvement will not be required in rock formations below 180 m (see Chapter 4).

9.4.1.1 Surface-Based Grouting

Formation grouting from the surface will likely be used during the pre-sink to control potential groundwater inflows in the top 20 m of the dolostones (excluding overburden). It may be possible to use this method of ground improvement for all upper formations to a depth of approximately 180 m.

Grouting will be accomplished using a drilled hole pattern and injecting cementitious grout into rock discontinuities. This creates a low permeability zone around the ground to be excavated. Cementitious grout (e.g., micro-fine cement) or chemical grout (e.g., sodium silicate suspension) would be injected through blind drilled holes or through a sleeve port grout pipe system that facilitates multiple grouting stages through the same grout holes depending upon observed performance. This method has been used extensively elsewhere during shaft sinking in a variety of rock types and provides the following:

- Decreases rock mass hydraulic conductivities (i.e., grouting of major fracture zones); and
- Control of seepage into the excavation during shaft construction.

Consideration for the removal of overburden for shaft collar construction will be taken into account in developing the drilling program for the grouting program.

9.4.1.2 Ground Freezing

If ground freezing is determined to be required it would be accomplished through vertical holes drilled from the surface around the perimeter of each shaft. Circulation pipes (denoted as freeze pipes) would be installed in each drill hole to permit flow of low temperature coolant (brine or liquid nitrogen). As the coolant circulates within the closed loop freezing system (all coolant is contained within the loop), it lowers the temperature of the groundwater outside of the pipe until it freezes.

The freeze hole pattern at the DGR site would be designed to create a virtually impermeable frozen ring around each of the planned shaft excavations. Probe holes would be drilled ahead of the advancing shaft bottom to verify the effectiveness of the freeze and to assess the inflow potential. The results of the probe holes would be used to direct the implementation of supplemental formation grouting.

9.4.1.3 Cover Grouting During Sinking

Cover grouting during sinking could be used as an alternative to, or in conjunction with, surface-base grouting. Cover grouting could also be used to supplement either surface-based grouting or ground freezing in the event some zones were not properly sealed by either of these surface-based ground improvement methods.

If surface-based ground improvement was used in advance of shaft sinking, then the probe holes would be used to verify the effectiveness of the grout curtain or ground freezing. The results of the probe holes would be used to direct the implementation of cover grouting at the shaft bottom.

Possible cover grouting arrangements consist of primary and secondary grout holes extending in front of the sinking face. Secondary grouting would be directed on the basis of inflow criteria applied to observed inflows from a second probe hole advanced after primary grouting is completed.

9.4.2 Preparation of Shaft Collars

Overburden will be removed to bedrock contact and engineered side slopes established using conventional excavation techniques (excavator, trucks, bulldozer, etc.).

The shaft collar construction will include the preparation of the headframe foundation and the pre-sink or collar development (approximately 20 m into bedrock) to provide sufficient space for the installation of the sinking Galloway (see Section 9.4.5.1). The development of the collar utilizes temporary equipment for drilling, mucking and hoisting requirements until such a time as the headframe and hoist houses are constructed and the Galloway is functional. The collar above bedrock will be constructed using conventional concrete forming practices and will incorporate the plenum construction for both shafts. The excavation will be backfilled with engineered fill as the collar elevation progresses towards ground level.

9.4.3 Erection of Main Shaft and Ventilation Shaft Headframes

Once the collar areas have been prepared and headframe foundations completed, headframe construction will begin at each shaft location. The collar house adjoining the ventilation shaft headframe will also be constructed (see Section 6.2.2.1). Similarly the WPRB will be used as a collar house; the electrical room and compressor building that adjoin the main shaft headframe will also be constructed.

The main shaft headframe structure will be constructed in its permanent configuration as described in Section 6.2.1.1. The headframe is a reinforced concrete structure and constructed using the slip-form technique. A forming set is advanced at a set advance rate with reinforcing and required openings installed in advance of the form. Concrete is supplied from the on-site ready mix batch plant and poured continuously to ensure a wet joint is maintained and provides the "slipping" requirements of the formwork. The main headframe is of consistent dimension from the collar to the hoist house level, which lends itself well to slip-form construction.

The main headframe design incorporates shaft sinking requirements (temporary hoist, Galloway and auxiliary hoist sheaves) such that there is no requirement for a temporary sinking headframe during construction (see drawing H333000-WP406-20-042-0003, Chapter 17). A temporary waste rock dumping facility and muck bay will be constructed beside the headframe for use during shaft sinking and will be removed following completion of the main shaft. Using the same structure for both sinking and operations will optimize the construction timing and provide a more efficient transition from sinking to operational configuration.

The ventilation shaft headframe will be constructed in its permanent configuration as described in Section 6.2.2.1. The ventilation shaft headframe design incorporates the sinking and permanent requirements with minimal modifications required to change over from sinking to permanent condition as the hoist is common to both phases (see drawing H333000-WP406-20-042-0008, Chapter 17).

The ventilation shaft headframe will be constructed at the same time as the main shaft headframe. Construction will be completed using conventional steel structure construction practices. Consideration has been given in the construction schedule for expected site wind restrictions and crane availability.

9.4.4 Installation of Temporary Hoisting Equipment

9.4.4.1 Main Shaft Sinking Hoist House

A double drum hoist will be installed in a nominal 13 m x 24 m and 12 m high building constructed as an insulated and clad steel frame structure. This building is required for the sinking phase only and houses a 3.66 m diameter double hoist drum for hoisting two 8-tonne buckets for waste rock removal, personnel and materials movement during sinking. An auxiliary hoist is provided to move personnel in the shaft when the buckets are in use, and is available for emergency egress. The building contains all the electrics and a control station. An 8-tonne monorail crane is installed for the installation and maintenance of the hoist. The hoist house is removed at the end of the sinking phase.

9.4.4.2 Temporary Winch Houses

Two sets of temporary winches are installed at both the main shaft and ventilation shaft (i.e., four winch houses in total). The winches are used to hoist the shaft sinking

Galloway stage (see Section 9.4.5.1). Each winch is a 2 m diameter single drum winch.

The buildings are nominally 10 m x 11 m and 9 m high containing all the electrics and a control station. An 8-tonne monorail crane is installed for the installation and maintenance of the hoists. The winch houses will be removed at the end of the sinking phase. Although there are two winch houses required for each shaft in the current design, there is an opportunity in detailed design to combine the winches into a single building. See Figure 9-3 for the arrangement of a typical shaft sinking hoist house.

9.4.5 Shaft Sinking

Once all required facilities for sinking of the two shafts have been commissioned (hoists, winches, sheave arrangements, Galloway, etc.), the two shafts will be sunk in parallel. The sequence for shaft sinking and the transition to lateral development are illustrated in Figure 9-4 to Figure 9-8 and further described in the following sections.

Figure 9-3: Typical Shaft Hoist House

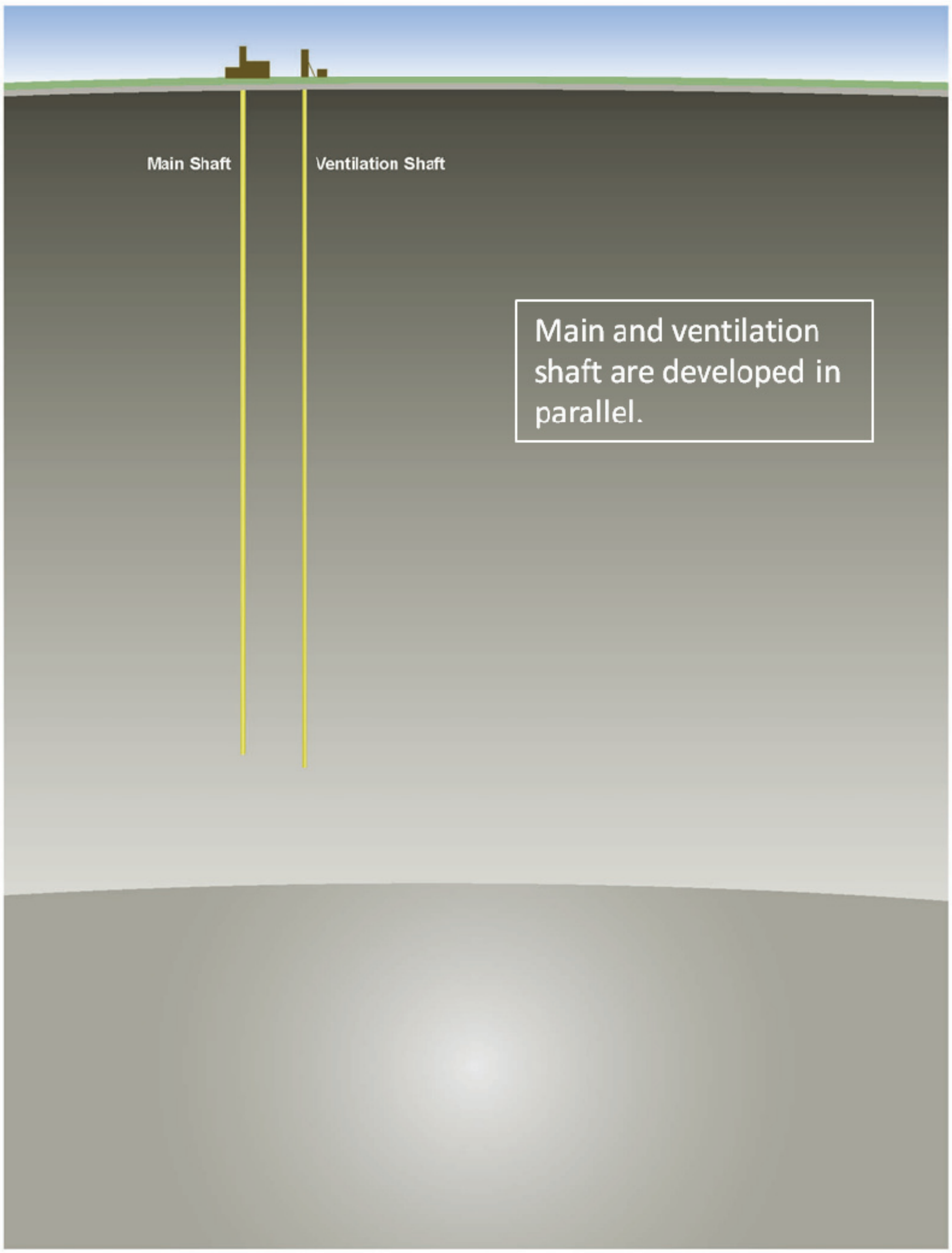


Figure 9-4: Initiation of Shaft Sinking

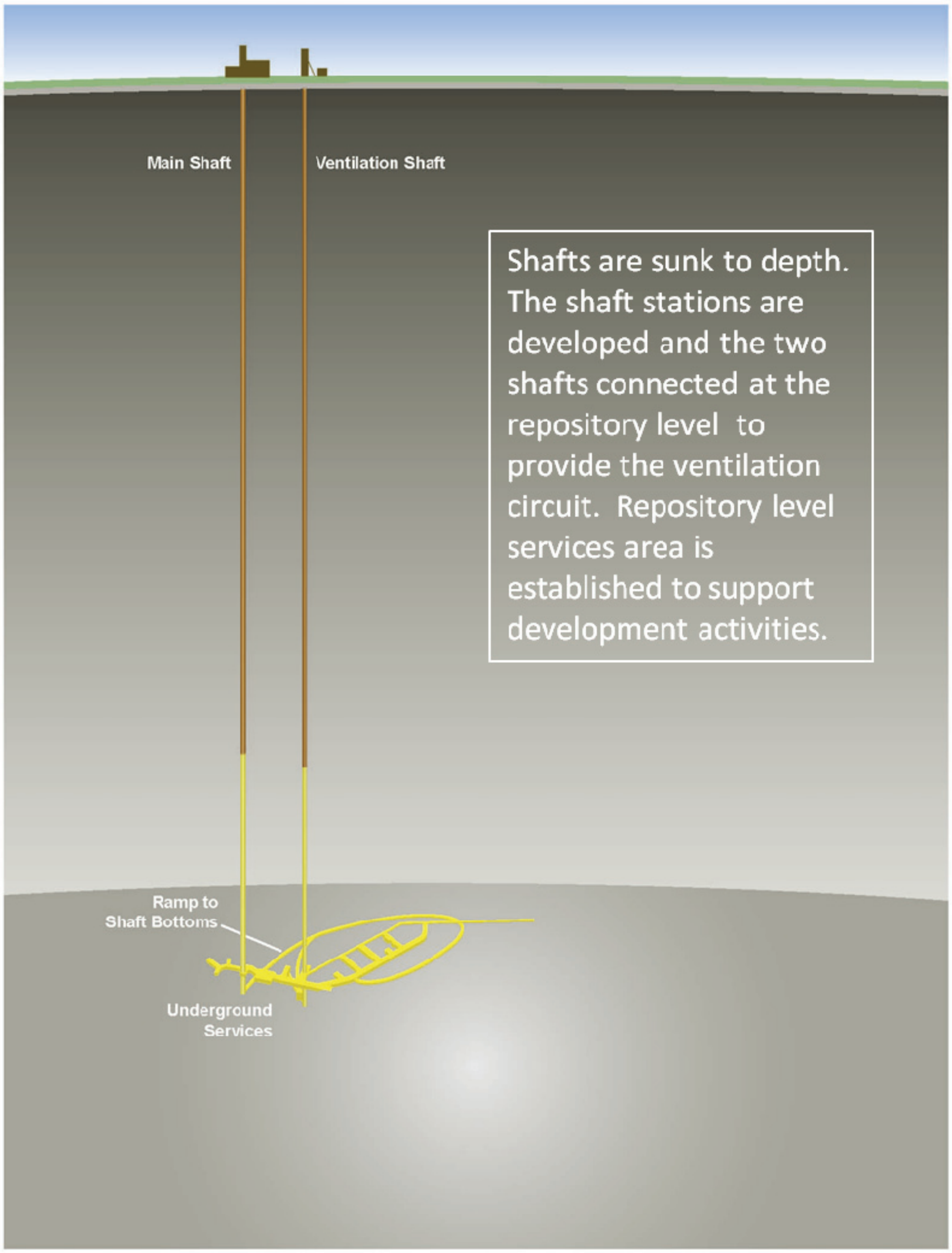


Figure 9-5: Complete Shaft Sinking and Establish Shaft Stations

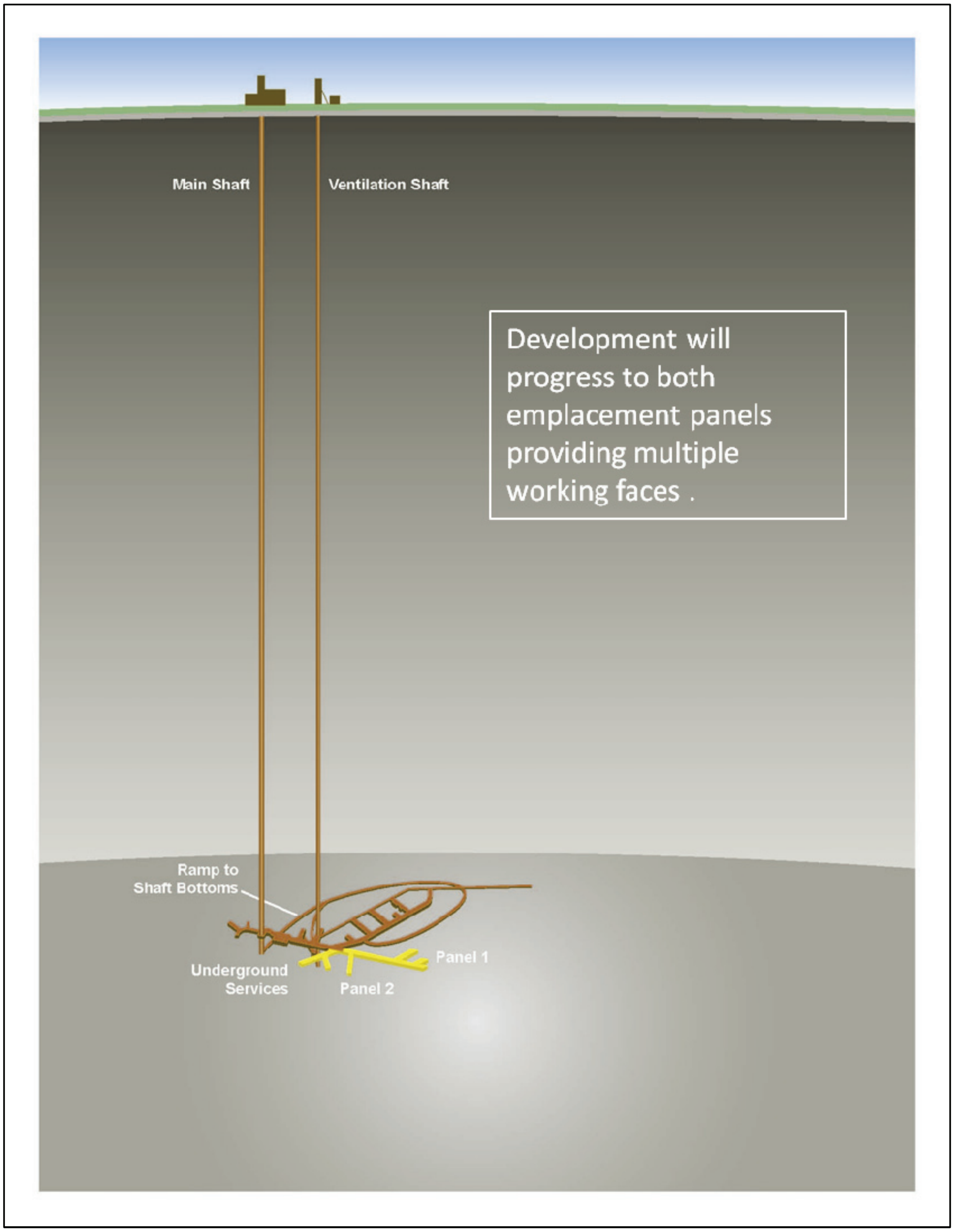


Figure 9-6: Initial Repository Level Development

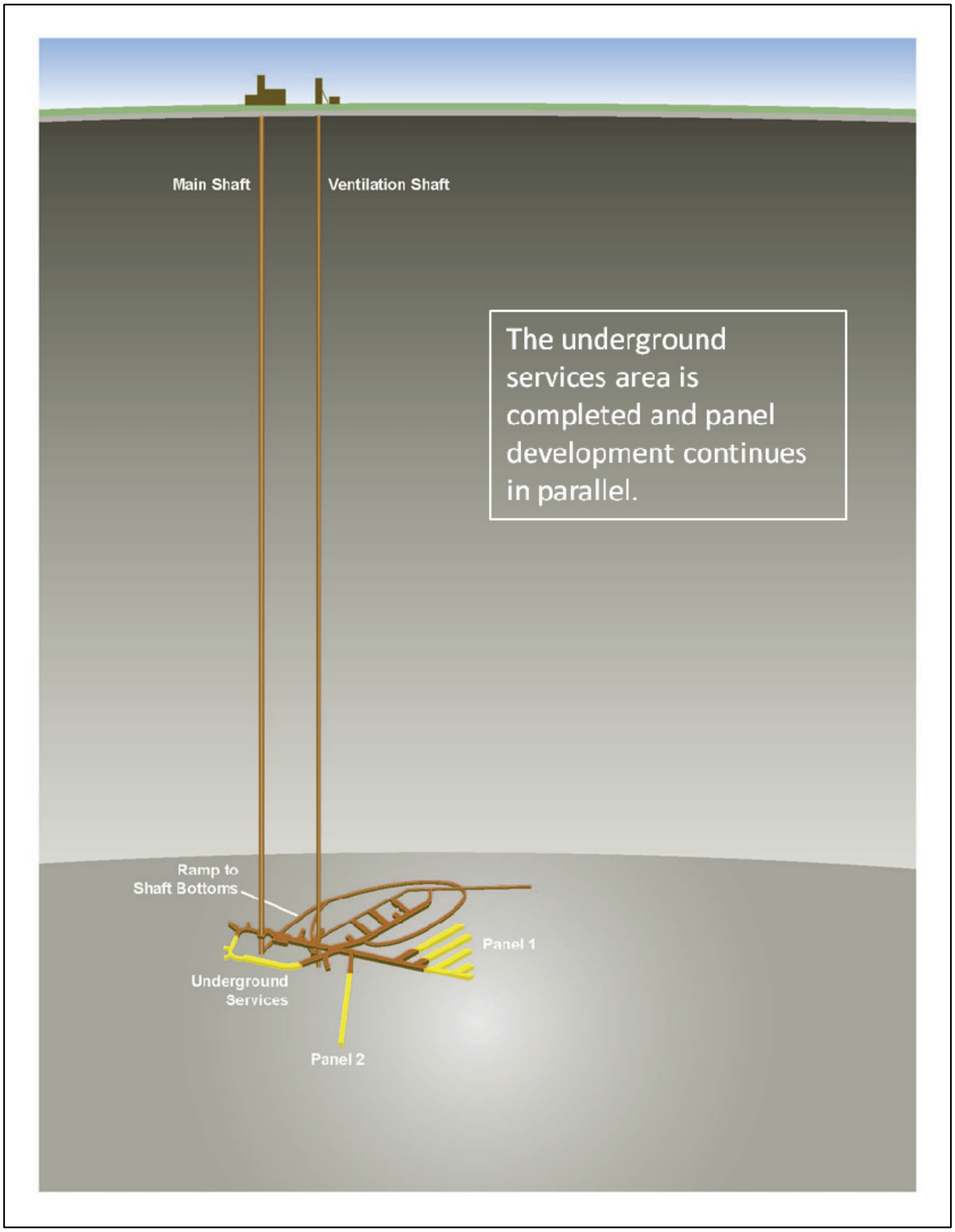


Figure 9-7: Complete Shaft Services Area and Establish Multiple Headings

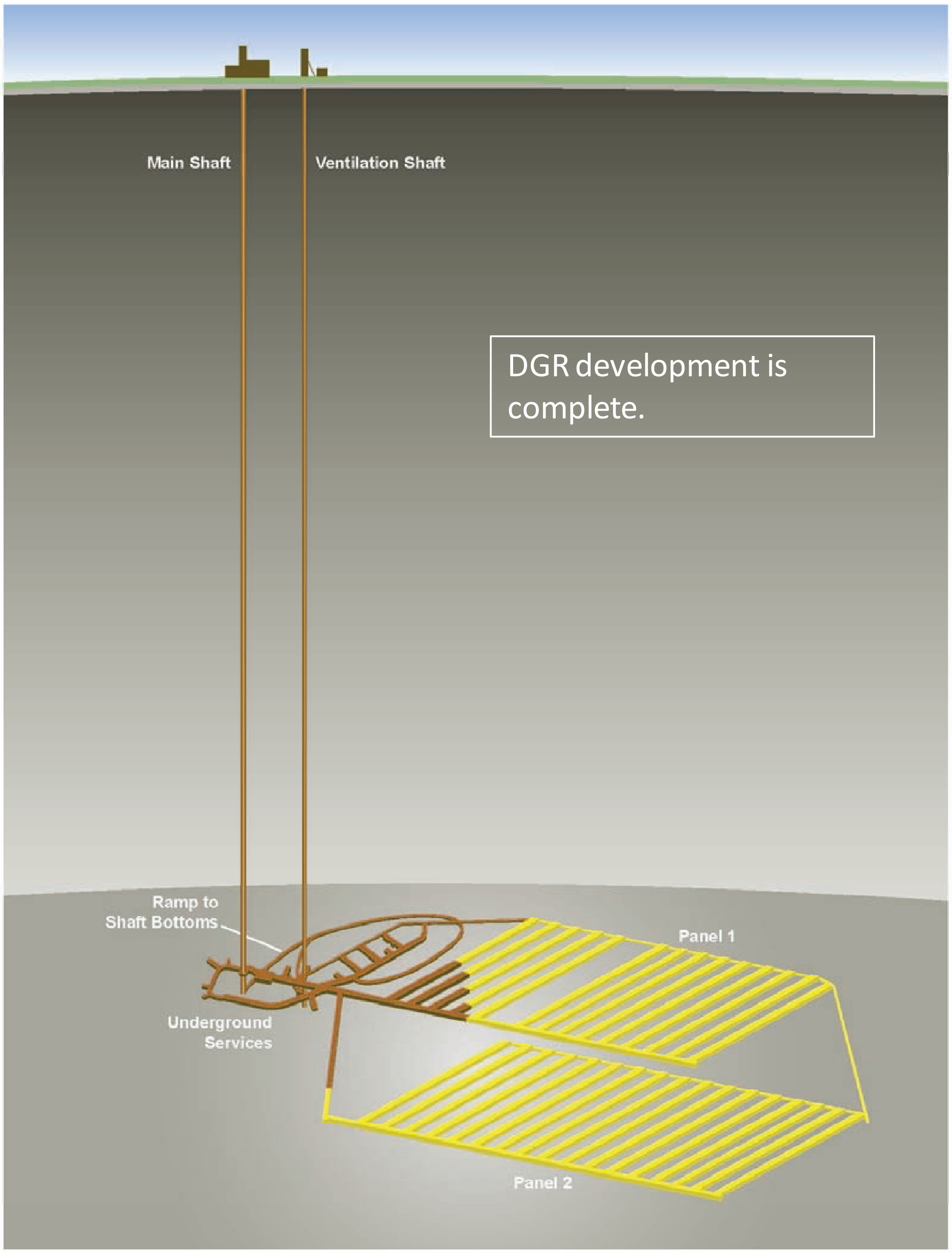


Figure 9-8: Complete Underground Development

9.4.5.1 Description of Shaft Sinking

The shafts will be developed using controlled drill and blast techniques. A sinking Galloway stage, designed and constructed specifically for each shaft, will be used providing multi-level working platforms (see Figure 9-9). The Galloways are lowered and raised, as required, within the shaft during sinking operations using the winches as described in Section 9.4.4.2.

The shaft sinking arrangement for the main shaft is shown in drawing H333000-WP405-20-035-0001 (see Chapter 17). The layout provides for two 8-tonne buckets to hoist waste rock and supplies, ventilation ducting, shaft sinking equipment (mucking machine, drill jumbo, etc.), services and personnel conveyance. The ultimate layout of the shaft for sinking will depend on the final Galloway configuration as designed and constructed by the sinking contractor. A typical shaft sinking layout is shown in Figure 9-10.

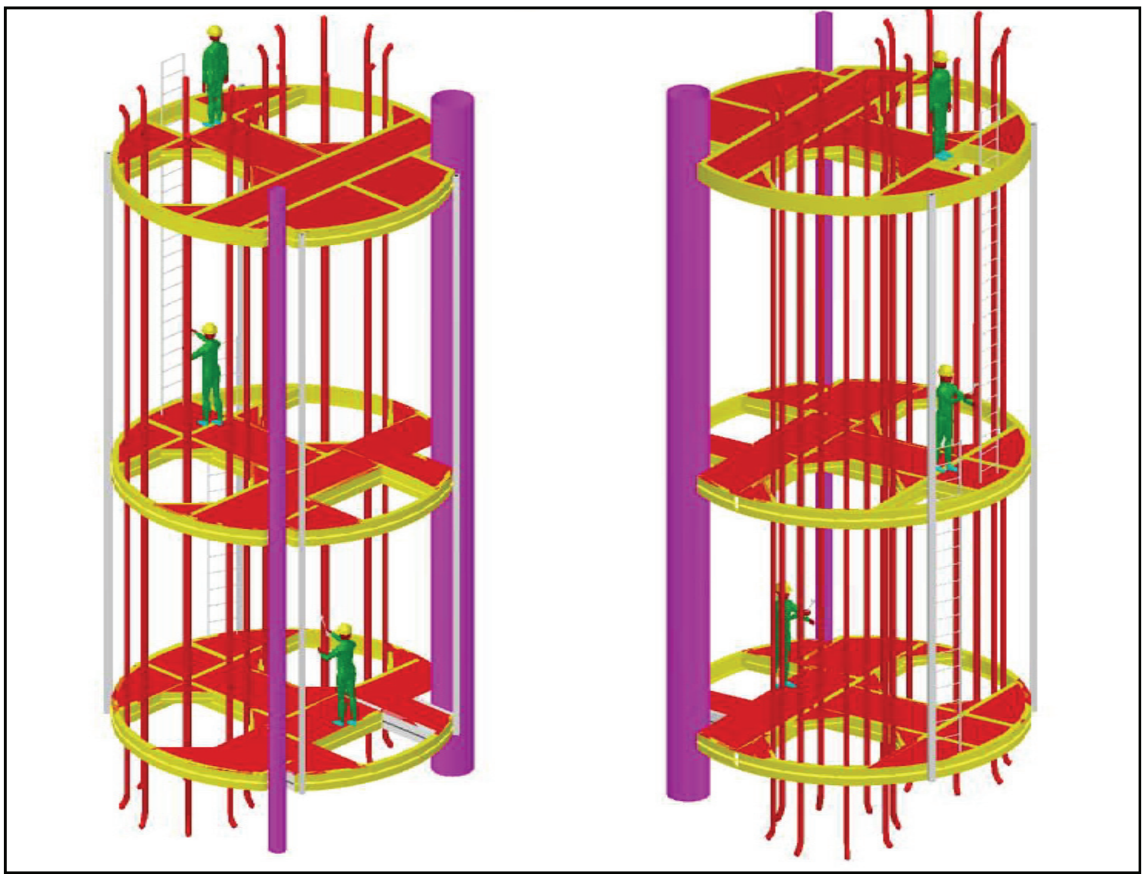


Figure 9-9: Typical Sinking Galloway

Figure 9-10: Typical Shaft Sinking Layout

The shaft development cycle (drill, blast, muck, support and services) is planned to produce a nominal 2.5 m advance per day for each shaft. Drill and blast patterns will be designed to minimize the damage, or overbreak, of the shaft walls. The designs will consider the type of material to be excavated and optimize drill patterns, explosive types and powder factors to achieve this requirement. In the event that groundwater inflows are detected from the probe holes, cover grouting will be employed as described in Section 9.4.1.3.

Mucking will be accomplished through the use of mechanical excavators, typically suspended from the bottom Galloway stage (e.g., brutus mucking machine) and the waste rock placed into the skipping bucket for transport to surface. At surface, the muck will be off-loaded at the headframe through the rock dumping facility (temporary at the main shaft and permanent at the ventilation shaft) and transferred to the muck bay. The waste rock will be handled at surface as described in Section 9.3.3. The majority of the waste rock generated from shaft sinking activities is expected to be reused in the construction of the site.

Ground or rock support for the shafts will be completed in stages (initial rock support and final liner construction) and is described in Sections 9.4.5.3 and 9.4.5.4.

9.4.5.2 Ventilation During Shaft Sinking

The majority of equipment for shaft sinking will be electric/hydraulic with some equipment being diesel-powered. Thus ventilation during shaft sinking is required to:

- Provide fresh air to the workers;
- Clear any residual dust generated primarily by mucking;
- Remove noxious gases after blasting;
- Remove heat generated from any diesel engines and, to a lesser extent, heat from electric/hydraulic equipment such as jumbos and bolters; and
- Dilute and remove any diesel engine exhaust to acceptable levels.

The ventilation system for shaft sinking is temporary and specific to each shaft. Temporary surface fans provide fresh air to the shaft bottom via temporary vent ducting. The exhaust air returns to surface through the shaft opening. Drawing H333000-WP410-20-030-0002 (see Chapter 17) shows the shaft sinking ventilation configuration. The sizing of the surface fans and ducting will depend on the ultimate equipment selection of the contractor. A temporary fresh air heater system will be installed to provide adequate working conditions (above freezing) during the winter months. Inspection, maintenance and monitoring of the ventilation system will be managed by the development contractor.

9.4.5.3 Description of Initial Rock Support

Initial rock support is required to provide safe working conditions for shaft sinking activities in advance of installing the final concrete liner. The design of the support will depend on the type of rock and ground conditions encountered as the shafts advance. In general, the support will include steel or fibreglass dowels (fibreglass dowels are required below a depth of about 180 mBGS to facilitate removal at time of shaft sealing) installed in a set pattern and either mechanically anchored or grouted to the rock. Installation of wire mesh, shotcrete or an initial concrete liner will follow bolting activities to limit the potential for rock spalling.

9.4.5.4 Final Liner Construction

The liner design for each shaft is described in Sections 6.3.1.2 and 6.3.2.2. Concrete will be supplied to the headframe collar houses from the on-site concrete batch plant as required for construction. Slick lines will advance with the services of the shaft to provide a conduit to the working areas in the shaft. Forming and concrete placement will be completed off of the Galloway stage.

Construction of the liner will depend on the detailed design. One possible approach would be installation of initial support and final shaft liner as the shaft advances from the top down. Alternatively, only the initial support could be installed as the shaft advances. In this case, when a large section of the shaft has been excavated, shaft sinking activities would be suspended and the final liner constructed from the bottom upwards by using a concrete slip-forming technique.

9.4.6 Shaft Equipping and Installing Permanent Hoisting Equipment

Upon completion of final liner construction in the shafts, the shaft steel and services will be installed in both shafts. The configuration of the steel sets are shown in Figure 6-8 and Figure 6-10 for the main and ventilation shafts respectively.

The majority of the permanent hoisting equipment will be installed within the main shaft headframe while the shaft is being sunk and commissioned upon the completion of shaft sinking. The main shaft permanent configuration will consist of the two (main and auxiliary) Koepe friction hoists and cages as described in Section 6.3.1.

Following the ventilation shaft sinking activities, the hoisting system will change over to the double drum configuration with two skips (open-top bins for handling rock) for waste rock handling.

9.4.7 Construction of Underground Facilities

There will be a portion of the underground developed off of the shafts utilizing the shaft sinking configuration (buckets) to bring waste rock to the surface while the main development infrastructure is being constructed (e.g., waste rock dump and loading pocket, shaft stations, electrical rooms, etc.).

The majority of the underground development will utilize the ventilation shaft for the transport of waste rock from the repository to surface as described above.

9.4.7.1 Excavation Methods and Installing Rock Support

Excavation of underground openings will be achieved by controlled drill and blast methods. This method involves the drilling of a series of parallel horizontal holes in a predetermined pattern and length. After drilling is completed, each hole is loaded with explosives and a time-delayed detonator, or blasting cap. The detonators in each hole are connected together using either electrical wire or non-electric shock cord. Once connected, the pattern or “round” is detonated from a safe distance in a defined sequence to control fragmentation. Use of delays will ensure proper fragmentation of the blasted rock, control vibrations and achieve smooth wall blasting. To further limit the damage to the side walls and minimize overbreak, a row of smaller diameter holes is drilled along the perimeter of the opening and loaded with lower powder factor explosives.

The blasted rock will be excavated at the face with Load Haul Dump (LHD) equipment and transferred to low-profile underground rock trucks. The trucks will transfer the rock to the waste rock dump on the repository level where it will be transferred by raise to the loading pocket. The waste rock raise provides surge capacity to allow a buffer between development and hoisting activities. The rock is loaded into the skips at the loading pocket. The loading pocket is configured with a control chute at the bottom of the rock raise, a vibrating feeder, diverter slide and two measuring flasks to control both volume and weight of the material being fed to the skip. The measuring flask then dumps the muck into the skip when it is in position. Installation of ground support completes the development cycle as shown in Figure 9-11.

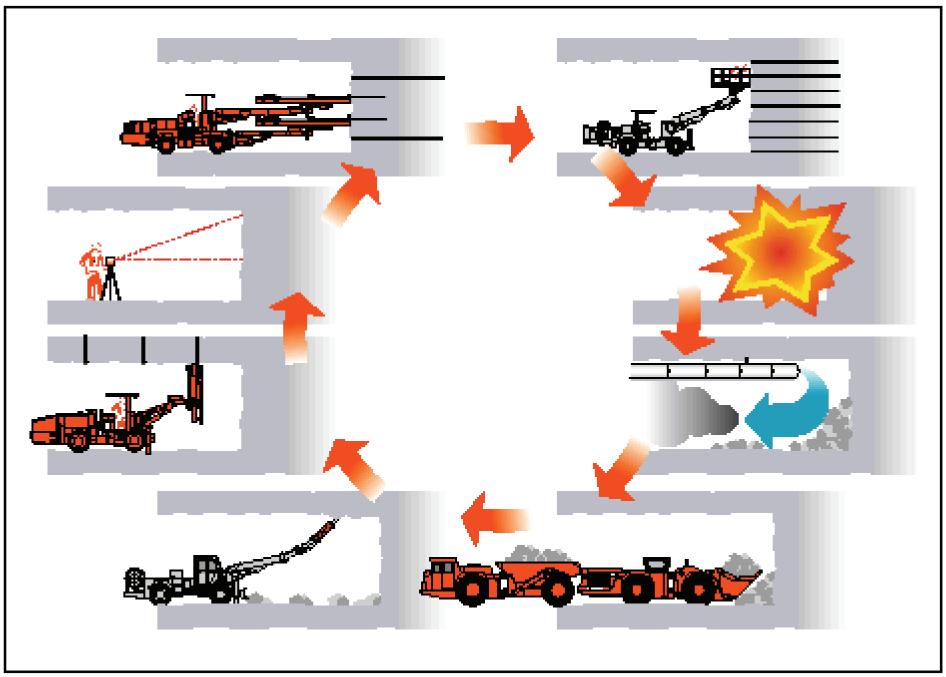


Figure 9-11: Typical Drill and Blast Excavation Cycle

Practical advance lengths and excavation opening sizes have been selected, which will permit an efficient excavation and support sequence. For project planning purposes, it is currently assumed that full-face excavation will be used for development of the emplacement room and access tunnel excavations and that advance lengths will be on the order of 4 m. In some areas, partial-face excavation methods may be required to reduce risk of rock falls. Partial-face excavation can be achieved by limiting the height and/or width of the excavation face.

Given the size of the openings for the DGR, rock support will be required between advance lengths in the DGR. The timing and magnitude of that rock support will depend upon in-situ stress levels, their orientation relative to the excavation opening, strength properties of the rock mass around the excavation, the size of the excavation and the time available for its installation. For a number of reasons, the most important of which is worker safety, timely installation of overhead rock support is required.

Initial rock support will consist of rock dowels mechanically anchored or grouted to the rock to minimize spalling. This will be performed using mechanical bolting equipment such that the operator is working under supported ground and working towards the excavation face. Additional support will likely be required to protect against spalling in between the bolts. Depending on the in-situ conditions, and the expected operating life of the opening, this could be either welded wire mesh or shotcrete. In areas of high stress and long operating requirements (e.g., shaft station, maintenance shop, etc.), cable bolting or metal strapping may be installed.

9.4.7.2 Development Sequence

The development sequence at the repository level is illustrated by Figure 9-4 to Figure 9-8. The initial construction will commence off of the shaft stations and develop the permanent electrical, communications, repository level rock handling, refuge station and pumping facilities. Connection of the two shafts at the repository level is critical to establish the main ventilation circuit. The main lateral development equipment (multiple boom jumbos, LHDs, trucks, etc.) will be mobilized to the repository level near the end of this construction stage.

Lateral development will have a progressive build-up of equipment as space becomes available. The construction of the remaining services areas will be completed early in this phase to provide support to the ongoing development program. Key areas include the explosives and cap magazines, mobile equipment maintenance area and underground refuelling station. The shaft bottom ramp is developed off both shaft bottoms and tie into the repository level.

As the services area is further developed, the panel access and exhaust ventilation tunnels will be developed to provide access for emplacement room development. This will allow for multiple development headings and optimizes advance rates. For the majority of the lateral development schedule, there will be a minimum of 6 development headings, with 4 headings advanced per day (approximately 2,100 tonnes of development waste rock produced and hoisted per day). As the tunnels advance, permanent services (power, air, water, communications, etc.) will be installed.

Once development of a series of emplacement rooms is complete, the concrete floor will be constructed in these rooms. Slicklines in the ventilation shaft will be used to transfer ready-mixed concrete from the surface batch plant to the ventilation shaft station at the repository level. Low profile transmixers will transport the concrete from the ventilation shaft to the emplacement rooms. Formwork will be used to control the placement and grade of the floor to maintain the required tolerances (0.25% grade towards the access tunnel) for package emplacement. Once a panel of emplacement rooms has been constructed the concrete floor in that panel access tunnel will be installed. Where required, steel rail will be embedded into the concrete floor of the emplacement rooms and access tunnels. All temporary services will be removed from the emplacement rooms and ventilation of rooms will be stopped. The emplacement rooms with the permanent metal ducting in place will be designated confined spaces until required for waste package placement.

9.4.7.3 Ventilation During Lateral Development

The ventilation system for the DGR development will require a staged approach until the exhaust ventilation tunnel is fully developed. In all stages, the ventilation system will be maintained through a formal inspection and maintenance program and monitored to ensure performance (e.g., alarms, flow rate and atmospheric monitoring, etc.).

During initial development, fresh air will flow down the main shaft and travel through the access tunnels. Fans and temporary ducting provide fresh air to the working areas as required. The exhaust air will be drawn through return ducting back to the ventilation shaft (see Figure 9-12).

A nominal 80 kW fan is mounted in the temporary exhaust duct near the entrance to the room. The duct on the inlet side of the fan will extend towards the room excavation face. However, due to potential for damage to the ducting by fly-rock during blasting, the duct will be kept a suitable distance from the excavation face. To provide fresh air directly to the excavation face, a smaller 40 kW fan will be located between the end of the exhaust duct and room entrance and a temporary flexible duct will be used to direct fresh air to the working area at the face. The construction ventilation system will be designed so that the exhaust system is capable of removing more air than what the fresh air delivery system can supply. This will prevent recirculation of "dirty air" in the room which may lead to unsatisfactory working conditions. Ventilation airflows will be maintained so as to not generate dust from the floors (nominally less than 6 m/s). Furthermore, air foggers and water sprays will be used to maintain appropriate moisture conditions at the muck pile to reduce fugitive dust during mucking.

Once the exhaust ventilation tunnel is completed, connecting both Panel 1 and Panel 2, the return ducting from the emplacement room can end at the closest room that is connected to the exhaust ventilation tunnel (flow-through).

As the excavation face progresses, the ducting for exhaust and fresh air are also advanced until the room is constructed and ties into the exhaust ventilation tunnel. The end-wall is constructed and the ventilation regulator installed. All temporary fans and ducting are removed from the room.

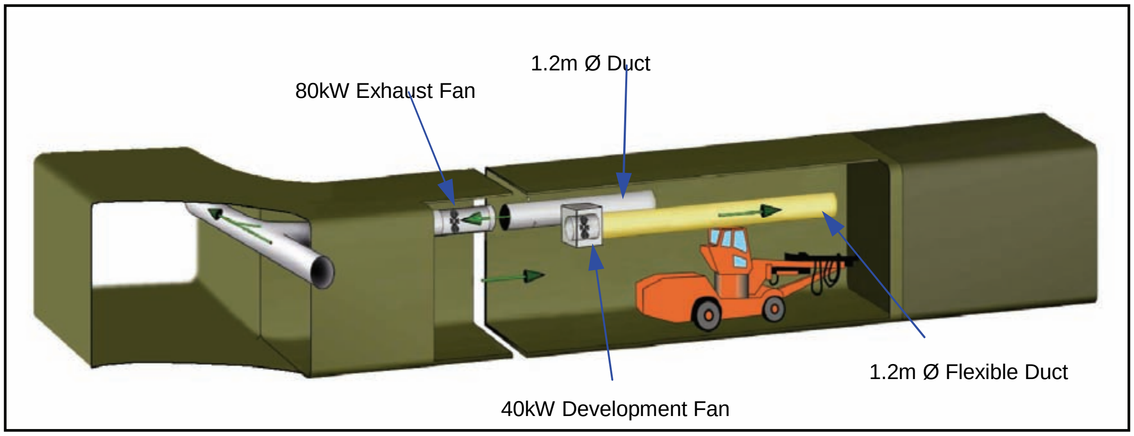


Figure 9-12: Ventilation During Construction

9.4.7.4 Underground Services During Construction

Services during underground construction and development (power, communications, water management and compressed air) will utilize the same services infrastructure that will be used during the operations phase (refer to Section 6.3.10) and temporary surface construction services as described in Section 9.3.1. Installation of services underground during construction, following completion of the shaft, will remain for the life of the facility.

During shaft sinking, temporary services will be provided in the shaft until the final liner is installed and the shaft is equipped. At the repository level, services are advanced from the shaft stations with the advancement of tunnel excavations.

Water management during shaft sinking and initial repository level construction will require temporary sumps for the collection and transfer of collected service water to surface via the shaft discharge column. This waste water stream is from the sinking equipment and is expected to require treatment prior to discharge into the surface facilities stormwater system as described in Section 6.3.10.4 and Section 9.3.2. The water treatment requirements will be provided by the selected contractor. As development progresses, the permanent sump system will be constructed and commissioned and the use of temporary sumps will no longer be required.

9.4.8 Surface Facilities Construction

An illustration of all permanent DGR surface facilities is shown in Figure 6-1 and the design of the various buildings and facilities are described in Section 6.2. Most of the surface facilities will be constructed in their permanent configuration around the time that shaft sinking is complete. Significant exceptions will be the WPRB, the office, control room and amenities building, permanent roadways and the crossing to the WWMF. The following sections describe construction activities on these buildings and facilities to occur near the end of DGR construction.

9.4.8.1 WPRB

The superstructure for the WPRB and maintenance and storage area will be erected around the time that the main shaft headframe is constructed and this building will act as the collar house during shaft sinking and lateral development. When underground lateral development work is largely complete, the WPRB will be converted from a collar house to a building that is equipped and ready for waste handling operations. Major equipment to be installed includes the 40-tonne overhead gantry crane, and mobile equipment to be used for handling waste packages.

9.4.8.2 Offices, Main Control Room and Amenities Building

The offices, main control room and amenities building will be constructed at the location of the main shaft rock dump area. Thus, this building can only be constructed after sinking of the main shaft is complete and main shaft headframe is changed over to its permanent hoisting configuration.

9.4.8.3 Roadways

Roadways will be developed as required to support construction activities. During construction, the majority of roads will be granular base construction. Roads will be maintained with graders and water trucks will be used to manage fugitive dust.

The majority of permanent roads around the shafts and WPRB will be paved. Permanent roads for the operations phase of the DGR will be constructed near the completion of construction to minimize the potential for damage from heavy construction equipment.

9.4.8.4 Crossing to WWMF

The crossing and connection to the WWMF will be one of the final construction activities to ensure the DGR project site will be maintained as a separate construction island. The crossing from the WWMF to the DGR is described in Section 6.2.4.7. Culverts will be installed in the north and south railway ditches of the abandoned rail bed to maintain existing stormwater flows. Excavated material from the shaft sinking and lateral development activities will be used to construct the embankment using conventional construction equipment (trucks, bulldozer, compactors, etc.). Granular subgrade will be used for the road and topped with asphalt pavement. The concrete barriers will be installed and the fencing to connect to the existing WWMF fenceline established.

The abandoned railbed crossing is planned at a location along the south railway ditch that is recognized as a fish habitat. The construction work necessary to install the crossing will be undertaken in a manner that limits the disturbance of this habitat.

9.4.8.5 Connection of Services

Site services are described in Section 9.3.1. All services, with the exception of sewage, will be installed as part of the site preparation activities. Connection of the DGR site sewage system to the existing Bruce Power system will be performed near the end of construction and this connection will only be used to support DGR operations.

9.4.9 Occupational Safety

9.4.9.1 Conventional Safety

As identified in Section 9.1.1, Ontario's Occupational Health and Safety Act will be applicable to the DGR construction staff. The likely conventional safety hazards posed by the site preparation and construction activities have been assessed and are reported in a separate report (NWMO11ac). Site specific safety procedures (e.g., personal protective equipment, lock and tag, etc.) are referenced as part of the Health and Safety Management Plan (see NWMO11a).

9.4.9.2 Radiological Safety

There are no anticipated radiological hazards during DGR site preparation and the above ground activities for the construction of the DGR. However, during construction of the shaft and the underground portion of the DGR, radon could potentially pose a radiological hazard. A radon assessment in the underground environment was conducted, based on the rock properties at the DGR site (NWMO11ae). Considering the Derived Working Limit from the Canadian Guidelines for Management of Naturally Occurring Radioactive Materials of 150 Bq/m^3, this assessment concluded that radon concentration is expected to be well below this limit, and there is no need for radon monitoring or development of an action level during construction. Radon will, however, be checked periodically during construction to confirm this conclusion.

Based on the above, there are no anticipated radiological hazards during DGR construction.

9.4.10 Contingency Plans during Construction

During the detailed design stage, a detailed risk assessment will be conducted based on typical hazards associated with construction projects involving activities, construction methods and equipment similar to those to be used for the DGR project. Once the risks have been determined, specific contingency plans will be developed. The Design and Construction Phase Management System document (NWMO11a) refers to the timing of when contingency plans for DGR construction will be prepared.

9.4.11 Commissioning

Commissioning plans for the DGR project will be developed in accordance with the commissioning program referred to in the Design and Construction Phase Management system document (NWMO11a). These plans will include human factors considerations.

Commissioning of the DGR project will be staged, with initial commissioning of key equipment and facilities occurring early in the construction program to support development of the repository. This would include the temporary main shaft hoists, the ventilation shaft hoists, ventilation systems, headframe and collar houses, etc. Commissioning plans will be developed after vendor selection, as they will incorporate vendor recommendations for selected equipment.

Although many of the systems are commissioned to support construction, the operating requirements for these equipment and facilities for operations will be different and will require a secondary commissioning plan. Commissioning during the construction phase brings the equipment and facilities to a point of mechanical completion. Commissioning activities will take place both underground and at surface as described in the following two sections.

9.4.11.1 Underground

At the end of underground construction, the waste rock handling system (dump through to the loading pocket) at the vent shaft will be put into a state suitable for extended care and maintenance. The system will be left in a state that will allow it to be reactivated, should it be required either for excavation maintenance or for potential future expansion of the repository. The explosives and detonator magazines will be decommissioned and the space prepared for general storage.

The concrete/shotcrete system in the ventilation shaft will also be prepared for extended care and maintenance for any potential future use. The ventilation system will be calibrated to support operations requirements and re-commissioned to these requirements.

The operating underground fleet of equipment for emplacement will be brought underground, assembled and commissioned. This includes both the gantry cranes and the rubber-tired mobile equipment. The contents of the various offices, geoscience laboratory and the workshop in the underground shaft and services area and any small equipment for the emplacement rooms will be brought underground and commissioned.

Once these tasks are complete, the underground repository will be mechanically ready for hand-over to OPG Operations.

9.4.11.2 Surface

The surface equipment and systems (hoists, electrical, ventilation, etc.) will be configured for operational requirements and commissioned accordingly. Material storage areas for key materials such as spare shaft sets, spare hoist ropes and reels and the one removed ventilation shaft skip, which will have been replaced by a cage, will be established as part of the surface yard restructuring.

9.5 Potential Environmental Effects and Monitoring Programs

Table 13-1 of the EIS (OPG11a) summarizes residual adverse effects during site preparation and construction in the areas of air quality; loss of a small portion of vegetation and habitat for some species within the DGR project footprint; noise levels at Baie du Doré. Although each of these effects was assessed to be not significant, monitoring programs (NWMO11at) are planned to confirm these predictions and assess the effectiveness of the mitigation measures. Monitoring programs are also planned to verify predictions that the DGR project will not result in adverse effects to the environment.

10. OPERATIONAL PROGRAMS

In support of the DGR, programs to protect the environment, and health and safety of the public and the workers during operation will be in place. These programs will assure compliance with applicable provincial and federal legislation, and applicable regulations.

10.1 Radiation Protection Program

OPG, as Operator of the DGR, will use its existing Radiation Protection Program (N-PROG-RA-0013) (OPGa) as required by Section 4 of the Radiation Protection Regulations (SOR/2000-203). The program will be used to manage radiological risks that could contribute to public and occupational radiation doses when the DGR Facility becomes operational.

The Radiation Protection Program will achieve and maintain high standards of radiation protection including the achievement of the objectives listed below.

a) Control occupational and public exposure by:

- Keeping individual doses below regulatory limits;
- Avoiding unplanned exposures;
- Keeping individual risk from lifetime radiation exposure to an acceptable level; and
- Keeping collective doses ALARA, social and economic factors taken into account.

b) Prevent the uncontrolled release of contamination or radioactive materials from the DGR site through the movement of people and materials.

c) Demonstrate the achievement of (a) and (b) through monitoring.

This program complies with the CNSC requirement that all licensees implement a radiation protection program and establish a quality program.

This program is designed to comply with the radiation protection program requirements of the following acts and regulations as applied to licensed OPG facilities and licensed OPG activities:

- Nuclear Safety and Control Act (NSCA97);
- General Nuclear Safety and Control Regulations (SOR/2000-202);
- Radiation Protection Regulations (SOR/2000-203);
- Class I Nuclear Facilities Regulations (SOR/2000-204); and
- Nuclear Substances and Radiation Devices Regulations (SOR/2000-207).

Occupational dose management will be through the use of personal dosimeters worn by workers. All radiation doses measured and assigned to workers at the DGR will be performed by a CNSC Licensed Dosimetry Service Provider.

Routine radiological surveys will be performed throughout the DGR to support the early discovery of unexpected hazards and to identify longer term trends in hazard conditions. The location, type and schedule of routine surveys will be approved by the Responsible Health Physicist.

Currently, OPG does not have a program that monitors for radon exposure to workers. Estimates of radon concentration have been made for the DGR and are detailed in the Radon Assessment report (NWMO11ae). It has been concluded that radon concentration will be well below the allowable limit of 150 Bq/m^3 (HC00).

As required by the Radiation Protection Regulations, action levels will be proposed for measured radiological parameters in support of the operating licence application, as appropriate.

10.1.1 Keeping Doses ALARA

Exposure to radiation is managed through the following processes:

- Limiting individual worker dose;
- Establishing facility design optimized on the basis of ALARA considerations;
- Assessing hazards for planning and to maintain knowledge of conditions; and
- Planning and performing radioactive work to keep exposures ALARA and avoid unplanned exposures.

The program elements listed in this section comply with the regulatory requirements to keep exposures ALARA, implement control of occupational and public exposure, and plan for unusual situations.

10.1.2 Control of Radiation Exposure and Contamination

A key practice in maintaining control of radiation exposure and contamination is through the use of zoning. As per OPG's procedure on Radiological Zoning, Personnel/Material Monitoring and Transfer Permits N-PROC-RA-0014 (OPGb), the zones are defined below.

- Zone 1 is a clean area inside the zoned area that is considered equivalent to the public domain.
- Zone 2 is an area inside the zoned area that is normally free of contamination but is subject to infrequent cross-contamination due to the movement of personnel and equipment from contaminated areas. This zone may contain enclosed, sealed radioactive systems and sources (i.e., active ventilation ducts, radioactive monitoring pipelines, and constancy check sources).
- Zone 3 is an area inside the zoned area that contains systems and equipment that may be sources of radiation or contamination.

Generally accessible areas outside the DGR Facility fence will be maintained at Zone 1 within the dose rate constraint ≤ 0.5 μSv/hr. All spaces within the DGR Facility perimeter will be classified in accordance with the potential for contamination. The movement of air due to ventilation will be generally from a lower to a higher zone in the DGR.

As described in Section 6.10.1, the surface facilities associated with handling of radioactive waste will be classified as Zone 2, including the crossing from the WWMF to the WPRB, the WPRB and the shafts. Office and amenities area will be designated as Zone 1, with a boundary between Zones 1 and 2 located within the amenities area for movement and tracking of personnel. Details of surface facilities zoning are illustrated in Figure 6-29. All underground facilities of the DGR will be classified as Zone 2; underground lunch room will require the use of the whole body and small article monitors. There will be no Zone 3 areas in the DGR either above ground or underground. Rubber areas[1] will be established, as required, within the Zone 2 areas, if a higher potential for contamination is predicted during a specific waste emplacement or maintenance activity, or if loose contamination is found.

Skill and knowledge of workers acquired through training and their diligence in keeping doses ALARA, and exercising good contamination control practices constitute the

[1] 'Rubber area' is an area set up with barriers and warning signs to prevent inadvertent access, to contain loose contamination, and to prevent its spread. Protective footwear is necessary in a rubber area.

primary basis for radiological safety during operation. These good practices may entail the use of additional effort in detecting and controlling contamination as prescribed by the OPG procedure on Radiological Zoning, Personnel/Material Monitoring and Transfer Permits, N-PROC-RA-0014 (OPGb). Furthermore, inter-zonal monitoring is the final barrier to the spread of radioactive contamination to the public domain.

10.2 Conventional Occupational Health and Safety Program

The operation of the DGR Facility will be regulated under the Occupational Health and Safety Act (Ontario) (OHSA90). The worker health and safety aspects under the Mines and Mining Plants Regulations (Reg. 854) under the Occupational Health and Safety Act will be applicable.

An overall Occupational Health and Safety Program will be implemented for the DGR that will meet the requirements of OPG's Environmental, Health and Safety Management Program W-PROG-ES-0001 (OPGc) applicable to its nuclear facilities. The program will also be consistent with the OPG Health and Safety Policy OPG-POL-0001 (OPGd) and the OPG Nuclear Safety Policy N-POL-0001 (OPGe).

The goal of OPG's conventional safety program is to ensure workers work safely in a healthy and injury-free workplace by managing and mitigating risks associated with activities, products and services of OPG operations. Risk reduction is primarily achieved through compliance, by competent workers, to effective operational controls, developed through effective risk assessment and safe work planning.

OPG's conventional safety program is intended to align with a number of internal and external specifications or standards. The conventional safety program is consistent with OPG's management system and British Standards Institution's Occupational Health and Safety Assessment Series (OHSAS) 18001 Management System Specification. The OPG management systems and OHSAS 18001 are based on a Plan→Do→Check→Review cycle.

OPG's conventional safety program results in:

- Compliance with applicable legislative, corporate and nuclear business requirements;
- Identification of continuous improvement opportunities, through a regular cycle of Plan→Do→Check→Review, and annual business objectives to ensure conventional safety risks are appropriately managed to achieve the annual safety performance targets; and
- Application of sound business management processes to the management of conventional safety risks.

10.2.1 Hazardous Materials Program

The DGR Facility will contain a variety of non-radiological materials typically found in industrial buildings. The handling of hazardous materials will be controlled and will meet provincial regulations, in particular the Occupational Health and Safety Act (OHSA90) and the Environmental Protection Act (Ontario) (EPA90) for non-radiological hazards).

Material Safety Data Sheets for hazardous materials will be readily available as required by Workplace Hazardous Materials Information System legislation. Persons who use or handle hazardous materials will be trained in the procedures for the safe use, storage, handling and disposal or the hazardous material.

10.2.2 Personal Protective Equipment

OPG will use their existing Safety Management System Program OPG-HR-SFTY-PROG-0001 (OPGf) to govern the selection, use and maintenance of PPE for the above ground portion of the DGR. For radiological hazards above ground, OPG's procedure N-PROC-RA-0025 (OPGg) on selection and use of radiological PPE will be used.

For underground operation, the requirements for PPE under the Mines and Mining Plants Regulations (Reg. 854) will be complied with.

10.3 Environmental Protection Program

Environmental protection policies, programs and procedures will be established and will meet the requirements of the:

- Environment Policy OPG-POL-0021 (OPGh);
- Biodiversity Policy OPG-POL-0002 (OPGj);
- Land Assessment and Remediation Policy OPG-POL-0016 (OPGk);
- Spills Management Policy OPG-POL-0020 (OPGm); and
- Policy for Use of Ozone Depleting Substances OPG-POL-0015 (OPGn).

Execution of the program will be accomplished through an integrated set of documented activities, typical of an Environmental Management System which will be aligned with the CNSC regulatory standard S-296 (CNSC06c) and the International Organization for Standardization (ISO) standard 14001 and will meet the requirements of OPG's Environmental, Health and Safety Management Program W-PROG-ES-0001 (OPGc).

10.4 Monitoring Program

The DGR site will be monitored during site preparation, construction, and operation. There will, therefore, be a substantial database of information on the impacts of the DGR on the environment before closure. A long-term monitoring plan for monitoring beyond closure will be considered, based on information and technologies then available, and in consultation with stakeholders. The monitoring plan will address radiological contaminants, chemical contaminants and physical stressors that may present a risk to either human health or non-human biota.

The objectives of the monitoring program during operation of the DGR will be:

- To assess performance of various structures, systems, equipment and components relative to design specifications and baseline conditions so safety of DGR workers can be ensured;
- To monitor changes in underground rock/excavation conditions (e.g., rock movement, stress) over time so that there is sufficient advance warning of any potential unstable rock conditions;
- To assess preclosure safety and environmental performance relative to defined standards or limits, and baseline conditions; and
- To monitor for changes in groundwater quality due to DGR operation so risks to human health or non-human biota can be determined and addressed.

10.4.1 Radiological Monitoring Program

As part of the Environmental Management System, a monitoring program that meets the intent of the CSA standard N288.4-10 for radiological monitoring (CSA10), will be implemented to:

- Estimate actual or potential doses to critical groups and populations from the presence of radiation fields or release of radioactive materials due to DGR operations in the environment;
- Provide data to confirm compliance of the DGR Facility with release guidelines, regulations and approval conditions to provide public assurance of compliance; and

- Provide a check, independent of effluent monitoring, on the effectiveness of containment and effluent control.

A pre-operational monitoring program will be carried out to determine background levels for later comparison.

The monitoring program for radiological contaminants will include direct measurements and/or environmental sampling to assess the most significant radionuclides and pathways. Monitoring for tritium, C-14 and gross-beta will be conducted at appropriate points, potentially including vent exhaust, surface water and groundwater.

Radiation monitoring is also discussed in Section 6.11.

The media sampled, the frequency of sampling, and the monitoring locations are documented in the DGR EA Follow-up Monitoring Program (NWMO11at).

10.4.1.1 Groundwater Monitoring Program

OPG has an existing groundwater monitoring program in place based on a network of groundwater sampling holes for the WWMF as described in Section 2.5.2.2. Under that program, a new network of groundwater sampling holes will be established for the DGR in a manner that allows distinguishing any groundwater impacts due to the operation of the DGR from any impacts due to the operation of the WWMF and the nuclear power plants on the Bruce nuclear site. A groundwater monitoring plan specific to the new groundwater sampling holes network for the DGR will be prepared prior to obtaining the operating licence.

10.4.1.2 Geotechnical Monitoring Program

A geotechnical monitoring program will be implemented during the operational phase of the DGR, as indicated in Section 6.3.9, to assess performance of openings and rock support systems.

The technical details of the program will be developed closer to obtaining the operating licence for the DGR.

10.4.1.3 Underground Air Quality Monitoring

The monitoring program for the DGR will include underground air quality monitoring to ensure that the health and safety of personnel within the DGR are not compromised. The monitoring program will ensure that airflows remain adequate for the equipment or activity in active work areas and there is no accumulation of toxic, asphyxiating, or radioactive gases, or flammable and explosive gases per Section 294 of Mines and Mining Regulations (Reg. 854).

Radon gas will be monitored underground periodically to confirm the conclusions of radon assessment (NWMO11ae) that the concentrations are well below the allowable limits. The frequency of monitoring will be determined based on the results of radon monitoring during construction.

Monitoring of airflow, CO, NO_2 and explosive gases including methane and hydrogen, will be conducted underground as described in Section 6.12. The monitoring installed at the main shaft during construction (measuring airflow, temperature, relative humidity, etc.) will be left in place for operations.

Air quality monitoring will be conducted prior to entering a confined space or for a waste-filled emplacement room that has been closed, using a temporary barrier, ventilation bulkhead or shielding wall during the preclosure period and prior to installation of access tunnel closure walls. All measurements will be monitored remotely on surface at the main control room and will also be available to be monitored underground.

10.5 Staffing and Training Program

A staffing and training program will be developed to ensure the presence of a sufficient number of qualified workers to carry out activities safely and in accordance with the Nuclear Safety and Control Act (NSCA97) and its regulations.

A minimum number of workers with specific qualifications (known as the minimum staff complement), will be identified by a systematic analysis to ensure that there are adequate staffing levels to successfully respond to all credible events, including the most resource-intensive conditions for the DGR Facility. The minimum staff complement may differ depending on the various operational activities of the DGR Facility.

Training, meeting the requirements of OPG's Training Program N-PROG-TR-0005 (OPGp), will be established and maintained to ensure:

- Provision of line input to nuclear programs and training to assist with development and implementation of training and qualification programs;
- Reinforcement of management expectations and standards; and
- Establishment of qualification requirements.

Only qualified staff will be assigned to work on tasks independently. Staff will be skilled and knowledgeable to perform the tasks assigned to them.

A systematic approach, based on analysis of job performance requirements, will be used to determine if training is required and to guide training development, design, implementation, and evaluation.

Effective performance-based training and qualifications based on systematic approach principles will include:

- Identification of position-specific training (based on analysis of job performance requirements and initial qualifications of trainee);
- Design of training curriculum with explicit learning objectives and appropriate trainee evaluation tools;
- Development of appropriate training content to meet the curriculum;
- Implementation of training as designed and developed;
- A mechanism to ensure that the trainee masters the learning objectives before working independently; and
- Training effectiveness evaluations to maintain and improve training and job performance.

10.6 Fire Protection Program

The DGR will use the NWMD Fire Protection Procedure W-PROC-ES-0011 (OPGq) to ensure compliance with the applicable national codes and standards included in the facility operating licence issued by the CNSC.

10.7 Emergency Preparedness and Emergency Response Program

Emergency response at the DGR will be conducted in co-operation with Bruce Power, as described in NWMD Employee Emergency Response Procedure W-PROC-ES-0002 (OPGr). OPG will ensure that an effective response can be made to address an emergency crisis affecting the life, safety and health of OPG employees, its business continuity, its property, contractors at OPG facilities, the environment, and the public.

Although the DGR is not considered to be a mine under the Occupational Health and Safety Act, an MRT will be required, which will co-ordinate its activities with the ERT for the Bruce nuclear site. The DGR rescue team will have its own dedicated equipment so that it can immediately respond to the need for mine rescue. The basis upon which mine rescue works is that another team from a mine in Ontario is to be available if required. MRTs are made up of volunteers from the facility work force and receive special training from Ontario Mine Rescue.

10.8 Inspection and Maintenance Program

Effective implementation and control of maintenance activities are primarily achieved by instituting an effective maintenance program consistent with requirements specified in OPG's Conduct of Operations and Maintenance Program, W-PROG-OM-0001 (OPGt). In compliance with Section 6(d) of Class I Nuclear Facilities Regulations, an inspection and maintenance program consisting of policies, processes, and procedures will be developed with an objective to maintain the structures, systems, equipment and components of the DGR as per design specifications. The program will cover a range of inspection and maintenance activities, including but not limited to monitoring, inspecting, testing, assessing, calibrating, servicing, repairing or replacement of parts.

Further to the Class I Nuclear Facilities Regulations requirements, the DGR will also be required to comply with Mines and Mining Plants Regulations (Reg. 854) for mining operations. Underground operations will require the development of inspection and maintenance plans which will include but are not limited to mobile equipment, ventilation systems, shaft and hoisting systems, excavations, etc.

10.9 Records and Document Control

During the operational phase, all records for OPG's nuclear facilities will be managed in accordance with OPG's Records and Document Control Program N-PROG-AS-0006 (OPGu). This program provides instructions for consistent management of records generated across OPG's nuclear facilities. It requires that records be classified, indexed and stored in an approved document management system.

This program also defines the expectations, roles and accountabilities of all employees and staff involved in records management. Further detail regarding the management for records in the areas of quality assurance, radiation protection and dose, licensing, and training is provided below.

- Records identified as controlled documents will be managed as per OPG's Controlled Document Management Procedure N-PROC-AS-0003 (OPGv) and will be revision controlled to ensure the most current revision is available and used. Past revisions are available as required. Quality assurance records will be managed as per the procedure.
- All dose records will be managed as per OPG's Creating and Maintaining Dose Records N-HPS-03413.1-0004 (OPGw).
- Records that are governed by the Radiation Protection Program will follow OPG's Radiation Protection Requirements N-RPP-03415.1-10001 (OPGx).
- Training Records are managed as per OPG's Records and Documentation Procedure N-PROC-TR-0012 (OPGy).

11. QUALITY ASSURANCE

11.1 Introduction

This chapter describes how DGR project activities in the RA Phase were conducted under an appropriate quality assurance framework, and in particular, how this framework was applied to project activities that were related to safety of workers and public, and environmental protection. This chapter also describes how activities during the Design and Construction (D&C) Phase will similarly be conducted under a quality assurance framework.

11.2 Regulatory Approvals Phase

A DGR PQP, DGR-PLAN-00120-0002 (NWMO09a), was established by the NWMO, approved by OPG and accepted by the CNSC for use in the RA Phase (CNSC09b). The PQP meets the requirements of both CAN/CSA-N286-05 (CSA05) and ISO 9001:2008 (ISO08). The DGR PQP was followed in the conduct of site characterization, design, safety assessment and EA activities documented in this PSR.

The quality program applied to all organizational units with responsibilities for the DGR project. The following management actions implemented the requirements:

- The managed system was described in a set of governing documents, prescribed controls and responsibilities to ensure activities were carried out in a safe, effective manner by qualified personnel;
- Individuals were held accountable for implementing and completing work in adherence to the managed system elements; and
- Evaluation and enhancement of the program elements was achieved through continuous improvement processes.

Suppliers and contractors were required to be qualified/approved to appropriate quality assurance standards. The designated suppliers and contractors were required to submit and obtain approval for a detailed quality assurance and inspection plan that was compliant with the DGR PQP. The suppliers and contractors completed work in accordance with their quality assurance plan and reported on these activities.

The quality program included systematic planned audits and assessments. These audits and assessments provided a comprehensive, critical and independent evaluation of project activities and they covered the overall quality program, sub-tier programs, and interfaces between programs. The audits and assessments monitored compliance with governing procedures, standards and technical requirements, and confirmed that quality program requirements were being effectively implemented. Audit and assessment results were documented, reported to and evaluated by senior management to ensure actions were taken to address the findings.

Additional oversight of activities was provided through self-assessment and the non-conformance and corrective action processes. The corrective action process ensured that non-conformance conditions were identified, documented, reported, evaluated and corrected in a timely manner.

The DGR PQP was supported by NWMO governance that established expectations for engineering and design, safety assessment, procurement, occupational health and safety, environmental protection, product and services approval, document control and records keeping.

The key elements of the DGR PQP which were implemented are described below.

- Project-specific quality objectives were established and used to evaluate results.
- Individuals working on the project were responsible for achieving and maintaining quality and management provided appropriate resources and evaluated the quality of the work.
- DGR project work was performed in accordance with applicable NWMO governing documents and established processes and procedures.
- All work was conducted by qualified individuals and organizations.
- The DGR project work performed by consultants/contractors was performed in compliance with ISO 9001:2008 or CAN/CSA N286-05 (as appropriate) and in compliance with an approved work specific quality plan and DGR project-specific governing documents.
- DGR work was verified using planned verification processes and procedures (for work conducted by contractors, appropriate verification procedures for deliverables, including verification process documentation, were included in approved project quality plans).
- DGR project staff observed and verified suppliers' quality processes and examined quality assurance documentation.
- Documents considered to be quality assurance records were obtained, organized and systematically placed into records.

- Planned targeted assessments of work were performed during the execution of the DGR project; work performed by NWMO staff was assessed for compliance with the DGR PQP and pertinent procedures; work performed by consultants' contractors and their subcontractors was assessed to confirm that the work was performed in compliance with their previously approved, work specific quality plans.

11.3 Design and Construction Phase

The NWMO will manage the engineering, site preparation and construction work for the L&ILW DGR project on behalf of OPG. Descriptions of the overall management systems within OPG and NWMO applicable to the D&C Phase of the DGR project are provided in OPG and NWMO D&C Management System documents (OPG11b, NWMO11a). As in the RA Phase, a project quality plan has been implemented for the D&C Phase. The D&C Phase PQP, DGR-PLAN-00120-0006 (NWMO10e) is compliant with CAN/CSA N286-05 and ISO 9001:2008 quality management standards, includes project-specific quality objectives and describes the quality requirements for the D&C Phase of the L&ILW DGR project. The D&C Phase will be executed sequentially in three stages: design stage, construction stage and commissioning stage.

The quality program applies to all organizational units with responsibilities for the DGR project. The following processes implement the requirements:

- A managed system consisting of governing documents that prescribe controls and responsibilities to ensure activities are carried out in a safe, effective manner by qualified personnel;

- Individual accountability for implementing and adhering to the managed system elements; and

- Evaluation and enhancement of the program elements through continuous improvement processes.

Suppliers and contractors are required to be qualified/approved to appropriate quality assurance standards. Each of these designated suppliers and contractors selected is required to submit a detailed quality assurance and inspection plan, consistent with the DGR D&C PQP for approval.

The quality program includes provisions for systematic planned audits and assessments designed to provide a comprehensive, critical and independent evaluation of project activities. These audits and assessments cover the overall quality program, sub-tier programs, and interfaces between programs. The audits and assessments monitor compliance with governing procedures, standards and technical requirements, and confirm that quality program requirements are being effectively implemented. Audit and assessment results are documented, reported to and evaluated by a level of management having sufficient breadth of responsibility to assure actions are taken to address the findings.

Additional oversight of activities is provided through self-assessment and the non-conformance and corrective action program. In particular, the corrective action program assures that non-conformance conditions are identified, documented, reported, evaluated and corrected in a timely manner.

The DGR D&C PQP is supported by NWMO governance that establishes expectations for engineering and design, safety assessment, procurement, occupational health and safety, environmental protection, product and services approval, construction and commissioning, document control and records keeping.

The key elements of the DGR D&C PQP are described below.

- Project-specific quality objectives are established.
- Each person working on the project is responsible for achieving and maintaining quality and management is responsible for providing adequate resources and evaluating the quality of the work.
- DGR project work performed is in accordance with applicable NWMO governing documents and established processes and procedures.
- All work is conducted by qualified individuals.
- When work within the scope of the DGR project is performed by another organization, the consultant/contractor performs work in compliance with ISO 9001:2008 or CAN/CSA N286-05, as appropriate, and in compliance with an approved work specific quality plan and DGR project-specific governing documents. When a consultant/contractor provides a specialized technical service, and their quality management system is not based on a recognized system, their quality management system may be accepted if it meets DGR project quality objectives and requirements.
- DGR work is verified via verification processes and procedures. Furthermore, for work conducted by contractors, project quality plans are approved and include appropriate verification procedures for deliverables including verification process documentation.
- DGR project staff has access to observe and verify suppliers' quality processes and examine quality assurance documentation.

- Documents considered to be quality assurance records are transmitted into records.

- Targeted periodic assessments of work are performed on the DGR project. Work performed by NWMO staff is assessed for compliance with the DGR D&C PQP and applicable procedures. Work performed by consultants/contractors and their subcontractors are assessed to confirm that it is being performed in compliance with their work specific quality plans.

11.3.1 Design Stage

In addition to the general requirements, there are specific quality assurance requirements specific to the design stage. The DGR D&C PQP requires that all design work be completed in accordance with the performance standards and guidelines established by the Professional Engineers of Ontario. In addition design work will be planned and executed in compliance with an engineering management plan that is prepared prior to the start of the work and is consistent with the requirements of the NWMO Design Management procedure, NWMO-PROC-EN-0001 (NWMOa).

The DGR D&C PQP also specifies the requirements for timing and nature of design reviews to ensure that quality continues to be integrated into final design decisions. For critical DGR design components, such as the hoist and ventilation systems, the designs will be verified by independent expert review. Complete design reviews will be completed at the thirty, fifty and eighty percent design completion milestones by knowledgeable engineers who were not directly involved in the design work. The fifty percent design review will be a Constructability, Operability, Maintainability and Safety (COMS) review. The eighty percent design review will include a Hazard and Operability (HAZOP) assessment. These structured and systematic examinations of the design and planned operation are completed in order to identify and evaluate problems that may represent risks to personnel or equipment, or prevent efficient operation.

The DGR D&C PQP also stipulates the minimum requirements for safety assessment which are completed in support of design as well as the minimum requirements to ensure all regulatory requirements are considered. Requirements for use of computer design tools and parameter values are also described.

Collectively the DGR D&C PQP requirements for design will ensure that quality continues to be integrated into final design decisions so that component configurations, materials specifications, functional performance, safety and constructability are optimized.

11.3.2 Construction Stage

In addition to the general requirements, there are specific quality assurance requirements applicable to the construction stage. The DGR D&C PQP requires that all construction and installation work will be completed in accordance with approved-for-construction drawings, specifications, construction procedures and installation instructions. A Construction Quality Assurance Plan will be prepared, appropriately approved and implemented by the construction organization. The Construction Quality Assurance Plan will reference detailed design and engineering requirements, precautions, installation requirements, sequential actions to be followed including co-ordinating construction and verification activities, special equipment/tools and processes required, specific document/drawing references, data report forms and records, cleanliness requirements and foreign material exclusion requirements. It will also include the necessary steps to ensure the correct and intended materials or items are used and installed as required.

Construction verification activities will be planned and integrated into the construction schedule. The planning for the verification activities will be completed prior to the start of construction and will include prerequisites, acceptance criteria, inspections, tests, test frequencies, hold and witness points, and documentation requirements. Construction verification activities performed by contractors will require pre-approval prior to use. The detailed requirements for the various in-the-field quality control activities, including sampling methodologies will be incorporated into a Field Quality Inspection Manual. The Field Quality Inspection Manual will be an approved, controlled document to be utilized by construction personnel to ensure verification activities are performed efficiently and effectively.

11.3.3 Commissioning Stage

The Commissioning Management Plan will define the commissioning process with detailed activities and schedule for the commissioning of the DGR. The Commissioning Management Plan will have two distinct stages because initial commissioning of the temporary main and ventilation shaft hoists and associated headframes will occur early in order to support development of the underground repository. Quality assurance requirements for commissioning activities will be included in the Commissioning Management Plan and will include specification of the required commissioning tests, definition of prerequisites, acceptance criteria for each test, necessary procedures, and final acceptance review. The plan will also describe the mechanism for identification and control of equipment and systems during commissioning. The design organization will also review and accept the Commissioning Management Plan to ensure structure, systems and components are systematically validated against design requirements. Final commissioning documents will be maintained as quality assurance records.

12. PUBLIC INFORMATION AND INVOLVEMENT PROGRAM

12.1 Overview

The Public Information and Involvement Program was initiated in 2002, following the signing of a Memorandum of Understanding between the Municipality of Kincardine and OPG, which set out the terms under which OPG, in consultation with the Municipality of Kincardine, would investigate the feasibility of various long-term management options for L&ILW at the Bruce nuclear site. Based the results of an Independent Assessment Study, the Municipality of Kincardine passed a resolution in 2004, which selected the DGR as its preferred course of study moving forward. A hosting agreement was signed by Kincardine Council and OPG in 2004, which was followed by a polling of Kincardine citizens, both permanent and seasonal, early in 2005, which indicated community support for the DGR. The letter of intent and the project description (OPG05b), submitted to the CNSC in December 2005, triggered the EA process. The Public Information and Involvement Program that has continued since 2002 through to the submission of supporting documents for OPG's site preparation and construction licence application is described in Chapter 2 of the EIS (OPG11a).

This chapter provides a summary of the planned Public Information and Involvement Program for OPG's DGR for L&ILW from Q1 2011 to approximately five years after obtaining the site preparation and construction licence. The current program has been designed in accordance with CNSC Regulatory Guide G-217 (CNSC04c) to meet the requirements for a public information program in the Class I Nuclear Facilities Regulations (SOR/2000-204).

The NWMO will support OPG in the development and delivery of the DGR Public Information and Involvement Program.

12.2 Public Information and Involvement Program

The delivery of the DGR Public Information and Involvement Program encompasses a broad approach, which is related to key milestones in the development of the DGR and the progression of the regulatory process. Annual communication plans define the communication objectives, communication strategy, spokespeople, target audience, key messages and communication activities. Communication plans are living documents that are adapted when necessary to reflect changing or emerging factors and issues related to the DGR.

This section describes the elements of the Public Information and Involvement Program; program objectives, target audience, public and media opinion, program description, evaluation process and contact information are provided.

12.2.1 Objectives

The objectives of the DGR Public Information and Involvement Program, in keeping with all relevant regulations, are to:

- Inform persons living in the vicinity of the site of the general nature and characteristics of the anticipated effects on the environment and health and safety of persons during site preparation and construction and subsequent phases of the project (including, but not limited to: site preparation and construction progress; results from follow-up monitoring; DGR milestones, decisions, and modifications);

- Provide a broad range of engagement opportunities for members of the general public and key stakeholders to become updated, ask questions, provide meaningful comment and raise concerns about the DGR;

- Provide a broad range of engagement opportunities for Aboriginal Peoples to become updated, ask questions, provide meaningful comment and raise concerns about the DGR;

- Continue to respond in a timely manner to issues raised by the community, key stakeholders, Aboriginal Peoples and general public; and

- Monitor, document and evaluate the DGR Public Information and Involvement Program.

12.2.2 Target Audience

12.2.2.1 Overview of the Target Audience

The target audience for the DGR Public Information and Involvement Program is regional in nature. It encompasses key stakeholders and the general population inhabiting the Municipalities of Arran-Elderslie, Brockton, Kincardine, Northern Bruce Peninsula, South Bruce; Towns of Saugeen Shores and South Bruce Peninsula; the Township of Huron-Kinloss; and the Aboriginal communities located within or asserting rights within Bruce County. Given the scope of the target audience, a broad and inclusive interpretation of persons living in the vicinity has been applied to ensure the information reaches all interested parties, consistent with G-217 (CNSC04c).

It is anticipated that there may be instances where information is unique to a specific geographic or study area and, as such, would only be communicated to affected stakeholders and members of the public.

12.2.2.2 Key Stakeholders

The DGR Public Information and Involvement Program includes (but is not limited to) a number of key stakeholders:

- Aboriginal Peoples (addressed in greater detail in Section 12.2.2.3);
- Elected federal, provincial and municipal representatives;
- Appointed government officials;
- General public;
- Officials with a regulator role;
- OPG/NWMO employees;
- Media;
- Local business groups;
- Chamber of Commerce groups;
- Service clubs;
- Women's groups;
- Agricultural organizations;
- Anglers and hunters;
- OPG/Bruce Power retiree associations;
- Beach associations;
- Tourism groups;
- Non-Governmental Organizations (NGOs);
- Educational sector;
- Source Water Protection Committee;
- Grey Bruce Health Unit;

- Scientific groups; and
- Representatives from the nuclear industry such as Bruce Power, Power Workers' Union and Society of Energy Professionals.

12.2.2.3 Aboriginal Peoples

Aboriginal Peoples are a significant target audience for the DGR Public Information and Involvement Program. OPG has encouraged the ongoing engagement of Aboriginal communities in the DGR to determine if the DGR has the potential to affect Aboriginal interests. OPG will continue to provide engagement opportunities for Aboriginal communities where they can become informed and updated, ask questions, provide meaningful comment and raise issues and concerns about key DGR activities, milestones and decisions. Aboriginal communities will be kept apprised of any significant environmental, safety or health issues, any significant changes to the DGR, and the results of any follow-up monitoring. The Aboriginal communities included in the DGR Public Information and Involvement Program are listed below.

- Saugeen Ojibway Nation (SON) – The collective name for the Chippewas of Nawash Unceded First Nation and the Chippewas of Saugeen First Nation. A Protocol Agreement, signed between SON, OPG and NWMO on March 9, 2009, provided a process to ensure that SON has the necessary resources to participate in the DGR's RA process.
- Historic Saugeen Métis (HSM) – The signing of a Participation Agreement between HSM, OPG and NWMO provides a process to ensure that HSM has the necessary resources to participate in the DGR's RA process.
- Métis Nation of Ontario (MNO) – The signing of a Participation Agreement between MNO, OPG and NWMO is anticipated, in order to provide MNO with the necessary resources and capacity to participate in the DGR's RA process. MNO represents the local interests of the Great Lakes Métis Council, the Georgian Bay Métis Council and the Moon River Métis Council on this project.

12.2.2.4 Michigan

Key stakeholders and interested members of the public in Michigan are also considered part of the target audience for the DGR, including, but not limited to: Michigan representatives of the U.S. Senate and Congress, The Macomb County Board of Commissioners, the St. Clair County Board of Commissioners, the Michigan Department of Environmental Quality and NGOs.

12.2.3 Public and Media Opinion

12.2.3.1 Public Opinion

From its inception, the DGR has received significant local community support from the host municipality as well as its neighbouring municipalities. This support extends to all of Bruce County and is documented in Section 2.2 of the EIS (OPG11a).

Of significance is the fact that an independent polling of both the permanent and seasonal residents occurred in 2005 whereby the Municipality of Kincardine confirmed that the majority of residents who participated in the poll supported moving forward with the DGR. This show of support for the DGR initiated the submission of the project description to the CNSC.

Public attitude research included activities described below.

- Telephone survey in June 2003 whereby 751 telephone interviews were conducted. Additional interviews were conducted in 2003 with tourists. These interviews were conducted as part of an Independent Assessment Study (which assessed the various options for the long term management of L&ILW).

- Independent public attitude research and a community leaders' survey in 2009, provided current evidence of strong community support for the DGR. Public attitude research and community leadership survey details and results are documented in Appendix H of the Socio-Economic Environment Technical Support Document (NWMO11h).

The public attitude research results from 2009 have shown that the majority of residents, living in Kincardine and the surrounding municipalities, have a high level of confidence in the DGR and do not anticipate that its presence will result in any change towards their attitudes or behavior as they pertain to:

- Level of commitment to living in their community;

- Level of satisfaction with living in the community;

- Feelings of personal health or sense of safety;

- Use and enjoyment of private property;

- Nature activities along shoreline; and

- Use of beaches and boating.

The community leaders' survey from 2009 reflected a strong support for the DGR at a rate, on average, of nine out of 10 with 23 community leaders from across Bruce County participating in the survey.

Public attitude research will continue throughout the licensing process of the DGR to ensure that public opinion remains positive.

The DGR has initiated key questions from individual Bruce County residents, and NGO groups in Ontario and Michigan (and some Michigan state politicians), which have given rise to concerns about the proximity of the DGR to Lake Huron and the ability of the DGR to protect the water quality of groundwater and the Great Lakes. It should be noted that similar questions have been expressed by supporters of the DGR.

Since 2005, OPG and NWMO have supported and attended a number of pow wows and community events hosted by Nawash Unceded First Nation and Saugeen First Nation. Comments and questions received at these events, for the most part, focused on concerns about the proximity of the DGR to Lake Huron and the ability of the DGR to protect the water quality of groundwater and the Great Lakes. Several individuals also expressed concerns about the perceived potential of the DGR to accommodate used fuel. Some questioned whether it is appropriate to place nuclear waste in Mother Earth. SON has expressed its historical objection to the siting of a nuclear power development within their traditional territories without proper consultation.

12.2.3.2 Media Opinion

Media coverage of the DGR has remained of interest primarily to local print media located within Kincardine and Saugeen Shores as well as local broadcast media who provide regional coverage across Bruce and Grey counties. Media articles are contained in Appendix D of the EIS (OPG11a).

Between 2002 and 2005, media coverage of the DGR peaked with close to several hundred media reports. The majority of the articles, which included editorials, columns, news stories, advertisements and broadcast reports, were from local media sources (i.e., Kincardine News, Kincardine Independent, Shoreline Beacon, Walkerton Herald Times, Owen Sound Sun Times, CKNX Radio and CFOS Radio). A limited number of articles also appeared in the national media, including Kitchener-Waterloo Record, Toronto Star, National Post, Ottawa Citizen and CBC Radio.

Reports during this period focused on key milestones leading up to the successful community poll. The majority of the media coverage was balanced and editorials supporting the DGR were published in several local papers prior to the independent polling early in 2005. Any negative media coverage, particularly with respect to letters to the editor, reflected concerns with the proximity of the DGR to Lake Huron, preservation of groundwater and Great Lakes' water quality and the perceived potential of the DGR to accommodate used fuel.

Media coverage of the DGR from 2006 onwards has fluctuated, with the exception of a flurry of coverage in Michigan during the 2008 public review of the draft DGR guidelines and draft Joint Review Panel Agreement. Most of the media coverage during this period has been gained through engagement activities such as media days, open houses, and attendance at public events and through the hosting of special initiatives.

The coverage in Michigan during the comment period for the draft guidelines was prompted by opposition from environmental NGOs who cited concerns with the ability of the DGR to protect the water quality of the Great Lakes. Reports were provided by print media (i.e., Detroit News, Macomb Daily, Port Huron Times Herald, and Port Huron News) and broadcast media including Detroit Today as well as several online publications. The Michigan coverage, for the most part, was balanced featuring comments from both OPG and the environmental NGOs. The coverage ceased with the end of the public comment period.

Media activities are anticipated to be similar to those described above. Coverage will likely continue and will be monitored in the future.

12.2.4 Program Description

The Public Information and Involvement Program in support of the licensing process will continue to be developed in a manner that ensures citizens are apprised of the general nature and characteristics of the anticipated effects on the environment and health and safety of persons during site preparation and construction and subsequent phases of the project. Program content will also include information about site preparation and construction progress; results from follow-up monitoring; DGR milestones, decisions, and modifications.

12.2.4.1 Communication Tools and Activities

A broad range of communication tools will be employed to provide the general public, key stakeholders and Aboriginal Peoples with information and opportunities for engagement. Communication tools to be used, at a minimum, are described below.

- Means of Notification – Public notifications will be prepared and distributed through a number of venues such as press releases, advertisements, web communications (www.opg.com/dgr), letters and face to face briefings.
- Stakeholder Briefings and Presentations – Regular briefings, on a frequency commensurate with key project activities, milestones and decisions, will continue to

be conducted to provide key stakeholders with updated information and opportunities to provide comments, ask questions and identify and discuss any concerns throughout. Established stakeholder groups which will continue to receive briefings are:

- Aboriginal Peoples;
- Huron-Bruce Member of Parliament;
- Huron-Bruce Member of Provincial Parliament;
- Kincardine DGR Community Consultation Committee (OPG and NWMO representatives, Mayor, Deputy-Mayor and two councilors);
- DGR Community Advisory Consultation Group (OPG and NWMO representatives, eight Bruce County Mayors and municipal administrators as required);
- Provincial and Federal Ministries;
- Local MOE Office;
- Grey-Bruce Medical Officer of Health;
- Inverhuron District Ratepayers' Association;
- Bruce Power; and
- Power Workers' Union and Society of Energy Professionals.

• DGR Project Newsletters – The established DGR project newsletters will continue to be issued on a frequency commensurate with key DGR activities, milestones and decisions. Currently three to four newsletters are distributed annually to about 30,000 Bruce County residences, businesses and interested parties on a designated mailing list, several of whom reside outside Canada, including Michigan.

• Fact Sheet/Brochure – Fact sheets or brochures will be utilized as a means of providing new information to the public about key DGR activities, milestones and decisions in the process.

• DGR Website – The DGR website can be accessed via the OPG website at www.opg.com/dgr. The website will continue to be updated, in a timely manner, with new information and copies of public notification products such as letters and press releases. Contact information is posted on the website to encourage two-way dialogue so members of the public have an opportunity to become informed and updated, ask questions, provide meaningful comment and identify and raise any concerns. The DGR information link is checked daily on weekdays, and a timely response is given to all enquiries.

- DGR Mobile Exhibit – The DGR mobile exhibit, used primarily at community and public events, will continue to be utilized as an engagement tool up to the public hearing for the DGR, and possibly afterwards on an "as needed" basis.

- Bruce County Marketplace Advertorial - The "Connecting With the DGR" column will be maintained to provide the public with information about key DGR activities, milestones and decisions.

- Open Houses/Community Information Sessions/Community Consultation Centre – Open houses, community information sessions and/or a community consultation centre will be planned for 2011, and possibly beyond, as vehicles to provide the public with opportunities to become informed and updated, make comment, ask questions and indentify and discuss any potential concerns. Other communication tools such as newsletters, advertising, press releases, etc., would be used in a support fashion to inform the public of these engagement opportunities.

- Media Briefings – Ongoing briefings with local media (print, broadcast and web-based) will continue to ensure media has accurate and current information regarding DGR activities, key milestones and decisions. Special events (press conferences, media days and radio talk shows) may also be employed.

- Telephone Communication – Contact information of public affairs staff will continue to be available to the public on written and electronic materials to facilitate any information requests and/or responses from the public.

- Employee Communication – Communications about the project will be provided to NWMO and OPG employees through employee publications, newsletters, intranet sites and Lunch and Learn events.

- Issue Management and Tracking - A DGR comment database will continue to be maintained to record and monitor all comments, correspondence and communications with the public and key stakeholders.

12.2.5 Program Evaluation Process

The DGR Public Information and Involvement Program will continue to be evaluated throughout to ensure the objectives of the program are being met.

Feedback from the public is encouraged by posting contact information on the DGR website, newsletters and all written publications. Comments about the DGR and the Public Information and Involvement Program will continue to be documented in a database using a tracking form, which identifies the source of the comment, date and type of communication (email, phone, letter or in person). It also provides a summary of the comment and response given with key information such as date of response, who provided the response and the manner in which the response was communicated.

Regular briefings and meetings with community leaders and established committees will continue to provide firsthand information about the effectiveness of the DGR Public Information and Involvement Program, and OPG and NWMO will continue to seek such feedback.

Public attitude research and community leadership surveys were undertaken as part of the socio-economic impact assessment for the DGR. Specific questions related to the effectiveness of the DGR Public Information and Involvement Program provided a scientific means of evaluating the program. Such methodology may again be employed in the future as part of follow-up monitoring, to evaluate and, if appropriate, adjust or revise the program.

Annual communication plans govern the delivery of the DGR Public Information and Involvement Plan. Prior to the development of each new plan, the objectives of the previous year's communication plan are discussed and evaluated. Subsequent plans are developed to include revisions as the result of any lessons learned.

12.2.6 Contact Information

Contact information is provided in all communication material.

13. PRELIMINARY DECOMMISSIONING PLANNING

The purpose of decommissioning is to retire the DGR Facility permanently from service and render it to a predetermined end-state condition. The Class I Nuclear Facilities Regulations (SOR/2000-204) require that the proposed plan for the decommissioning of the nuclear facility or of the site be included in a licence application. A Preliminary Decommissioning Plan (PDP) has been prepared for the DGR Facility to meet the expectations of CNSC guidance and the CSA standard. The PDP (NWMO11au) contains, in its appendix, a mapping of how this plan meets:

- CNSC Regulatory Guide G-219 Decommissioning Planning for Licensed Activities (CNSC00c); and
- CSA N294-09 Decommissioning of Facilities Containing Nuclear Substances (CSA09a).

It describes the site location and characteristics, the facility, the decommissioning plan, decommissioning activities, the cost estimate and financial guarantee, human factors considerations, waste management, environmental impacts, health and safety, emergency response planning, security, quality assurance, records, and period review of the plan. This chapter summarizes the key decommissioning aspects of the PDP.

13.1 Objectives of Decommissioning

The objectives of decommissioning are described below.

1. Permanently retire the DGR Facility from service at the end of its service life in a manner that ensures that the health, safety and security of workers, the public and the environment are protected.
2. Install passive engineering features (containment, sealing, structural stability and a design to minimize inadvertent intrusion) that ensure that:
    - The waste is contained until most of the radioactivity, and especially that associated with shorter-lived radionuclides, has decayed;
    - The waste is isolated from the biosphere; substantially reducing the likelihood of inadvertent human intrusion into the waste;
    - Significant migration of radionuclides to the biosphere is delayed until a time in the far future when much of the radioactivity will have decayed; and

- Levels of radionuclides eventually reaching the biosphere are such that possible radiological impacts in the future are acceptably low.

3. Restore the site to the desired end-state, described below.

13.2 End-State of Decommissioning

Decommissioning will be complete when the waste is permanently secured in the repository, the shafts have been sealed and secured, the surface facilities have been dismantled and removed, and the surface landscape has been rehabilitated. This will constitute the final end-state. After completion of the decommissioning work, the site will be subject to a period of institutional controls.

13.3 Planning Assumptions

Planning for decommissioning is an ongoing process and planning assumptions are expected to change with evolving technologies, international and operational experience, regulations, and cost estimates. The PDP describes the preliminary plan as it exists at the time of writing. This document will be reviewed and revised periodically in order to incorporate any changes in the planning assumptions.

The Bruce nuclear site contains a number of other licensed nuclear facilities such as the WWMF and the NGSs, in the immediate vicinity of the DGR Facility location. Although the DGR Facility is located within the existing Bruce nuclear site boundary, the planning envelope for other facilities is different. Decommissioning of the other licensed nuclear facilities is outside the scope of the DGR Facility PDP.

Planning for decommissioning of the DGR Facility is based on the following fundamental assumptions:

- Decommissioning will start following the end of waste emplacement operations and a period of monitoring and surveys;
- Underground facilities will be sealed from entry and waste emplaced in the DGR will remain in the facility emplacement rooms in perpetuity;
- The ventilation shaft infrastructure will be dismantled and the shaft will be sealed;
- The main shaft infrastructure will be dismantled and the shaft will be sealed;
- Surface infrastructure and buildings will be dismantled and removed;
- OPG will retain ownership of the DGR Facility site area during all stages of and following decommissioning; and

- A period of institutional controls, assumed to last up to 300 years, will follow the decommissioning work.

13.4 Decommissioning Overview

Preparation for decommissioning is envisaged to include the following activities.

- Submit a notification of the intent to decommission the facility to the CNSC.
- Inform the public, key stakeholders and the host community on the initiation of the decommissioning, and obtaining their input for the development of the Detailed Decommissioning Plan (DDP). The public involvement process will help to identify potential socio-economic impacts and other issues associated with the decommissioning activities and appropriate impact management strategies.
- Complete an EA of the decommissioning project.
- Complete the documentation that will be used in support of the licence application such as the detailed design of the shaft seal system, a safety assessment, and the DDP. The detailed design of the shaft seal system, safety assessment and decommissioning plan will take into account experience and knowledge gained during construction and operation of the facility.
- Submit to the CNSC a DDP and project documentation in support of the application for a decommissioning licence.
- Obtain the decommissioning licence and any permits required for the decommissioning work.
- Confirm that appropriate resources are in place to assume the responsibility for decommissioning.

Decommissioning will begin following a period of monitoring after all of the waste has been emplaced and a decommissioning licence has been obtained.

The scope of decommissioning work for the DGR Facility will include decommissioning of underground facilities, sealing of shafts, demolition and removal of surface facilities, and restoration of the site.

The decommissioning work will be considered complete when the planned end-state of the DGR Facility, as described in the application for decommissioning licence, has been reached and the CNSC has agreed that the decommissioning work has been completed. The decommissioning work is expected to take approximately five years to complete.

13.5 Decommissioning Strategy

The reference decommissioning strategy following operation of the facility is based upon the fundamental assumption that no radioactive wastes emplaced in the DGR will be removed as part of the decommissioning.

The decommissioning strategy will be based on a combination of prompt decommissioning and in-situ confinement as defined in CSA N294-09 (CSA09a). The approach for surface infrastructure and buildings as well as selected underground facilities will be prompt decommissioning. Infrastructure supporting the main shaft and the ventilation shaft will be dismantled and emplaced in the repository if contaminated and where possible. The approach for the wastes emplaced in the underground structures will be in-situ confinement. A concrete monolith will then be installed at the base of the main shaft and the ventilation shaft sealing the access ways from access. The main shaft and the ventilation shaft will both be permanently sealed. Surface infrastructure and buildings will be dismantled and removed. The decommissioning work will be followed by a period of institutional controls.

The DGR Facility is unique in that it combines aspects of mining with a nuclear facility. Even though the DGR Facility does not meet the legal definition of a mine, Mine Development and Closure Regulations under Part VII of the Mining Act (Reg. 240/00) do provide for the installation of concrete caps atop decommissioned mine shafts. In general, a reinforced concrete cap, certified by a qualified professional engineer, is placed atop decommissioned mine shafts. The caps installed atop the main and ventilation shafts will be consistent with the requirements given in Reg. 240/00.

13.6 Decommissioning of the DGR Facility

The PDP is submitted in support of the site preparation and construction licence application for the DGR Facility and it will be revised periodically, as required, until it is replaced by a DDP closer to the time of decommissioning.

The PDP describes the areas to be decommissioned and the general structure and sequence of the principle decommissioning work packages envisioned, including:

- Underground repository;
- Ventilation shaft and main shaft; and
- Surface facilities.

A brief overview of preliminary decommissioning planning activities is presented here. Further details are presented in the PDP for the DGR Facility.

The DDP will be developed and submitted to the CNSC for appropriate licensing action prior to beginning decommissioning activities. This detailed plan will provide further procedural and organizational details to the preliminary plan.

13.6.1 Preparation for Shaft Sealing

The WPRB will be inspected, tested for contamination and, if necessary, decontaminated for use during shaft sealing operations. The WPRB will act as a warehouse for the shaft sealing materials and as a maintenance workshop for the seal construction equipment.

At the ventilation shaft, a set of temporary stage winches will be installed from which a working platform (also called a "stage") will be suspended on wire ropes. The platform will enable the placement of sealing materials within the shafts. The existing second egress hoist (i.e., hoist used during operations) within the ventilation shaft will be used as the primary means of travel between surface and the shaft bottom for workers, equipment and materials. However, once the steelwork has been stripped, the conveyances will use the stage ropes as guides, rather than the steel or timber guides. Shaft infrastructure, such as ventilation, will be removed on a phased basis in a manner that ensures the provision of required services to the shaft during shaft sealing.

Prior to the start of main shaft decommissioning, the main and auxiliary Koepe friction hoists (hoists used in main shaft during operations) will be removed and stage winches installed to suspend the working platform. In addition a temporary single drum hoist will be installed for worker, material and equipment access during shaft decommissioning and sealing work. Then decommissioning and sealing of main shaft will proceed in a manner similar to that described for the ventilation shaft.

13.6.2 Decommissioning of the Underground Services Area

Decommissioning at the repository level will largely consist of preparing the underground services area for the construction of the concrete monolith. All other parts of the underground repository will have already been isolated by the closure walls during the operations phase, as described in Section 6.13.

Decommissioning of the services area will involve assessing equipment and materials to determine what can or should be removed prior to repository sealing. Particular attention would be given to areas where potentially hazardous materials such as any waste fluids from mobile equipment, may exist. It is currently assumed that most permanent equipment and materials will remain within the repository and that only mobile equipment, which has been tested and does not contain any residual radioactive contamination, will be removed to the surface.

Once the underground openings in the services area have been prepared, the construction of the concrete monolith would be carried out. The monolith will be constructed in two stages.

Concrete Monolith

A concrete monolith will be constructed at the base of the two shafts. The concrete monolith will provide a stable foundation for the overlying seal materials and a high degree of support to the shaft station rock openings. Low heat of hydration concrete will be used to minimize heat generation and shrinkage. The concrete will be placed to a distance of about 60 m beyond the circumference of the excavated shaft diameter (see Figure 13-1).

The large lateral extent of the monolith provides:

- Margin of safety against loads during glaciation events;
- Extended roof rock support around shaft to ensure that any roof collapse within the panel area does not propagate horizontally into rock around the shafts;
- Controls or limits the propagation of gas from waste-filled panels to the base of the shaft; and
- Greater certainty that bentonite/sand mixture in shafts will not be pushed or creep down from shaft into the repository void space.

The monolith will be created in two stages, one for the ventilation shaft, followed by another for the main shaft. However, they will form one contiguous mass concrete structure and there will be no structural reinforcement within the concrete; monolith will be a mass concrete structure. All services and utilities will be stripped out of the excavations to be filled by the monolith so as to remove potential voids. Bulkheads will be constructed at the maximum limit of the monolith in the underground services area tunnels and other openings prior to placement of concrete. The concrete monolith will be created by filling the shaft sumps, ramps to the shaft bottom, shaft stations, and the tunnels or peripheral rooms with concrete.

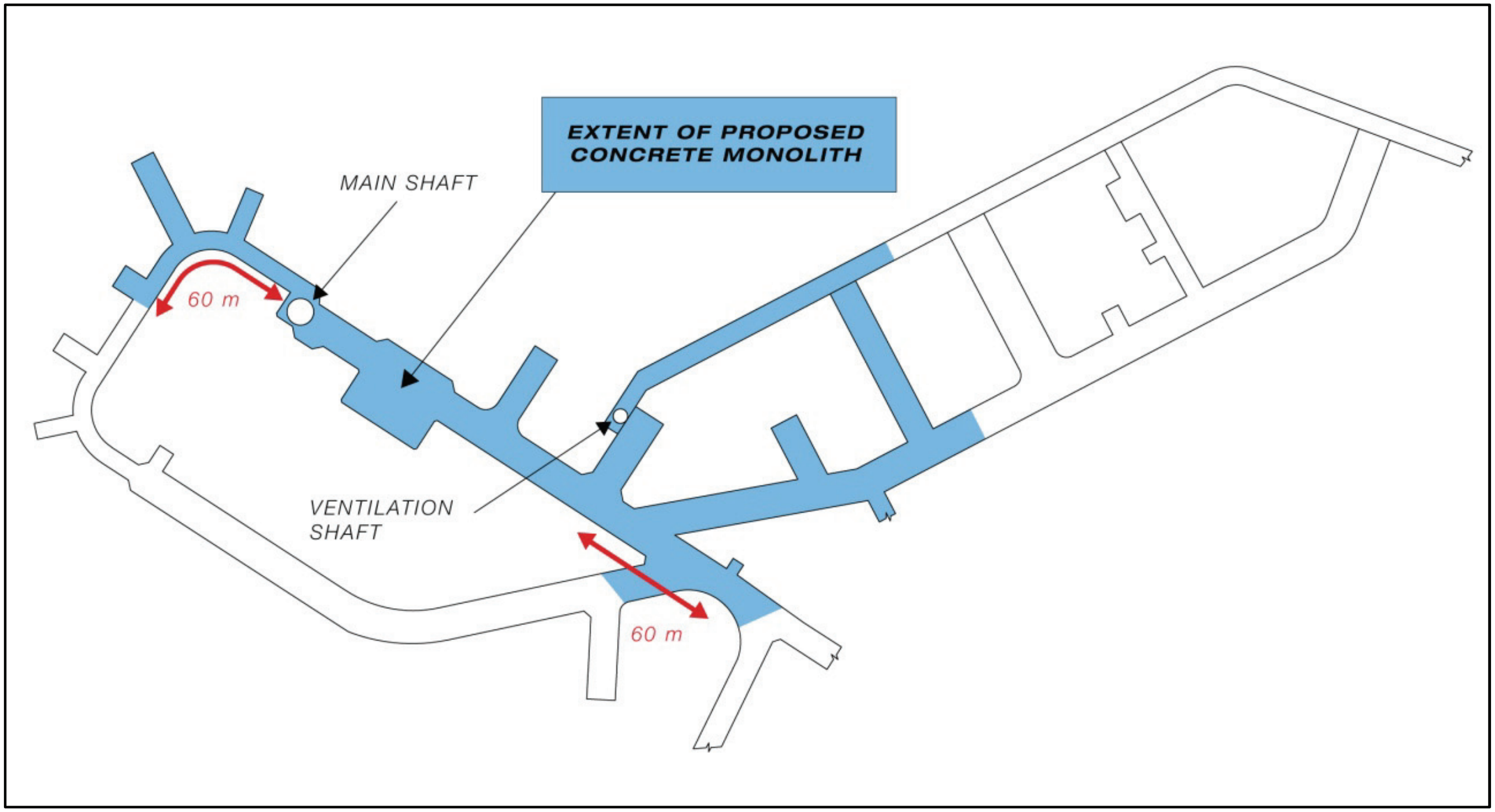


Figure 13-1: Extent of Proposed Monolith

Concrete will be mixed on the surface in a temporary batch plant and then delivered into either shaft through the use of a slickline and header. A slickline is essentially a steel pipe secured to the shaft wall to transport fluid concrete from the surface to the required depth. At the base of the slickline, a slightly larger steel pipe called a header diverts the downward flow of material 45 degrees, dissipating the impact energy of the falling material. A flexible hose is then connected to the header enabling exact placement of the concrete. For concrete that needs to be transferred to locations at a large distance from the base of either shaft, a pumping system will be used to transfer this concrete.

The installation of this monolith could potentially generate large amounts of heat during the concrete curing process. Mass concrete construction procedures will be developed and followed to control heat build-up.

13.6.3 Decommissioning of Shafts

Decommissioning of the shafts will consist of sequential removal of shaft infrastructure and installation of the shaft seal materials. All internal shaft support structures (e.g., steel sets) and infrastructure connections (e.g., power, ventilation, water) will be disconnected and removed before sealing work begins in a shaft. A new ventilation system will be established in each shaft to allow the workers to safely decommission the shafts. The decommissioning of the ventilation shaft would start first followed by main shaft decommissioning with some of the shaft decommissioning work being performed concurrently in each shaft (see schedule in Section 13.9). The design and construction of the shaft seal system are described in the following section.

13.6.3.1 Design and Construction of Shaft Seal

The approach for the shaft seal design and construction has focused on the use of simple, relatively well understood and durable materials, and use of proven methodologies for emplacement. It is similar in concept to the WIPP facility shaft seal design (HANSEN00). The arrangement of the shaft sealing system, selected components and their relative location is shown in Figure 13-2.

Prior to placing shaft seal materials, a concrete monolith would be constructed at the base of each shaft, as described in Section 13.6.2. The monolith will provide a stable foundation for the overlying seal materials and a high degree of support to the repository station openings. The concrete monolith is then overlain by a column of compacted bentonite/sand. An asphalt column is placed above the first bentonite/sand layer to provide a redundant low permeability sealing material against upward or downward fluid flow. A series of bentonite/sand columns are separated by concrete bulkheads to provide structural components to the column and provide additional sealing capability.

Concrete, bentonite/sand mixture and asphalt will be the sealing materials used in each shaft. An engineered fill material based on rock excavated during shaft sinking or some other suitable material will be used in the upper portion of each shaft (i.e., see Figure 13-2).

Since the shaft seals will not be constructed for several decades, there is time to incorporate new information learned during operation of the DGR Facility as well as experience gained from international repository projects. Therefore, the design is intended to provide a reasonable assurance that a competent shaft seal can be constructed using currently available materials and methods, but is not necessarily the final design.

The arrangement of shaft sealing materials for the ventilation shaft and main shaft is described below.

Removal of Shaft Infrastructure

Throughout all seal sections up to the top bulkhead in Figure 13-2, shaft support structures and concrete liners will be removed to ensure a complete seal of the shaft column to the surrounding low permeability host rock. Also, it is assumed that an additional 500 mm of host rock will be excavated beyond the initial shaft diameter to remove any damaged rock that may have formed during shaft sinking and the operational period of the DGR.

All shaft infrastructures will be mechanically cut from the shaft in a series of controlled lifts which are expected to be about 10 m to 20 m in length. Rock bolts will be installed, as required, to support concrete liner and newly exposed rock where the liner has been removed to allow workers to safely place seal materials. Each section of removal will be closely followed by backfilling of the lift with shaft sealing materials.

Bentonite/Sand Mixture

The column of sealing materials in each shaft is largely comprised of a compacted bentonite/sand mixture (Figure 13-2). Once saturated, the compacted bentonite/sand materials will act as a durable low permeability barrier to retard the movement of radionuclides out of the repository and minimize the potential for groundwater flow down into the repository. Compacted clays or clay/sand mixtures are the most commonly proposed sealing materials for nuclear waste repositories.

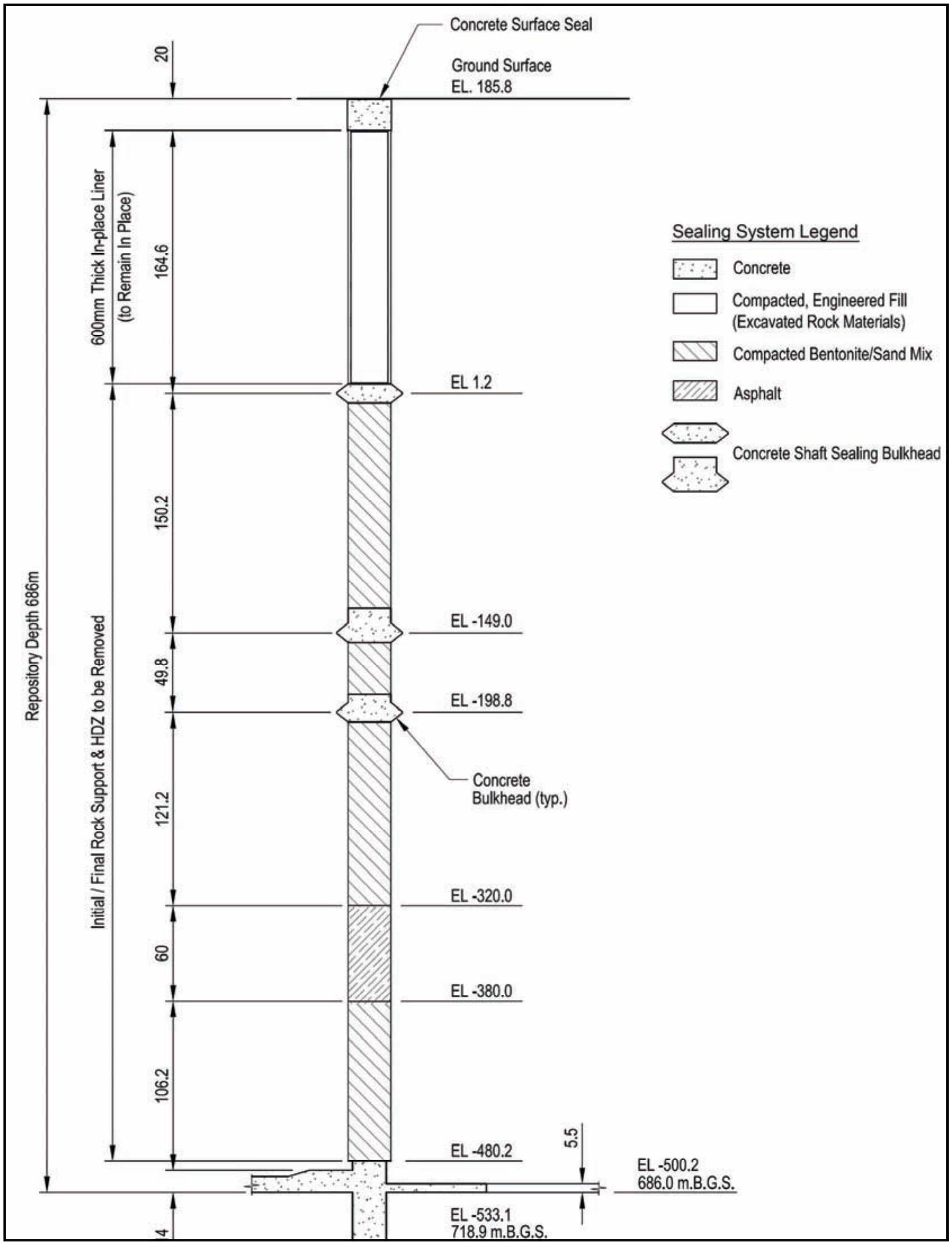


Figure 13-2: Arrangement of Shaft Seal Components

As the compacted bentonite/sand materials saturate with groundwater from the surrounding rock, they will generate swelling pressures which will aid in the development of a tight seal against the shaft wall and provide a confining pressure to the rock surface.

Sand will be added to the bentonite to act as a filler without compromising the hydraulic conductivity and swelling potential of the bentonite dominant material. The use of sand will improve workability during placement, ease compaction and dust control.

Bentonite will be mixed with sand to a 70:30 mix through the use of a temporary batch plant. The plant would have two hoppers, one holding the sand component and the other holding the bentonite, along with a tank that holds water. The sand and bentonite are fed onto a conveyor belt that feeds a screw-auger which mixes the materials as they approach the discharge spout. Water is applied to the mixture, as required, as it enters the screw-auger. The materials would then be transported directly into the shaft, or stored temporarily within plastic bags to retain moisture levels, and then transported into the shaft.

The bentonite/sand mixture will be placed loose via a slickline and header (similar to that used for placing the concrete), and then compacted in-situ to a dry density of approximately 1600 kg/m^3. Compaction of these materials can be performed using vibratory plate compactors, and sheepsfoot rollers. Seal materials will be placed in roughly 150 mm thick lifts to ensure compaction over full depth of each lift. Smaller compaction equipment will be used in proximity to the shaft walls in order to ensure adequate compaction in this area.

Asphalt

Asphalt was selected because it has the ability to flow and make good contact with host rock. Immediately upon emplacement the asphalt will create an effective barrier to water flow. The use of another low permeability sealing material provides an additional level of redundancy to the sealing system against upward or downward fluid flow. A 60 m thick asphalt column will be placed above the lowermost bentonite/sand column. The asphalt column extends over a length of the Georgian Bay Formation (see Section 4.1) to just above the Queenston/Georgian Bay contact.

The reference asphalt mixture is based on a mix of asphalt compounds and aggregate, combined with a small porosity fraction to ensure low permeability. The asphalt mix will be prepared on the surface with the use of a temporary pug mill and heated to a temperature that would allow delivery by slickline. Following mixing, asphalt will be pumped to the shaft and placed through the use of a slickline and header. The slickline will require heating in order to maintain the asphalt's viscous state. Asphalt will be placed in controlled lifts. Following placement of an asphalt lift, placement operations will be ceased to allow for cooling of the asphalt and to ensure a safe environment for workers starting placement of the next layer of asphalt (or placement of bentonite/sand mixture at the top of the asphalt column). In order to promote cooling and to remove any hazardous fumes, ventilation into the shaft will be maintained during this period. Air temperature and quality will be remotely monitored at a location 1-3 m above the asphalt column to establish when it would be safe to re-enter the shaft and resume shaft sealing operations.

Concrete Bulkheads

Leading up to the top bulkhead, there are two higher permeability units within the surrounding geosphere: the Guelph Formation and the upper 4 m of Salina A1 carbonate unit (see Section 4.1). The Guelph Formation has a hydraulic conductivity which is 2 to 3 orders of magnitude greater than adjacent formations. Due to the expected lateral flow along this unit, a concrete cylinder will be placed along the full extent (approximately 6 m) of this unit. In order to ensure structural stability, the underlying concrete structure will be constructed to a height slightly larger than diameter of the excavated shaft. In order to maintain structural stability, the concrete bulkhead will be keyed into the surrounding host rock. The concrete mix will be similar to that selected for the concrete monolith. The concrete/rock interface will also be pressure-grouted to minimize groundwater flow along the interface.

A concrete bulkhead will be installed at the upper 4 m of the Salina A1 carbonate unit and the design will be the same as that proposed for the Guelph Formation.

Salina Unit F represents a lower (at least one order of magnitude) permeability zone within the dolostones (an aquitard) between a fresh water aquifer above and more saline water-bearing formations below. To prevent movement of the poor quality, saline groundwater from the lower Salina Formation upwards through the shaft cross-section into the upper fresh water aquifer, a concrete bulkhead will be constructed at this location.

As with the monolith, concrete for the bulkheads will be placed in mass and with no reinforcing steel, and using measures to control heat build-up. Contact/seal grouting will be applied around the bulkheads in order to minimize the potential impacts of shrinkage at the interface with the host rock formation. Concrete will be poured directly onto the bentonite/sand columns located below each bulkhead.

Because the three concrete bulkheads will be keyed into the adjacent rock, they will provide structural support for the overlying seal materials and confinement of the swelling seal materials between the bulkheads. The location and the need for additional bulkheads will be assessed in future design phases taking into consideration new information from field observations and geomechanical field testing. These confirmatory investigations will assist in the determination of final requirements for bulkheads to be installed during shaft seal construction.

Engineered Fill

The uppermost portion of each shaft will be filled with an engineered fill (e.g., 'Granular A' material), possibly created from crushed rock obtained during shaft excavation and/or concrete. The fill material will be engineered and compacted. It will not be necessary to remove the concrete liner throughout the section where fill is to be placed. Therefore, it will be left in place to avoid safety risks to workers and the cost associated with its removal; however, the state of the liner and the possibility of removal will be re-examined prior to seal construction to determine if liner removal has any significant benefits.

The rock materials will be crushed and screened prior to placement. This material will then be graded on the surface and hydrated in order to obtain an optimal moisture content for compaction. Fill materials will then be transported into the shaft, via a slickline and header, in a manner similar to that proposed for concrete. Following placement, compaction of engineered fill will be completed in the same manner as for bentonite/sand, with the exception that compaction can be accomplished in larger lifts (e.g., 300 mm).

Concrete Cap

The engineered fill will be topped by a surficial concrete cap, representing the final element of the seal system. The cap will serve to:

- Further reduce the potential for subsidence, as concrete is stronger than compacted fill;
- Provide a marker for the shaft locations; and
- Reduce the potential for inadvertent human entry by providing a restrictive barrier at the surface.

The surficial cap will be constructed, at a minimum, of concrete to meet CSA A23.1 (CSA09b). Air entrainment within the concrete is required to minimize adverse effects of freeze/thaw action on the concrete cap. The surficial cap will be constructed through staged pours, in approximately 3 m lifts. It is currently assumed that structural elements are not required within the concrete.

13.6.3.2 Quality Assurance

The seal components will be tested as required during their installation to ensure that they meet specifications.

Appropriate testing requirements to confirm that the properties and integrity of shaft seal components will be established during the detailed design of the shaft seal system.

13.6.4 Decommissioning of Surface Facilities

The majority of surface facility decommissioning will occur following completion of shaft sealing, because these facilities will be required to maintain service to the shafts during the installation of the shaft seals.

The expected order of surface facility decommissioning is as follows:

- Heater building;
- Exhaust fans;
- Ventilation shaft headframe and hoist house, temporary hoisting system and temporary stage winches (installed for shaft stripping and sealing), and collar house;
- Main shaft headframe, temporary hoist system and temporary stage winches (installed for shaft stripping and sealing) as well as the associated buildings, including the WPRB; and
- Electrical substation and emergency generator.

13.6.5 WRMA

The waste rock remaining on the site will be covered by a soil cap and vegetation. The rock pile will be capped with a minimum of 150 mm of soil and topsoil that is suited to the requirements of the local flora. Prior to capping, the waste rock surface will be scarified or ripped in those areas where the rock has been compacted by vehicle traffic. Surface materials will be stabilized and the surface will be contoured as required to promote drainage and to minimize erosion. Wind breaks will be added, if required, for erosion control until such time that the vegetation has taken hold. Any additional rock added during decommissioning will also be capped and vegetated.

In addition, the waste rock pile will be inspected for physical stability during all stages of closure until the site is closed out. The pile would be inspected for tension cracks at the crest of any slopes, signs of new or ongoing failure, and rill or gully erosion both on the rock pile and on the soil cap. The soil cap vegetation will be inspected until it is fully established.

The stormwater pond and drainage ditches will be decommissioned during general site restoration work. They will not be maintained or monitored following decommissioning of the DGR Facility.

13.7 Site Restoration

Following the removal of all surface facilities, the site will be graded and revegetated. Revegetation will be carried out so as to enhance natural vegetation growth and establish self-sustainable vegetation growth. The location of the shafts will be secured to ensure that the possibility of an accidental disturbance is minimized.

Prior to revegetating the site, the ground surface will be scarified or ripped in those areas where the surface has been compacted from vehicle traffic and construction activity. Surface materials will be stabilized and the surface will be contoured as required to blend into the surrounding areas, to promote natural drainage and to minimize erosion. Wind breaks will be added if required for erosion control until such time that the vegetation has taken hold.

A final environmental survey will be performed as part of site restoration work to confirm that there are no residual radioactive or hazardous materials remaining on the site. If necessary, appropriate actions will be taken to remove any radioactive or hazardous materials from the site and to transfer these materials to a licensed disposal facility.

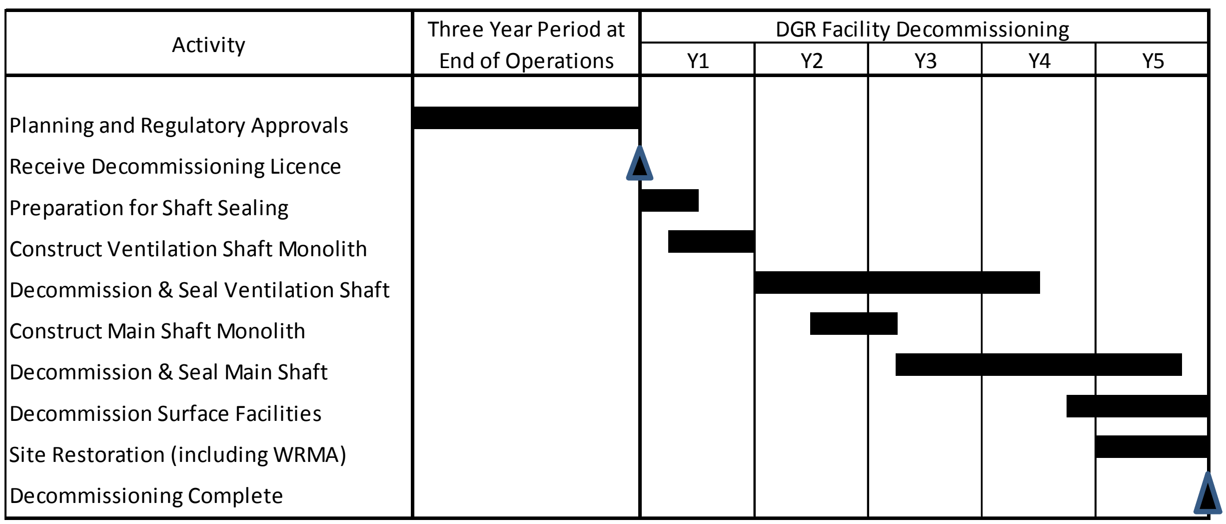
By the end of the site restoration phase, the site areas will be free of industrial hazards. All radioactive contamination (if applicable) in excess of the established clearance levels and all other hazardous materials will have been removed from each site area.

13.8 Decommissioning End-State Report

Following the completion of decommissioning work, an end-state report will be prepared. The report will describe the decommissioning work that has been performed, the outcome of that work, the results of the final surveys that were performed and the interpretation of those results. Other information required by the applicable regulations will also be included. The end-state report will be filed with the CNSC to demonstrate that the intended end-state has been achieved in accordance with the DDP and regulatory requirements.

13.9 Decommissioning Schedule

The outline schedule for completing the five-year decommissioning work program is presented in Figure 13-3 (NWMO11au).



Note:
Figure is from the Preliminary Decommissioning Plan (NWMO11au)

Figure 13-3: Schedule for the End-of-Operations Decommissioning Work

13.10 Waste Management During Decommissioning

13.10.1 Radioactive Waste

During operations, all waste packages will be checked for contamination, and decontaminated if necessary, before they are placed in storage. Abnormal operating occurrences may result in some contamination events during the course of operations; however, it is anticipated that any such contamination will be removed whenever it is discovered. It is expected that there will be little or no radioactive contamination on facility structures, systems and equipment. Consequently, the volume of radioactive waste generated during the decommissioning is roughly estimated to be 10 m^3, in addition to the waste in Section 13.10.2. Operational experience and radiological surveys will be used to prepare a revised estimate for the DDP.

13.10.2 Demolition Waste

Wherever appropriate, equipment and materials from demolition of surface and underground facilities will be recycled or reused elsewhere to minimize requirements for disposal. Those materials that are not recycled will be disposed of in a licensed disposal facility. Any materials or equipment in surface facilities that would be considered radioactive waste will be removed near the start of decommissioning and placed in the underground repository prior to the start of shaft sealing.

It is currently assumed that underground mobile equipment, which has been tested and does not contain any residual radioactive contamination, will be removed to the surface. Once at surface, it is possible that some of the equipment and materials could be salvaged for reuse or for its scrap metal. Alternatively, if the equipment has no value, space is available and approval is received to do so, then the mobile equipment could remain underground. In these instances, all fluids (e.g., fuel, lubricants, hydraulic fluids, etc.) and any other hazardous materials (e.g., batteries) would be removed prior to leaving any equipment underground.

Waste materials resulting from the removal of ventilation shaft and main shaft infrastructure (such as shaft steelwork and concrete lining) will be brought to the surface and reused/recycled wherever possible. Similarly waste rock resulting from excavation of any damaged rock in two shafts will be reused on site wherever possible (e.g., as engineered fill in upper portion of shafts; see Figure 13-2) or could be placed in the WRMA, as noted in Section 13.6.5. Materials from decommissioning of ventilation shaft and main shaft that cannot be reused or recycled will be sent to a licensed disposal facility.

Table 13-1 presents the estimated quantity of waste materials that would arise from the decommissioning of the DGR Facility. As noted above, it is assumed that contamination that could occur during abnormal operational occurrences would have been decontaminated during the specific operational activity.

13.10.3 Hazardous Waste

Conventional and hazardous waste material would be produced during DGR Facility operations and similar wastes would also be produced during decommissioning of the DGR Facility. These wastes would consist of consumable materials, namely rags and coveralls used in maintenance and clean-up operations, solids generated from underground sanitary facilities and other miscellaneous wastes. All waste materials would be collected in waste bins or totes and wastes would be sent for treatment (if necessary) and disposed of at licensed facilities. The projected range of conventional and hazardous waste materials that would be produced by the DGR Facility during decommissioning is shown in Table 13-2.

Table 13-1: Waste Materials Arising from Decommissioning

Structure	Material Type	Quantity[a]
Ventilation shaft	Steel	490 tonnes
	Concrete	5,600 m^3 [b]
	Waste Rock (HDZ)[c]	7,000 m^3
Ventilation shaft headframe	Steel	520 tonnes
	Concrete	260 m^3
Main shaft	Steel	780 tonnes
	Concrete	9,100 m^3
	Waste Rock (HDZ)[c]	8,800 m^3
Main shaft headframe and WPRB	Steel	380 tonnes
	Concrete	8,700 m^3
Other items such as miscellaneous cabling, panels, and other equipment		

Notes:
a. Volumes (in m^3) of material are bulked volumes.
b. It is assumed that less than 10% of the ventilation shaft concrete could be contaminated. However, it would be impractical to separate the contaminated concrete from the remainder of the concrete liner.

Table 13-2: Projected Range of Conventional and Hazardous Wastes

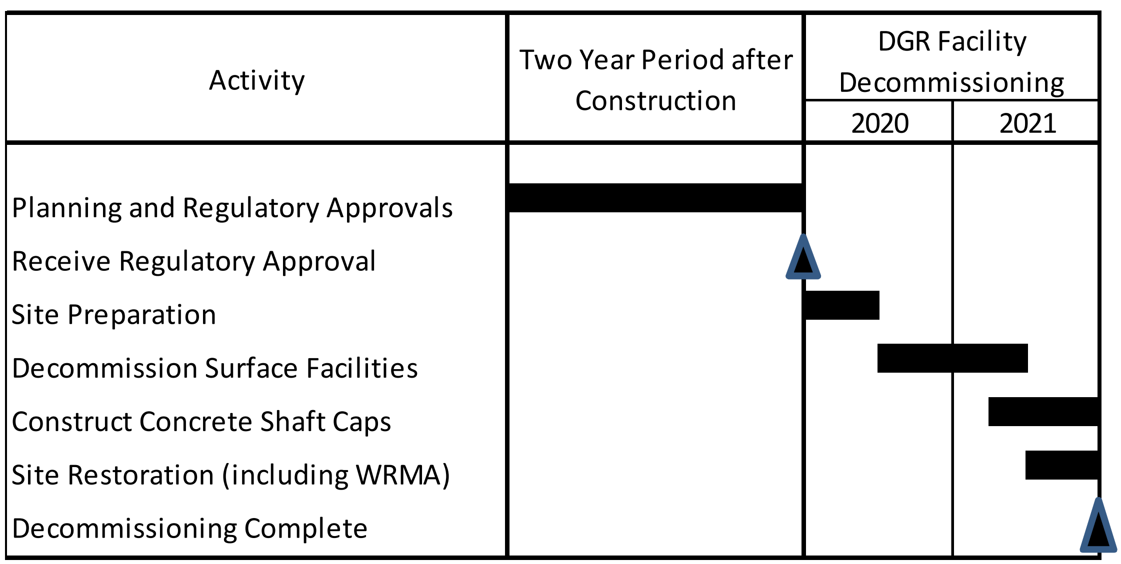
Waste Material	Projected Range of Output
Oils and grease	15,000 – 18,000 L per year
Batteries	60 – 80 kg per year
Solvents	1,500 – 2,500 L per year
Domestic waste (in addition to waste in Table 13-1)	25,000 – 35,000 kg per year
Sanitary waste	8,000 – 12,000 kg per year

13.11 Decommissioning at the End of Construction

Appendix B of the PDP (NWMO11au) describes the decommissioning of the DGR Facility in the event that it is decided to suspend construction prior to completion or to shutdown the facility prior to the beginning of waste emplacement operations. In either case, it is assumed that waste emplacement operations have not begun, and that there is no intention to begin them prior to decommissioning. Decommissioning during or following construction will be similar to decommissioning following operation with one significant difference – the absence of any radioactive materials in the emplacement rooms. Since, in this scenario, no radioactive waste will have been handled at the DGR Facility, decommissioning will not involve any radiological considerations.

The decommissioning process will follow the general steps outlined in Chapter 5 of the PDP; however, the process will be much simpler and will exclude the construction of the concrete monolith and the shaft seal system. No radioactive surveys will be performed, as there will not be any radioactive material in the DGR post-construction. After decommissioning is complete, and regulatory approvals have been received, the site will be abandoned in-situ and available for other uses. Decommissioning of the DGR post-construction will not involve a period of institutional control.

The schedule, including the decommissioning activities required at the end of construction, is presented in Figure 13-4 (NWMO11au).



Note:
Figure is from the Preliminary Decommissioning Plan (NWMO11au)

Figure 13-4: Schedule for the End-of-Construction Decommissioning Work

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14. CONCLUSION

14.1 Overview

The purpose of this chapter is to draw conclusions about whether the DGR can meet the need for safe long-term management of the L&ILW currently stored at the WWMF, as well as the future operational and refurbishment L&ILW produced as a result of operation of OPG-owned or operated nuclear reactors. The information presented in this PSR provides the basis for confirming whether the DGR safety objective stated in Chapter 1 as follows, is met:

"to provide safe long-term management of low and intermediate level waste without posing unreasonable risk to the environment or health and safety of humans."

To facilitate understanding how the information presented in the PSR contributes towards conclusions on DGR safety drawn later in this Chapter, an overview of this information is presented below.

Chapter 1 provided the context for the PSR and the regulatory and international requirements and standards applicable to the DGR. The safety objective for the DGR was defined and the conditions for how it can be demonstrated that this objective can be met, were stated in this Chapter. The strategies for design, safety assessment and management of the DGR were presented and the contents of the PSR were introduced in Chapter 1.

Chapter 2 described the existing site to set the context for the environment and the DGR vicinity – an established nuclear complex with the convenience of available site services necessary for the DGR, a Class I nuclear facility, such as security, emergency response, water, power and telecommunications.

Chapter 3 provided details on how the geosphere at the DGR site was evaluated and characterized, and introduced seven hypotheses that specified geoscientific site attributes and characteristics favourable for safe implementation of the DGR project.

Chapter 4 summarized a large number of site-specific and regional geoscientific studies conducted, the results of which were used to test the hypotheses introduced in Chapter 3. Multiple lines of evidence that support the seven hypotheses were summarized at the end of Chapter 4.

Chapter 5 provided an overview of the physical, radiological and chemical characteristics of the waste to be stored, and provided a description of the containers to be used for emplacement of L&ILW into the DGR.

Chapter 6 provided the key design criteria and described the preliminary DGR design that meets those criteria, providing details of structures and features that contribute toward achieving the safety objective for the DGR.

Chapter 7 provided results of the preclosure safety assessment that included a conservative safety assessment of normal operation, as well as malfunctions and accidents. The assessment demonstrated that wastes can be handled and emplaced in the DGR without undue risk to workers or the general public.

Chapter 8 provided a conservative assessment of the DGR's ability to perform in the postclosure period in a manner that will protect human health and the environment. The assessment included pathway analysis of contaminant releases, contaminant transport, receptor exposure, and potential effects for the expected (normal) evolution scenario. To test the robustness of the DGR, assessment results for unlikely or low probability postulated "what if" scenarios that could disrupt or bypass the repository barriers, were also presented. These scenarios included the human intrusion scenario, the severe shaft seal failure scenario, the poorly sealed borehole scenario, and the vertical fault scenario. Chapter 8 concluded that acceptance criteria for DGR safety in the postclosure period can be met.

Chapter 9 presented information to demonstrate that the DGR site can be prepared and DGR construction can be carried out safely. It described the activities to be undertaken for site preparation for, and construction of, the DGR, and the technology and methods to be used for conducting these activities safely.

Chapter 10 demonstrated that OPG has the governing framework in place and is qualified to develop and implement appropriate operational programs to safely operate the DGR post-construction, after obtaining an operating licence.

Chapter 11 described the quality assurance aspects of the DGR project and how it is being ensured that the implementation of this project will be in accordance with established quality standards.

Chapter 12 presented the public information and involvement programs in place to ensure that the community will be kept informed of the progress of the DGR project and communication channels will remain open to respond to public questions and enquiries about the project.

Chapter 13 provided a summary of the Preliminary Decommissioning Plan to safely decommission the DGR at the end of either its construction or its operational life. This plan demonstrated that appropriate methodology, materials and technology are available to meet the end-state objectives for the DGR, so human health and the environment remain protected in the postclosure period.

Finally, this chapter provides an overview of the operating experience from facilities similar to the DGR, summarizes confidence-building arguments from the evidence provided in the PSR, and uses these arguments to confirm compliance with the DGR safety objective stated in Chapter 1. In accordance with the regulatory requirement stated in G-320, and international practice, this exercise is conducted through building a safety case for the DGR, presented in Section 14.3.

14.2 International L&ILW Deep Geologic Repositories

The DGR would be the first deep geologic repository for L&ILW in Canada and there are no directly comparable Canadian facilities. There is, however, in the U.S. and overseas, good operating experience with geologic repositories for similar wastes. Current repositories are listed in Table 14-1.

Table 14-1: Characteristics of Current International L&ILW Repositories

Repository Location	Design	Waste Capacity and Type	Start of Operation
Forsmark, Sweden	50 m deep, granite	63,000 m^3 L&ILW	1988
Olkiluoto, Finland	70–100 m deep, granite	8,000 m^3 L&ILW	1992
Loviisa, Finland	110 m deep, granite	7,400 m^3 L&ILW	1997
Waste Isolation Pilot Plant (WIPP), U.S.	655 m deep, salt	170,000 m^3 transuranic waste	1999
Konrad, Germany	800-1300 m deep, iron-oolithic limestones surrounded by shales	300,000 m^3 L&ILW	Licensed; under construction

The U.S. WIPP is particularly relevant as it is situated in a sedimentary setting at a depth similar to the DGR, and OPG has gained valuable insight into the construction and operation of its DGR through many visits to WIPP and interactions with WIPP staff.

WIPP is located 26 miles outside of Carlsbad, New Mexico. It began operation in 1999. By 2010, the facility had processed 9,000 shipments of waste. WIPP is expected to operate until 2070.

Waste is placed in rooms 655 m (2,150 feet) deep underground that have been excavated within a 1000 m (3,000 feet) thick massive salt formation (>99% NaCl) that has been stable for more than 250 million years.

WIPP accepts mixed waste that contains both radioactive and hazardous constituents. Radioactive material emplaced at WIPP is regulated by the federal government (United States Environmental Protection Agency), but hazardous components are regulated by the state (New Mexico Environment Department).

WIPP achieved a key milestone on November 18, 2010 when it was recertified by the United States Environmental Protection Agency, for a second time since it started operating in 1999. On December 1, 2010, WIPP also received its first renewal of the original Hazardous Waste Facility Permit received in October 1999. In the area of occupational safety, WIPP recently surpassed 4 million hours without an injury causing days away from work. This marked the third time since opening in 1999 that WIPP exceeded the 4 million hour mark (WIPP11).

14.3 Safety Case

'Safety case' has been defined by the NEA as "*an integration of arguments and evidence that describe, quantify and substantiate the safety, and the level of confidence in the safety, of the deep geological repository system."* (NEA04)

Detailed arguments and evidence that constitute the DGR safety case are presented in this section in Table 14-3. The conclusions drawn from the DGR safety case are presented in Section 14.4. However, to provide context, a summary of the national and international guidance on what constitutes the safety case for a long-term waste management facility, is presented first.

CNSC Guide G-320 states: "Demonstrating long term safety consists of providing reasonable assurance that waste management will be conducted in a manner that protects human health and the environment. This is achieved through the development of a safety case, which includes a safety assessment complemented by various additional arguments based on:

1 Appropriate selection and application of assessment strategies;

2 Demonstration of system robustness;

3 The use of complementary indicators of safety; and

4 Any other evidence that is available to provide confidence in the long-term safety of radioactive waste management."

The safety case presented here has been constituted to provide reasonable assurance that the DGR will meet its safety objective in the long term. This is consistent with CNSC Guide G-320 statement above, which takes into account the ICRP (ICRP00) position that: "*Proof that the disposal system satisfies criteria cannot be absolute because of the inherent uncertainties, especially in understanding the evolution of the geologic setting, biosphere, and engineered barriers over the long term. .....a decision on the acceptability of a disposal system should be based on reasonable assurance rather than on an absolute demonstration of compliance."*

Additionally, the safety case presented here also demonstrates how the DGR safety objective will be met in the near term. This approach is consistent with the IAEA Safety Standard WS-R-4, paragraph 3.46 (IAEA06b) that states, "*the safety case for a geological facility addresses both operational safety and post-closure safety*".

An iterative approach has provided a good basis for progressive strengthening of the safety case as new information under different programs becomes available. Also mentioned earlier in Chapter 1, this approach is shown in Table 14-2.

Table 14-2: Iterative Approach for Development of the Safety Case

Site Characterization	Inventory	Design	Safety Assessment	Safety Case
Generic Data (non-site)	Reference Inventory Report (draft)	Early Conceptual Design	V0 "Dry Run"	Early Draft Preliminary Safety Case
Phase I Geosynthesis	Reference Inventory Report (2008)	Conceptual Design	V1 Peer Review	Draft Preliminary Safety Case
Phase II a & b Geosynthesis	Reference Inventory Report (2010)	Preliminary Design	V2 Site Preparation and Construction Licence	Preliminary Safety Case

Table 14-3: Arguments and Evidence for DGR Safety[1]

1. THE DGR PROVIDES LONG-TERM ISOLATION AND CONTAINMENT	
Relevant Arguments and Evidence	**Reference**
1-1 The DGR is situated deep underground.	
• The repository is located at a depth of approximately 680 mBGS, within the argillaceous limestone of the Cobourg Formation.	PSR Sections 1.2 and 4.1.2.1
• Lake Huron is the only major water body in the vicinity of the Bruce nuclear site and the DGR is located at a safe distance from Lake Huron. The DGR site is located approximately 1 km away (laterally) from the Lake Huron shoreline, and will be sited at a depth that is approximately 450 m below the deepest point of the lake.	PSR Section 2.1.1 and Figure 4-3
1-2 The DGR is enclosed by multiple natural barriers.	
• The deep DGR-series boreholes confirm that the Bruce nuclear site is underlain by 34 bedrock formations, members and units comprising layered carbonate, shale, evaporite, siltstone and sandstone, with a total sedimentary thickness of approximately 840 m above the Precambrian crystalline rock basement.	PSR Section 4.1.2.1 and Figure 4-9
• The thickness and orientation of bedrock formations encountered beneath the Bruce nuclear site are highly consistent and predictable. Within an area of approximately 1.5 km^2 enclosing the DGR footprint, information derived from the deep drilling and coring program confirms that Ordovician formation thickness variations are on the order of meters and do not exceed 5%. Formation dips are uniformly 0.59° +/- 0.08° (≈10 m/km) to the southwest towards the Michigan Basin.	PSR Section 4.1.2.2 and Table 4.2
• The results of the two-dimensional seismic reflection survey, including nine survey lines totalling 19.7 km, provide evidence for the continuous and undeformed nature of the bedrock stratigraphy. The inclined drilling and continuous coring activities, which targeted potential sub-vertical faults or fault zone structures in proximity to the DGR footprint, did not encounter any evidence for faulting or stratigraphic offset through the target intervals. This result was confirmed through core logging,	PSR Section 4.1.2.3

[1] Main arguments are in the dark colour box, the supporting arguments are in the lighter colour box, the evidence is presented using bullets, and the associated references from the PSR are in the right-hand column.

1. THE DGR PROVIDES LONG-TERM ISOLATION AND CONTAINMENT	
Relevant Arguments and Evidence	**Reference**
geophysical logging and in-situ hydraulic testing of the inclined boreholes. Evidence supporting vertical fault displacement, or the occurrence of steeply-oriented linear and elongate hydrothermally dolomitized reservoirs within the Ordovician carbonate rocks, is absent with no proximal deep seated fault system identified.	
• Within the Ordovician sediments that host and enclose the proposed DGR there are numerous units characterized as aquicludes that posses extremely low rock mass permeabilities. The host Cobourg Formation has a very low horizontal hydraulic conductivity (K_H) ≈ 10^{-14} m/s. The overlying > 200 m of Ordovician shales (3 formations) have rock mass horizontal hydraulic conductivities <10^{-13} m/s. The underlying 150 m of Ordovician carbonates (5 formations) have K_H values ranging from ≈10^{-15} to 10^{-10} m/s. Above the Ordovician sediments, the Silurian sediments have K_H values, which are on the order of <10^{-11} m/s.	PSR Section 4.4.1
• Observed abnormal hydraulic heads in the Ordovician and Cambrian rocks and high vertical hydraulic gradients strongly suggest: i) extremely low rock mass hydraulic conductivities at formation scale; and ii) that vertical transmissive connectivity across bedrock aquitards/aquicludes is highly unlikely.	PSR Section 4.4.4.3
• The long-term barrier integrity of the Ordovician shale cap rock is, in part, demonstrated by analogue with hydrocarbon cap rock seals located in the Appalachian and Michigan basins, and with observations from the Bruce nuclear site. These observations include the occurrence of paleo under-pressures, sealed fractures, low formation permeabilities, and a relatively low degree of thermal maturation at the onset of the oil window.	PSR Sections 4.1.1.2, 4.1.2.2, 4.1.2.3 and 4.4.4.3
• No geochemical evidence has been found for the infiltration of glacial or recent meteoric recharge water into the Ordovician host or bounding formations. The stable water isotopes (^{18}O and ^{2}H) indicate that the maximum depth of glacial meltwater infiltration is 328.5 mBGS (reference depth in DGR-1/2) within the Salina A1 carbonate aquifer. Further, the results of numerical simulations – paleohydrogeology – provide insight into long-term groundwater system performance and indicate that: 1) glacial perturbations do not alter the governing solute transport mechanisms within the deep groundwater system; and 2) single and multiple glaciation scenarios, when modelled using regional and site specific parameters, do not result in the infiltration of glacial meltwater into the deep groundwater system.	PSR Sections 4.3.2.3 and 4.4.4.2

1. THE DGR PROVIDES LONG-TERM ISOLATION AND CONTAINMENT	
Relevant Arguments and Evidence	**Reference**
1-3 The DGR is positioned within a stable deep diffusion dominant groundwater system.	
• Horizontal hydraulic conductivities (K_H) within the Cobourg Formation (DGR host rock), the overlying Ordovician shales (Georgian Bay, Blue Mountain and Queenston formations, and the Collingwood Member), and underlying Ordovician limestones and dolostones (Sherman Fall, Kirkfield, Coboconk, Gull River, and Shadow Lake formations) are extremely low (≈10^{-15} to 10^{-10} m/s). Vertical hydraulic conductivities (K_V) within the same formations are lower. Such conditions are consistent with a diffusion dominated regime.	PSR Sections 4.4.1 and 4.4.4.3
• The effective diffusion coefficient (D_e) for HTO in the Ordovician shales is on the order of 10^{-12} m^2/s, and in the carbonates 10^{-13} to 10^{-12} m^2/s. D_e values obtained with HTO are on average 1.9 times greater than D_e values obtained with an iodide tracer. This difference is attributed to the influence of anion exclusion in lowering the tracer-accessible porosity for iodide. The low D_e values, coupled with the low hydraulic conductivities of the Ordovician sediments, indicate that solute migration is diffusion dominated in the deep groundwater system.	PSR Sections 4.3.2.4 and 4.4.4.1
• The occurrence of isotopically distinct types of methane and helium in separate zones (one zone in the Upper Ordovician shale and Cobourg Formation, another zone in the underlying Middle Ordovician carbonates) demonstrates that there has been little to no cross-formational mixing (advective or diffusive) while these gases were resident in the system. The occurrence of radiogenic $^{87}Sr/^{86}Sr$ ratios in the Middle and Upper Ordovician porewater are interpreted to result from a combination of water-rock interaction, in-situ ^{87}Rb decay, and diffusive transport upward from the Precambrian bedrock below the site. These mechanisms indicate extremely long residence times.	PSR Section 4.3.2.3
• The chemistries of the deep brines indicate that they were formed by evaporation of seawater, which was subsequently modified by fluid-rock interaction processes. The Cl/Br and Na/Cl ratios, as well as the stable water isotope data, suggest that the deep groundwater system contains evolved ancient sedimentary brines at, or near, halite saturation. The nature of the brines, in particular the high salinities and the enriched $\delta^{18}O$ values (enriched in ^{18}O with respect to the GMWL) in the porewaters, indicate that the deep system is isolated from the shallow groundwater system and that the porewaters have resided in the system for geologic time periods.	PSR Section 4.3.2.3
• Host and enclosing bedrock formation mineralogy strongly suggest that the groundwater geochemical environment is reducing, and that oxygenated	PSR Sections 4.3.2.2 and

1. THE DGR PROVIDES LONG-TERM ISOLATION AND CONTAINMENT	
Relevant Arguments and Evidence	**Reference**
groundwaters have not penetrated to repository depths over geologic time.	4.3.2.3
1-4 The DGR is situated in a seismically quiet region.	
• The Bruce nuclear site is located within the tectonically stable interior of the North American continent, which is characterized by low rates of seismicity. No earthquake exceeding magnitude 5 has been observed in the regional monitoring area in 180 years of record. The maximum earthquake within the 150 km radius study area is a M4.3 event at 99 km from the site with a focal depth of 11 km. This is consistent with the seismic hazard information provided in the 2005 National Building Code of Canada.	PSR Section 4.5.2.1
• Based on the results of a Probabilistic Seismic Hazard Assessment performed for the Bruce nuclear site, the far field/regional seismic sources are the dominant contributors to the hazard for the site at ground level. The estimated surface bedrock peak ground motions are 18.7% and 60.1% g for annual probabilities of 10^{-5} and 10^{-6} events, respectively. Historical evidence from underground structures worldwide shows that strong ground motions are reduced at depth due to surface effects.	PSR Section 4.5.2.1
• The micro-seismic monitoring network installed and commissioned in August 2007 confirms the lack of low level seismicity (> M1.0) within the vicinity of the Bruce nuclear site implying no seismogenic structures or faults within or in close proximity to the DGR footprint.	PSR Section 4.5.2.1
• Field-based neotectonic and geologic investigations in the DGR area, including outcrop and Quaternary paleoseismic mapping and deep drilling have found no evidence for the presence of structural features that would indicate a higher seismic hazard near the Bruce nuclear site than that estimated from the regional rate of earthquake occurrence.	PSR Sections 4.5.2.1 and 4.5.2.2
1-5 DGR openings are geomechanically stable.	
• Precedent construction experience with the excavation of underground openings in the Ordovician sediments indicates that excavated openings in either the Ordovician shale or Ordovician limestone are likely to be virtually dry and stable.	PSR Section 4.2.2
• The laboratory testing of the Cobourg Formation core rock samples reveals a high strength argillaceous limestone with an average uniaxial compressive strength (UCS) value of 113 MPa. These rock strength conditions compare favorably with	PSR Section 4.2.2

1. THE DGR PROVIDES LONG-TERM ISOLATION AND CONTAINMENT	
Relevant Arguments and Evidence	**Reference**
other sedimentary formations considered internationally for long-term radioactive waste management purposes.	
• No borehole breakouts were observed in the deep DGR boreholes, which provides a constraint on the possible range of the in-situ stress magnitudes. At the repository horizon, the range of stress ratios is estimated to be: σ_H/σ_v from 1.5 to 2.0; σ_h/σ_v from 1.0 to 1.2. Observed borehole deformation over time frames to 16 months strongly suggests that the orientation of maximum horizontal stress is similar to that of the Michigan Basin, a NE to ENE direction.	PSR Sections 4.2.3 and 4.2.4
• Numerical simulations of repository evolution illustrate under varied long-term rock mass properties and loading scenarios (i.e., glacial ice sheet, seismic ground motions and repository gas pressure), that the barrier integrity of the enclosing Ordovician bedrock formations is unaffected.	PSRSection 4.5.5
• A 3-dimensional numerical simulation that explored DGR shaft stability for a range of observed geomechanical formation properties under various loading scenarios was undertaken. Due to the vertical geometry of the shaft, glacial loading has only a minor effect on differential ground stress in horizontal plane. The effect of damage zone (HDZ and EDZ) along the shaft is minor. Similarly, pore pressure and seismic shaking will not significantly increase the predicted damage zone around the shaft.	PSR Sections 4.5.5 and 8.6.4
1-6 Natural resource potential is low, reducing potential for human intrusion.	
• The DGR is situated approximately 680 mBGS within an extremely low permeability, saline aquiclude system. The groundwater at the repository depth is not potable (TDS > 200 g/L) and extremely low bedrock formation hydraulic conductivities (< 10^{-13} m/s) cannot yield groundwater.	PSR Sections 1.2, 4.3.2.1 and 4.4.1
• The near surface groundwater system (0-100 mBGS) is potable and permeable. With increasing depth groundwater quality becomes increasingly saline and formation specific yield decreases. These long-lived natural conditions (extremely high salinities and low hydraulic conductivities) would discourage deep drilling for groundwater resources.	PSR Sections 4.3.2.1 and 4.4.1
• No commercial hydrocarbon accumulations were encountered during site characterization activities or known from hydrocarbon exploration in the surroundings of the Bruce nuclear site. No structural, lithological, chemical or hydrological evidence suggests that the Bruce nuclear site is proximal to an	PSR Sections 4.5.3.1, 4.1.2.2, 4.1.2.3 and 4.4..4.3

1. THE DGR PROVIDES LONG-TERM ISOLATION AND CONTAINMENT	
Relevant Arguments and Evidence	**Reference**
ancient hydrothermal dolomite system.	
• The potential for future shale gas in the Blue Mountain Formation and Collingwood Member is very low based on site specific information: - The shales contain low average total organic carbon (TOC) of < 1 % with a local maximum of 2.5 %. - The formations did not pass through the gas generation window during burial and diagenesis. The sediments did not reach depths that would yield high enough temperatures to allow for the thermal cracking of kerogen, which is necessary to generate natural gas.	PSR Sections 4.1.2.2, 4.1.2.3 and 4.5.3.1
• Historical oil and gas records indicate that no commercially viable oil and gas reserves have been discovered or developed within a 40 km radius of the DGR site.	PSR Section 4.5.3.1
• Site specific drilling and coring has confirmed that the commercial salt deposits occurring further to the south of the site (in Kincardine and Goderich) are absent beneath the Bruce nuclear site.	PSR Section 4.5.3
1-7 Chemical and hydrogeologic conditions limit contaminant mobility at the repository depth.	
• The large volume of limestone host rock (calcium carbonate) provides a chemical buffering capacity which will act to maintain conditions within the repository around approximately neutral pH.	PSR Section 8.6.1
• The bentonite-sand shaft seals and the host rocks have capacity to chemically sorb radionuclides, especially the argillaceous component of these materials. Under high water salinities at the depths of the DGR, sorption would occur through surface complexation processes, not ion exchange.	PSR Table 8-6
1-8 Resaturation of the repository with groundwater will be very slow.	
• The full resaturation of the repository is gradual, taking more than one million years due to the low permeability of the host rock and shaft seals, and the generation of gas within the repository. Delay in resaturation limits the releases from the wastes to groundwater in the repository, and subsequent migration into the porewaters in the geosphere.	PSR Sections 8.6.4.2 and 8.8.2.1.

1. THE DGR PROVIDES LONG-TERM ISOLATION AND CONTAINMENT	
Relevant Arguments and Evidence	**Reference**
1-9 Shaft design provides long-term isolation and DGR integrity.	
• The shaft concrete liner and adjacent rock in the excavation damaged zone (EDZ) is removed on closure in order to provide a better shaft seal.	PSR Section 13.6.3
• The primary seal material is a bentonite-sand mixture. Bentonite is natural clay, which has the ability to swell and self seal when exposed to water. Sufficient clay will be used to ensure swelling under the DGR saline conditions.	PSR Sections 8.6.2.9 and 13.6.3.1
• Bentonite and sand are durable natural materials, typically millions of years old. The thick bentonite-sand seals are expected to be substantially unchanged under DGR conditions, in part due to the low water flow rate and the low temperatures.	PSR Section 8.6.2.9 (Box 2)
• In the reference design, the bentonite-sand seals are augmented with an asphalt seal layer. This asphalt layer provides an additional, independent sealing capability and provides containment during bentonite swelling.	PSR Section 13.6.3.1
1-10 Radioactivity of waste will decrease with time due to radioactive decay.	
• By volume, approximately 80% of waste emplaced in the DGR is LLW. LLW contains primarily short-lived radionuclides with half lives shorter than or equal to 30 years.	PSR Table 5-6, 5-7, 5-8 and Figure 5-1
• Key radionuclides at closure in terms of total inventory and mobility are tritium and C-14. Tritium will decay within a few hundred years, while C-14 will significantly decay within 60,000 years. After these have decayed, the remaining radionuclides are primarily only mobile within groundwater, which is a very slow process at the DGR site.	PSR Sections 5.9, 5.10 and 8.6.4.1
• The total amount of radioactivity remaining in the repository after about 10,000 years is less than that of the natural radioactivity in the overlying rock at the Bruce nuclear site. The residual radioactivity after about 100,000 years is less than that of the natural radioactivity in the rock directly above the repository footprint.	PSR Section 8.6.4.1
1-11 Corrosion resistant ILW degrades very slowly.	
• Most of the long-lived radionuclides are embedded within stainless steel and Zircaloy components. These cannot be released until the component corrodes, which will be slow process in these corrosion-resistant alloys.	PSR Table 5-8, Section 8.6.4.1 and Table 8-6

2. PRECLOSURE AND POSTCLOSURE SAFETY CRITERIA ARE MET	
Supporting Arguments and Evidence	**Reference**
2-1 Potential impacts to humans and non-human biota during preclosure period (operations) will be below the acceptance criteria.	
• During normal operations, the maximum dose from DGR emissions to the general public due to normal operations will be about 0.1% of the regulatory dose limit of 1 mSv/year.	PSR Sections 7.4.2.3 and 7.7.1
• The estimated doses from potential accidents at the DGR are below the public radiological dose criterion of 1 mSv.	PSR Sections 7.5.4 and 7.7.2
• During normal operations the maximum dose to workers will be below OPG's occupational dose target of 10 mSv/year.	PSR Sections 7.4.4.2 and 7.7.1
• During normal operations, the maximum dose to non-NEWs on the site will be below OPG's Radiation Protection Requirement of 0.5 µSv/hr at the facility fence (based on 2000 hr/year and annual dose rate limit of 1 mSv).	PSR Sections 7.4.4.2 and 7.7.1
• The estimated doses to a worker from potential accidents at the DGR are below the worker radiological dose criterion of 50 mSv.	PSR Sections 7.5.4 and 7.7.2
• The estimated doses to non-human biota during normal operation and due to potential accident scenarios are below the acceptance criteria.	PSR Section 7.1.2.1
• The estimated non-radiological impacts on public from potential waste package accidents are below the acceptance criteria.	PSR Sections 7.5.4 and 7.7.2
• The estimated non-radiological impacts on workers from potential waste package accidents are below the acceptance criteria. The conventional occupational health and safety program will ensure worker safety through effective risk assessment and safe work planning.	PSR Sections 7.5.4, 7.7.2 and 10.2
• The non-radiological impacts to non-human biota during normal operations and accident scenarios are below the acceptance criteria.	PSR Section 7.1.2.1
2-2 Under the normal evolution scenario, future (postclosure) impacts to humans and non-human biota will be insignificant.	
• The calculated peak annual doses to humans for the normal evolution scenario	PSR Sections 8.6.4.4

2. PRECLOSURE AND POSTCLOSURE SAFETY CRITERIA ARE MET	
Supporting Arguments and Evidence	**Reference**
are more than five orders of magnitude smaller than the 0.3 mSv/year public dose criterion and the typical natural background radiation dose (2 mSv/year).	and 8.9
• For the normal evolution scenario, concentrations of radionuclides and of non-radionuclide contaminants in surface media are well below the relevant environmental protection criteria.	PSR Section 8.6.4.4

3. THE DGR SYSTEM IS ROBUST	
Supporting Arguments and Evidence	**Reference**
3-1 The geology is robust.	
• The DGR is located deep within old, stable, laterally extensive rocks that provide multiple natural barriers. The host rock formation has remained stable after millions of years of tectonics, seismic activity and glaciation.	See supporting arguments 1-1 to 1-8
3-2 There is low risk of impacts even under disruptive postclosure scenarios.	
Human Intrusion Scenario: • If a borehole is inadvertently drilled into the repository in the future, and gases and material from the repository are brought to surface and not appropriately contained (inconsistent with general practice), the calculated doses could be about 1 mSv for the drill crew and about 1 mSv/year for a future person farming on the contaminated site. These are similar to the natural background radiation dose rate. No effect is expected. However, this scenario is unlikely, and the scenario meets the DGR risk criterion. • The potential dose from this scenario decreases significantly after about 60,000 years due to decay of C-14 and Nb-94.	PSR Section 8.7.1
Severe Shaft Seal Failure: • If the entire shaft seals were to unexpectedly fail to about 2-3 orders of magnitude from their design value (i.e. to about 10^{-9} m/s hydraulic conductivity), calculated doses to a person living on top of the repository shaft would reach around 1 mSv/a, which is the allowable public dose criterion.	PSR Section 8.7.2

3. THE DGR SYSTEM IS ROBUST	
Supporting Arguments and Evidence	**Reference**
• The main risk from shaft seal failure is C-14 in gas. The total amount of C-14 in the repository is approximately equal to the site annual Derived Release Limit for C-14 release to air. Even if the entire C-14 inventory was released in gas, the dose to people living around the current Bruce nuclear site would be around 1 mSv. • The potential dose from this scenario decreases significantly after about 60,000 years due to decay of C-14.	
Poorly Sealed Borehole: • If a deep site characterization borehole around the repository were not sealed properly, the peak calculated dose to a person living on the repository site is very small, orders of magnitude less than the 1 mSv/year public dose criterion.	PSR Section 8.7.3
Vertical Fault: • If a vertical fault were to exist at 100 m from the repository, the peak calculated dose to a person living on or near the repository site is very small, orders of magnitude less than the 1 mSv/year public dose criterion.	PSR Section 8.7.4
Large Earthquake: • The DGR is located in a stable region of the North American continent. Large earthquakes are very unlikely. • Analyses indicate that a large earthquake would have little effect on the repository, mostly causing some rockfall within emplacement rooms. This is included in the Normal Evolution Scenario safety assessment. • The effects of a large earthquake are also bounded by the above Severe Shaft Failure and the Vertical Fault scenarios.	PSR Sections 4.5.2.1 and 8.6.1
3-3 Natural features would act to delay contaminant release.	
• The low permeability of the geosphere limits the rate of water inleakage into the repository, which limits the rate at which contaminants can get into water at the repository depth. Calculations indicate that the full resaturation of the repository is gradual, taking more than one million years.	PSR Sections 8.6.1 and 8.8.2.1
• The surrounding limestone (carbonate) rock would act as a chemical buffer, tending to equilibrate conditions within the repository towards neutral pH. Under this condition, chemical reactions are generally less aggressive. For example, container corrosion rates would generally be slower.	PSR Section 8.6.1

3. THE DGR SYSTEM IS ROBUST	
Supporting Arguments and Evidence	**Reference**
• The permeable Guelph and Salina A1 rock formations provide a natural lateral pathway that tends to divert laterally any gas or groundwater moving up through the shaft. This provides more dispersion and decay before any release to the shallow groundwater system.	PSR Sections 4.4.4.3, 8.8.2.5 and 8.8.2.9
• The groundwater is brine below 180 m depth. This means the deeper groundwaters are dense, which is stabilizing against vertical flow.	PSR Sections 4. 3.1 and 4.3.5
• The low hydraulic heads in the overlying Ordovician shales provide a "sink" for groundwater. Groundwater is pulled into these formations, including downwards from the higher rock formations.	PSR Section 4.4.1
3-4 Gas pressure tends towards the natural steady-state hydraulic pressure.	
• Detailed modelling results from a wide range of calculation cases show that the long-term gas pressure within repository tends towards 7-9 MPa - at or slightly above the natural steady-state hydraulic pressure of 7-8 MPa. At higher pressures, gas and water seep out from the repository, while at lower pressures, gas and water seep into the repository.	PSR Section 8.8.2.4
• The shale caprock at the site is able to hold natural gas pressures of 70% lithostatic for millions of years. At this site, 70% lithostatic is 12 MPa.	PSR Section 4.1.2.3
3-5 Large safety assessment margin exists.	
• The calculated dose results for all Normal Evolution Scenario cases are many orders of magnitude below the dose criterion. The peak doses do not occur for hundreds of thousands to millions of years.	PSR Section 8.9
• Uncertainties have been addressed through a wide range of calculation cases. The resulting dose impacts remain orders of magnitude below the dose criterion.	PSR Section 8.8 and 8.9
• Safety assessment models include a range of conservatisms, such as: - Containers do not provide any barrier to contaminant release; - Instant release of most radionuclides on contact of waste with water; - Tritium and C-14 are released as gases even of there is little water; - Water consumption by chemical reactions is not included in the repository	PSR Sections 8.6.2.1

3. THE DGR SYSTEM IS ROBUST	
Supporting Arguments and Evidence	**Reference**
water balance; - Solubilities and sorptions are either neglected or conservative values assumed; - Wastes degrade completely, with maximum production of gas; - Shaft EDZ is based on the largest shaft EDZ dimension calculated by geomechanical modelling; - EDZ does not self-seal with time, such as from creep or precipitation processes; - The Cambrian overpressure is steady throughout the calculation; while many calculations neglect the Ordovician underpressures; - No horizontal groundwater flow in the Guelph and Salina A1 upper carbonate formations; - A self-sufficient farming family is living directly on top of the repository, drawing water from a groundwater well placed downstream from the repository; and - Damage or degradation occurs at time of closure – e.g., all rockfall, container failure, concrete degradation, and shaft seal degradation.	
• Glaciation is a large potential perturbation to the site. However, the first ice sheet coverage of the site is probably not until after 60,000 years, which is after most of the C-14 has decayed. The remaining radionuclides are not volitile, and are limited to move with groundwater. However, the deep groundwater system is stagnant and diffusion dominated.	PSR Section 8.8.1 Argument 1-3

4. THE DGR CAN BE CONSTRUCTED, OPERATED AND DECOMMISSIONED SAFELY	
Supporting Arguments and Evidence	**Reference**
4-1 Strength and geomechanical properties of the host rock are favourable for construction and operation of underground facilities.	
• Geomechanical testing on site-specific core samples of the Bruce nuclear site indicates an UCS of approximately 113 MPa.	PSR Section 4.2.2
• The low hydraulic conductivity of the Cobourg Formation and enclosing formations gives strong indication that the DGR openings (tunnels and shafts)	PSR Section 4.4.1

4. THE DGR CAN BE CONSTRUCTED, OPERATED AND DECOMMISSIONED SAFELY	
Supporting Arguments and Evidence	**Reference**
will remain virtually dry during construction and operations. • The orientation of the emplacement rooms in accordance with the local stress regime is intended to maximize opening stability.	PSR Section 6.3
4-2 The DGR has been designed for safe construction, operation and decommissioning, incorporating good engineering practices and use of known technologies.	
• Safety design features (cage chairing system, conveyance arresting system, refuge stations, fire suppression and detection system, ventilation system (low to high) and capacity (sufficient); emergency power, ventilation shaft as second emergency egress for personnel).	PSR Section 6.3
• Use of existing mining practices and technologies. - Liners and rock support to prevent rock fall.	PSR Section 6.3
• Construction of all the emplacement rooms and access tunnels will be carried out prior to, not concurrently with, operations.	PSR Figure 9-1
4-3 Experience with facilities similar to the DGR demonstrate a strong operational record.	
• Operation of the WIPP, which is a facility for disposal of radioactive waste in the U.S., demonstrates an excellent operational safety record. Safe transportation, handling and deep geologic disposal of radioactive wastes has been demonstrated on a daily basis at the WIPP for more than 10 years.	PSR Section 14.2
• Sweden (Forsmark) has a repository for radioactive operational waste constructed at a depth of 50 m under the Baltic sea near Forsmark. The repository has a waste capacity of 63,000 m^3, which has been operating since 1988.	PSR Section 14.2
• Finland has two granite repositories for L&ILW. - The repository in Olkiluoto is 70-100 m deep with waste capacity of 8,000 m^3 (began operation in 1992). - The repository in Loviisa is 110 m deep with a capacity of 7,400 m^3 (began operation in 1997).	PSR Section 14.2
• OPG's Western Waste Management Facility (WWMF) has received positive safety ratings from the CNSC in each of the safety areas which support health and safety of workers and the public, environmental protection and security	Commission Member Document CMD 07-

4. THE DGR CAN BE CONSTRUCTED, OPERATED AND DECOMMISSIONED SAFELY	
Supporting Arguments and Evidence	**Reference**
(i.e., operations and maintenance, radiation protection, environmental protection quality management and fire protection). Based on these ratings, the WWMF received an extended licence in 2007, valid for 10 years. The WWMF meets all CNSC requirements and performance objectives.	H3A.
4-4 Operational programs and controls ensure that emplacement activities can be safely performed.	
• Radiation Protection Program: Worker exposure will be managed by proven OPG radiation practices, including application of ALARA during the design stage and as operational experience is accumulated. Procedures exist for contamination control and dose control.	PSR Section 10.1
• Conventional Occupational Health and Safety Program: Programs to achieve worker safety, such as Hazardous Materials program, PPE program and specific procedures to ensure worker safety exist. Effective operational controls are developed through effective risk assessment and safe work planning.	PSR Section 10.2
• Environmental Protection Program: Under the OPG's Environment, Health and Safety Management program, environmental policies have been established that will be complied with when project-specific procedures are developed.	PSR Section 10.3
• Monitoring Program: Environmental, radiological, geotechnical and underground air quality monitoring programs will be in place for the DGR.	PSR Section 10.4
• Staffing and Training Programs are in place.	PSR Section 10.5
• Fire Protection Program: Minimization of fire risk will be achieved by using closed non-combustible waste packages and by minimizing combustible materials underground.	PSR Sections 10.6 and 6.8.1
• Emergency Preparedness and Emergency Response Program: Emergency response is available on the Bruce nuclear site. MRT will be available.	PSR Section 10.7
• Inspection and Maintenance Program: Conduct of Operations and Maintenance program is in place. Specific program for inspection and maintenance will be developed for the DGR project.	PSR Section 10.8
• Records Management: Records and document control will be in accordance with the records and document control program.	PSR Section 10.9

4. THE DGR CAN BE CONSTRUCTED, OPERATED AND DECOMMISSIONED SAFELY	
Supporting Arguments and Evidence	**Reference**
• Safety Management System: Quality Assurance, Performance Assurance (indicators/targets i.e., monitoring etc.), procedural adherence, event free tools, operational experience, change control, etc., will be in place for the DGR project.	PSR Chapter 11
• Operational Limits and Conditions: Waste packages must meet the DGR waste acceptance criteria or they will not be handled at the DGR. This specifically includes limits on contamination and dose rate.	PSR Section 5.5

14.4 Summary and Conclusion

This section presents the conclusions drawn from the safety case presented in Table 14-3. In Table 14-4 below, the arguments from Table 14-3 are examined against the compliance conditions for the safety objective set out in Section 1.6.

It is evident from Table 14-4 that the arguments for DGR safety supported by sound evidence provided in Table 14-3 constitute a strong case for DGR safety. The arguments consist of multiple lines of reasoning drawn from the information in the PSR and provide confidence that:

- The DGR provides good long-term isolation and containment, which are the two safety functions identified in Section 1.6;
- The preclosure and postclosure safety criteria are met; all doses are below the regulatory limits and environmental impacts are below the acceptance criteria;
- The DGR system is robust - even under disruptive scenarios; and
- The DGR can be constructed, operated and decommissioned safely.

Table 14-4: Summary of Arguments for DGR Safety

Compliance Conditions for the Safety Objective	Summary of Arguments
1. The DGR provides long-term isolation and containment	1-1 The DGR is situated deep underground.
	1-2 The DGR is enclosed by multiple natural barriers.
	1-3 The DGR is positioned within a stable deep diffusion dominant groundwater system.
	1-4 The DGR is situated in a seismically quiet region.
	1-5 DGR openings are geomechanically stable.
	1-6 Natural resource potential is low, reducing potential for human intrusion.
	1-7 Chemical and hydrogeologic conditions limit contaminant mobility at the repository depth.
	1-8 Resaturation of the repository with groundwater will be very slow.
	1-9 Shaft design provides long-term isolation and DGR integrity.
	1-10 Radioactivity of waste will decrease with time due to radioactive decay.
	1-11 Corrosion resistant ILW degrades very slowly.
2. Preclosure and postclosure safety criteria are met.	2-1 Potential impacts to humans and non-human biota during the preclosure period (operations) will be below the acceptance criteria.
	2-2 Under the normal evolution scenario, future impacts (postclosure) to humans and non-human biota will be insignificant.
3. The DGR system is robust.	3-1 The geology is robust.
	3-2 There is low risk of impacts even under disruptive postclosure scenarios.
	3-3 Natural features would act to delay contaminant release.
	3-4 Gas pressure tends towards the natural steady-state hydraulic pressure
	3-5 Large safety assessment margin exists.
4. The DGR can be constructed, operated and decommissioned safely.	4-1 Strength and geomechanical properties of the host rock are favourable for construction and operation of underground facilities
	4-2 The DGR has been designed for safe construction, operation and decommissioning, incorporating good engineering practices and use of known technologies.
	4-3 Experience with facilities similar to the DGR demonstrate a strong operational record.
	4-4 Well established operational programs and controls ensure that emplacement activities can be safely performed.

The conditions for compliance with the DGR safety objective stipulated in Section 1.6, are, therefore, achieved.

The long-term safety case is examined further against other national and international criteria (identified in Section 14.3) below.

As stated in Section 14.3, CNSC Guidance G-320 states that demonstrating long-term safety consists of providing reasonable assurance that waste management will be conducted in a manner that protects human health and the environment. It further states that this is achieved through the development of a safety case, which includes a safety assessment complemented by various additional arguments based on factors listed in the left-hand column of Table 14-5. In this table, the safety case for the DGR is examined against these factors, which form the basis for establishing if reasonable assurance has been provided.

Table 14-5: Long-Term Safety Case vs. CNSC Guidance G-320

CNSC Guidance G-320	Long-Term Safety Case
1. Appropriate selection and application of assessment strategies.	The assessment strategy outlined in Section 1.8.4 clearly demonstrates that its various elements were selected carefully and Section 8.2 shows that the strategy was applied appropriately to the various assessments conducted to expose weaknesses in the DGR system.
2. Demonstration of system robustness.	System robustness is demonstrated through the arguments 3-1 to 3-5 in Table 14-3, with associated evidence based on the analyses presented in Chapters 4 and 8.
3. The use of complementary indicators of safety.	Section 8.6. to 8.8 use performance indicators (e.g., contaminant amounts and fluxes) as well as the safety indicators such as environmental concentrations. These are complementary indicators of safety.
4. Any other evidence that is available to provide confidence in the long-term safety of radioactive waste management.	Sufficient "other evidence" is provided in the safety case including the geological characteristics, lack of availability of natural resources at and near the DGR site, isolation of groundwater, multiple natural barriers, engineered barriers, international experience with repositories, and natural analogues.

Table 14-5 demonstrates that reasonable assurance of long-term safety has been achieved for the DGR.

In the area of international guidance, the IAEA safety standard WS-R-4 states that "*the safety case for a geological facility addresses both operational safety and post-closure safety*". Arguments 4-2 to 4-5 in Table 14-3 demonstrate that the DGR is safe to operate.

Based on the information provided in the PSR, **it is concluded that the DGR meets the safety objective: *"to provide safe long-term management of low and intermediate level waste without posing unreasonable risk to the environment or health and safety of humans".***

In addition to the safety case, an important consideration for the DGR that needs to be reiterated is its ideal location at the Bruce nuclear site with the following advantages:

- Near WWMF to minimize waste transportation distance;
- Nuclear site, with considerable infrastructure; and
- Community supportive of the DGR.

OPG is qualified and has programs and plans in place to safely prepare the site, construct and operate the proposed DGR, in accordance with the regulatory requirements and established quality standards.

Based on the above synthesis of the information presented in the PSR, the overall conclusion is that, it will be safe to prepare the site, construct, operate and decommission the proposed DGR at the Bruce nuclear site. Furthermore, there is reasonable assurance that the DGR will provide safe long-term management of OPG's L&ILW.

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15. REFERENCES

ADAMS91	Adams J., and Basham P.W., "The Seismicity and Seismotectonics of Eastern Canada" in The Geology of North America, Decade Map Vol. 1, Neotectonics of North America, edited by B. Slemmons et al., p. 261-275, Boulder, ColoMcHone and Butler, 1991.
AL10a	Al T., Xiang Y., and Cavé L., "Technical Report: Measurement of Diffusion Properties by X-Ray Radiography and by Through-Diffusion Techniques Using Iodide and Tritium Tracers: Core Samples from OS-1 and DGR-2", TR-07-17, 2010.
AL10b	Al T., Xiang Y., Loomer D., and Cavé L., "Technical Report: Measurement of Diffusion Properties by X-Ray Radiography and by Through-Diffusion Techniques Using Iodide and Tritium Tracers: Core Samples from DGR3 and DGR4", TR-08-27, May 2010.
ANDERSON05	Anderson R., Garamszeghy M., Rodrigues F., and Nicholls E., "Installing and Commissioning a New Radioactive Waste Tracking System – Lessons Learned", Proceedings of the Canadian Nuclear Society Conference - Waste Management, Decommissioning and Environmental Restoration for Canada's Nuclear Activities: Current Practices and Future Needs, Ottawa, Canada, May 8-11 2005, INEEL/CON-04-02346 (Preprint).
ANDJELKOVIC98	Andjelkovic D., and Cruden A.R., "Relationships Between Fractures in Paleozoic Cover Rocks and Structures in the Precambrian Basement, South Central Ontario: in Summary of Field Work and Other Activities 1998", Ontario Geological Survey, Misc. Paper 169, p.275-280, 1998.
ARAVENA95	Aravena R., Wassenaar L.I, and Plummer L.N., "Estimating 14C Groundwater Ages in a Methanogenic Aquifer", Water Resources Research, Vol. 31, No. 9, p.2307-2317, April 1995.
ARMSTRONG93	Armstrong D.K., "Paleozoic Geology of the Central Bruce Peninsula", Ontario Geological Survey, Open File Report 5856, Ontario Ministry of Northern Development and Mines, Toronto, Ontario, 1993.
ARMSTRONG06	Armstrong D.K., and Carter T.R., "An Updated Guide to the Subsurface Paleozoic Stratigraphy of Southern Ontario", Ontario Geological Survey, Open File Report 6191, p.214, 2006.

ARMSTRONG10	Armstrong D.K., and Carter T.R., "The Subsurface Paleozoic Stratigraphy of Southern Ontario", Ontario Geological Survey, Special Volume 7, p.301, 2010.
ASSEL03	Assel R., Cronk K., and Norton D., "Recent Trends in Laurentian Great Lakes Ice Cover", Climatic Change Vol. 57, p.185-204, 2003.
BACKBLOM02	Bäckblom G. and Munier R., "Technical Report: Effects of Earthquakes on the Deep Repository for Spent Fuel in Sweden Based on Case Studies and Preliminary Model Results", SKB, TR-02-24, Stockholm, Sweden, 2002.
BAHR94	Bahr J. M., Moline G. R., and Nadon G. C., "Anomalous Pressures in the Deep Michigan Basin", in: P. Ortoleva (Ed.), Basin Compartments and Seals, American Association of Petroleum Geologists, Memoir, 61, 153-165, 1994.
BAILEY84a	Bailey S.M.B., and Cochrane R.O., "Evaluation of the Conventional and Potential Oil and Gas Reserves of the Cambrian of Ontario", Ontario Geologic Survey, Open File Report 5499, 1984.
BAILEY84b	Bailey S.M.B., and Cochrane R.O., "Evaluation of the Conventional and Potential Oil and Gas Reserves of the Ordovician of Ontario", Ontario Geologic Survey, Open File Report 5498, 1984.
BARKER84	Barker J.F., and Pollock S.J., "The Geochemistry and Origin of Natural Gases in Southern Ontario", Bulletin of Canadian Petroleum Geology, Vol. 32, No. 3, p.313-326, September 1984.
BARNETT84	Barnett R.L., Arima, M., Blackwell J.D., Winder C.G., Palmer H.C., and Hayatsu A., "The Picton and Varty Lake Ultramafic Dikes: Jurassic Magmatism in the St. Lawrence Platform near Belleville, Ontario" Canadian Journal of Earth Sciences, Vol. 21, p.1460–1472, December 1984.
BARNETT92	Barnett, P.J., "Quaternary Geology of Ontario", in Geology of Ontario, Ontario Geological Survey, Special Vol. 4, Part 2, p.1011-1088, 1992.
BEAR88	Bear J., "Dynamics of Fluids in Porous Media", Dover edition, Dover Publications Inc., 1988.

BETHKE90	Bethke C. M., and Marshak S., "Brine Migrations Across North America; the Plate Tectonics of Groundwater", Annual Review of Earth and Planetary Sciences, Vol.18, p.287-315, May 1990.
BOTTOMLEY99	Bottomley D.J., Katz A., Chan L.H., Starinsky A., Douglas M., Clark I.D., and Raven K.G., "The Origin and Evolution of Canadian Shield Brines: Evaporation or Freezing of Seawater? New Lithium Isotope and Geochemical Evidence from the Slave Craton", Chemical Geology, Vol.155, p.295–320, 1999.
BOTTOMLEY02	Bottomley D.J., Renaud R., Kotzer T. and Clark I.D., "Iodine-129 Constraints on Residence Times of Deep Marine Brines in the Canadian Shield", Geology, Vol.30, No. 7, p.587-590, 2002.
BOTTOMLEY03	Bottomley D.J., Chan L.H., Katz A., Starinsky A. and Clark I.D., "Lithium Isotope Geochemistry and Origin of Canadian Shield Brines", Ground Water, Vol.61, p.847-856, 2003.
BOTTOMLEY04	Bottomley D.J., and Clark I. D., "Potassium and Boron Co-Depletion in Canadian Shield Brines: Evidence for Diagenetic Interactions Between Marine Brines and Basin Sediments", Chemical Geology, Vol.203, p.225-236, 2004.
BOTTOMLEY05	Bottomley D.J., Clark I. D., Battye N., and Kotzer T., "Geochemical and Isotopic Evidence for a Genetic Link Between Canadian Shield Brines, Dolomitization in the Western Canada Sedimentary Basin, and Devonian Calcium-Chloridic Seawater", Canadian Journal of Earth Sciences, Vol.42, p.2059–2071, 2005.
BOUBEL94	Boubel R. W., Fox D.L., and Turner D.B., Fundamentals of Air Pollution 3rd Edition. San Diego: Academic Press, 1994.
BOYCE02	Boyce J.J., and Morris W.A., "Basement-Controlled Faulting of Paleozoic Strata in Southern Ontario, Canada: New Evidence from Geophysical Lineament Mapping", Tectonophysics, Vol.353, No. 1-4, p.151-171, August 2002.
BP02	"Environmental Assessment Study Report – Volume 1: Main Report, Bruce A Units 3 & 4 Restart Environmental Assessment", Bruce Power, August 2002.
BP04	"New Fuel Project for Bruce B Environmental Assessment Study Report", Bruce Power, October 2004.

BP05a	"Bruce A Refurbishment for Life Extension and Continued Operations Project Environmental Assessment – Environmental Assessment Study Report, Volume1: Main Report", Bruce Power, December 2005.
BP05b	"Report on Results of Storm Water Monitoring Program, Bruce Power Site, Tiverton, Ontario", Golder Associates Limited., 021-1243, January 2005.
BP07	"Annual Summary and Assessment of Environmental Radiological Data for 2006", Bruce Power, B-REP-03419-00007 R000, April 2007.
BP08	"Annual Summary and Assessment of Environmental Radiological Data for 2007", Bruce Power, B-REP-03419-00008 R000, April 2008.
BP09	"Annual Summary and Assessment of Environmental Radiological Data for 2008", Bruce Power, B-REP-03419-00009-R001, June 2009.
BP10	"Annual Summary and Assessment of Environmental Radiological Data for 2009", Bruce Power, B-REP-03419-00010 R000, April 2010.
BRACKE08	Bracke G., and Miller W., "Contribution to a More Realistic Approach in Assessing the Release of C-14 from Low-Level Radioactive Waste Repositories", Journal of Contaminant Hydrology, Vol. 102, p.210-216, December 2008.
BREDEHOEFT63	Bredehoeft J.D., Blyth C.R., White W.A., and Maxey G.B, "Possible Mechanism for Concentration of Brines in Subsurface Formations", Bulletin of the American Association of Petroleum Geologists, Vol.47, No.2, p.257-269, February 1963.
BRIGHAM71	Brigham R.J., "Structural Geology of Southwestern Ontario and Southeastern Michigan", Ontario Department of Mines and Northern Affairs - Petroleum Resources Section, Paper 71-2, 1971.
BROGLY98	Brogly P.J., Martini I.P. and Middleton G.V., "The Queenston Formation: Shale-Dominated, Mixed Terrigenous Carbonate Deposits of Upper Ordovician, Semiarid, Muddy Shores in Ontario, Canada", Canadian Journal of Earth Sciences, Vol. 35, p.702-719, February 1998.
BROOKFIELD88	Brookfield M.E., and Brett C.E., "Paleoenvironments of the Mid-Ordovician (Upper Caradocian) Trenton Limestones of Southern Ontario, Canada: Storm Sedimentation on a Shoal-Basin Shelf Model", Sedimentary Geology, Vol.57, p.75-105, May 1988.

BRUNTON08	Brunton F.R., and Dodge, J.E.P., "Karst of Southern Ontario and Manitoulin Island", Ontario Geological Survey, Groundwater Resources Study 5, 2008.
BUDAI91	Budai J.M., and Wilson W.L., "Diagenetic History of the Trenton and Black River Formations in the Michigan Basin", Geological Society of America, Special Paper 256, p.73-88, 1991.
CARPENTER78	Carpenter A. B., "Origin and Chemical Evolution of Brines in Sedimentary Basins", Oklahoma Geological Survey, Circular, Vol.79, p.60-77, 1978.
CARR00	Carr S. D., Easton R. M., Jamieson R. A., and Culshaw N. G., "Geologic Transect Across the Grenville Orogen of Ontario and New York", Canadian Journal of Earth Sciences, Vol.37, No.2-3, p.193–216, 2000.
CARTER90a	Carter T.R., and Easton R.M., "Extension of Grenville Basement Beneath Southwestern Ontario: Lithology and Tectonic Subdivisions", in Subsurface Geology of Southwestern Ontario: a Core Workshop, 1990.
CARTER90b	Carter T.R., "Subsurface Geology of Southwestern Ontario; a Core Workshop", America Association of Petroleum Geologists, 1990 Eastern Section Meeting, hosted by Ontario Petroleum Institute, London, Ontario.
CARTER96	Carter T.R., Trevail R.A., and Easton R.M., "Basement Controls on Some Hydrocarbon Traps in Southern Ontario, Canada"; Geological Society of America, Special Papers, Vol.308, p.95-107, 1996.
CAVÉ09	Cavé L.C., Al T.A., Xiang Y. and Vilks, P., "A Technique for Estimating One-Dimensional Diffusion Coefficients in Low Permeability Sedimentary Rock Using X-Ray Radiography: Comparison with Through-Diffusion Measurements", Journal of Contaminant Hydrology Vol.103, p.1-12, January 2009.
CCME07	"Canadian Environmental Quality Guidelines", Canadian Council of the Ministers of Environment, http://ceqg-rcqe.ccme.ca/, Site Accessed: December 23, 2010.
CEAA92	Canadian Environmental Assessment Act, S.C. 1992, c. 37.
CEAA09	"Guidelines for the Preparation of the Environmental Impact Statement for the Deep Geologic Repository of Low- and Intermediate-Level Radioactive Wastes", Canadian Environmental Assessment Agency and Canadian Nuclear Safety Commission, January 2009.

CHACKO08 Chacko T. and Deines P., "Theoretical Calculation of Oxygen Isotope Fractionation Factors in Carbonate Systems", Geochimica et Cosmochimica Acta, Vol.72, p.3642–3660, 2008.

CHAPMAN84 Chapman L.J., and Putman D.F., "The Physiography of Southern Ontario", Ontario Geological Survey, Special Vol. 2, p.270, 1984.

CLARK97 Clark I.D. and Fritz P. Environmental Isotopes in Hydrogeology. Boca Raton, Florida, USA: Lewis Publishers, 1997.

CLARK10a Clark I., Mohapatra R., Mohammadzadeh H. and Kotzer T., "Technical Report: Pore Water and Gas Analyses in DGR-1 & DGR-2 Core", Intera Engineering TR-07-21, 2010.

CLARK10b Clark I., Liu I., Mohammadzadeh H., Mohapatra R., Zhang P., and Wilk M., "Technical Report: Pore Water and Gas Analyses in DGR-3 & DGR-4 Core", Intera Engineering TR-08-19, 2010.

CNSC00a "Regulatory Policy P-119: Policy on Human Factors", Canadian Nuclear Safety Commission, October 2000.

CNSC00b "Regulatory Guide G-206: Financial Guarantees for the Decommissioning of Licensed Activities", Canadian Nuclear Safety Commission, June 2000.

CNSC00c "Regulatory Guide G-219: Decommissioning Planning for Licensed Activities", Canadian Nuclear Safety Commission, June 2000.

CNSC01a "Regulatory Policy P-211: Compliance", Canadian Nuclear Safety Commission, May 2001.

CNSC01b "Regulatory Policy P-223: Protection of the Environment", Canadian Nuclear Safety Commission, February 2001.

CNSC01c "Regulatory Guide G-225: Emergency Planning at Class I Nuclear Facilities and Uranium Mines and Mills", Canadian Nuclear Safety Commission, August 2001.

CNSC03a "Regulatory Guide G-221: A Guide to Ventilation Requirements for Uranium Mines and Mills", Canadian Nuclear Safety Commission, June 2003.

CNSC03b "Regulatory Guide G-276: Human Factors Engineering Program Plans", Canadian Nuclear Safety Commission, June 2003.

CNSC03c “Regulatory Guide G-278: Human Factors Verification and Validation Plans”, Canadian Nuclear Safety Commission, June 2003.

CNSC03d “Regulatory Guide G-4: Measuring Airborne Radon Progeny at Uranium Mines and Mills”, Canadian Nuclear Safety Commission, June 2003.

CNSC04a “Regulatory Policy P-290: Managing Radioactive Waste”, Canadian Nuclear Safety Commission, July 2004.

CNSC04b “Regulatory Guide G-129: Keeping Radiation Exposure and Doses ‘As Low as Reasonably Achievable (ALARA)’”, Rev. 1, Canadian Nuclear Safety Commission, October 2004.

CNSC04c “Regulatory Guide G-217: Licensee Public Information Programs”, Canadian Nuclear Safety Commission, January 2004.

CNSC04d “Draft Regulatory Guide G-224: Environmental Monitoring Program at Class I Nuclear Facilities and Uranium Mines and Mills” Canadian Nuclear Safety Commission, July 2004.

CNSC05 “Siting and Site Characterization for Long-term Radioactive Waste Containment Facilities”, Attachment to CNSC Letter from Klassen, K. to Nash, K.E., “Geologic Repository for Low and Intermediate Level Waste – Site Characterization”, W-CORR-00531-00186, May 2005.

CNSC06a “Regulatory Guide G-320: Assessing the Long Term Safety of Radioactive Waste Management”, Canadian Nuclear Safety Commission, December 2006.

CNSC06b “Regulatory Guide G-296: Developing Environmental Protection Policies, Programs and Procedures at Class I Nuclear Facilities and Uranium Mines and Mills”, Canadian Nuclear Safety Commission, March 2006.

CNSC06c “Regulatory Standard S-296: Environmental Protection Policies, Programs and Procedures at Class I Nuclear Facilities and Uranium Mines and Mills”, Canadian Nuclear Safety Commission, March 2006.

CNSC08 CNSC Letter from Klassen K. to Nash K.E., “Deep Geologic Repository (DGR) – Proposed Acceptance Criteria for Postclosure Safety Assessment”, 00216-CORR-00531-00051, August 20, 2008.

CNSC09a CNSC Letter from Klassen K. to King F., "DGR Project for OPG's LILW - Proposed Acceptance Criteria for Postclosure Safety Assessment of Radiological Impacts on Non-Human Biota and for Non-Radiological Impacts on Human and Non-Human Biota", CD# DGR-CORR-00531-0021, August 17, 2009.

CNSC09b CNSC Letter from Klassen K. to King F., "DGR Project for OPG's LILW DGR Project Quality Plan", CD# DGR-CORR-00531-0025, October 23, 2009.

CNSC10 CNSC Letter from Klassen K. to King F., "DGR Project for OPG's LILW – Updates to Acceptance Criteria for Postclosure Safety Assessment of Non-Radiological Impacts on Human and Non-Human Biota", CD# DGR-CORR-00531-0043, May 31 2010.

COAKLEY95 Coakley B., and Gurnis, M., "Far Field Tilting of Laurentia during the Ordovician and Constraints on the Evolution of a Slab Under an Ancient Continent", Journal of Geophysical Research, Vol.100, p.6313-6327, 1995.

CONIGLIO92 Coniglio M., and William-Jones A.E., "Diagenesis of Ordovician Carbonates from the Northeast Michigan Basin, Manitoulin Island Area, Ontario: Evidence from Petrography, Stable Isotopes and Fluid Inclusions", Sedimentology, Vol. 39, No. 5, p.813-836, October 1992.

CONIGLIO94 Coniglio M., Sherlock R., Williams-Jones A.E., Middleton K., and Frape S.K., "Burial and Hydrothermal Diagenesis of Ordovician Carbonates from the Michigan Basin, Ontario, Canada", International Association of Sedimentologists, Special Publication 21, p.231-254, 1994.

CONIGLIO03 Coniglio M., Zheng Q., and Carter T.R., "Dolomitization and Recrystallization of Middle Silurian Reefs and Platformal Carbonates of the Guelph Formation, Michigan Basin, Southwestern Ontario", Bulletin of Canadian Petroleum Geology, Vol. 51, No. 2, p.177-199, June 2003.

CRC83 CRC Handbook of Chemistry and Physics 64th Edition. Boca Raton, Florida, USA: CRC Press, 1983.

CROUGH81 Crough S.T., "Mesozoic Hotspot Epeirogeny in Eastern North America", Geology, Vol.9, No.1, p.2-6, January 1981.

CSA91	"Guidelines for Calculating Radiation Doses to the Public from a Release of Airborne Radioactive Material under Hypothetical Accident Conditions in Nuclear Reactors", Canadian Standards Association, CAN/CSA N-288.2-M91 (R2008), April 1991.
CSA05	"Management System Requirements for Nuclear Power Plants", Canadian Standards Association, CAN/CSA N286-05, February 2005.
CSA08a	"Management of Low- and Intermediate- Level Radioactive Waste", Canadian Standards Association, CSA N292.3, March 2008.
CSA08b	"Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities", Canadian Standard Association, CSA N288.1, September 2008.
CSA09a	"Decommissioning of Facilities Containing Nuclear Substances", Canadian Standards Association, CSA N294-09, 2009.
CSA09b	"Concrete Materials and Methods of Concrete Construction", Canadian Standards Association, CSA A23.1-09, 2009.
CSA10	"Environmental Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills", Canadian Standards Association, N288.4-10, May 2010.
DAMJANAC10	Damjanac B., and Fairhurst C., "Evidence for a Long-Term Strength Threshold in Crystalline Rock", Rock Mechanics and Rock Engineering, Vol.43, p.513-531, 2010.
DAVIES06	Davies G.R., and Smith L.B., "Structurally Controlled Hydrothermal Dolomite Reservoir Facies: An Overview", American Association of Petroleum Geologists Bulletin, Vol.90, No. 11, p.1641-1690, November 2006.
DIXON02	Dixon, D., Chandler N., Graham J., and Gray M.N., "Two Large-Scale Sealing Tests Conducted at Atomic Energy of Canada's Underground Research Laboratory: The Buffer-Container Experiment and the Isothermal Test", Canadian Geotechnical Journal, Vol.39, p. 503-518, 2002.

DOLLAR88 — Dollar P.S., "Geochemistry of Formation Waters, Southwestern Ontario, Canada and Southern Michigan, U.S.A.: Implications for Origin and Evolution", M.Sc. Thesis, University of Waterloo, 1988.

DOLLAR91 — Dollar P.S., Frape S.K., and McNutt R.H., "Geochemistry of Formation Waters, Southwestern Ontario, Canada and Southern Michigan U.S.A.: Implications for Origin and Evolution", Ontario Geological Survey, Open File Report 5743, 1991.

DOUCHE07 — Douche C., "Interim Storage for Tritiated Wastes", Proceedings of the 8th International Conference on Tritium Science and Technology, Rochester, USA, September 16-21 2007.

DOUGLAS00 — Douglas M., Clark I.D., Raven K. and Bottomley D., "Groundwater Mixing Dynamics at a Canadian Shield Mine", Journal of Hydrology, Vol.235, p.88–103, 2000.

DREVER79 — Drever, J. I., Lawrence J.R. and Antweiler R.C. "Gypsum and Halite from the Mid-Atlantic Ridge, DSDP Site 395", Earth and Planetary Science Letters, Vol.42, p.98-102, 1979.

EASTON92 — Easton R.M., "The Grenville Province and the Proterozoic History of Central and Southern Ontario; Geology of Ontario", Ontario Geological Survey, Special Volume 4, Part 2, p.714-904, 1992.

EASTON95 — Easton R.M., and Carter T.R., "Geology of the Precambrian Basement Beneath the Paleozoic of Southwestern Ontario", in Ojakangas, R.W., Dickas, A.B., and Green, J.C., (eds), Basement Tectonics 10, Kluwer Academic Publishers, The Netherlands,p. 221-264, 1995.

EC03 — "Priority Substances List Assessment Report – Releases of Radionuclides from Nuclear Facilities (Impact on Non-Human Biota) - Canadian Environmental Protection Act", Environment Canada and Health Canada. 2003.

EC10a — Environment Canada, "National Climate Data and Information Archive - Canadian Climate Normals 1961-1990", http://climate.weatheroffice.gc.ca/climate_normals/results_1961_1990_e.html?stationID=1032&province=ALL&stationName=Wiarton&searchType=BeginsWith. Site Accessed: November 17, 2010.

EC10b	Environment Canada, "Map of Canada's Lightning Hot Spots", http://www.ec.gc.ca/meteo-weather/default.asp?lang=En&n=9FAED4CF-1. Site Accessed: November 5, 2010.
ECKERMANN96	Eckermann K.F., and Leggett R.W., "DCFPAK: Dose Coefficient Data File Package for Sandia National Laboratory", Oak Ridge National Laboratory, Report ORNL/TM-13347, 1996.
ENGELDER80	Engelder T., and Geiser P., "On the Use of Regional Joint Sets as Trajectories of Paleostress Fields during the Development of the Appalachian Plateau", Journal of Geophysical Research, Vol. 85, No. B11, p.6319-6341, 1980.
ENGELDER06	Engelder T., and Whitaker, A., "Early Jointing in Coal and Black Shale: Evidence for an Appalachian-Wide Stress Field as a Prelude to the Alleghanian Orogeny", Geology, Vol.34, 581-584, 2006.
EPA90	Environmental Protection Act (Ontario), R.S.O. 1990 Chapter E.19.
FARQUHAR87	Farquhar R.M., Haynes S.J., Mostaghel M.A., Tworo A.G., MacQueen R.W., and Fletcher I.R., "Lead Isotope Ratios in Niagara Escarpment Rocks and Galena: Implications for Primary and Secondary Sulphide Deposition", Canadian Journal of Earth Sciences, Vol. 24, No. 8, p.1625-1633, 1987.
FITZGERALD09	Fitzgerald W.R., "Ministry of Culture Stage 2 Archaeological Assessment: Bruce New Build Project and Deep Geologic Repository Project Bruce Nuclear Site. Part Lots 11-31 Concession A (Lake Range) Bruce Township Municipality of Kincardine Bruce County", May 2009.
FRAPE84	Frape S. K., Fritz P., and McNutt R. H., "The Role of Water-Rock Interaction in the Chemical Evolution of Groundwaters from the Canadian Shield", Geochimica et Cosmochimica Acta, Vol.48, p.1617-1627, 1984.
FRAPE87	Frape S.K. and Fritz P., "Geochemical Trends for Groundwaters from the Canadian Shield", in: Fritz, P., Frape, S.K. (Eds.), Saline Water and Gases in Crystalline Rocks, Geological Association of Canada Special Paper 33, 211-233, 1987.
FRITZ82	Fritz, P., and Frape S. K., "Saline Groundwaters in the Canadian Shield – A First Overview", Chemical Geology, Vol.36, p.179-190, 1982.

FRITZ87	Fritz P., Drimmie R.J., Frape S.K., and O'Shea K., "The Isotopic Composition of Precipitation and Groundwater in Canada", in IAEA International Symposium on the Use of Isotope Techniques in Water Resources Development, IAEA-SM-299/017: 14, 1987.
GOLDER03	"Report on LLW Geotechnical Feasibility Study, Western Waste Management Facility, Bruce Site, Tiverton, Ontario", Golder Associates Limited, 021-1570, January 2003.
GOLDER05	"Hydrocarbon Resource Assessment of the Trenton-Black River Hydrothermal Dolomite Play in Ontario", Ontario Oil, Gas and Salt Resources Library, p.35 and four appendices, Golder Associates Limited, 2005.
GOLDER06	"Phase II: Assessment of the Crayfish Species and Population Investigation in Project and Representative Offsite Areas at the Western Waste Management Facility", Golder Associates Limited, 06-1112-026, October 2006.
GOLDER10a	"Annual Follow-up Monitoring Program Report, 2008 - Bruce A Refurbishment for Life Extension and Continued Operations Environmental Assessment", Golder Associates Limited, 09-1112-0038, April 2010.
GOLDER10b	"Annual Follow-up Monitoring Program Report, 2009 - Bruce A Refurbishment for Life Extension and Continued Operations Environmental Assessment", Golder Associates Limited, 10-1151-0175, September 2010.
GRAF65	Graf D.L., Friedman I. and Meents W., "The Origin of Saline Formation Waters, II: Isotopic Fractionation by Shale Micropore Systems", Illinois State Geological Survey, Circular 393, 1965.
GREENE08	Greene S., Battye N., Clark I., Kotzer T. and Bottomley D., "Canadian Shield Brine from the Con Mine, Yellowknife, NT, Canada: Noble Gas Evidence for an Evaporated Palaeozoic Seawater Origin Mixed with Glacial Meltwater and Holocene Recharge", Geochimica et Cosmochimica Acta, 72, 4008–4019, 2008.
GRASTY04	Grasty R.L., and LaMarre J.R., "The Annual Effective Dose from Natural Sources of Ionising Radiation in Canada", Radiation Protection Dosimetry, Vol. 108, No. 3, p.215-226, 2004.

GROSS91	Gross M.R., and Engelder T., "A Case for Neotectonic Joints Along the Niagara Escarpment", Tectonics, Vol. 10, No. 3, p. 631-641, 1991.
GROSS92	Gross M.R., Engelder T., and Poulson S.R., "Veins in the Lockport Dolostone: Evidence for an Acadian Fluid Circulation System", Geology, Vol.20, p.971-974, November 1992.
GROVE06	Grove Software, MicroSkyshine Version 2.10, Lynchburg, USA, 2006.
GROVE09	Grove Software, MicroShield Version 8.02, Lynchburg, USA. 2009.
HAMBLIN06	Hamblin A., "The "Shale Gas" Concept in Canada - A Preliminary Inventory of Possibilities", Geological Survey of Canada, Open File 5384, 2006.
HAMBLIN08	Hamblin A., "Hydrocarbon Potential of the Paleozoic Succession of Southwestern Ontario: Preliminary Conceptual Synthesis of Background Data", Geological Survey of Canada, Open File 5730, 2008.
HANCOCK85	Hancock P.L., "Brittle Microtectonics: Principles and Practice", Journal of Structural Geology, Vol.7, No.3-4, p.437-457, 1985.
HANCOCK89	Hancock P.L., and Engelder T., "Neotectonic Joints", Geological Society of America Bulletin, Vol.101, p.1197-1208, 1989.
HANMER92	Hanmer S., and McEachern S.J., "Kinematical and Rheological Evolution of a Crustal-Scale Ductile Thrust Zone, Central Metasedimentary Belt, Grenville Orogen, Ontario", Canadian Journal of Earth Sciences, Vol.29, p.1779-1790, 1992.
HANOR79	Hanor J.S. Geochemistry of Hydrothermal Ore Deposits. New York, United States: John Wiley & Sons, 1979.
HANSEN00	Hansen F.D., and Knowles M.K., "Design and Analysis of a Shaft Seal System for the Waste Isolation Pilot Plant", Reliability Engineering and System Safety, Vol.69, No. 1-3, p.87-98, September 2000.
HARPER95	Harper D.A., Longstaffe F.J., Wadleigh M.A., and McNutt R.H., "Secondary K-Feldspar at the Precambrian-Paleozoic Unconformity, Southwestern Ontario", Canadian Journal of Earth Sciences, Vol.32, No. 9, p.1432-1450, September 1995.

HAYEK08	Hayek S.J., Drysdale J.A., Adams, J., Peci V., Halchuck and Street P., "Seismic Monitoring Annual Report 2007", December 2008.
HAYEK09	Hayek S.J., Drydale J.A., Adams J., Peci V., Halchuk S., Woodgold C., and Street P., "Seismic Monitoring Annual Report 2008", July 2009.
HAYEK10	Hayek S.J., Drydale J.A., Adams J., Peci V., Halchuk S., and Street P "Seismic Monitoring Annual Report 2009", December 2010.
HC00	"Canadian Guidelines for the Management of Naturally Occurring Radioactive Materials (NORM)", Health Canada, October 2000.
HC10	"Guidelines for Canadian Drinking Water Quality – Summary Table", Health Canada, December 2010.
HEAGLE10	Heagle D., and Pinder L., "Opportunistic Groundwater Sampling in DGR-3 and DGR-4", TR-08-18, June 2010.
HEAMAN00	Heaman L.M. and Kjarsgaard B.A., "Timing of Eastern North American Kimberlite Magmatism: Continental Extension of the Great Meteor Hotspot Track", Earth and Planetary Science Letters, Vol.178, No.3-4, p.253-268, May 2000.
HEIDBACH08	Heidbach O., Tingay M., Barth A., Reinecker J., Kurfeß D., and Müller B., "The World Stress Map Database Release 2008", doi:10.1594/GFZ.WSM.Rel2008, 2008.
HERWEGH08	Herwegh M., and Mazurek M., "Feasibility SEM Study: Primary and Secondary Salts and Sulfates in the Paleozoic of the DGR-2 Borehole, Bruce, Southwestern Ontario", Progress Report PR 08-03, Institute of Geological Sciences, University of Bern, Switzerland, September 26, 2008.
HOLST82	Holst T.M., "Regional Jointing in the Michigan Basin", Geology, Vol.10, No. 5, p.273-277, 1982.
HOWELL90	Howell P.D., and van der Pluijm B.A., "Early History of the Michigan Basin: Subsidence and Appalachian Tectonics", Geology, Vol.18, No.12, p.1195-1198, December 1990.
HOWELL99	Howell P.D., and van der Pluijm B.A., "Structural Sequences and Styles of Subsidence in the Michigan Basin", Geological Society of America Bulletin, Vol.111, No.7, p.974-991, July 1999.

HUFF92	Huff .W. D., Bergstrom S.M., and Kolata D.R., "Gigantic Ordovician Volcanic Ash Fall in North America and Europe; Biological, Tectonomagmatic, and Event-Stratigraphic Significance", Geology, Vol.20, No.10, p.875-878, October 1992.
HUNT96	Hunt, J.M. Petroleum Geochemistry and Geology. New York, New York, USA: W.H. Freeman & Co., 1996.
HURLEY90	Hurley N.F., and Budros R., "Albion-Scipio and Stoney Points Fields – USA. Michigan Basin", in: E.A. Beaumont and N.H. Foster (Eds.), Stratigraphic Traps I: Treatise of Petroleum Geology, Atlas of Oil and Gas Fields, American Association of Petroleum Geologists, 1-32, 1990.
HUYSMANS05	Huysmans M., and Dassargues, A. "Review of the Use of Péclet Numbers to Determine the Relative Importance of Advection and Diffusion in Low Permeability Environments", Hydrogeology Journal, Vol.13, p.895–904, 2005.
IAEA94	"IAEA Safety Guide: Siting of Geological Disposal Facilities", International Atomic Energy Agency, IAEA Safety Series No. 111-G-4 1, May 1994.
IAEA95	"IAEA Safety Fundamentals: The Principles of Radioactive Waste Management", International Atomic Energy Agency, IAEA Safety Series No. 111-F, September 1995.
IAEA04a	"Safety Assessment Methodologies for Near Surface Disposal Facilities – Results of a Co-ordinated Research Project, Volume 2 – Test Cases", International Atomic Energy Agency, IAEA-ISAM-1, July 2004.
IAEA04b	"Safety Assessment Methodologies for Near Surface Disposal Facilities - Results of a Co-ordinated Research Project, Volume 1 – Review and Enhancement of Safety Assessment Approaches and Tools", International Atomic Energy Agency, IAEA-ISAM-1, July 2004.
IAEA06a	"IAEA Safety Fundamentals: Fundamental Safety Principles", International Atomic Energy Agency, IAEA Safety Standard Series No. SF-1, November 2006.
IAEA06b	"IAEA Safety Requirements: Geological Disposal of Radioactive Waste", International Atomic Energy Agency, IAEA Safety Standards Series No. WS-R-4, May 2006.

IAEA06c	"IAEA Safety Standards: Disposal of Radioactive Waste", International Atomic Energy Agency, Draft Safety Requirements IAEA DS 354, 2006.
IAEA07	"IAEA Safety Standards: Geological Disposal of Radioactive Waste", International Atomic Energy Agency, Draft Safety Guide DS 334, 2007.
IAEA08a	"IAEA Safety Standards: The Safety Case and Safety Assessment for Radioactive Waste Disposal", International Atomic Energy Agency, Draft Safety Guide IAEA DS-355.
IAEA08b	"IAEA Safety Guide: Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material Safety Guide", International Atomic Energy Agency, IAEA Safety Standards Series No. TS-G-1.1 (Rev.1), 2008.
IAEA09	"Determination and Use of Scaling Factors for Waste Characterization in Nuclear Power Plants", International Atomic Energy Agency, IAEA Nuclear Energy Series No. NW-T-1.18, April 2009.
ICRP95	"Dose Coefficients for Intakes of Radionuclides by Workers", International Commission on Radiological Protection, ICRP Publication 68, July 1995.
ICRP00	"Radiation Protection Recommendations as Applied to the Disposal of Long-Lived Solid Radioactive Waste", International Commission on Radiological Protection, ICRP Publication 81, 2000.
ICRP07	"The 2007 Recommendations of the ICRP", International Commission on Radiological Protection, ICRP Publication 103, Annals of the ICRP, Vol. 37, No. 2-4, 2007.
ISO07	"Nuclear Energy - Nuclear Fuel Technology - Scaling Factor Method to Determine the Radioactivity of Low- and Intermediate-Level Radioactive Waste Packages Generated at Nuclear Power Plants", International Organization for Standardization, ISO 21238:2007, April 2007.
ISO08	"Quality Management Systems – Requirements", International Organization for Standardization, ISO 9001:2008, 2008.
JACKSON09	Jackson R., "Organic Geochemistry and Clay Mineralogy of DGR-3 and DGR-4 Cores", TR-08-29, November 2009.
JACKSON10	Jackson R., and Heagle D., "Opportunistic Groundwater Sampling in DGR-1 and DGR-2", TR-07-11, 2010.

JACOBI93 — Jacobi R., and Fountain J., "The Southern Extension and Reactivations of the Clarendon-Linden Fault System", Géographie Physique et Quaternaire, Vol.47, No.3, p.285-302, 1993.

JOHNSON92 — Johnson M.D., Armstrong D.K., Sanford V.V., Telford P.G., and Rutka M.A., "Paleozoic and Mesozoic Geology of Ontario", in Geology of Ontario, Ontario Geologic Survey, Special Vol. 4, Part 2, p.907-1008, 1992.

KC04 — "The Municipality of Kincardine Council Resolution #2004-232 Regarding Selection of the Deep Rock Vault Option", the Council of the Municipality of Kincardine, April 21, 2004.

KECOJEVIC07 — Kecojevic V., Komljenovic D., Groves W., and Radomsky M, "An Analysis of Equipment-Related Fatal Accidents in U.S. Mining Operations: 1995-2005", Safety Science, Vol. 45, No. 8, p.864-874, September 2007.

KEIL09 — Keil C.B., Simmons C.E., and Anthony T.R." Mathematical Models for Estimating Occupational Exposure to Chemicals. Fairfax: USA: American Industrial Hygiene Association Press, 2009 (2nd Edition).

KESLER02 — Kesler S.E., and Carrigan C.W., "Discussion on "Mississippi Valley-Type Lead-Zinc Deposits Through Geological Time: Implications from Recent Age-Dating Research", by Leach D.L., Bradley D., Lewchuk M.T., Symons D.T.A., de Marsily, G., and Brannon J. (2001), Mineralium Deposita, Vol.36, p.711-740, 2002.

KETCHUM00 — Ketchum J.W.F., and Davidson A., "Crustal Architecture and Tectonic Assembly of the Central Gneiss Belt, Southwestern Grenville Province, Canada: A New Interpretation", Canadian Journal of Earth Sciences, Vol.37, No.2-3, p.217–234, 2000.

KHARAKA73 — Kharaka Y.K., and Berry F.A.F., "Simultaneous Flow of Water and Solutes Through Geological Membranes I: Experimental Investigation", Geochimica et Cosmochimica Acta, Vol.37, No.12, p.2577-2603, December 1973.

KINECTRICS03 — "Chemical, Radiochemical and Microbiological Characterisation of the Contents of Resin Liners Sampled at the Western Waste Management Facility in 2002", Kinectrics Inc., 9820-090-RA-0001-R00, February 2003.

KINECTRICS05a	"Western Waste Management Facility Integrated EA Follow-up Program. Phase III – Operational Phase (October 2003 - -May 2005)", Kinectrics Inc., K-010669-001-RA-0001-R00, June 2005.
KINECTRICS05b	"Bruce Heavy Water Plant Monitoring and Follow-up Program: Phase 1 Pre-demolition", Kinectrics Inc., Technical Memorandum K-010293, March 2005 (Revised November 2006).
KINECTRICS05c	"Bruce Heavy Water Plant Monitoring and Follow-up Program: Phase 2 – Demolition", Kinectrics Inc., Technical Memorandum K-011611, October 2005 (Revised November 2006).
KOLATA98	Kolata D.R., Huff W.D., and Bergström S.M., "Nature and Regional Significance of Unconformities Associated with the Middle Ordovician Hagan K-bentonite Complex in the North American Midcontinent", Geological Society of America Bulletin, Vol.110, No.6, p.723-739, June 1998.
KOROLEVA09	Koroleva M., de Haller A., Mader U., Waber H.N., and Mazurek M., "Technical Report: Borehole DGR-2 Pore-Water Investigations", TR-08-06, Revision 0, August 4, Rock Water Interaction (RWI), Institute of Geological Sciences, University of Bern, Bern, Switzerland, 2009.
KUMARAPELI66	Kumarapeli P.S., and Saul V.A., "The St. Lawrence Rift Valley System: A North American Equivalent of the East African Rift Valley System", Canadian Journal of Earth Science, Vol.3, No.5, p.639-658, 1966.
KUMARAPELI76	Kumarapeli P.S., "The St. Lawrence Rift System, Related Metallogeny, and Plate Tectonic Models of Appalachian Evolution", Geological Association of Canada, Special Paper, p.301-319, 1976.
KUMARAPELI85	Kumarapeli P.S., "Vestiges of Iapetan Rifting in the Craton West of the Northern Appalachians", Geoscience Canada, Vol.12, p.54-57, 1985.
KYSER03	Kyser K., and Hiatt E.E., "Fluids in Sedimentary Basins: An Introduction", Journal of Geochemical Exploration, Vol.80, No.2-3, p.139-149, September 2003.
LAINE10	Laine H., and Karttunen P., "Long-Term Stability of Bentonite, A Literature Review", Posiva Working Report 2010-53, July 2010.

LAM07	Lam T.M., Martin C.D. and McCreath D., "Characterising the Geomechanics Properties of the Sedimentary Rock for DGR Excavations", Canadian Geotechnical Conference, Ottawa, 2007.
LAMPE09	Lampe D.C., "Hydrogeologic Framework of Bedrock Units and Initial Salinity Distribution for a Simulation of Groundwater Flow for the Lake Michigan Basin", U.S. Geological Survey, Scientific Investigations Report 2009-5060, 2009.
LANL05	Los Alamos National Laboratory (LANL), MCNP5 Version 1.40, 2005.
LEE98	Lee H.T., Bakowsky W.D., Riley J., Bowles J., Puddister M., Uhlig P., and McMurray S., Ecological Land Classification for Southern Ontario. First Approximation and its Application. Northbay, Ontario: Ontario Ministry of Natural Resources (Southcentral Science Section), SCSS Field Guide FG-02, September 1998.
LEGALL81	Legall F.D., Barnes C.R., and MacQueen R.W., "Thermal Maturation, Burial History and Hotspot Development, Paleozoic Strata of Southern Ontario-Quebec, From Conodont and Acritarch Colour Alteration Studies", Bulletin of Canadian Petroleum Geology, Vol.29, No.4, p.492-539, 1981.
LEHMANN95	Lehmann D., Brett C.E., Cole R., and Baird G., "Distal Sedimentation in a Peripheral Foreland Basin, Ordovician Black Shales and Associated Flysch of the Western Taconic Foreland, New York State and Ontario", Geological Society of America Bulletin, Vol.107, p.708-724, 1995.
LGL02	"Bruce Nuclear Power Development Bio-Inventory Study – Final Report", LGL Limited., Project No. TA2522, DGR-REF-07500-23629, October 2002.
LIBERTY71	Liberty B.A., and Bolton T.E., "Paleozoic Geology of the Bruce Peninsula Area, Ontario", Geological Survey of Canada, Memoir 360, p.163, 1971.
LINDHOLM78	Lindholm R. C., "Triassic-Jurassic Faulting in Eastern North America – A Model Based on Pre-Triassic Structures", Geology, Vol.6, No.6, p.365-368, June 1978.
LUMBERS90	Lumbers S.B., Heaman L.M., Vertolli V.M., and Wu T.W., "Nature and Timing of Middle Proterozoic Magmatism in the Central Metasedimentary Belt Grenville Province, Ontario", in Mid-Proterozoic Laurentia – Baltica, Geological Association of Canada, Special Paper 38, p.243-276, 1990.

LUND09	Lund B., Schmidt P., and Hieronymus C., "Stress Evolution and Fault Stability during the Weichselian Glacial Cycle", SKB Technical Report TR-09-15, Svensk Kärnbränslehantering AB, 2009.
MA06	Ma S., and Atkinson, G.M., "Focal Depths for Small to Moderate Earthquakes (Mn≥2.8) in Western Quebec, Southern Ontario, and Northern New York", Bulletin of the Seismological Society of America, Vol.96, No.2, April 2006.
MARIVOET08	Marivoet, J., Beuth T., Alonso J., Becker D.A., "Safety Functions, Definition and Assessment of Scenarios, Uncertainty Management and Uncertainty Analysis, Safety Indicators and Performance/Function Indicators", European Commission, PAMINA D-No. 1.1.1, March 2008.
MARTINO07	Martino J., Dixon D., Kozak E., Gascoyne M., Vignal B., Sugita Y., Fujita T., and Masumoto K., "The Tunnel Sealing Experiment: An International Study of Full-Scale Seals", Physics and Chemistry of the Earth, Parts A/B/C, Vol.32, No.1-7, p.93-107, 2007.
MARSHAK89	Marshak S., and Tabor J.R., "Structure of the Kingston Orocline in the Appalachian Fold-Thrust Belt, New York", Geological Society of America Bulletin, Vol.101, No. 5, p.683-701, May 1989.
MAZUREK03	Mazurek M., Pearson F.J., Volckaert G., and Bock H., "FEPCAT Project: Features, Events and Processes Evaluation Catalogue for Argillaceous Media", OECD/NEA, 2003.
MCCAFFREY87	McCaffrey M.A., Lazar B. and Holland H.D, "The Evaporation Path of Seawater and the Coprecipitation of Br^- and K^+ with Halite", Journal of Sedimentary Petrology, Vol.57, No. 5, p.928-937, September 1987.
MCHONE84	McHone J.G., and Butler J.R., "Mesozoic Igneous Provinces of New England and the Opening of the North Atlantic Ocean", Geological Society of America Bulletin, Vol.95, p.757-765, 1984.
MCINTOSH05	McIntosh J.C., and Walter L.M., "Volumetrically Significant Recharge of Pleistocene Glacial Meltwaters into Epicratonic Basins: Constraints Imposed by Solute Mass Balances", Journal of Chemical Geology, Vol. 222, No. 3-4, p.292-309, November 2005.

MCINTOSH06	McIntosh J.C., and Walter L.M., "Paleowaters in the Silurian-Devonian Carbonate Aquifers: Geochemical Evolution of Groundwater in the Great Lakes Region Since the Late Pleistocene", Geochimica et Cosmochimica Acta, Vol.70, No. 10, p.2454-2479, 2006.
MCKENNA92	McKenna C.M., McNutt R.H. and Frape S.K., "Lead and Strontium Isotopic Data on Brines from the Michigan Basin, Ontario and Michigan", in: Kharaka Y.K., and Maest A.S. (Eds.), Proceedings 7th International Water-Rock Interaction Conference, Park City, Utah, U.S.A, 1992.
MCNUTT87	McNutt R.H., Frape S.K., and Dollar P., "A Strontium, Oxygen and Hydrogen Isotopic Composition of Brines, Michigan and Appalachian Basins, Ontario and Michigan", Applied Geochemistry, Vol. 2, No. 5-6, p.495-505, December 1987.
MCWILLIAMS07	McWilliams C. K., Wintsch R.P. and Kunk M.J., "Scales of Equilibrium and Disequilibrium during Cleavage Formation in Chlorite and Biotite-Grade Phyllites, SE Vermont", Journal of Metamorphic Geology, Vol.25, No.8, p.895-913, October 2007.
MELCHIN94	Melchin M.J., Brookfield M.E., Armstrong D.K. and Coniglio M., "Stratigraphy, Sedimentology and Biostratigraphy of the Ordovician Rocks of the Lake Simcoe Area, South-Central Ontario", Geological Association of Canada–Mineralogical Association of Canada, Joint Annual Meeting, Waterloo, Ontario, Guidebook for Field Trip A4, p.101, 1994.
MIDDLETON93	Middleton K., Coniglio M., Sherlock R., and Frape S.K., "Dolomitization of Middle Ordovician Carbonate Reservoirs, Southwestern Ontario", Bulletin of Canadian Petroleum Geology, Vol. 41, No. 2, p.150-163, June 1993.
MILKEREIT92	Milkereit B., Forsyth D.A., Green A.G., Davidson A., Hanmer S., Hutchinson D.R., Hinze W.J. and Mereu R.F., "Seismic Images of a Grenvillian Terrane Boundary", Geology, Vol.20, No.11, p.1027-1030, November 1992.
MISHIMA08	Mishima J., Foppe T., and Laul J.C., "Proposed Beryllium Metal Bounding Airborne Release Fractions (ARFs)/Rates (ARRs) and Respirable Fractions (RFs) for DOE Facility Accidents Analyses", Journal of Chemical Health & Safety, Vol. 15, No. 4, p.26-45, July – August 2008.

MOE09a "Agreement to Establish a Joint Review Panel for the Deep Geologic Repository Project by Ontario Power Generation Inc. within the Municipality of Kincardine, Ontario, between the Minister of the Environment and the Canadian Nuclear Safety Commission", January 2009.

MOE09b "Soil, Groundwater and Sediment Standards for Use under Part XV.1 of the Environmental Protection Act", Ontario Ministry of the Environment, July 27, 2009.

MOEE94 "Water Management, Policies Guidelines Provincial Water Quality Objectives of the Ministry of Environment and Energy", Ontario Ministry of Environment and Energy, 1994.

MTO09 "Regional Tourism Profiles, CD 41: Bruce County 2007", Ministry of Tourism Ontario, 2009.

NBCC05 National Building Code of Canada, 2005.

NEA99 "Safety Assessment of Radioactive Waste Repositories – Systematic Approaches to Scenario Development – An International Database of Features, Events and Processes. A Report of the NEA Working Group on Development of a Database of Features, Events and Processes Relevant to the Assessment of the Post-closure Safety of Radioactive Waste Repositories", Nuclear Energy Agency, 1999.

NEA04 "Post-Closure Safety Case for Geological Repositories Nature and Purpose", Nuclear Energy Agency, NEA No.3679, 2004.

NEA10 "Self-Sealing of Fractures in Argillaceous Formations in the Context of Geological Disposal of Radioactive Waste, Review and Synthesis", Nuclear Energy Agency (Organisation for Economic Co-Operation and Development, NEA No. 6184, 2010.

NILSEN09 Nilsen A.R., and Log T., "Results from Three Models Compared to Full-Scale Tunnel Fires Tests", Fire Safety Journal, Vol. 44, p.33-49, January 2009.

NIOSH05 "2004 NIOSH Respirator Selection Logic", National Institute for Occupational Health and Safety, NOISH Publication No. 2005-100, October 2004.

NOAA08	"Water Levels of the Great Lakes", National Oceanic and Atmospheric Administration (NOAA) Great Lakes Environmental Research Laboratory, January 2008.
NORMANI07	Normani S.D., Park Y.J., Sykes J.F., and Sudicky E.A., "Sub-Regional Modelling Case Study 2005-2006 Status Report", TR-2007-07., 2007.
NORMANI09	Normani S.D., "Paleoevolution of Pore Fluids in Glaciated Geologic Settings", Ph.D. Thesis, University of Waterloo, Waterloo, Ontario, Canada, 2009.
NRCAN96	"Policy Framework for Radioactive Waste", Natural Resources Canada, News Release 96/79, July 10 1996.
NSCA97	Nuclear Safety and Control Act, S.C. 1997, c. 9.
NWMOa	"Design Management", Nuclear Waste Management Organization, NWMO-PROC-EN-0001; the most recent approved version.
NWMO09a	"L&ILW Deep Geologic Repository Project Quality Plan", Nuclear Waste Management Organization, DGR-PLAN-00120-0002, Rev. 002, June 2009.
NWMO09b	NWMO Letter from King F., to Klassen K., "Deep Geologic Repository for OPG's Low and Intermediate Level Waste – Proposed Acceptance Criteria for Radiological Impacts on Non-Human Biota and for Non-Radiological Impacts on Human and Non-Human Biota", DGR-CORR-00531-0011, June 2, 2009.
NWMO10a	"L&ILW Deep Geologic Repository – Project Glossary", Nuclear Waste Management Organization, DGR-LIST-00120-0001 R001, March 2010.
NWMO10b	"Ontario Power Generation's Deep Geologic Repository for Low and Intermediate-Level Waste Project Requirements", Nuclear Waste Management Organization, DGR-PDR-00120-0001, September 2010.
NWMO10c	NWMO Letter from King F. to Klassen K., "OPG's Deep Geologic Repository for L&ILW - Updates to Acceptance Criteria for Non-Radiological Impacts on Humans and Non-Human Biota", DGR-CORR-00531-0031, March 29, 2010.
NWMO10d	"Technical Computing Software Procedure", Nuclear Waste Management Organization, NWMO-PROC-EN-0002, March 2010.

NWMO10e "Design and Construction Phase Project Quality Plan for OPG's DGR for L&ILW", Nuclear Waste Management Organization, DGR-PLAN-00120-0006 R000, October, 2010.

NWMO11a "Design and Construction Phase Management System", Nuclear Waste Management Organization, DGR-PD-EN-0001 R000, 2011.

NWMO11b "Maximum Flood Hazard Assessment", Nuclear Waste Management Organization, NWMO DGR-TR-2011-35 R000, March 2011.

NWMO11c "Geosynthesis", Nuclear Waste Management Organization, NWMO DGR-TR-2011-11 R000, March 2011.

NWMO11d "Terrestrial Environment Technical Support Document", Nuclear Waste Management Organization, NWMO DGR-TR-2011-05 R000, March 2011.

NWMO11e "Radiation and Radioactivity Technical Support Document", Nuclear Waste Management Organization, NWMO DGR-TR-2011-06 R000, March 2011.

NWMO11f "Atmospheric Environment Technical Support Document", Nuclear Waste Management Organization, NWMO DGR-TR-2011-02 R000, March 2011.

NWMO11g "Aquatic Environment Technical Support Document", Nuclear Waste Management Organization, NWMO DGR-TR-2011-01 R000, March 2011.

NWMO11h "Socio-economic Environment Technical Support Document", Nuclear Waste Management Organization, NWMO DGR-TR-2011-08 R000, March 2011.

NWMO11j "Aboriginal Interests Technical Support Document", Nuclear Waste Management Organization, NWMO DGR-TR-2011-09 R000, March 2011.

NWMO11k "Descriptive Geosphere Site Model", Nuclear Waste Management Organization, NWMO DGR-TR-2011-24 R000, March 2011.

NWMO11m "Regional Geology – Southern Ontario", Nuclear Waste Management Organization, NWMO DGR-TR-2011-15 R000, March 2011.

NWMO11n "Regional Geomechanics – Southern Ontario", Nuclear Waste Management Organization, NWMO DGR-TR-2011-13 R000, March 2011.

NWMO11p "Hydrogeologic Modelling", Nuclear Waste Management Organization, NWMO DGR-TR-2011-16 R000, March 2011.

NWMO11q "Regional Hydrogeochemistry – Southern Ontario", Nuclear Waste Management Organization, NWMO DGR-TR-2011-12 R000, March 2011.

NWMO11r "Long Term Climate Change", Nuclear Waste Management Organization, NWMO DGR-TR-2011-14 R000, March 2011.

NWMO11t "Long-Term Geomechanical Stability Analysis", Nuclear Waste Management Organization, NWMO DGR-TR-2011-17 R000, March 2011.

NWMO11u "Glacial Erosion Assessment", Nuclear Waste Management Organization, NWMO DGR-TR-2011-18 R000, March 2011.

NWMO11v "Neotectonic Features and Landforms Assessment", Nuclear Waste Management Organization, NWMO DGR-TR-2011-19 R000, March 2011.

NWMO11w "Seismic Hazard Assessment", Nuclear Waste Management Organization, NWMO DGR-TR-2011-20 R000, March 2011.

NWMO11x "Excavation Damaged Zones Assessment", Nuclear Waste Management Organization, NWMO DGR-TR-2011-21 R000, March 2011.

NWMO11y "Analogue Study of Shale Cap Rock Barrier Integrity", Nuclear Waste Management Organization, NWMO DGR-TR-2011-23 R000, March 2011.

NWMO11z "Karst Assessment", Nuclear Waste Management Organization, NWMO DGR-TR-2011-22 R000, March 2011.

NWMO11aa "Three-Dimensional Geological Framework Model", Nuclear Waste Management Organization, NWMO DGR-TR-2011-42 R000, March 2011.

NWMO11ab "Outcrop Fracture Mapping". Nuclear Waste Management Organization, NWMO DGR-TR-2011-43 R000, March 2011.

NWMO11ac "Preliminary Conventional Safety Assessment", Nuclear Waste Management Organization, NWMO DGR-TR-2011-37 R000, March 2011.

NWMO11ad "Malfunctions, Accidents, and Malevolent Acts Technical Support Document", Nuclear Waste Management Organization, NWMO DGR-TR-2011-07 R000, March 2011.

NWMO11ae	"Radon Assessment", Nuclear Waste Management Organization, NWMO DGR-TR-2011-34 R000, March 2011.
NWMO11af	"Postclosure Safety Assessment Report", Nuclear Waste Management Organization, NWMO DGR-TR-2011-25 R000, March 2011.
NWMO11ag	"Postclosure Safety Assessment: Features, Events and Processes", Nuclear Waste Management Organization, NWMO DGR-TR-2011-29 R000, March 2011.
NWMO11ah	"Postclosure Safety Assessment: System and Its Evolution", Nuclear Waste Management Organization, NWMO Report DGR-TR-2011-28 R000, March 2011.
NWMO11aj	"Postclosure Safety Assessment: Gas Modelling", Nuclear Waste Management Organization, NWMO DGR-TR-2011-31 R000, March 2011.
NWMO11ak	"Postclosure Safety Assessment: Analysis of the Normal Evolution Scenario", Nuclear Waste Management Organization, NWMO DGR-TR-2011-26 R000, March 2011.
NWMO11am	"Postclosure Safety Assessment: Data", Nuclear Waste Management Organization, NWMO DGR-TR-2011-32 R000, March 2011.
NWMO11an	"Postclosure Safety Assessment: Groundwater Modelling", Nuclear Waste Management Organization, NWMO DGR-TR-2011-30 R000, March 2011.
NWMO11ap	"T2GGM Version 2: Gas Generation and Transport Code". Nuclear Waste Management Organization, NWMO DGR-TR-2011-33 R000, March 2011.
NWMO11aq	"Postclosure Safety Assessment: Analysis of Human Intrusion and Other Disruptive Scenarios", Nuclear Waste Management Organization, NWMO DGR-TR-2011-27 R000, March 2011.
NWMO11ar	"Geoscientific Verification Plan", Nuclear Waste Management Organization, NWMO DGR-TR-2011-38, March 2011.
NWMO11at	"DGR EA Follow-Up Monitoring Program", Nuclear Waste Management Organization, NWMO DGR-TR-2011-10 R000, March 2011.
NWMO11au	"Preliminary Decommissioning Plan", Nuclear Waste Management Organization, NWMO DGR-TR-2011-39 R000, March 2011.

OBERMAJER96 Obermajer M., Fowler M.G., Goodarzi F. and Snowdon L.R., "Assessing Thermal Maturity of Palaeozoic rocks from Reflectance of Chitinozoa as Constrained by Geothermal Indicators: An Example from Southern Ontario, Canada", Marine and Petroleum Geology, Vol.13, p.907-919, 1996.

OBERMAJER00 Obermajer M., Fowler M.G., Snowdon L.R. and Macqueen R.W., "Are the Silurian Reef-Hosted Oils Locally Sourced in Ontario?", American Association of Petroleum Geologists Bulletin, Vol.84, No.9, p.1390-1391, 2000.

OECD09 "Natural Tracer Profiles across Argillaceous Formations: The CLAYTRAC Project", Organisation for Economic Co-operation and Development – Nuclear Energy Agency, OECD/NEA No. 6253, 2009.

OGS91 Ontario Geological Survey, "Bedrock Geology of Ontario, Southern Sheet", Ontario Geological Survey, Map 2544, 1991.

OGSR04 Ontario Oil, Gas and Salt Resources Library, " Cumulative Oil and Gas Production in Ontario to the End of 2004", Excel Format Data, in Members Package Dataset, Petroleum Resources Centre, Ontario Ministry of Natural Resources Oil, Ontario Gas & Salt Resources Library, 2004.

OGSR06 Ontario Oil, Gas and Salt Resources Library, "Oil and Gas Pools and Pipelines of Southern Ontario, Revised October 2006", Petroleum Resources Centre, Ontario Ministry of Natural Resources Oil, Gas & Salt Resources Library, UTM NAD83, 2006.

OH86 "Summary of Ontario Hydro's Tunnelling Experience at Bruce, Darlington and Wesleyville", Ontario Hydro, Report No. 86009, January 1986.

OH97 "Bruce Used Fuel Dry Storage Facility Environmental Assessment – A Submission to the Atomic Energy Control Board in Fulfilment of the Requirements Under the Canadian Environmental Assessment Act", Ontario Hydro, CD# 01098-REP-00531-00001-R001, December 1997.

OH98 "Addendum to Bruce Used Fuel Dry Storage Facility Environmental Assessment Report – A Submission to the Atomic Energy Control Board in Fulfilment of the Requirements for Comprehensive Study under the Canadian Environmental Assessment Act", Ontario Hydro, CD# 01098-REP-00531-00001-ADD01-R001, July 1998.

OHSA90 Occupational Health and Safety Act (Ontario), R.S.O. 1990, Chapter O.1.

OPGa "Radiation Protection Program", Ontario Power Generation, OPG Program Document N-PROG-RA-0013; the most recent approved version.

OPGb "Radiological Zoning, Personnel/Material Monitoring and Transfer Permits", Ontario Power Generation, OPG Procedure N-PROC-RA-0014; the most recent approved version.

OPGc "Environmental, Health and Safety Management Program", Ontario Power Generation, OPG Program Document W-PROG-ES-0001; the most recent approved version.

OPGd "Health and Safety Policy", Ontario Power Generation, OPG Policy OPG-POL-0001; the most recent approved version.

OPGe "Nuclear Safety Policy", Ontario Power Generation, OPG Policy N-POL-0001; the most recent approved version.

OPGf "Safety Management System Program", Ontario Power Generation, OPG Program Document OPG-HR-SFTY-PROG-0001; the most recent approved version.

OPGg "Selection of Radiation Personal Protective Equipment", Ontario Power Generation, OPG Procedure N-PROC-RA-0025; the most recent approved version.

OPGh "Environment Policy", Ontario Power Generation, OPG Policy OPG-POL-0021; the most recent approved version.

OPGj "Biodiversity Policy", Ontario Power Generation, OPG Policy OPG-POL-0002; the most recent approved version.

OPGk "Land Assessment and Remediation Policy", Ontario Power Generation, OPG Policy OPG-POL-0016; the most recent approved version.

OPGm "Spills Management Policy", Ontario Power Generation, OPG Policy OPG-POL-0020; the most recent approved version.

OPGn "Policy for Use of Ozone Depleting Substances", Ontario Power Generation, OPG Policy OPG-POL-0015; the most recent approved version.

OPGp "Training Program", Ontario Power Generation, OPG Program Document N-PROG-TR-0005; the most recent approved version.

OPGq	"Fire Protection Procedure", Ontario Power Generation, OPG Procedure W-PROC-ES-0011; the most recent approved version.
OPGr	"Employee Emergency Response Procedure", Ontario Power Generation, OPG Procedure W-PROC-ES-0002; the most recent approved version.
OPGt	"Conduct of Operations and Maintenance Program", OPG Program Document W-PROG-OM-0001; the most recent approved version.
OPGu	"Records and Document Control", Ontario Power Generation, OPG Program Document N-PROG-AS-0006; the most recent approved version.
OPGv	"Controlled Document Management", Ontario Power Generation, OPG Procedure N-PROC-AS-0003; the most recent approved version.
OPGw	"Creating and Maintaining Dose Records", Ontario Power Generation, OPG Document N-HPS-03413.1-0004; the most recent approved version.
OPGx	"Radiation Protection Requirements – Nuclear Facilities", Ontario Power Generation, OPG Document N-RPP-03415.1-10001; the most recent approved version.
OPGy	"Records and Documentation", Ontario Power Generation, OPG Procedure N-PROC-TR-0012; the most recent approved version.
OPG99	"Investigations into Tritium Transfer Between Low Level Storage Buildings and the Middle Sand Aquifer at WSH 231", Ontario Power Generation, 0125(WS)-79811-98040(NWO), March 1999.
OPG00a	"Bruce Radioactive Waste Operations Site 2 – Environmental Assessment Report for Additional Storage of Low and Intermediate Level Waste – A Submission to the Canadian Nuclear Safety Commission (September 2000)", Attachment to OPG Letter from Nash K.E., to Klassen K., "Bruce Radioactive Waste Operations Site 2 (RWOS2) – Submission of Environmental Assessment for Additional Storage of Low and Intermediate Level Waste", CD# 0125-CORR-00531-00081, September 14, 2000.
OPG00b	"Bruce Nuclear Power Development Ecological Effects Review", Ontario Power Generation, N-REP-07015-10001, December 2000.
OPG00c	"Carbon-14 Monitoring at RWOS 2 – Status Updated", Ontario Power Generation, 0125-IR-79100-00031, September 2000.

OPG01a OPG Letter from Nash K.E., to Klassen K., "Bruce Radioactive Waste Operations Site 2 (RWOS2) – Submission of Addendum to the Draft Environmental Assessment on Additional Facilities for Storage of Low and Intermediate Level Waste", CD# 0125-CORR-00531-00130, April 18, 2001.

OPG01b "Radiation Protection Requirements – Nuclear Facilities", Ontario Power Generation, N-RPP-03415.1-10001-R07, June 2001.

OPG03a "Transportation Package Maintenance Building at the Western Waste Management Facility - Environmental Assessment Study Report", Ontario Power Generation, CD# 0125-REP-07720-00001 R01, June 2003.

OPG03b "Derived Release Limits for the Western Waste Management Facility", Ontario Power Generation, 0125-REP-03482-00002-R00, July 2003.

OPG03c OPG Letter from Nash K.E. to Klassen K., "Summary Report: Carbon-14 Management at the WWMF 1998 – 2002", Ontario Power Generation, CD# 0125-CORR-00531-00313, June 2003.

OPG04 "Low Level Storage Buildings 9, 10 & 11 Environmental Assessment Study Report ", Ontario Power Generation, CD# 01098-REP-07701-00001 R00, March 2004.

OPG05a "Western Waste Management Facility Refurbishment Waste Storage Project Environmental Assessment Study Report", Ontario Power Generation, CD# 01098-REP-07701-00002 R01, October 2005.

OPG05b OPG Letter from Nash K.E. to Howden B., "Intent to Construct a Geologic Repository for Low and Intermediate Level Waste", CD# 00216-CORR-00531-00001, December 2, 2005.

OPG06a "Western Waste Management Facility Safety Report", Ontario Power Generation, W-SR-01320-00001 R01, July 2006.

OPG06b "OPG's Deep Geologic Repository for Low and Intermediate Level Waste Geoscientific Site Characterization Plan", Ontario Power Generation, 00216-REP-03902-00002-R00, April 2006.

OPG06c "Dose Limits and Exposure Control", Ontario Power Generation, N-PROC-RA-0019 R004, 2006.

OPG07a OPG Letter from Nash K.E., to Howden B., "Deep Geologic Repository for Low and Intermediate Level Waste (DGR) – Application for the Site Preparation and Construction Licence", CD# 00216-CORR-00531-00033, August 13, 2007.

OPG07b "Refurbishment Waste Storage Project EA Follow-up Monitoring Report", Ontario Power Generation, 01098-REP-07701.8-10004 R000, May 2007

OPG07c An Overview of Tritium Release and Mitigation for the Western Waste Management Facility", Ontario Power Generation, W-REP-00531-00005-R0, February 2007.

OPG08a "OPG's Deep Geologic Repository for Low and Intermediate Level Waste Phase 2 Geoscientific Site Characterization Plan", Ontario Power Generation, 00216-REP-03902-00006-R00, April 2008

OPG08b OPG Letter from Nash K.E., to Klassen K., "Deep Geologic Repository for Low and Intermediate Level Waste - Proposed Acceptance Criteria", CD# 00216-CORR-00531-00044, May 20, 2008.

OPG09a "Nuclear Waste Management Division Western Waste Operations - Quarterly Operations Report", Ontario Power Generation, W-REP-00531.1-00051-R000, June 2009.

OPG09b "Nuclear Waste Management Division Western Waste Operations - Quarterly Operations Report", Ontario Power Generation, W-REP-00531.1-00052-R000, September 2009.

OPG10a "Reference Low and Intermediate Waste Inventory for the Deep Geologic Repository", Ontario Power Generation, 00216-REP-03902-00003 R03, December 2010.

OPG10b "Nuclear Waste Management Division Western Waste Operations - Quarterly Operations Report", Ontario Power Generation, W-REP-00531.1-00054-R000, March 2010.

OPG11a "Deep Geologic Repository for Low and Intermediate Level Waste – Environmental Impact Statement", Ontario Power Generation, 00216-REP-07701-00001 R000, March 2011.

OPG11b "Deep Geologic Repository Project Management System", Ontario Power Generation, 00216-CHAR-0001 R000, 2011.

PARK94 Park R.G., and Jaroszewski W., "Craton Tectonics, Stress and Seismicity", in Hancock P.L., Ed., Continental Deformation, p.200-222, 1994.

PARK09 Park Y.J., Sudicky E. and Sykes J., "Effects of Shield Brine on the Safe Disposal of Waste in Deep Geologic Environments", Advances in Water Resources, Vol. 32, p.1352–1358, 2009.

PEARSON87 Pearson F.J. Jr., "Models of Mineral Controls on the Composition of Saline Groundwaters in the Canadian Shield", in: Fritz, P., Frape, S.K. (Eds.), Saline Waters and Gases in Crystalline Rocks, Geological Association of Canada Special Paper, Vol.33, p.39–51, 1987.

PERCIVAL07 Percival J.A. and Easton R.M., "Geology of the Canadian Shield in Ontario: An Update", Geologic Survey of Canada, June 2007.

PERRY07 Perry R.H. and Green D.W. Perry's Chemical Engineers' Handbook. New York: USA: McGraw Hill, October 2007 (8th Edition).

PERSON07 Person M., McIntosh J., Bense V., and Remenda V.H, "Pleistocene Hydrology of North America: The Role of Ice Sheets in Reorganizing Groundwater Flow Systems", Reviews of Geophysics, Vol. 45, 2007.

PHILLIPS90 Phillips, D. The Climates of Canada. ISBN 0-660-13459-4, 1990.

PKF01 "Tourism Business Opportunities Lake Huron Shoreline Tourism Investment Area (Prepared for Bruce Community Development Corporation)", Pannell Kerr Forester Consulting Inc., March 2001.

POWELL84 Powell T.G., MacQueen R.W., Barker J.F., and Bree D.G., "Geochemical Character and Origin of Ontario Oils", Bulletin of Canadian Petroleum Geology, Vol.32, No. 3, p.289-312, September 1984.

POWER98 Power M.S., Rosidi D. and Kaneshiro, J.Y., "Seismic Vulnerability of Tunnels and Underground Structures Revisited", p.243-250, 1998.

PRICE99 Price G.D., "The Evidence and Implications of Polar Ice during the Mesozoic", Earth-Science Reviews, Vol.48, p.183–210, 1999.

PROUTY88 Prouty C.E., "Trenton Exploration and Wrench Tectonics; Michigan Basin and Environs", in Keith B.D., Ed., The Trenton Group (Upper Ordovician Series) of Eastern North America, American Society of Petroleum Geologists, Studies in Geology, Vol.29, p.207-236, 1988.

PRUESS99 — Pruess, K., Oldenburg C., and Moridis G., "TOUGH2 Users Guide Version 2.0", Technical Report LBNL-43134, Earth Sciences Division, Lawrence Berkeley National Laboratory, 1999.

QUINLAN84 — Quinlan, G. and Beaumont C., "Appalachian Thrusting, Lithospheric Flexure and the Paleozoic Stratigraphy of the Eastern Interior of North America", Canadian Journal of Earth Sciences, Vol.21, No. 9, p.973-996, 1984.

QUINTESSA09 — "AMBER 5.3 Reference Guide", Quintessa Limited, QE-AMBER-1, Version 5.3, 2009.

QUINTESSA10 — "Postclosure Safety Assessment Services for OPG's Proposed Deep Geologic Repository: Project Quality Plan", Quintessa Limited, QRS-1335B-PQP Version 3.0, 2010.

Reg. 169/03 — Safe Drinking Water Act 2002, Ontario Regulation 169/03 – Ontario Drinking Water Quality Standards.

Reg. 213/91 — Occupational Health and Safety Act, Ontario Regulation 213/91 – Construction Projects Regulations.

Reg. 240/00 — Mining Act, Ontario Regulation 240/00 - Mine Development and Closure under Part VII of the Mining Act.

Reg. 851 — Occupational Health and Safety Act R.R.O 1990, Regulation 851 – Industrial Establishments.

Reg. 854 — Occupational Health and Safety Act R.R.O. 1990, Regulation 854 – Mines and Mining Plants.

ROBERTSON06 — Robertson N. The History of the County of Bruce and of the Minor Municipalities Therein. Toronto: William Briggs, 1906.

ROSE70 — Rose E.R., Sanford B.V. and Hacquebard C.A., "Economic Minerals of Southeastern Canada Chapter VII", in Douglas R.J.W., Ed., Geology and Economic Minerals of Canada, Geological Survey of Canada Economic Geology Report, p.307-364, 1970.

SAGE91 — Sage R.P., "Paleozoic and Mesozoic Geology of Ontario", in Geology of Ontario, Ontario Geological Survey, Special Volume 4, Part 1, p.683-709, 1991.

SANFORD85	Sanford B.V., Thompson F.J., and McFall G.H., "Plate Tectonics – A Possible Controlling Mechanism in the Development of Hydrocarbon Traps in Southwestern Ontario", Bulletin of Canadian Petroleum Geology, Vol.33, No.1, p.52-71, March 1985.
SANFORD93	Sanford B.V., "St. Lawrence Platform: Economic Geology", in Stott D.F. and Aitken J.D., Eds., Sedimentary Cover of the Craton in Canada, Geological Survey of Canada, Geology of Canada Series, No.5, p.787-798, 1993.
SANGSTER71	Sangster D.F., and Liberty B.A., "Sphalerite Concretions from Bruce Peninsula, Southern Ontario, Canada", Economic Geology, Vol. 66, p. 1145-1152, 1971.
SAVAGE05	Savage D., "The Effects of High Salinity Groundwater on the Performance of Clay Barriers", SKI Report 2005:54, August 2005.
SCHANDL09	Schandl E., "Petrography of DGR-1 and DGR-2 Core", TR-07-12, April 2009.
SCOTT98	Scott W.B., and Crossman E.J., Freshwater Fishes of Canada. Oakville: Galt House Publications Limited, 1998.
SERAGELDIN09	Serageldin M., and Reeves D., "Development of Welding Emission Factors for Cr and Cr(VI) with a Confidence Level", Journal of Air and Waste Management Association, Vol. 59, p.619-626, May 2009.
SHARPE79	Sharpe D.R., and Edwards W.A.D., "Quaternary Geology of the Chesley-Tiverton Area, Southern Ontario", Ontario Geological Survey, Preliminary Map 2314, 1:50,000 Scale, 1979.
SHERWOOD94	Sherwood-Lollar B., Weise S.M., Frape S.K., and Barker J.F., "Isotopic Constraints on the Migration of Hydrocarbon and Helium Gases of Southwestern Ontario", Bulletin of Canadian Petroleum Geology, Vol.42, No. 3, p.283-295, September 1994.
SKOWRON09	Skowron, A. and E. Hoffman, "Geochemical and SEM/EDS Analysis of DGR-1 and DGR-2 Core", Technical Report TR-08-02, Intera Engineering, April 2009.
SLOSS82	Sloss L.L., "The Michigan Basin; Selected Structural Basins of the Midcontinent, USA", UMR Journal, Vol.3, p.25-29, 1982.

SOR/98-180	Ontario Hydro Nuclear Facilities Exclusion from Part II of the Canada Labour Code Regulations (Occupational Health and Safety), SOR/98-180.
SOR/2000-202	General Nuclear Safety and Control Regulations, SOR/2000-202.
SOR/2000-203	Radiation Protection Regulations, SOR/2000-203.
SOR/2000-204	Class I Nuclear Facilities Regulations, SOR/2000-204.
SOR/2000-206	Uranium Mines and Mills Regulations, SOR/2000-206.
SOR/2000-207	Nuclear Substances and Radiation Devices Regulations, SOR/2000-207
STATSCAN07a	Statistics Canada 2007, "Bruce, Ontario (table) - Community Profiles", 2006 Census. Statistics Canada Catalogue No. 92-591-WXE. Ottawa. Released March 13, 2007. http://www12.statcan.ca/census-recensement/2006/dp-pd/prof/92-591/index.cfm?Lang=E. Site Accessed: November 8, 2010.
STATSCAN07b	Statistics Canada 2007. "Bruce, Ontario (table) - Aboriginal Population Profile" 2006 Census. Statistics Canada Catalogue No. 92-594-XWE. Ottawa. Released January 15, 2008. http://www12.statcan.ca/census-recensement/2006/dp-pd/prof/92-594/index.cfm?Lang=E Site Accessed: November 19, 2010.
STERLING10	Sterling S., "Technical Report: Bedrock Formations in DGR-1 and DGR-2, TR-07-05", Intera Engineering Limited 2010.
STROESS08	Stroess-Gascoyne S., and Hamon C.J., "Preliminary Microbial Analysis of Limestone and Shale Rock Samples", Nuclear Waste Management Organization Report, NWMO TR-2008-09, 2008.
SUTTER85	Sutter J.F., Ratcliffe N.M., and Mukasa S.B., "40Ar/39Ar and K-Ar Data Bearing on the Metamorphic and Tectonic History of Western New England", Geological Society of America Bulletin, Vol.96, No. 1, p.123-136, January 1985.
TAM86	Tam Y.T., "A Preliminary Study of the Dispersion Climatology of the Bruce Area". Ontario Hydro Research Division, Report No. 86-6-K (N-08800-1415-P), March 1986.

THERRIEN10	Therrien, R., McLaren R. G., Sudicky E. A., Panday S.M., and Guvanasen V., "FRAC3DVS-OPG: A Three-Dimensional Numerical Model Describing Subsurface Flow and Solute Transport. User's Guide", University of Waterloo - Groundwater Simulations Group, 2010.
THOMAS06	Thomas W. A., "Tectonic Inheritance at a Continental Margin", GSA Today, Vol.16, No.2, p.4-11, February 2006.
TIDWELL00	Tidwell V.C., Meigs L.C., Christian-Frear T. and Boney C.M., "Effects of Spatially Heterogeneous Porosity on Matrix Diffusion as Investigated by X-Ray Absorption Imaging", Journal of Contaminant Hydrology, Vol.42, No.2-4, p.285–302, March 2000.
TOTTMAN94	Tottman R., "Background to the Nawash Saugeen Breach of Trust Suit", December 1994.
UNSCEAR96	Sources and Effects of Ionizing Radiation - United Nations Scientific Committee on the Effects of Atomic Radiation UNSCEAR 1996 Report to the General Assembly, with Scientific Annex. New York: United Nations Publication, 1996.
USDOE94	"Airborne Release Fractions/Rates and Respirable Fractions for Non-Reactor Nuclear Facilities", U.S. Department of Energy, DOE-HDBK-3010-94, December 1994.
USDOE05	"Waste Isolation Pilot Plant Contact Handled (CH) Waste Documented Safety Analysis", U.S. Department of Energy, DOE/WIPP-95-2065, Rev. 9, September 2005.
USDOE06	"Accident Analysis for Aircraft Crash into Hazardous Facilities", U.S. Department of Energy, DOE-STD-3014-2006, May 2006.
USDOE07	"Preparation of Safety Basis Documents for Transuranic Waste Facilities", U.S. Department of Energy, DOE-STD-5506-2007, April 2007.
USDOE10	"Protective Action Criteria (PAC) with AEGLs, ERPGs, & TEELs: Rev. 26 for Chemicals of Concern", U.S. Department of Energy, September 2010.
USDOT06	"Traffic Safety Facts 2006: A Compilation of Motor Vehicle Crash Data", U.S. Department of Transport, DOT HS 810 818, 2006.
USEPA97	"Exposure Handbook Volume 1 - General Factors", U.S. Environmental Protection Agency, EPA/600/P-95.002Fa, 1997.

USMSHA09 "Report of Investigation August 28, 2000 (Underground Coal Mine), Fatal Hoisting Accident (Double Fatality), Mine No. 2 (I.D. No. 15-09571) Excel Mining LLC Pilgrim, Martin County, Kentucky, Hei-Tech (I.D. No. TWN) Tazewell, Virginia" U.S. Mine Safety and Health Administration, http://www.msha.gov/fatals/2000/ftl00c2324.htm. Site Accessed: May 2009.

USNRC04 "Fire Dynamics Tools (FDTs) Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program", U.S. Nuclear Regulatory Commission, NUREG-1805, December 2004.

VALLEY10 Valley B. and Maloney S., "Technical Report: Analysis of DGR-1, DGR-2, DGR-3 and DGR-4 Borehole Images for Stress Characterization", MIRARCO/ Geomechanics Research Centre - Laurentian University, TR-08-35, 2010.

VANDERPLUIJM04 Van der Pluijm, B. and S. Marshak. Earth Structure: An Introduction to Structural Geology and Tectonics. 2nd edition. New York, New York, USA: W.W. Norton, 2004.

VANLOON03 Van Loon L.R., Soler J.M., and Bradbury. M.H., "Diffusion of HTO, $^{36}Cl^-$ and $^{125}I^-$ in Opalinus Clay Samples from Mont Terri. Effect of Confining Pressure", Journal of Contaminant Hydrology, Vol.61, No. 1-4, p.73-83, March 2003.

VANLOON07 Van Loon L.R., Glaus M.A. and Muller W., "Anion Exclusion Effects in Compacted Bentonites: Towards a Better Understanding of Anion Diffusion", Applied Geochemistry, Vol.22, No.11, p.2536-2552, November 2007.

VANSCHMUS92 Van Schmus W.R., "Tectonic Setting of the Midcontinent Rift System", Tectonophysics, Vol.213, No.1-2, p.1-15, October 1992.

VASCONCELOS05 Vascocelos C., McKenzie J.A., Warthmann R. and Bernasconi S.M., "Calibration of the ^{18}O Paleothermometer for Dolomite Precipitated in Microbial Cultures and Natural Environments", Geology, Vol.33, p.317-320, 2005.

VEIZER05 Veizer J., and MacKenzie F.T., "Evolution of Sedimentary Rocks", Sedimentary Rocks: Treatise on Geochemistry, Vol. 7, p.369-407, 2005.

WALLACH98 Wallach J.L., Mohajer A.A., and Thomas R.L., "Linear Zones, Seismicity, and the Possibility of a Major Earthquake in the Intraplate Western Lake Ontario Area of Eastern North America", Canadian Journal of Earth Sciences, Vol.35, No. 7, p.762-786, February 1998.

WANG94 Wang H.F., Crowley K.D., and Nadon G.C., "Thermal History of the Michigan Basin from Apatite Fission-Track Analysis and Vitrinite Reflectance", in P. J. Ortoleva, Ed., Basin Compartments and Seals, American Association of Petroleum Geologists, Memoir 61, p.167-178, 1994.

WATTS09 Watts, M., Schieck D., and Coniglio M., "Technical Report: 2D Seismic Survey of Bruce Site", Nuclear Waste Management Organization, NWMO DGR-REP-01330-0001-R000, February 2009.

WCS07 "Application for License to Authorize Near-Surface Land Disposal of Low-Level Radioactive Waste, Appendix 8.0-5: Accident Analysis", Waste Control Specialists LLC, Rev 12a, March 2007

WEAVER95 Weaver T.R., Frape S.K., Cherry J.A., "Recent Cross-Formational Fluid Flow and Mixing in the Shallow Michigan Basin", Geological Society of America Bulletin, Vol.107, No. 6, p.697-707, June 1995.

WHEELER95 Wheeler R.L., "Earthquakes and the Cratonward Limit of Iapetan Faulting in Eastern North America", Geology, Vol.23, No.2, p.105-108, February 1995.

WHITE00 White D.J., Forsyth D.A., Asudeh I., Carr S.D., Wu H., Easton R.M. and Mereu R.F., "A Seismic-Based Cross-Section of the Grenville Orogen in Southern Ontario and Western Quebec", Canadian Journal of Earth Science, Vol. 37, p.183–192, 2000.

WHITICAR99 Whiticar M.J., "Carbon and Hydrogen Isotope Systematics of Bacterial Formation and Oxidation of Methane", Chemical Geology, Vol.161, No.1-3, p.291–314, September 1999.

WIGSTON09 Wigston, A., and Heagle. D.,"Technical Report. Bedrock Formations in DGR-1, DGR-2, DGR-3 and DGR-4", TR-08-12, Intera Engineering Limited, 2009.

WILSON93a Wilson T.P., and, Long D.T., "Geochemistry and Isotope Chemistry of Michigan Basin Brines: Devonian Formations", Applied Geochemistry, Vol.8, No. 1, p.81-100, January 1993.

WILSON93b	Wilson T.P., and Long D.T., "Geochemistry and Isotope Chemistry Ca-Na-Cl Brines in Silurian Strata, Michigan Basin, U.S.A.", Applied Geochemistry, Vol.8, No. 5, p.507-524, September 1993.
WINTER95	Winter B.L., Johnson C.M., Simo J.A. and Valley J.W., "Paleozoic Fluid History of the Michigan Basin: Evidence from Dolomite Geochemistry in the Middle Ordovician St. Peter Sandstone", Journal of Sedimentary Research, A65(2), p.306-320, 1995.
WIPP11	Waste Isolation Pilot Plant. http://www.wipp.energy.gov/ Site Accessed: January 5, 2011. .
WMO86	"Manual for Estimation of Probable Maximum Precipitation", World Meteorological Organization, Operational Hydrology Report No. 1, 2nd Edition, 1986.
ZHENG99	Zheng Q., "Carbonate Diagenesis and Porosity Evolution in the Guelph Formation, Southwestern Ontario", Ph.D. Thesis, University of Waterloo, Waterloo, Ontario, Canada, 265p, 1999.
ZIEGLER00a	Ziegler K., and Longstaffe F.J., "Multiple Episodes of Clay Alteration at the Precambrian/Paleozoic Unconformity, Appalachian Basin: Isotopic Evidence for Long-Distance and Local Fluid Migrations", Clays and Clay Minerals, Vol. 48, No. 4, p.474-493, August 2000.
ZIEGLER00b	Ziegler K., and Longstaffe F.J., "Clay Mineral Authigenesis Along a Mid-Continent Scale Fluid Conduit in Paleozoic Sedimentary Rocks from Southern Ontario, Canada", Clay Minerals, Vol. 35, No. 1, p.239-260, March 2000.

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16. SPECIAL TERMS

16.1 Units

a	annum
Bq	becquerel
C	Celsius
cm	centimetre
cm^2	square centimetre
d	deuterium excess
dBA	decibels
dm	decimetre
g	gram
ha	hectare
hr	hour
ka	thousand years
kBq	kilobequerel
kg	kilogram
km	kilometer
km^2	square kilometre
kPa	kilopascal
kV	kilovolt
kVA	kilovolt-ampere
kW	kilowatt
L	litre
lux	illuminance
m	metre
m^2	square metre
m^3	cubic metre
M	Nuttli magnitude

Ma	million years
MaBP	million years before present
mAGS	metres above ground surface
mASL	metres above sea level
mBGS	metres below ground surface
MBq	megabequerel
mg	milligram
Mg	megagram
min	minute
mLBGS	metres length below ground surface
mlx	millilux
mm	millimetre
mmol/kgw	millimole per kilogram (water)
mol	mole
mol/kgw	mole per kilogram (water)
MPa	megapascal
mSv	millisievert
MW	megawatt
s	second
Sv	sievert
TBq	terabequerel
V	volt
VAC	volt (alternating current)
W	watt
wt %	mass percentage
^{o}	degrees
µCi	microcurie
µg	microgram
µSv	microsievert

‰	parts per thousand
%	percent
% g	percent of gravitational acceleration

16.2 Abbreviations and Acronyms

2D	Two-Dimensional
3D	Three-Dimensional
3DGF	Three-Dimensional Geological Framework
ADF	Atmospheric Dilution Factors
AECL	Atomic Energy of Canada Limited
ALARA	As Low As Reasonably Achievable
ALW	Active Liquid Waste
ARF	Airborne Release Fraction
ATHEL	Alternative Tile Hole Equivalent Liner
ATV	Acoustic Televiewer
BHWP	Bruce Heavy Water Plant
BMb	Bruce Megablock
BNPD	Bruce Nuclear Power Development
CANDU	Canada Deuterium Uranium
CCME	Canadian Council of Ministers of the Environment
Cd	Concrete Degradation
CHIS	Canadian Hazards Information Services
CI	Crack Initiation
CMBBZ	Central Metasedimentary Belt Boundary Zone
CNSC	Canadian Nuclear Safety Commission
COMS	Constructability, Operability, Maintainability and Safety
CPS	Counts Per Second
CSA	Canadian Standards Association
D_a	Apparent Diffusion Coefficient

D&C	Design and Construction
D_e	Effective Diffusion Coefficient
DAC	Derived Air Concentration
DDP	Detailed Decommissioning Plan
DGR	Deep Geologic Repository
DGSM	Descriptive Geosphere Site Model
DO	Dissolved Oxygen
DOE	Department of Energy (U.S)
DR	Damage Ratio
DRL	Derived Release Limits
DSC	Dry Storage Container
DY	Seismic Loading
δ	Delta
E	East
E	Elastic Modulus (Chapter 4)
EA	Environmental Assessment
EDTA	Ethylenediaminetetraacetic acid
EdZ	Excavation Disturbed Zone
EDZ	Excavation Damaged Zone
E_h	Redox Potential
EIS	Environmental Impact Statement
ELC	Ecological Land Classification
ENE	East-Northeast
ENEV	Estimated No Effect Values
EPD	Electronic Personal Dosimeters
ERA	Ecological Risk Assessment
ERT	Emergency Response Team
ESE	East-Southeast
ETH	Encapsulated Tile Hole

FEPs	Features, Events and Processes
FRAC3DVS	FRACtured Three Dimensional Variably Saturated
ζ	One-Dimensional Loading Efficiency
GFTZ	Grenville Front Tectonic Zone
GL	Glacial Loading
GGM	Gas Generation Model
GMWL	Global Meteoric Water Line
GRG	Geoscience Review Group
GSCP	Geoscientific Site Characterization Plan
HAT	Headspace Air Turnover
HAZOP	Hazard and Operability
HDZ	Highly Damaged Zone
HSM	Historic Saugeen Métis
HTD	Hydrothermal Dolomite
Θ	Porosity
IAEA	International Atomic Energy Agency
IC	In-Ground Container
ICP	Inductively Coupled Plasma
ICRP	International Commission on Radiological Protection
IDLH	Immediately Dangerous to Life and Health
ILW	Intermediate Level Waste
ISAM	Improvement of Safety Assessment Methodologies
ISO	International Organization for Standardization
IX	Ion-Exchange
K	Hydraulic Conductivity
K_H	Horizontal Hydraulic Conductivity
K_V	Vertical Hydraulic Conductivity
L_{eq}	Equivalent Continuous Noise Level
LHHPC	Low Heat High Performance Cement

LGM	Last Glacial Maximum
LIS	Laurentide Ice Sheet
LLSB	Low Level Storage Building
LLW	Low Level Waste
LHD	Load Haul Dump
L&ILW	Low and Intermediate Level Waste
LPF	Leakpath Factor
LSD	Long-term Strength Degradation
MAC	Maximum Acceptable Concentration
MAR	Material At Risk
MASHA	Mines and Aggregates Safety and Health Association
MCC	Main Control Centre
MCNP	Monte Carlo N-Particle
MLE	Mean Life Expectancy
MNO	Métis Nation of Ontario
MOE	Ministry of Environment
MOL	Ministry of Labour
MP	Massively Parallel
MPC	Mine Power Centres
MRT	Mine Rescue Team
MVT	Mississippi Valley Type
N	North
NE	Northeast
NNE	North-Northeast
NEA	Nuclear Energy Agency
NEC	No-Effect Concentration
NE-GG1	Increase Gas Generation Case
NE-NM	Methanogenic Reaction Case
NE-RC	DGR Reference Case

NE-SBC	Steady-State Cambrian Overpressure Case
NEW	Nuclear Energy Worker
NGO	Non-Governmental Organization
NGS	Nuclear Generating Station
NHF	Natural Hydraulic Fractures
NMb	Niagara Megablock
NNW	North-Northwest
NW	Northwest
NIOSH	National Institute for Occupational Safety and Health
NSCA	Nuclear Safety and Control Act
nSIGHTS	Sandia National Laboratories Numerical Hydraulic-Test Simulator
NNW	North-Northwest
NW	Northwest
NWMD	Nuclear Waste Management Division
NWMO	Nuclear Waste Management Organization
ν	Poisson's Ratio
OD	Outer Diameter
OECD	Organization for Economic Co-operation and Development
OGSR	Oil, Gas, and Salt Resources Library
OHSAS	British Standards Institution's Occupational Health and Safety Assessment Series
OL	Outer Length
OPG	Ontario Power Generation
PAC	Protective Action Criteria
PAH	Polycyclic Aromatic Hydrocarbons
PCB	Polychlorinated Biphenyls
PDP	Preliminary Decommissioning Plan
PHT	Primary Heat Transport
PMF	Probable Maximum Flood
PMP	Probable Maximum Precipitation

PP	Pore Pressure
PPE	Personal Protective Equipment
PQP	Project Quality Plan
PSR	Preliminary Safety Report
PSHA	Probabilistic Seismic Hazard Assessment
R_a	Isotope Ratio in Air
RA	Regulatory Approvals
r_b	Bulkhead Radius
r_s	Preclosure Shaft Radius
r_s'	Postclosure Shaft Radius
R_s	Isotope Ratio in the Sample
REMP	Radiological Environmental Monitoring Program
RF	Respirable Fraction
RQD	Rock Quality Designation
RWC(EF)	Retube Waste Containers (for end fittings)
RWC(PT)	Retube Waste Containers (for pressure tubes, calandria tubes, calandria tube inserts)
RWSB	Refurbishment Waste Storage Building
ρ	Fluid Density
S	South
SE	Southeast
SEM/EDS	Scanning Electron Microscope/Energy Dispersive Spectral
SON	Saugeen Ojibway Nation
S_s	Specific Storage
SSE	South-Southeast
SSW	South-Southwest
SVCA	Saugeen Valley Conservation Authority
SW	Southwest
σ_v	Vertical Compressive Stress
σ_H	Maximum Horizontal Stress

σ_h	Minimum Horizontal Stress
TD	Time Dependent Strength Degradation
TDS	Total Dissolved Solids
T-H-E	Tile Hole Equivalent
TIBL	Thermal Inversion Boundary Layer
TOC	Total Organic Carbon
TOUGH2	Transport of Unsaturated Groundwater and Heat Version 2
TSS	Total Suspended Solids
TU	Tritium Units
τ	Tortuosity
UCS	Uniaxial Compressive Strength
UFDS	Used Fuel Dry Storage
UofT GSM	University of Toronto Glacial Systems Model
VFD	Variable Frequency Drive
VoIP	Voice over Internet Protocol
W	West
WHMIS	Workplace Hazardous Materials Information System
WIPP	Waste Isolation Pilot Plant (U.S.)
WNW	West-Northwest
WPRB	Waste Package Receiving Building
WRMA	Waste Rock Management Area
WSW	West-Southwest
WVRB	Waste Volume Reduction Building
WWMF	Western Waste Management Facility
XRD	X-Ray Diffraction

16.3 Glossary of Terms

Advection – A process by which dissolved or suspended substances (natural constituents, artificial tracers, contaminants), are transported by the bulk motion of a fluid medium (water, air).

Aerobic – Commonly used to describe the presence of air (oxygen), the term aerobic is often used interchangeably with the term *oxic*. However, aerobic can also be used more generally to describe environments in which one or more redox couples control the redox potential (Eh) at relatively positive values.

Algonquin Arch – A northeast trending crystalline basement doming (high) that separates the *Michigan Basin* from the *Appalachian Basin*.

Anaerobic – Commonly used to describe the absence of air (oxygen), the term anaerobic is often used interchangeably with the term anoxic. However, anaerobic can also be used more generally to describe environments in which one or more redox couples control the redox potential (Eh) at relatively negative values.

Analogue (Geosphere) – An investigation or quantitative analysis of the natural evolution of a repository site that conveys an understanding of long-term geologic and hydrogeologic stability relevant to demonstrating concepts of long-term waste isolation and containment.

Anhydrite – A mineral consisting of anhydrous calcium sulphate: $CaSO_4$. It represents gypsum without its water of crystallization, and it alters readily to gypsum, from which it differs in crystal form and in being harder and slightly less soluble. Anhydrite usually occurs in white or slightly colored, granular to compact masses, forming large beds or seams in sedimentary rocks or associated with gypsum or halite in evaporites.

Anisotropy – The condition of having properties that vary with direction at a given point location (e.g., a glacial till or clay, in which the hydraulic conductivities could be orders of magnitude different in the x, y, and z directions).

Appalachian Basin – An elongated *sedimentary basin* on the North American continent, with a maximum depth of 12 km. In southern Ontario, sedimentary rocks of both the Appalachian Basin and *Michigan Basin* overlie the Precambrian crystalline basement, with a maximum thickness of approximately 1.5 km.

Aquiclude – A medium with very low values of hydraulic conductivity (permeability) which, although it may be saturated with groundwater, is almost impermeable with respect to groundwater flow. Such geologic media will act as boundaries to aquifers and may form confining strata.

Aquifer – A geological formation or structure that is sufficiently porous and permeable to store, transmit, and yield significant or economic quantities of groundwater to wells and springs. A confined aquifer is bound by low permeability formations such that it is under pressure. An unconfined aquifer is one whose upper groundwater surface (water table) is at atmospheric pressure.

Aquitard – A confining bed and/or formation composed of rock or sediment that retards but does not prevent the flow of water to or from an adjacent aquifer. It does not readily yield water to wells or springs, but stores groundwater.

Argillaceous – Pertaining to, largely composed of, or containing clay-size particles (< 4 microns) or clay minerals.

Argillaceous Limestone – A limestone containing an appreciable amount (but < 50 percent) of clay.

Backfill – An engineered material formulated and placed to fill the excavated openings in a repository as part of sealing and closure. See also *Grout.*

Basement (rock) – The crust of the Earth (Precambrian igneous and metamorphic complex) underlying the sedimentary deposits.

Bathymetry – The measurement of water depth at various locations within a body of water. Bathymetry maps enable estimates of the topography and elevation of ground surface within areas covered by bodies of water.

Bedding – The natural arrangement of sedimentary rocks into layers of varying thickness and character.

Biosphere – The physical media (atmosphere, soil, surface waters and associated sediments) and the living organisms (including humans) that interact with them.

Borehole Breakout – The spalling at the edge of a borehole as a result of the concentration of the maximum horizontal stress. The stress concentration is so large that induced differential stress causes shear fractures within the rock next to the borehole wall. Spalling releases the fractured rock to create a deformation or elongation of the borehole wall in the direction of the least horizontal stress.

Bounding Assessment – An assessment designed to provide limiting estimates, based on simplification of the processes being simulated or the use of data limits (such as maximum possible precipitation, or thermodynamic solubility limits).

Brackish Water – Water with a salinity between freshwater and seawater (i.e., water that contains between 1 and 10 g/L total dissolved solids. See also *Brine* and *Saline Water*.

Breccia – A coarse-grained clastic rock, composed of angular or broken rock fragments, and held together by a mineral cement or fine-grained matrix.

Brine – Water with a salinity greater than 100 g/L total dissolved solids. See also *Brackish Water* and *Saline Water*.

Bruce Megablock – A regional subdivision of Southern Ontario based upon characteristics of an interpreted fracture framework, developed by Sanford (1985). It extends from the top of the *Algonquin Arch* to Georgian Bay to the north.

Bruce nuclear site – The 932 hectare (9.32 km^2) parcel of land located within the administrative boundaries of the Municipality of Kincardine in Bruce County. Two operating nuclear stations are located on the site. The site is owned by OPG but has been leased to Bruce Power since May 2001. However, parts of the site, including land on which WWMF is located, have been retained by OPG. See also *OPG-retained lands*.

Bruce Power – The licensed operator of the Bruce A and Bruce B nuclear generating stations.

Calcareous – Term referring to a rock, mud, or cement that is mostly or partly composed of calcium carbonate (typically >50%).

Cambrian – The earliest period of the Paleozoic era extending from 543 to 490 million years ago; also, refers to rocks formed, or sediments laid down, during this period (e.g., Cambrian sandstones).

Canadian Nuclear Safety Commission (CNSC) – The Canadian federal agency responsible for regulating nuclear facilities and materials, including management of all radioactive waste in Canada.

Canadian Shield – A large plateau that occupies most of eastern and central Canada and consists of exposed Precambrian basement rocks in a stable craton. It is surrounded by younger sedimentary rocks.

CANDECON Waste – CANDECON is a chemical decontamination process for nuclear heat transport systems. Wastes produced from this process are contaminated resins and filters, which contain high levels of chelating agents such as EDTA.

Cap rock – Refers to the thick sequence of Ordovician shales that act as a barrier to fluid movement and overlie the DGR host rock.

Capillary Pressure – The difference in pressure across two immiscible fluid phases jointly occupying the interstices of a rock.

Chatham Sag – A narrow topographic low within the Precambrian crystalline basement surface that separates the Algonquin and Findlay Arches; located in the vicinity of Lake St. Clair in southwestern Ontario.

Clastic – Refers to rock or sediment that is composed primarily of broken fragments derived from pre-existing rocks or minerals, which have been transported some distance from their place of origin and accumulated.

Closure – The administrative and technical actions directed at a repository at the end of its operating lifetime. For example covering the waste (for a near surface repository), backfilling and/or sealing of rooms, tunnels and/or shafts (for a geological repository), and termination or completion of activities in any associated structures.

Compaction of Waste –Compaction by medium force of waste, such as light metal objects, insulation materials, hoses, cables, metal fillings and turnings, with a contact dose rate less than 2 mSv/h (200 mrem/h).

Conceptual Model – A set of qualitative and/or quantitative assumptions used to describe a system or subsystem for a given purpose. At a minimum, these assumptions concern the geometry and dimensionality of the system, temporal and spatial boundary conditions, and the nature of the relevant physical and chemical processes. The assumptions should be consistent with one another and with existing information within the context of the given purpose.

Containment (Safety Case) – Limiting the release of hazardous materials to the biosphere.

Crack Initiation Stress – Represents the threshold marking the onset of stable crack growth in brittle rock under loading, which is the lower bound for the in situ rock strength, and is identifiable as the point where the lateral strain curve of a test rock sample departs from linearity (or the initiation of acoustic emission response of the sample to loading).

Craton – A large portion of a continental plate that has remained relatively tectonically stable since the Precambrian era.

Critical Group – A group of members of the public which is reasonably homogeneous with respect to its exposure for a given contamination source and given exposure pathway, and is typical of individuals receiving the highest health impacts by the given exposure pathway from the source.

Decommissioning – Those actions taken, in the interest of health, safety, security and protection of the environment, to retire a licensed activity/facility permanently from service and render it to a predetermined end-state condition.

Deep Geologic Repository (or DGR, or Repository) – The underground portion of the deep geologic repository facility for low- and intermediate-level waste. Initially, the repository includes the access-ways (shafts, ramps and/or tunnels), underground service areas and installations, and emplacement rooms. In the postclosure phase it also includes the engineered barrier systems. The repository includes the waste emplaced within the rooms and excludes the excavation damage zone.

Deep Geologic Repository Facility (or DGR Facility, or Repository Facility) – The deep geologic repository for low- and intermediate-level waste, and the various surface and underground support facilities. The support facilities include equipment, materials and infrastructure for receiving, inspecting and handling waste packages, for transferring waste packages from the surface to the repository horizon, for handling the waste packages in the repository, for emplacing waste packages, for excavating the repository (during operations), for constructing room shield walls, and for material storage. The repository facility excludes the waste emplaced within the rooms and any zones of damaged rock around underground openings.

Deep Geologic Repository Site (or DGR Site, or Repository Site) – The physical location of the deep geologic repository on the Bruce nuclear site. It is characterized by such features as its proximity to other human developments, geology, hydrogeology and geotechnical conditions, adjacent land use patterns, and meteorological and seismic conditions.

Deep Geologic Repository System (or DGR System, or Repository System) – The deep geologic repository facility for low and intermediate-level waste, its geological setting, and the surrounding surface environment. The system includes the wastes, and the engineered and natural barriers that provide isolation and containment of the waste.

Deformation – A general term for the process of folding, faulting, shearing, or fabric development of the rocks as a result of Earth stresses; or the change in geometry of a body of rock as a consequence of stress(es).

Descriptive Geosphere Site Model – A description of the present day 3-dimensional physical and chemical characteristics of a specific site as they relate to implementation of the Deep Geologic Repository concept. The model is based on the integration of multi-disciplinary geoscientific data that, in part, relies on multiple lines of evidence to constrain uncertainty and/or non-uniqueness in interpretation. See also *Geosynthesis.*

Design Basis – Identifies specific functions to be performed by a system, structure, equipment, component or software; and the specific values or range of values chosen for controlling parameters as reference bounds for the design.

Design Life – The period during which a structure, system or component will perform while still meeting original design specifications, including routine maintenance but without major repair or refurbishment.

Deuterium – Refers to 'heavy hydrogen', ^2H, the stable isotope of hydrogen that has an atomic mass of two, as opposed to the common isotope of hydrogen, ^1H, which has an atomic mass of one.

Devonian – The fourth period of the Paleozoic Era extending from 417 to 354 million years ago; also refers to rocks formed, or sediments laid down, during this period (eg., Devonian shales).

Diffusion – The process by which both ionic and molecular species dissolved in water move from areas of higher concentration to areas of lower concentration. Movement is random and is proportional to the gradient of concentration. The process tends to distribute the particles more uniformly. See also *Advection* and *Dispersion*.

Diffusion Coefficient – The diffusion coefficient D is the constant of proportionality relating the solute flux J_i to the solute concentration gradient in a given co-ordinate direction $\partial C/\partial x_i$ as described by Fick's First Law: $J_i = -D\ \partial C/\partial x_i$

Apparent Diffusion Coefficient (D_a) – The diffusion coefficient for a specific solute in a porous medium that accounts for the 3-dimensional geometry of the pore space, as well as the sorption behaviour of the solute. It is related to the *effective diffusion coefficient* D_e and the porous medium capacity factor α as follows: $D_a = D_e / \alpha$

Effective Diffusion Coefficient (D_e) – The diffusion coefficient for a specific solute in a porous medium that accounts for the 3-dimensional geometry of the pore space, including tortuosity, constrictivity and diffusion-accessible porosity. It is the product of the diffusion-accessible porosity ϕ_{diff}, the tortuosity factor τ_f, and the free-water diffusion coefficient D_0 as follows: $D_e = \phi_{diff} \cdot \tau_f \cdot D_0$

Digital Elevation model (DEM) – A representation of the topography of the land surface in a digital format (also digital terrain model). Data files consist of elevation data related to rectangular grid coordinates.

Dip – The maximum angle that a geological structural surface (bedding plane, fault, etc.) makes with the horizontal; measured in the vertical plane, perpendicular to the strike of the structure.

Discontinuity – Any interruption in sedimentation (unconformity), for whatever cause or length of time. Typically, discontinuities represent time periods of non-deposition or erosion. May also refer to any naturally occurring fracture (break) in logging rock core samples.

Dispersion – A small scale, spreading and mixing process resulting from dissolved substances traveling at different velocities along and between flow paths through a porous or fractured medium. The spreading of the dissolved substance in the direction of bulk flow is known as longitudinal dispersion. Spreading in directions perpendicular to bulk flow is known as transverse dispersion.

Disposal – The emplacement of waste in an appropriate facility without the intention of retrieval.

Dolostone – A sedimentary rock of which more than 50 percent by weight consists of the mineral dolomite (magnesium carbonate). Dolostone is generally thought to form when magnesium ions replace some of the calcium ions in limestone by the process of dolomitization. Migrating fluids along some faults and fractures may locally dolomitize limestone, the resulting rock being more porous may become a host for oil and gas deposits.

Dose – A measure of the energy deposited by radiation in a tissue. Also referred to as absorbed dose, committed equivalent dose, committed effective dose, effective dose, equivalent dose or organ dose, depending on the context.

Drilling Fluid – A fluid used to lubricate and cool the drill bit, to carry cuttings from the bottom, and to maintain a hydrostatic pressure in the borehole offsetting pressures of fluids that may exist in the formation. For the DGR, water from Lake Huron was employed to drill the upper rock sequence above the Salina Formation (where fresh groundwater is encountered) and a brine-based fluid was used to drill the Salina and underlying formations (where saline groundwaters are present).

DRL (**Derived Release Limit**) - The limit at which release of a radionuclide occurring from a nuclear station or a facility will not result in dose to individual members of the public exceeding the dose limits set by the CNSC.

Dyke – A planar injection of magmatic or sedimentary material that cuts across the pre-existing fabric of a rock. Dykes can be formed by the filling of a crack/fissure from above, below, or laterally by forcible injection, or intrusion, under abnormal pressures.

Earthquake – A shaking or trembling of the earth resulting from subterranean movement usually along faults.

Elastic Modulus – A measurement of material stiffness. The modulus represents the ratio of the stress applied to a body to the strain that results in the body in response to the stress. All moduli of elasticity determined in DGR testing are tangent Young's moduli, which are computed based on the stress-strain curve at a fixed stress level of 40% of the peak strength of the material.

Emplaced Volume (Waste) – The external volume of the *waste package* for emplacement in the DGR, which includes the waste, storage container, overpack, and/or shield.

Emplacement Room – A portion of the underground repository into which waste packages are permanently placed. Rooms are bounded by the host rock for floor, ceiling and walls on most sides, and by a wall or access tunnel on one side.

Engineered Barrier – A physical obstruction that has been constructed to prevent or delay water seepage and/or radionuclide migration and/or migration of other materials between components in the repository, or between the repository and the surface environment.

Environmental Isotopes – Naturally occurring stable and radioactive *isotopes* of elements found in the environment. The principal elements of hydrogeological, geological and biological systems are hydrogen, oxygen, carbon, nitrogen and sulphur. Less abundant elements include helium, argon and krypton. Environmental isotopes permit quantitative determinations of the origin, age and flow paths of groundwaters on a regional scale.

Equivalent Sound Level (Leq) – Average weighted sound level over a specified period of time.

Eustasy/Eustatic – Refers to sea-level changes which occur on a global scale. Eustasy results from either a change in the volume of seawater, or a change in the size of the ocean basin that contains the water. Causes of eustatic sea level change include glaciations and deglaciation, tectonic activity, and continental drift.

Excavation Damaged Zone (EDZ) – The region of rock around repository openings that has been physically or chemically affected as a result of the excavation process, with significant changes in flow and transport properties (i.e., permeability of the rock increased by at least one order of magnitude). See also *Highly Damaged Zone* and *Excavation Disturbed Zone*.

Excavation Disturbed Zone (EdZ) – The region of rock surrounding the EDZ with possible stress or flow changes as a result of the excavation, but without significant changes in flow and transport properties (i.e. permeabilities with the rock materially unchanged). See also *Highly Damaged Zone* and *Excavation Damaged Zone*.

Exposure Pathway – A route by which contaminants can reach humans or biota and cause exposure. An exposure pathway may be very simple, for example external exposure from airborne contaminants, or involve a more complex chain, for example internal exposure from drinking milk from cows that ate grass contaminated with deposited contaminants.

Facies Change – A lateral or vertical variation in the lithologic or paleontologic characteristics of contemporaneous sedimentary deposits. It is caused by, or reflects, a change in the depositional environment.

Fault – A discrete surface or zone of discrete surfaces separating two rock masses across which one mass has slid past the other. Any faults in the DGR region would most likely be vertical/sub-vertical with probable vertical displacements propagating from the Precambrian surface into the overlying sedimentary rocks.

Feldspars – A group of abundant rock-forming minerals, generally rich in potassium, sodium, calcium, barium, rubidium, and strontium, as well as silicon and aluminum. Feldspars constitute approximately 60% of the Earth's crust.

FEPs (Features, Events and Processes) - FEPs are all relevant factors that describe the current state and possible future evolution of a system. They are used as input for scenario development and subsequent consequence analysis regarding health, safety and environment.

Filter Waste – Depending on each specific station system, filter waste may consist of disposable vessels along with the exhausted filter cartridges contained therein, or filter cartridges from systems employing permanent vessels.

Focal Depth – The depth at which an earthquake originates (the focal depth can be measured with respect to mean sea level, or with respect to the average ground surface elevation for all seismic stations that record a given seismic event).

Fracture - A general term for any surface within a material across which there is no cohesion, including cracks, joints, faults, and bedding partings.

Geosphere – The rock around the repository, and extending up to the biosphere. It can consist of both an unsaturated zone (which is above the groundwater table) and the saturated zone (which is below the groundwater table).

Geosynthesis – The assembly of all the geologically-based evidence relevant to the repository safety case; the integration of multi-disciplinary geoscientific data relevant to the development of a descriptive conceptual geosphere model; explanation of a site-specific descriptive conceptual geosphere model within a systematic and structured framework. See also *Descriptive Geosphere Site Model*.

Glacial Perturbations – Changes in geological, hydrological or geochemical systems as a result of glacial processes that include glacial isostacy, permafrost and ice sheet history.

Glaciation – The formation, movement, and recession of glaciers or ice sheets.

Graben – An elongate geological depression bounded on both sides by high-angle normal faults that dip toward one another.

Grenville Front Tectonic Zone (GFTZ) – That part of the Central Gneiss Belt (a subdivision of the Precambrian Grenville Province) that lies within 20-30 km of the Grenville Front boundary fault, consists of deformed and metamorphosed rocks, and is characterized by northeasterly trending shear zones (several kilometers wide) and foliation.

Grenville Orogeny – A major plutonic, metamorphic, and deformational event during the Precambrian era, 800 to 1,000 million years ago, which affected a broad province along the southeastern border of the Canadian Shield. The Grenville orogeny is thought to be the consequence of a Himalayan-type continental collision during the assembly of a supercontinent (Rodinia).

Groundwater (or Ground water) – In general, water contained in geologic formations below the Earth's surface. In the context of the DGR, the term is specifically applied to water that is relatively unconstrained by low permeability media and therefore free to flow under the influence of hydraulic gradients. This includes water within the connected pore space between mineral grains in unconsolidated sediment or in a fractured or porous rock matrix, as well as water in permeable, connected structures in the subsurface. See also *Porewater.*

Grout – A fluid mixture of cementitious materials, aggregates, additives and/or clay and water that will flow without segregation of the constituents into small spaces, and will form a low-permeability fill material to resist groundwater flow. In the DGR context, grouting applies to filling of fractures within the rock, or pore spaces within waste containers. See also *Backfill*.

Highly Damaged-Zone (HDZ) – The zone of rock around an excavation where macro-scale fracturing or spalling may occur, thereby inducing changes in flow and transport through the interconnected fracture system (i.e. permeabilities within the rock increased by at least 2 orders of magnitude). See also *Excavation Damaged Zone* and *Excavation Disturbed Zone*.

Human Intrusion – Human actions that modify the performance of engineered and/or natural barriers leading to the creation of a route by which humans (potentially both the intruder(s) and public) could be exposed to radionuclides derived from the repository.

Hydraulic Conductivity – The capacity of a rock to transmit a fluid. It is expressed as the volume of water at a given kinematic viscosity that will move in unit time under a unit hydraulic gradient through a unit area measured at right angles to the direction of flow.

Hydraulic Gradient – The rate of change of pressure (pressure head) per unit of distance. Typical hydraulic gradients in natural groundwater flow systems are on the order of 0.01 to 0.001.

Hydraulic Head – Fluid mechanical energy per unit weight of fluid, which correlates to the elevation that water will rise in a well.

Environmental Head – The sum of the elevation head and the pressure head calculated using the average density of the water over the entire vertical water column. This is used for calculating vertical hydraulic gradients.

Freshwater Head – The sum of the elevation head and the pressure head calculated using the density of fresh water (1000 kg m^{-3}). This is used for calculating horizontal hydraulic gradients.

Hydrogeology – The science that deals with subsurface waters and related aspects of surface waters. Hydrogeology is the study of the law governing 1) the movement of groundwater, 2) mechanical, chemical, and thermal interaction of groundwater and the porous medium, and 3) the transport of energy and chemical constituents by flow of groundwater.

Iapetus Ocean – The ocean that existed east of North America before Europe and Africa collided with North America during the Carboniferous and Permian periods (320-250 million years ago).

IC-18 – An in-ground storage structure used for intermediate level waste, primarily ion exchange resins, with a capacity of 18 m^3. See *In-Ground Storage*.

Incinerable Waste – Radioactive waste materials generally consisting of paper, plastic, wood, cardboard etc. which can be incinerated. The contact dose rate of such waste is less than 0.6 mSv/hr (60 mrem/hr).

In-Ground Storage – Storage of waste in in-ground containers (ICs); generally used for intermediate level waste. All ICs with the exception of those used for heat exchangers consist of steel liners fixed with concrete inside boreholes in the ground. IC-HXs use limestone gravel for the backfill.

In-Service Date – The date on which the facility is put into service or made available for operation.

In-Situ Stress – The natural or virgin state of stress in a rock mass that was derived from a pervasive force field imposed by geological perturbations such as tectonic activity.

Institutional Control – Control of a deep geologic repository by an authority or institution designated under the laws of a country or state. This control may be active (monitoring, surveillance, remedial work) or passive (land use control).

Intermediate-Level Waste (ILW) – Radioactive non-fuel waste, containing significant quantities of long-lived radionuclides (generally refers to half-lives greater than 30 years).

Intracratonic Basin – A basin formed in the interior region of a continental *craton* (away from plate boundaries) due to subsidence of some part of the craton.

Ion – An atom or molecule that has an unbalanced charge (i.e. the number of protons is not equal to the number of electrons). A cation is an ion with a net positive charge (e.g. Ca^{2+}, Na^{+}) and an anion is an ion with a net negative charge (e.g. Cl^{-}, SO_4^{2-}).

Irradiated Core Components – Radioactive waste such as flux detectors and liquid zone control rods resulting from the routine replacement of core components during the operation of nuclear reactors.

Isolation (Safety Case) – Making human encounter with the waste unlikely.

Isotope – An isotope is one of two or more species of the same element that have the same number of protons in the nucleus but a different number of neutrons, which results in small variations in the atomic mass (e.g., oxygen has 8 protons, but the atomic masses of naturally occurring oxygen isotopes range between ^{16}O, ^{17}O and ^{18}O). See also *radioisotope.*

IX Resin – Ion-exchange resin used to maintain the water quality in station process systems (e.g., moderator and Primary Heat Transport heavy water systems, and light water auxiliary systems such as the Active Liquid Waste Treatment System).

Joint – A planar fracture, crack, or parting in a rock, without shear displacement. Often occurs with parallel joints to form part of a joint set.

Karst – A type of topography that is formed in limestone, gypsum or other rocks, primarily by dissolution, and that is characterized by sinkholes, caves and underground drainage. The most common type of karst is associated with the dissolution of limestone by meteoric waters when the carbonate rocks are exposed to the atmosphere at the Earth's surface, forming an unconfined aquifer. This most commonly occurs when shallow-marine limestones have become exposed due to a fall in sea-level. Karst can also be formed in coastal settings where fresh and marine waters mix, or as a result of limestone dissolution by sulphuric acid during deep burial of sediments.

Kimberlite – A mantle-derived *ultramafic* igneous rock containing at least 35% olivine, does not contain leucite, and contains one or more of the following: monticellite, carbonate, serpentine, diopside, or phlogopite.

L&ILW – Low and Intermediate Level Waste.

Limestone – A sedimentary rock composed of the mineral calcite (calcium carbonate). Where it contains appreciable magnesium carbonate it is called dolomitic limestone. The primary source of this calcite is usually the shells of marine organisms. See also *Dolostone*.

Lineament – An extensive linear geologic or topographic surface feature. Some examples are straight stream courses, fault lines, and straight escarpments.

Lithofacies – A lateral, mappable, subdivision of a stratigraphic unit, distinguished from adjacent subdivisions on the basis of lithology (mineralogy, petrography, paleontology – appearance, composition, and texture).

Lithology – Describes the physical character of a rock, including color, grain size, and mineralogy.

Low Level Storage Building (LLSB) - Refers to a series of buildings at OPG's Western Waste Management Facility for the interim storage of low-level waste.

Low-Level Waste (LLW) – Radioactive waste in which the concentration or quantity of radionuclides is above the clearance levels established by the regulatory body (CNSC), and which contains primarily short-lived radionuclides (half-lives shorter than or equal to 30-years).

Mafic – General term for igneous rocks composed primarily of ferromagnesian (iron- and magnesium-rich), dark-colored, minerals.

Marker (bed) – An easily recognized stratigraphic feature having characteristics distinctive enough for it to serve as a reference point or datum, and that is traceable over long distances, especially in the subsurface (i.e. unconformities, salt beds, etc.).

Mesozoic – An era of geologic time covering the time span from 248 to 65 million years ago, that lies above the *Paleozoic* and below the Cenozoic. This is the era when dinosaurs roamed on earth.

Meteoric Recharge – Surface water that has recently been a part of the atmospheric portion of the hydrologic cycle, which has infiltrated into the sub-surface.

Methanogenesis – The generation of methane (CH_4) as a result of biogenic (microbial) activity.

Michigan Basin – A nearly-circular intracratonic *sedimentary basin* with a diameter of between 500 and 600 km, centered in Michigan, with a maximum depth of over 4 km. In southern Ontario, sedimentary rocks from edges of both the Michigan Basin and the *Appalachian Basin* are present. The maximum thickness of the sedimentary rocks in southern Ontario is approximately 1.5 km.

Microseismicity – Very low level seismic activity, generally considered to be seismic events of M3 or less. The three borehole seismographs installed in 2007 in the vicinity of the *Bruce nuclear site* are capable of measuring microseismic events of less than M1.

Mississippi Valley-type (MVT) deposit – A strata-bound hydrothermal deposit of lead and/or zinc minerals in carbonate rocks, together with associated minerals fluorite and barite. These deposits characteristically have relatively simple mineralogy, occur as veins and replacement bodies, are at moderate to shallow depths, show little post-ore deformation, are marginal to sedimentary basins, and are without an obvious source of mineralization.

Near- field Rock – The rock adjacent to the repository that may have experienced changes in flow, mechanical, chemical or microbial characteristics as a consequence of the excavation, operation, decommissioning and closure of the repository. See also *Highly Damaged Zone, Excavation Damaged Zone* and *Excavation Disturbed Zone*.

Neo- – Prefix used when referring to something 'new' or 'recent'.

Neotectonic - Tectonic activity that had occurred since the last *glaciation*, in the last 12,000 years.

Net Volume (Waste) – The internal volume of the container in which waste is stored.

Non-Processible Waste – Wastes that are neither incinerable nor compactible, such as heavy gauge metal objects, glass, concrete, tools, heavy slings and cables. Maximum dose rate is 10 mSv/h (1 rem/hr) at 30 cm for storage in LLSBs. Higher dose rate wastes are stored in shielded structures, notably trenches or ICs.

OPG-retained Land – The parcels of land on the Bruce nuclear site for which control has been retained by OPG. This includes the WWMF, certain landfills, and the Heavy Water Plant Lands.

Ordovician – The second period of the *Paleozoic* Era extending from 490 to 443 million years ago; also refers to rocks formed, or sediments laid down, during this period (eg., Ordovician carbonates).

Orogeny – A period of mountain building that lasts for several to tens of millions of years.

Osmosis – The movement of water across a semi-permeable membrane in order to reduce the difference in solution concentration. Water moves from a volume of low solute concentration to a volume of high solute concentration - essentially diluting the fluid of high solute concentration by the addition of water, and concentrating the fluid of low solute concentration by the removal of water.

Outcrop – An exposure of bedrock at the surface of the Earth. Specifically, an outcrop is the part of a geologic (rock) formation or structure that appears at or above the surface of the surrounding land.

Overcoring – Rock coring directly over an existing smaller diameter borehole to relieve the in situ stresses present in the smaller borehole. Used to measure the magnitude and direction of in situ stresses.

Overpack – An enclosure used to provide physical and/or radiological protection or convenience in handling of a waste package, or to combine two or more waste packages.

Oxic – Often used interchangeably with the term *aerobic*, oxic strictly means the presence of oxygen.

Paleo- – Prefix used when referring to something 'ancient' or 'old' (e.g. *Paleozoic* refers to 'ancient/old life'), or which involved ancient conditions (e.g. paleoclimate).

Paleohydrogeology – The hydrogeologic study (physical/chemical) of the evolution of a site or flow domain based on knowledge of its current state and external perturbations that have acted upon it in geologic time.

Paleozoic – The time span covering approximately from 540 to 250 million years ago.

Periglacial – The conditions, processes and landforms associated with non-glacial cold climate conditions. Periglacial environments are those where frost action or permafrost processes dominate.

Permafrost – Ground that has been below 0°C for at least 2 years. It is not necessarily frozen because the freezing point of any included water may be depressed by pressure or salinity, or moisture may not be present. A continuous layer of permafrost is found where the annual mean temperature is below about -5°C.

Permeability – The ease with which a porous medium can transmit water or other fluids. The intrinsic permeability [m^2] of medium is independent of the type of fluid present.

Phanerozoic – Includes the *Paleozoic*, *Mesozoic*, and Cenozoic eras, and represents the time-frame from 540 million years ago to present.

Pinnacle Reef – A small reef patch, consisting of coral growing sharply upwards (with slopes ranging from 45° to nearly vertical). In southern Ontario, ancient, fossilized pinnacle reefs occur in the Guelph Formation and can become oil and gas traps when they are capped by anhydrite or shale.

Pleistocene – The earlier of two epochs comprising the Quaternary Period covering the time span from 1.8 million years to 11.5 thousand years before present.

Poisson's Ratio – The ratio of the lateral strain (perpendicular to the applied load) to the axial strain (in the direction of the applied load) in a body that has been stressed longitudinally within its elastic limit.

Porewater (or Pore water) – Water within the connected pore space between mineral grains in low-permeability sediments or rocks in which flow under the influence of hydraulic gradients is inhibited. In contrast with groundwater, which flows into or can be sampled from boreholes over time scales of days to months, laboratory techniques are generally required to extract porewaters from the sediment or rock matrix. See also *Groundwater*.

Porosity – **Physical Porosity** – The volume of pores per total volume of sample. Pores are defined as everything which is not solid. Interlayer water of clays is considered as part of the pore space.

Diffusion (Accessible) Porosity – The volume of pores, per total volume, accessible for a given solute. Typically determined from diffusion experiments. Solute specific.

Transport Porosity (also Effective porosity) – The proportion of the *physical porosity* of a rock or soil in which transport of fluids (e.g., gases, water) occurs.

Water Loss Porosity – The volume of pores per total volume of sample, derived from water extraction at 105°C (additional specification if extracted e.g., under vacuum). In argillaceous rocks, water loss porosity at 105°C is usually somewhat smaller than the *physical porosity*, because the bound water is only partially released at this temperature.

Postclosure Monitoring – Monitoring during the time period following closure of the repository.

Postclosure Phase – The period of time following closure of the deep geologic repository.

Potentiometric surface – An imaginary surface that represents the total hydraulic head in an aquifer. It represents the height above a datum plane at which the water level stands in tightly cased wells that penetrate the aquifer.

Precambrian – All geologic time before the beginning of the Paleozoic Era, preceding 543 million years ago; also refers to rocks formed, or sediments laid down, during this period (eg., Precambrian gneiss).

Preclosure Phase – The period of time that includes all activities from siting through to decommissioning and closure of all components of the deep geologic repository.

Preliminary Design – A design product that is sufficiently developed so that management can determine the merit of completing the design based on financial, safety and regulatory criteria.

PSR – Preliminary Safety Report. See *Safety Report.*

Quadricell – An above ground storage structure used for intermediate level waste, primarily ion exchange resins.

Quaternary – The upper time period of the Cenozoic era, extending from 1.8 million years ago and continuing into the present. It contains two epochs: the *Pleistocene* and the Holocene.

Radioactive Waste – Any material (liquid, gaseous or solid) that contains a radioactive "nuclear substance" as defined in Section 2 of Nuclear Safety and Control Act, and which the owner has declared to be waste. In addition to containing nuclear substances, radioactive waste may also contain non-radioactive "hazardous substances", as defined in Section 1 of the CNSC's General Nuclear Safety and Control Regulations.

Radioisotope – A radioactive *isotope*. See also *radionuclide.*

Radionuclide – A radionuclide is an atom with an unstable nucleus which can undergo radioactive decay by the emission of gamma ray(s) and/or subatomic particles. The resulting emission(s) is defined as radiation. See also *radioisotope.*

Ramp – An inclined excavated passageway that connects the surface with an underground workplace or connects one underground workplace to another at a different elevation. Also called inclines or declines.

Receptor – Any person or environmental entity that is exposed to radiation, or a hazardous substance, or both. A receptor is usually an organism or a population, but it could also be an abiotic entity such as surface water or sediment.

Redox – A shorthand notation used to describe chemical reduction-oxidation reactions. Such reactions involve a change in the oxidation state of the atoms or molecules involved.

Retrieval – 1) The accessing and removal of waste containers from storage facilities for the purpose of transferring to another facility (e.g. a repository).
2) The accessing and removal of waste containers from either closed emplacement rooms (i.e., prior to decommissioning and closure of the repository), or from a sealed deep geologic repository (i.e., after the decommissioning and closure of the underground excavations).

Retrievability – The ability to remove waste packages from where they have been emplaced. Conditions may necessitate the use of different equipment and procedures from those used during emplacement of waste packages.

Retubing Waste – Radioactive waste produced from the fuel channel replacement (retubing) program i.e., pressure tubes, calandria tubes, calandria tube inserts, end fittings, yokes and studs.

Risk – A multi-attribute quantity expressing hazard, danger or chance of harmful or injurious consequences associated with actual or potential exposures. It relates to quantities such as the probability that specific deleterious consequences may arise and the magnitude and character of such consequences.

Rock Mass – An assemblage of blocks or layers of rock material bounded by discontinuities in which groundwater may be present.

Rock Quality Designation (RQD) – The cumulative length of drilled core pieces longer than 100 mm in a run, divided by the total length of the run, expressed as a percentage. Mechanical breaks caused by the drilling process or extracting the core from the core barrel are ignored, but lost or missing core is included in the total core-run length.

Safety Analysis – A calculation performed, with or without the assistance of computer software, to address a specific safety issue or as part of a safety assessment.

Safety Assessment – The process of systematically analyzing the hazards associated with the facility, and the ability of the site and design to provide the safety functions and meet technical requirements.

Safety Case – An integration of arguments and evidence that describe, quantify and substantiate the safety, and the level of confidence in the safety, of the geological disposal facility.

Safety Functions – The functions that the DGR must perform to ensure that the safety objective is achieved. These functions are *Isolation* and *Containment.*

Safety Indicator – A quantity used in safety assessments as a measure of the impact of a source, or of the performance of protection and safety provisions.

Safety Objective – The safety objective of the DGR is to prevent unreasonable risk to the health and safety of the public and the workers, and the environment.

Safety Report – A key licensing document which provides an overview of the facility design and operations, summarizes the integrated results of individual safety assessments, and demonstrates that a facility can be constructed, operated, or continue to be operated, without undue risk to health and safety of the workers and the public, and the environment.

Preliminary Safety Report (PSR) is the Safety Report submitted to CNSC in support of an application for a Site Preparation/Construction Licence.

Final Safety Report (FSR) is the Safety Report submitted to CNSC in support of an application for a Licence to Operate.

Saline Water – Water with a salinity between 10 to 100 g/L total dissolved solids. See also *Brackish Water* and *Brine*.

Sandstone – A medium-grained *clastic* sedimentary rock composed of abundant sand size particles with or without a fine-grained matrix (clay or silt) and cemented (commonly silica, iron oxide or calcium carbonate), the consolidated equivalent of sand. May be deposited by water or wind.

Saturated – A state of being completely wet, or in which the rock mass has absorbed and is retaining the greatest possible amount of fluid and can hold no more.

Scenarios – A postulated or assumed set of conditions or events. They are most commonly used in analysis or assessment to represent possible future conditions or events to be modelled, such as the possible future evolution of a repository and its surroundings.

Sealing (Fracture) – A reduction of fracture permeability by any hydromechanical, hydrochemical or biochemical process.

Sealing System – A low-permeability system, typically comprising clay and/or cementitious materials, placed to fill and seal rooms, tunnels, shafts and/or boreholes when they are no longer needed, in order to inhibit groundwater movement and contaminant transport.

Sedimentary Basin – A low area in the earth's crust in which sediments have accumulated over geologic time and subsequently transformed into sedimentary rock, such as the *Michigan Basin* or the *Appalachian Basin*.

Sedimentary Rock – A layered rock made of compacted and cemented sediments such as fragments of other rocks, minerals and/or organic remains (fossils), or precipitated out of solution. *Limestone*, *dolostone*, *shale* and *sandstone* are examples.

Seismicity – The frequency or magnitude of earthquake activity in a given area. See also *microseismicity*.

Seismic Reflection – A surface geophysical method recording seismic waves reflected from geologic strata, giving an estimate of their depth and thickness.

Seismograph – An instrument that detects, magnifies, and records vibrations of the Earth, either earthquake or those generated for applied seismology purposes. Also called a seismometer.

Sensitivity Analysis – A quantitative examination of how the behaviour of a simulated system (e.g., a computer model) varies with change, usually in the values of its parameters.

Shaft – A vertical or near-vertical excavated passageway that connects the surface with an underground workplace or connects two or more underground workplaces at different elevations.

Shale – A fine-grained detrital sedimentary rock, formed by the compaction and cementation of clay, silt, or mud. It may have a fine laminated structure which gives it a fissility along which the rock splits readily.

Shear Strength - The capacity to resist deformation resulting from stresses that cause contiguous parts of a body to slide relatively to each other in a direction parallel to their plane of contact.

Silurian – The third period of the *Paleozoic* Era extending from 443 to 417 million years ago, also refers to rocks formed, or sediments laid down, during this period (eg., Silurian evaporites).

Slickenside – Term to denote lineated fault surfaces, which also may consist of grooves and/or fibrous minerals. The general definition refers to a rock surface that has been scratched or polished by the effects of friction during structural changes. The term can also refer to changes in the appearance of swelling clays that have been subject to large changes in water content, and to diagenetic features formed as a result of differential compaction of layered sediments.

Solute – A substance that is dissolved in another (e.g. dissolving salt in water: salt is the solute, water is the solvent, and the result is a saline solution).

Specific storage – The volume of water that a rock mass (or aquifer) releases, per unit volume of rock mass, per unit decline in pressure head, while remaining fully saturated. Essentially, the volume of water that a confined unit (or aquifer) will release due to a given change in pressure head.

Stakeholder – Any person or organization that has an interest in a particular aspect of the project.

Stored Volume (Waste) – (also As-stored waste volume) The external volume of the storage container in which the waste is currently stored. This volume does not include overpacks or concrete shields which may be required for repository emplacement. See also *Net Volume* and *Emplaced Volume*.

Straddle Packers – A straddle packer is a system of two packers separated by a fixed length into which fluid is injected, after packer inflation, to test the hydraulic properties of the bedrock in a borehole.

Strain – To alter the relations between the parts of a structure or shape by applying an external force.

Stratigraphy – The study of the age relation of rock strata, including the original succession (order of emplacement), form, distribution, composition, fossil content, geophysical and geochemical properties, and the environment of origin and geologic history, of a rock mass. The science primarily involves the description of rock bodies, and their organization into distinctive, mappable units based on their properties and features.

Strength – The ability to withstand differential stress, expressed in the units of stress. See also *stress*.

Stress – In a solid, the force per unit area, acting on any surface within it.

Strike – The direction or trend taken by a structural surface as it intersects the horizontal; measured with respect to the horizontal plane.

Stylolite – A surface or contact, usually in carbonate rocks, marked by an irregular and interlocking penetration of the two sides: the columns, teeth, and pits on one side, fit into their counterparts on the other side. Stylolites resemble a suture, or 'seam', in the rock, and the 'seams' are usually parallel to bedding surfaces and consist of insoluble rock constituents (clay, iron oxides).

Subsurface characterization – All activities carried out in the shafts, tunnels and rooms of the repository and via deep boreholes in the vicinity of the repository for the purpose of gathering geoscience data for the development of a repository design and the associated safety case. Examples of characterization activities are mapping and testing of rock formations during underground excavation, monitoring of groundwater pressures and chemistry via boreholes and within the repository, and in situ testing to measure rock properties.

Technical Computing Software – Software used by technical specialists for design, analysis or simulation of engineered systems. Examples include finite element stress analysis software, waste site safety analysis software, radiation shielding software, and waste inventory database software.

Tectonic – Said of or pertaining to the forces involved in, or the resulting structures or features of, *tectonics.*

Tectonics – A branch of geology dealing with a broad architecture of the outer part of the earth, that is, the regional assembling of structural or deformational features, a study of their mutual relations, origin, and historical evolution.

Tensile Strength - The capacity of a material to resist a normal stress that tends to pull apart the material on the opposite sides of the plane on which it acts.

Thermal Maturity – A measure of the state of a rock in terms of hydrocarbon generation. The sedimentary rock type, physical environment, and temperature of the environment will determine thermal maturity. Rocks that have been exposed to high temperatures, resulting in a different distribution of the various compounds (e.g. the alteration of organic molecules and petroleum to hydrocarbons - oil and/or gas) are defined as mature, and the extent of such alteration determines the level of maturity.

Tortuosity (τ) – A geometric factor that accounts for the effective transport path length for solute transport within a porous medium (L_e) compared to the shortest straight-line transport path length (L) between two points, as follows: $\tau = (L_e / L)^2$. Note that $\tau \geq 1$.

Transfer Fault – A strike-slip fault that links two segments of a rift that are offset relative to one another.

Transmissivity – The product of *hydraulic conductivity* and aquifer thickness; a measure of a volume of water to move through an aquifer. Transmissivity is a measure of the subsurface's ability to transmit groundwater through its entire saturated thickness and affects the potential yield of wells.

Ultramafic – Term to describe an igneous rock composed of > 90% *mafic* minerals.

Uncertainty Analysis – An analysis of the amount of variation in the results of assessments or analyses due to incomplete knowledge about the current and future states of a system.

Unconformity – An erosion surface separating two rock masses or strata of different ages, indicating that sediment deposition was not continuous. An unconformity refers to any substantial break in the geologic record, where a rock unit is overlain by another that is not the next in the stratigraphic succession.

Underground Service Areas – Any excavations within the deep geologic repository that provide the space for the infrastructure to characterize, demonstrate, construct, operate, monitor and decommission a deep geologic repository. Service areas include all excavations in a deep geologic repository that are not classified as tunnels, shafts, ramps, emplacement rooms or boreholes.

Uniaxial Compressive Strength - Represents the capacity of a material to withstand applied mechanical compressive forces; also is that value of uniaxial compressive stress reached when the material fails completely. The strength is usually expressed in units of stress.

Validation (Model) – The process of building confidence that a model adequately represents a real system for a specific purpose.

Valued Ecosystem Component (VEC) – VECs are features of the environment selected to be a focus of the environmental assessment because of their ecological, social, or economic value, and their potential vulnerability to the effects of the DGR project.

Verification (Model) – The process of determining whether a computer model correctly implements the intended conceptual or mathematical model.

Waste Acceptance Criteria – Formal criteria which define the qualities of waste packages (including the waste) that are accepted for emplacement in the repository.

Waste Arisings – The amount of waste produced at the stations, prior to any *waste conditioning*.

Waste Characterization – Activities to define the physical, chemical and radiological characteristics of the radioactive waste.

Waste Conditioning – Those operations that produce a waste package suitable for handling, transport, storage and/or disposal. Conditioning may include the conversion of the waste to a solid waste form, enclosure of the waste in containers, and, if necessary, providing an overpack.

Waste Package – The waste material, the container, and any external barriers (e.g. shielding material), as prepared in accordance with requirements for handling, transfer and emplacement in the repository. It is a discrete unit that can be individually identified and handled at the repository facility. See also *Waste Packaging*.

Waste Packaging – The container and any external barriers (e.g., *overpack*, shielding material), used for handling, transfer and disposal of the waste. It does not include the waste itself. See also *Waste Package*.

Water Content – Also known as volumetric water content. Identical to *water loss porosity* for a fully saturated rock sample.

Water Loss Porosity – Refers to the ratio of the water-filled pore volume in a rock sample with respect to the total volume of the rock sample, and is typically measured during the heating and drying of the sample.

Water table (groundwater table) – The top water surface of an unconfined aquifer at atmospheric pressure.

Westbay Casing – A multi-level modular groundwater monitoring, sampling and testing system, consisting of multiple inflatable packers, valved ports, blank pipe segments and couplings to seal and provide access to multiple monitoring zones in one borehole. Monitoring, sampling and testing are carried out with the use of several available types of wireline operated probes.

Wetting phase – The preference of a solid to contact one liquid or gas, known as the wetting phase, rather than another. The wetting phase will tend to spread on the solid surface and a porous solid will tend to imbibe the wetting phase, in both cases displacing the non-wetting phase. Rocks can be water-wet, oil-wet or intermediate-wet. The intermediate state between water-wet and oil-wet can be caused by a mixed-wet system, in which some surfaces or grains are water-wet and others are oil-wet, or a neutral-wet system, in which the surfaces are not strongly wet by either water or oil. Both water and oil wet most materials in preference to gas, but gas can wet sulphur, graphite and coal.

WPRB (Waste Package Receiving Building) – The building at the DGR surface where waste packages arrive for transfer underground.

WVRB (Waste Volume Reduction Building) – The building at WWMF containing waste volume reduction equipment.

WWMF (Western Waste Management Facility) – The centralized processing and storage facility on the Bruce nuclear site for OPG's L&ILW and for the dry storage of used fuel from Bruce nuclear generating stations.

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17. ENGINEERING DRAWINGS

Section First Mentioned	Document Number	Title
6.2	11T1076-C-SK-1	Operations Layout: General Arrangement
6.2.3	H333000-WP404-10-042-0001	Waste Rock Management Area: Site Grading and Drainage
6.2.3	H333000-WP404-10-042-0003	Waste Rock Management Area: Base Case
9.4.3	H333000-WP406-20-042-0003	Main Shaft: Headframe – Sinking Condition General Arrangement
9.4.3	H333000-WP406-20-042-0008	Ventilation Shaft: Headframe – Sinking Condition General Arrangement
9.4.5.1	H333000-WP405-20-035-0001	Shaft Hoisting Systems: Dia. 6500 – Main Shaft Sinking Stage and Bucket Proposed Layout
9.4.5.2	H333000-WP410-20-030-0002	Ventilation System: Typical Shaft Sinking Ventilation Process Flow Diagram

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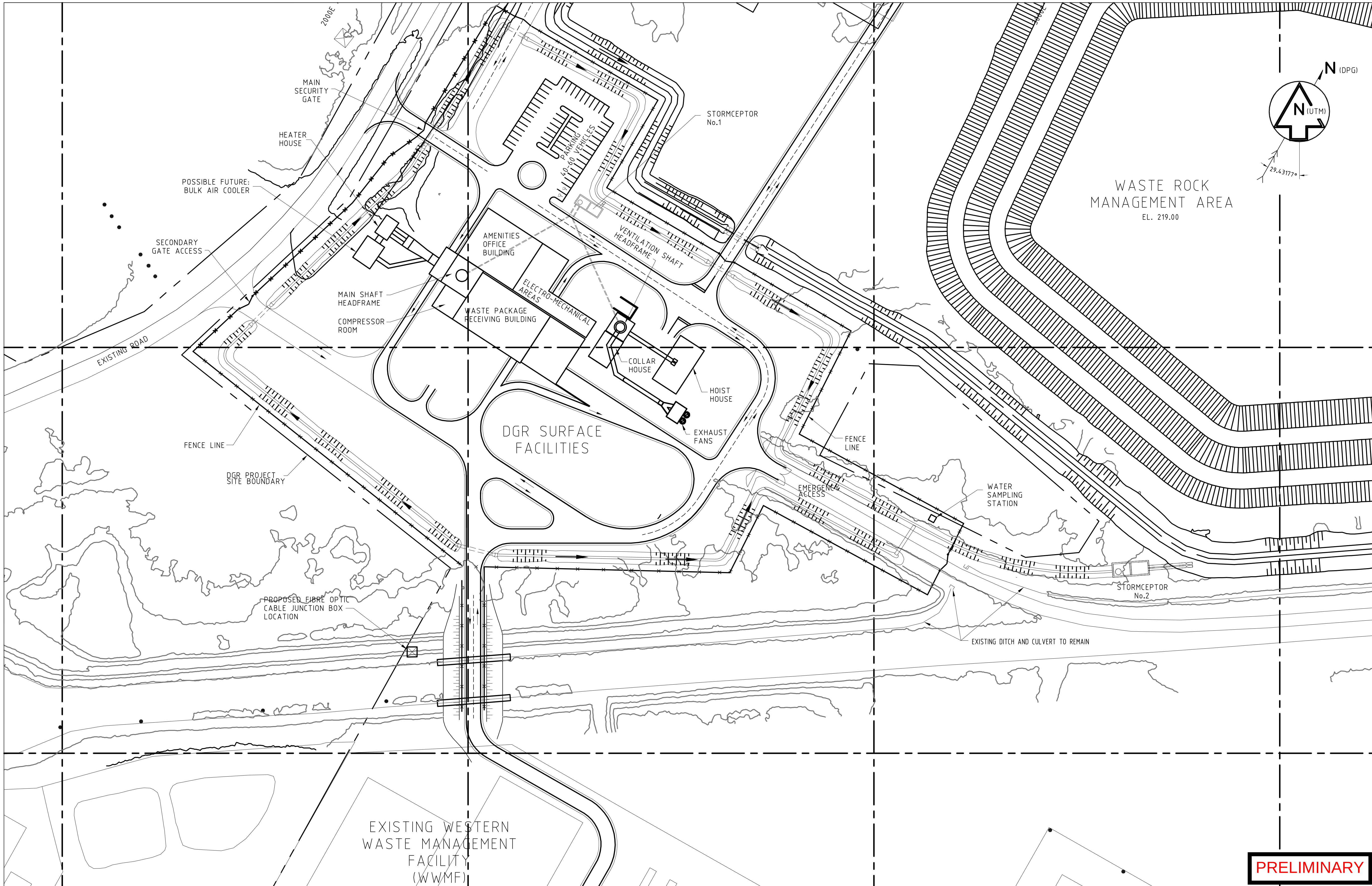

PRELIMINARY

OPG DWG. NO.	DWG. NO.	REFERENCE DRAWINGS	REV	REVISIONS	DATE (MM/DD/YY)	BY	CHK'D	REV'D	APP'D

ENGINEER AND PERMIT STAMPS

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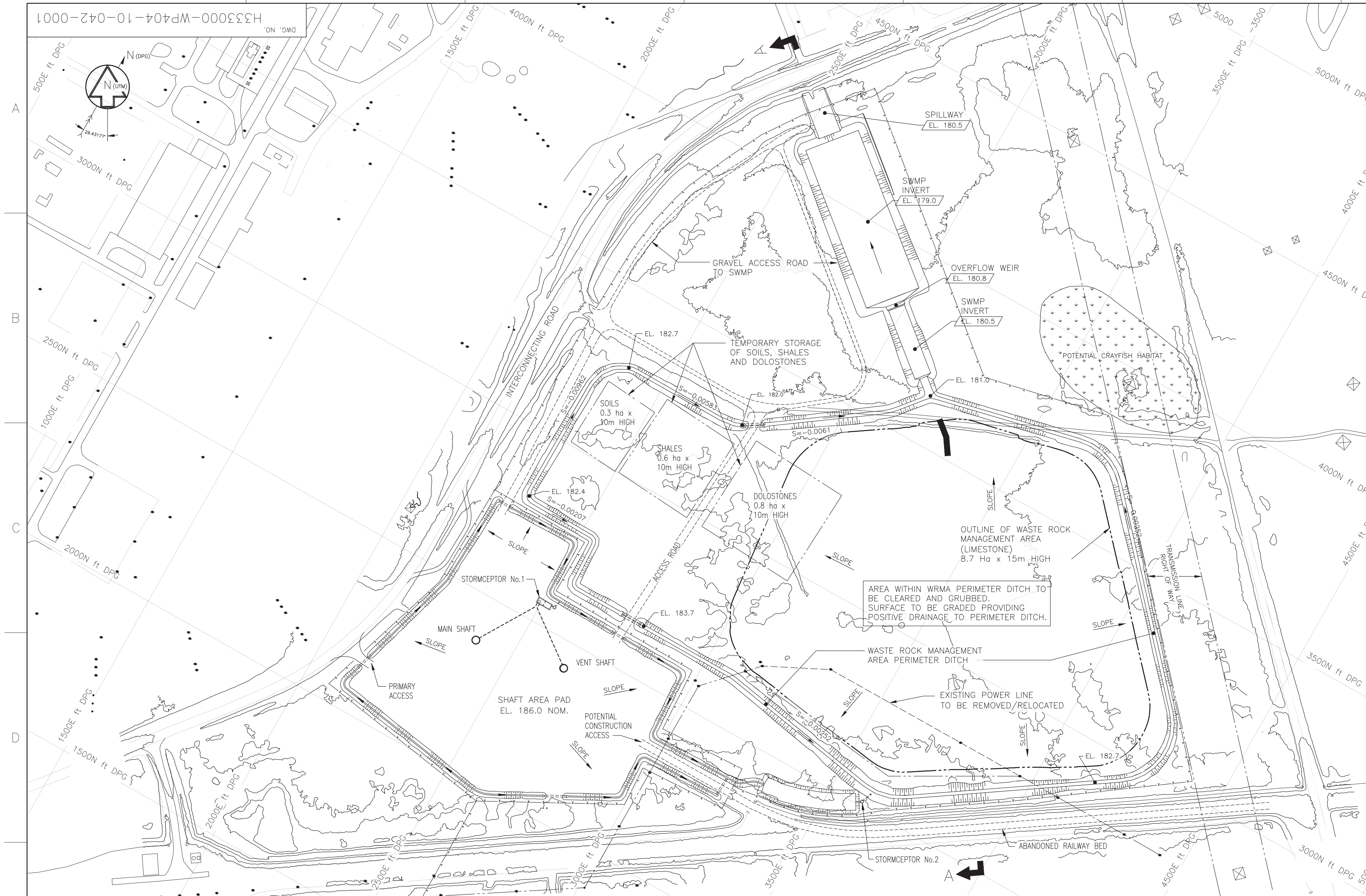
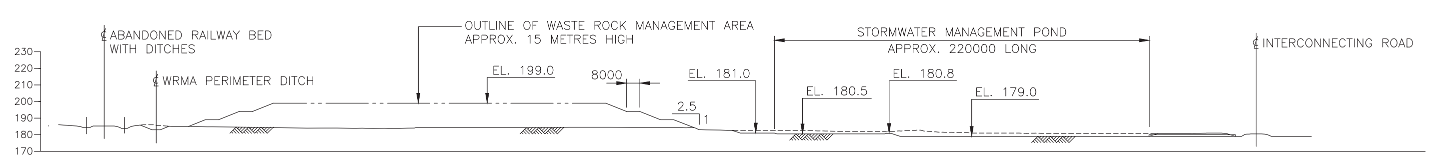
resources & energy

PROJECT OPG's DEEP GEOLOGIC REPOSITORY ROJECT FOR LOW & INTERMEDIATE LEVEL WASTE
SURFACE FACILITIES PACKAGE 1

TITLE OPERATIONS LAYOUT GENERAL ARRANGEMENT

JOB NO.	SCALE	DWG. NO.
11T1076	NTS	11T1076-C-SK1



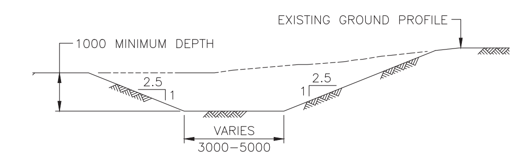


DIMENSIONS SHOWN IN MILLIMETRES
ELEVATIONS SHOWN IN METRES
COORDINATES SHOWN IN FEET

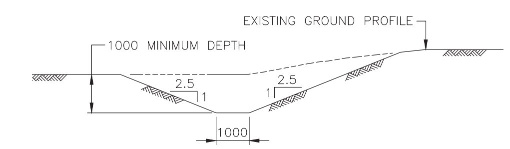
NOTES:

1. HORIZONTAL DATUM: COORDINATE SYSTEM SHOWN IS BASED ON 1959 DOUGLAS POINT GRID (DPG). GRID SHOWN IN FEET AND CAN BE CONVERTED TO METRES BY MULTIPLYING BY 0.3048.
   COORDINATES CAN BE TRANSFORMED FROM DPG TO UTM NAD83 ZONE 17 (OR REVERSE) USING THE FOLLOWING EQUATIONS:
   $EASTING_{UTM} = 452408.341 + (0.8705935 \times EASTING_{DPG}) + (0.4911905 \times NORTHING_{DPG})$
   $NORTHING_{UTM} = 4907963.328 + (0.8705935 \times NORTHING_{DPG}) - (0.4911905 \times EASTING_{DPG})$
   DPG COORDINATES MUST BE CONVERTED TO METRES BEFORE USING THIS TRANSFORMATION.
   EQUATIONS PROVIDED BY 4DM INC. IN "DGR PROJECT COORDINATE TRANSFORMATION FROM DPG TO UTM WPD–SC34 WORK REQUEST 3", FEBRUARY 5, 2008.

2. VERTICAL DATUM: STATION 0011972U188
   ELEVATION 183.330 (CGVD28).
   LOCATION: BRUCE NUCLEAR POWER DEVELOPMENT (AUXILIARY STEAM PLANT) CONCRETE CHIMNEY, TABLET IN EAST FACE OF CHIMNEY, 6.7m CLOCKWISE OF A STEEL DOOR, 4.5m COUNTER–CLOCKWISE OF A STEEL LADDER AND 30cm ABOVE CONCRETE BASE.



WRMA PERIMETER DITCH
TYPICAL SECTION
1:100



DGR SURFACE FACILITIES PERIMETER DITCH
TYPICAL SECTION
1:100

1:100 0 2 4 6 8 10 METRES
1:2000 0 40 80 120 160 200 METRES

PRELIMINARY

DRAWING NO.	DRAWING TITLE
REFERENCE DRAWINGS	

NO.	DESCRIPTION	CHK'D	APP'D	DATE
REVISIONS				

REV.	ISSUE FOR	AUTH. BY	DATE
ISSUE AUTHORIZATION			

DESIGNED BY	DRAWN BY
DATE	DATE
CHECKED BY	DISCIP. ENGR.
DATE	DATE
PROJ. DES. COORD.	PROJ. ENGR.
DATE	DATE
PROJ. MGR.	
DATE	

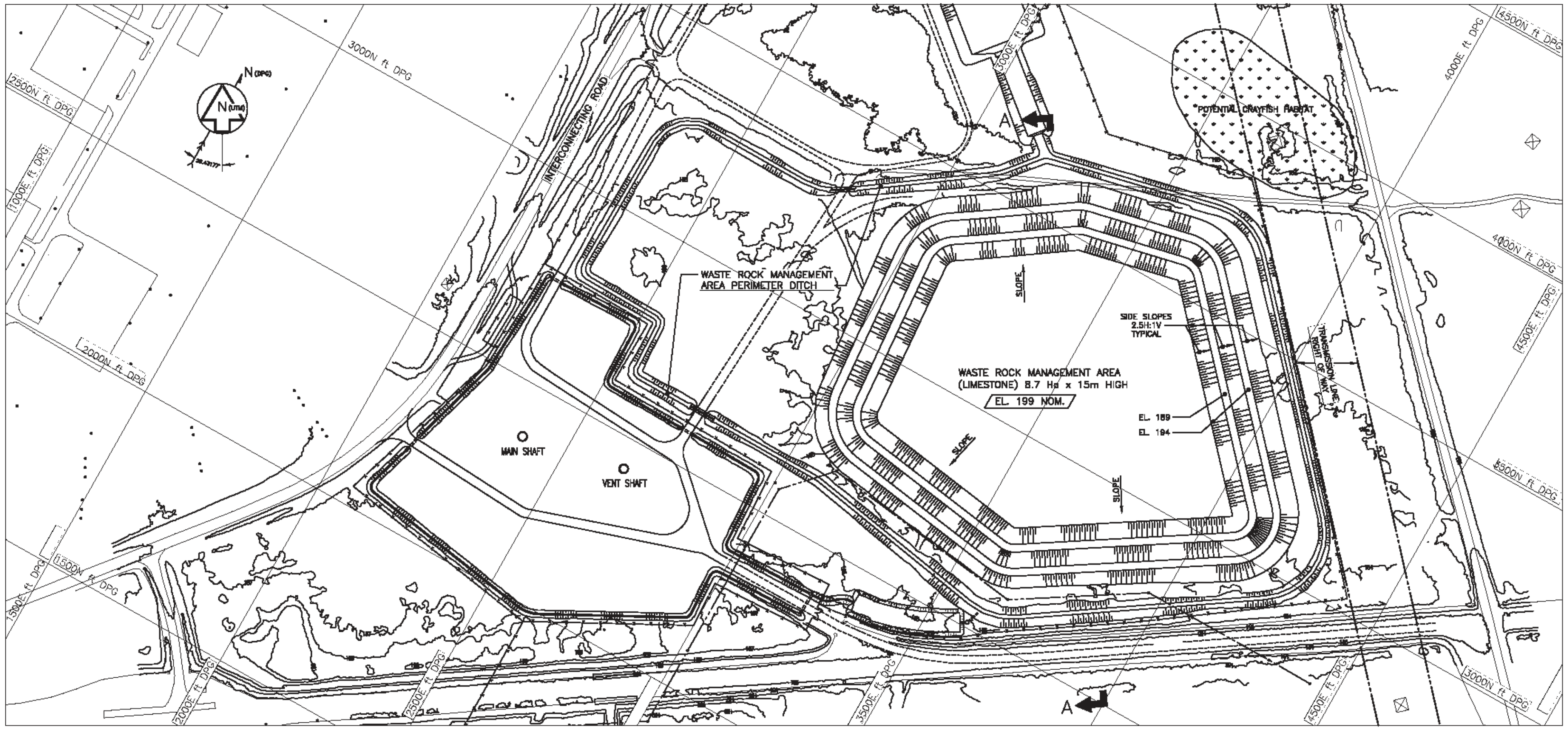
NUCLEAR WASTE MANAGEMENT ORGANIZATION
DEEP GEOLOGIC REPOSITORY PROJECT

WASTE ROCK MANAGEMENT AREA

SITE GRADING AND DRAINAGE

SCALE	DWG. NO.	REV.
OR AS NOTED	H333000–WP404–10–042–0001	00

DWG. NO. H333000-WP404-10-042-0003



GENERAL ARRANGEMENT
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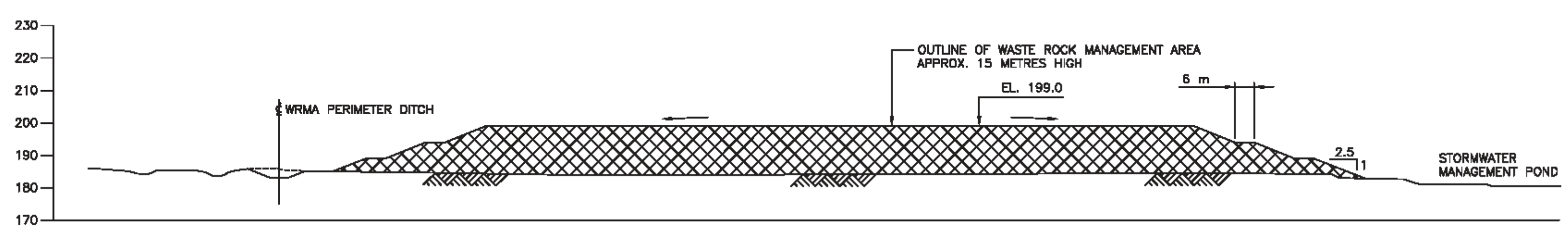


SECTION A–A

NOTE:

1. HORIZONTAL DATUM: COORDINATE SYSTEM SHOWN IS BASED ON 1959 DOUGLAS POINT GRID (DPG). GRID SHOWN IN FEET AND CAN BE CONVERTED TO METRES BY MULITILYING BY 0.3048.
2. VERTICAL DATUM: STATION 0011972U188 ELEVATION 183.330 (CGVD28). LOCATION: BRUCE NUCLEAR POWER DEVELOPMENT (AUXILIARY STEAM PLANT) CONCRETE CHIMNEY, TABLET IN EAST FACE OF CHIMNEY, 6.7m CLOCKWISE OF A STEEL DOOR, 4.5m COUNTER–CLOCKWISE OF A STEEL LADDER AND 30cm ABOVE CONCRETE BASE.
3. CONTOURS PRODUCED BY HATCH LTD. FROM A DIGITAL ELEVATION MODEL (DEM). THE DEM WAS PROVIDED AND PREPARED BY 4DM INC. BASED ON A LIDAR SURVEY FLOWN BY TERRAPOINT CANADA, JULY 19, 2007.

1:1000 0 20 40 60 80 100 METRES
1:2000 0 40 80 120 160 200 METRES

PRELIMINARY

DRAWING NO.	DRAWING TITLE
REFERENCE DRAWINGS	

NO.	DESCRIPTION	CHK'D	APP'D	DATE
REVISIONS				

REV.	ISSUE FOR	AUTH. BY	DATE
ISSUE AUTHORIZATION			

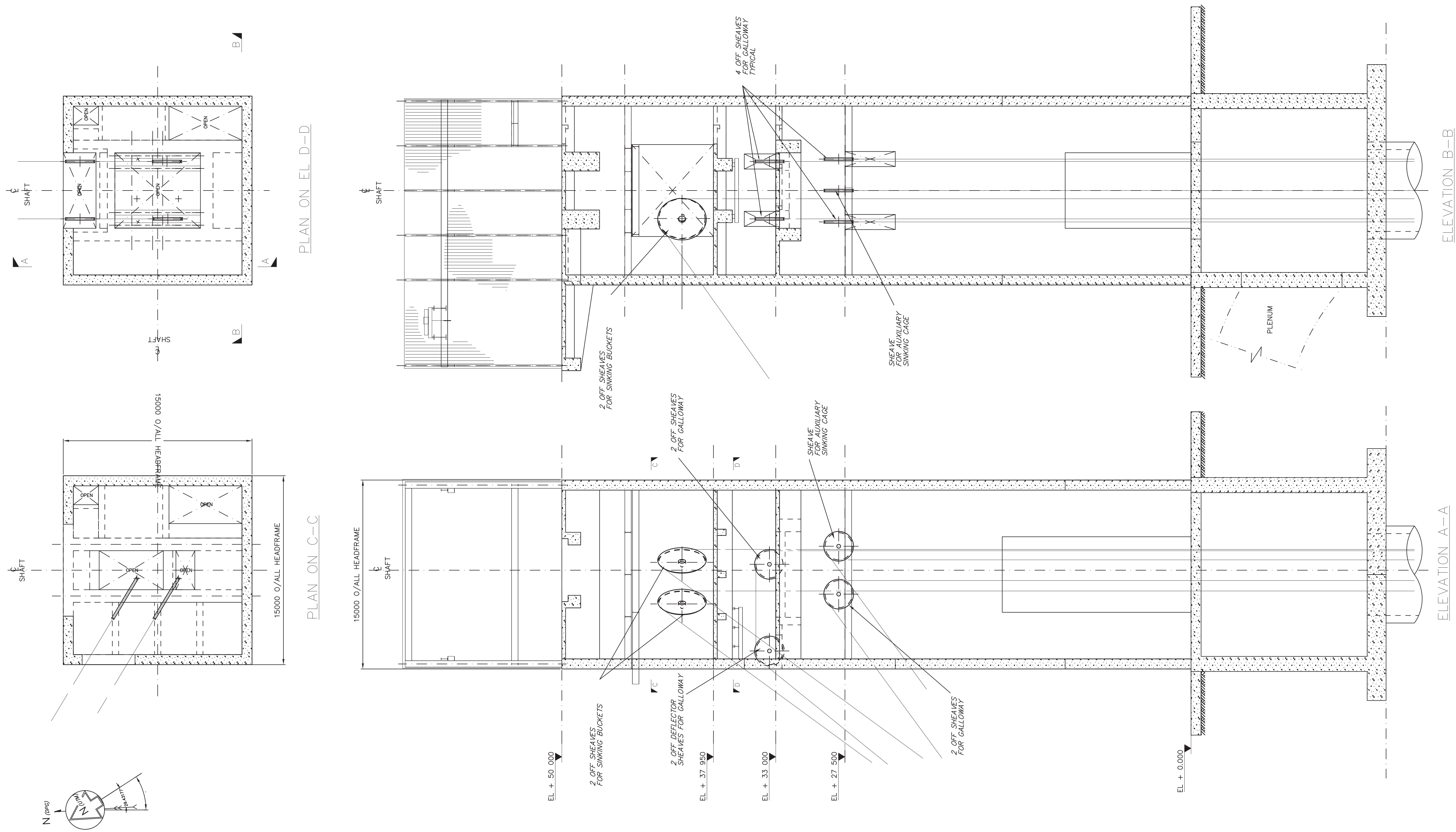
DESIGNED BY / DATE	DRAWN BY / DATE
CHECKED BY / DATE	DISCIP. ENGR. / DATE
PROJ. DES. COORD. / DATE	PROJ. ENGR. / DATE
PROJ. MGR. / DATE	

NUCLEAR WASTE MANAGEMENT ORGANIZATION
DEEP GEOLOGIC REPOSITORY PROJECT

WASTE ROCK MANAGEMENT AREA

BASE CASE

SCALE	DWG. NO.	REV.
OR AS NOTED	H333000-WP404-10-042-0003	01



PRELIMINARY

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-	-								

ENGINEERING AND PERMIT STAMPS (As Required)

SCALE

PROJECT No.

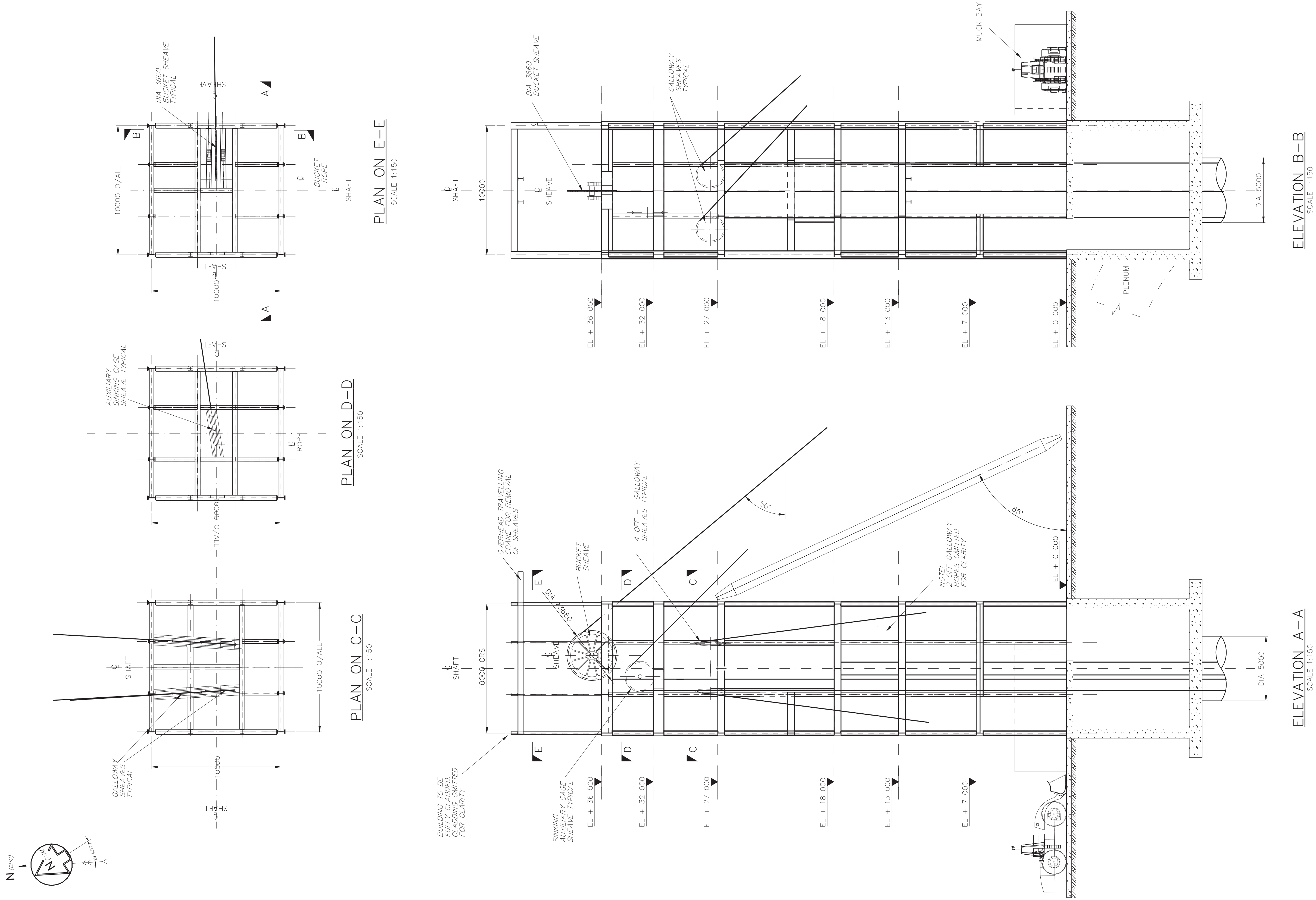
NWMO WBS NO.

OPG DWG NO. REV

OPG'S DEEP GEOLOGIC REPOSITORY FOR LOW & INTERMEDIATE LEVEL WASTE

MAIN SHAFT
HEADFRAME - SINKING CONDITION GENERAL ARRRANGEMENT

DWG No. H333000-WP406-20-042-0003 REV



PRELIMINARY

OPG DWG NO	REF. DRAWING NO	REFERENCE DRAWING TITLE	REVISION DESCRIPTION	DESIGNED	CHECKED	REVIEWED	APPROVED	REV	DATE

ENGINEERING AND PERMIT STAMPS (As Required)

SCALE

PROJECT No.

NWMO WBS NO.

OPG DWG NO.

REV

OPG'S DEEP GEOLOGIC REPOSITORY FOR LOW & INTERMEDIATE LEVEL WASTE

VENTILATION SHAFT
HEADFRAME - SINKING CONDITION GENERAL ARRRANGEMENT

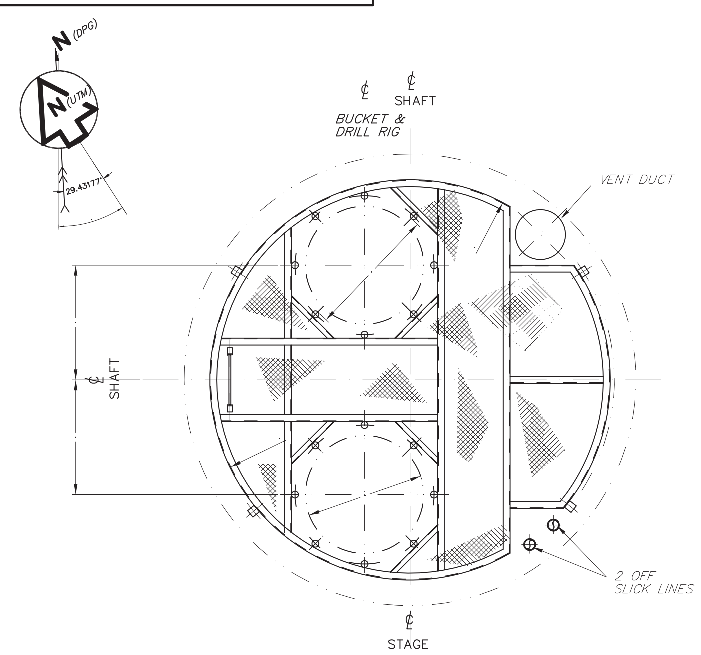
DWG No.

H333000-WP406-20-042-0008

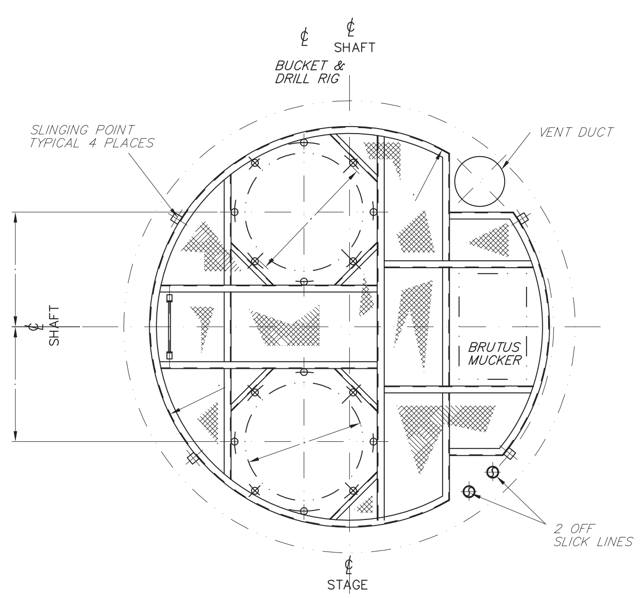
WPC-A1 VERSION

PLOT DATE & TIME 31/12/2010 10:41:18 AM
SAVE DATE & TIME 6/1/2011 11:10:31 AM
USER NAME
LOCATION C:\DOCUMENTS AND SETTINGS\GLTA\DESKTOP\CONC HEADFRAME.DWG

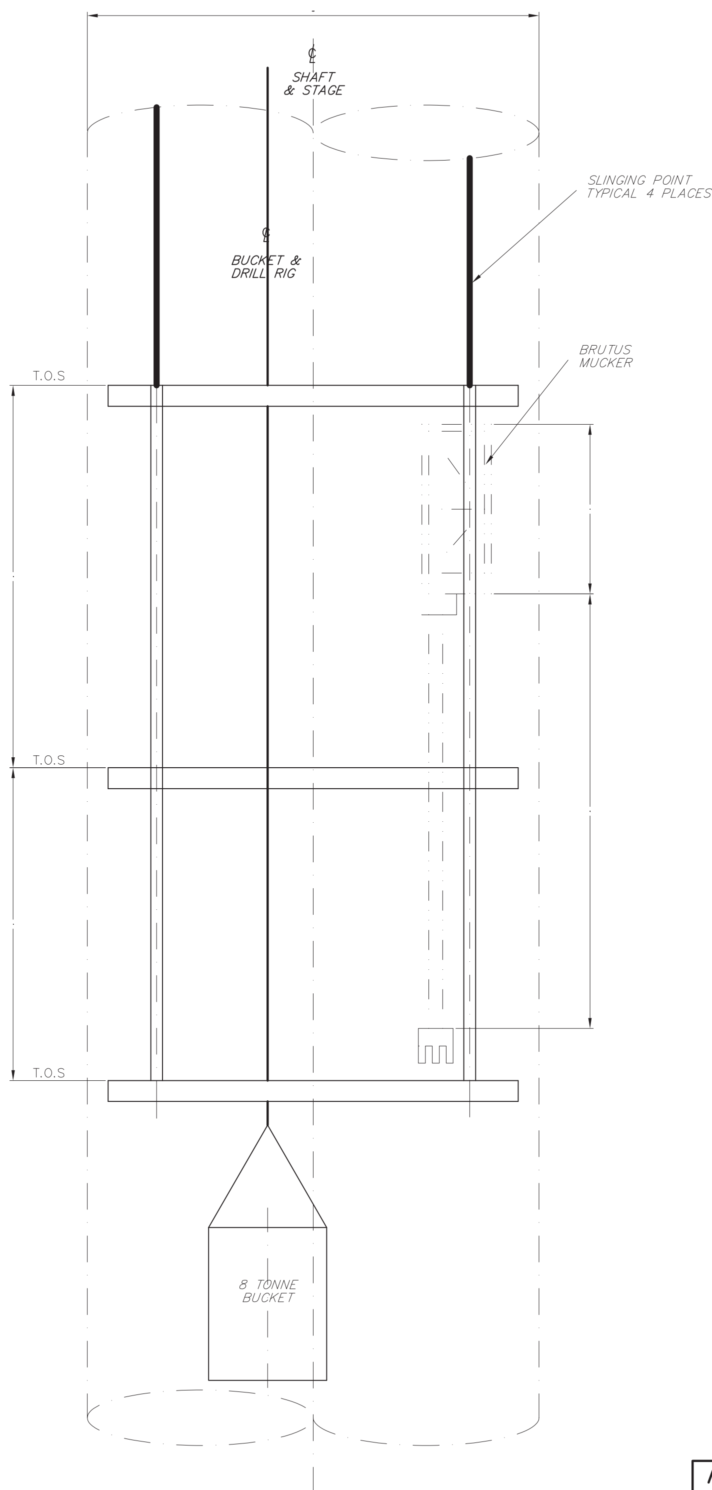
H333000-WP405-20-035-0001
DWG. NO.



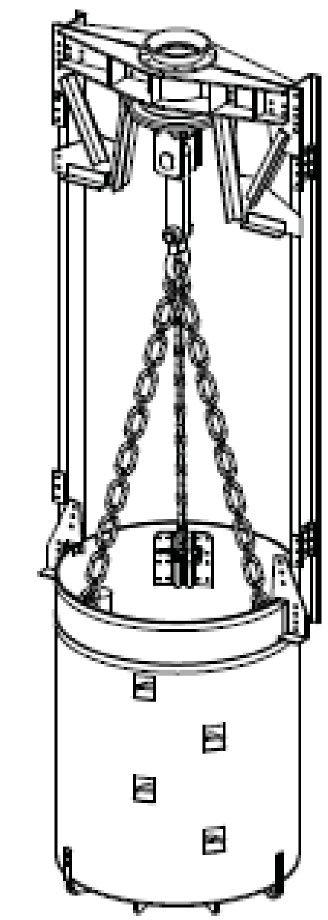
PLAN ON TOP DECK
SCALE 1:50



PLAN ON MIDDLE & BOTTOM DECKS
SCALE 1:50



ELEVATION ON SINKING STAGE
SCALE 1:50



LAYOUT OF SINKING BUCKET
SCALE 1:25

NOTE!
DUE TO THE LIMITED FLOOR AREA ON THE GALLOWAY, IT HAS BEEN ASSUMED THAT THE JUMBO DRILL RIG WILL BE REMOVED TO SURFACE VIA THE BUCKET OPENING WHEN NOT IN USE.

PRELIMINARY

NOTE:
DRAWING TO BE USED FOR PREIMINARY DESIGN ONLY
NOT TO BE USED FOR CONSTRUCTION

DRAWING NO.	DRAWING TITLE
REFERENCE DRAWINGS	

NO.	DESCRIPTION	CHK'D	APP'D	DATE
REVISIONS				

REV.	ISSUE FOR	AUTH. BY	DATE
ISSUE AUTHORIZATION			

DESIGNED BY / DATE — DRAWN BY / DATE
CHECKED BY / DATE — DISCIP. ENGR. / DATE
PROJ. DES. COORD. / DATE — PROJ. ENGR. / DATE
PROJ. MGR. / DATE

NUCLEAR WASTE MANAGEMENT ORGANIZATION
DEEP GEOLOGIC REPOSITORY PROJECT

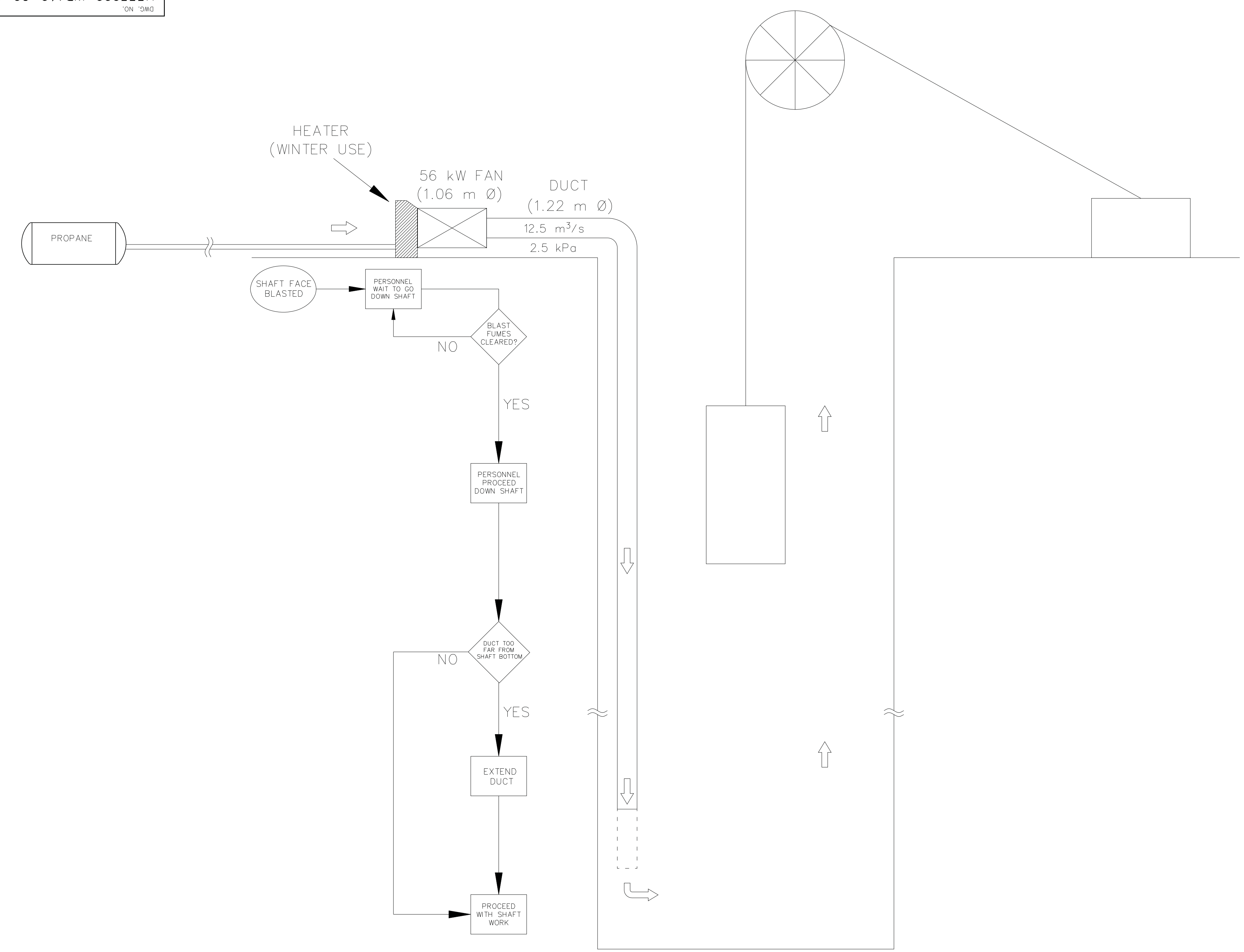
SHAFT HOISTING SYSTEMS

DIA 6500 – MAIN SHAFT
SINKING STAGE AND BUCKET
PROPOSED LAYOUT

SCALE 1 : 50 OR AS NOTED
DWG. NO. H333000-WP405-20-035-0001
REV.

$FILE$ $DATE$ $TIME$ $USERNAME$

H333000-WP410-20-030-0002
DWG. NO.



PRELIMINARY

DRAWING NO.	DRAWING TITLE

REFERENCE DRAWINGS

NO.	DESCRIPTION	CHK'D	APP'D	DATE

REVISIONS

NOT FOR CONSTRUCTION
This drawing has been reviewed for the purposes of DGR Preliminary Engineering only and must not be used for other purposes.
Work Package Owner
Digitally signed by Kingsley Hortin
Area Lead
Digitally signed by G.R. Davidge Date: 2010.01.12 12:34:39 -05'00'
Engineering Manager
Sherwin Aarons 2010.01.12 14:28:47 -05'00'
Project Manager
Sherwin Aarons 2010.01.12 14:28:53 -05'00'
Accepted by Owner
Digitally signed by Derek Wilson DN: cn=Derek Wilson, o=NWMO, ou, email=dwilson@nwmo.ca, c=CA Date: 2010.01.19 09:37:48 -05'00'

REV.	ISSUE FOR	AUTH. BY	DATE
00	APPROVED FOR USE	GRD SAA	2010-01-11

ISSUE AUTHORIZATION

HATCH™

DESIGNED BY K. HORTIN DATE	DRAWN BY AKB DATE 2009-11-11
CHECKED BY DATE	DISCIP. ENGR. DATE
PROJ. DES. COORD. DATE	PROJ. ENGR. DATE
PROJ. MGR. M. DAWBORN DATE 2010-01-11	

NUCLEAR WASTE MANAGEMENT ORGANIZATION
DEEP GEOLOGIC REPOSITORY PROJECT

VENTILATION SYSTEM

TYPICAL SHAFT SINKING VENTILATION PROCESS FLOW DIAGRAM

SCALE N.T.S OR AS NOTED | DWG. NO. H333000-WP410-20-030-0002 | REV. 00

P:\NUCLEAR\333000\CAD\F\DGN\WP4-10\H333000-WP410-20-030-0002 Shaft Sinking Process Flow Diagram.dgn
1/11/2010 11:32:52 AM
bout59734

A. PRECLOSURE SAFETY ASSESSMENT ACCIDENT CALCULATIONS

A.1 Introduction

This appendix provides details of Preclosure Safety Assessment calculation results for each identified accident scenario (Section 7.5.1) for the radiological and non-radiological species of potential concern (Section 7.5.2). The methodology to calculate the accident consequence is given in Section 7.5.3.

The following sections discuss each identified accident assessed under the topics listed below.

- Scenario Title.
- Scenario Description – a sequence of events potentially leading to this scenario.
- Source Term – describes the scenario-specific parameters selected based on rationale provided in Section 7.5.3.1.
- Dispersion and Consequence – describes relevant equations used for the dispersion and consequence of airborne releases of radionuclide and non-radiological species for each accident assessment based on methodology described in Section 7.5.3.3 to Section 7.5.3.4; for non-fire scenarios, external radiation modelling assumptions are also described.
- Results – provides results based on consequence analysis for above ground scenarios (Section A.2) and underground scenarios (Section A.3) of calculated radionuclide doses and non-radiological species exposure to workers and the public.

In addition, the results for the ventilation system failure scenario are given in Section A.4. In the following tables, “N/A” means “not applicable”; “N/D” means “not developed”; and zero value means that the radiological or non-radiological species is either not present or in insignificant amount.

A sample calculation is given in Section A.5 for outdoor LLW fire scenario.

A.2 Above Ground Accident Consequence

A.2.1 Fire

A.2.1.1 Outdoor Waste Package Fire

<u>Scenario Description</u>

A vehicle accident or equipment failure (e.g., electric malfunction) in a transportation truck or forklift may ignite a fire in the equipment that may in turn propagate and set transported LLW packages or unshielded resin liners on fire.

Compacted waste (boxed) and non-processible waste (boxed and drummed) were identified as the combustible LLWs to be assessed. Unshielded resin liners with moderator IX resin were identified as the combustible ILWs to be assessed for this scenario.

<u>Source Term</u>

Parameters used for the calculation of the source term amounts (Equation 7-5) and release rates (Equation 7-6) due to this fire accident scenario are summarized in Table A-1 for the waste categories assessed.

Table A-1: Source Term Parameters - Outdoor Waste Package Fire

Selected Waste Category	# of Packages	DR	ARF	RF	LPF	T_{FD} (hr)
Box Compacted	8	0.5	0.001	1	1	5.0
Non-Processible Boxed	8	0.5	0.001	1	1	0.7
Non-Processible Drummed	8	0.5	0.001	1	1	0.7
Moderator Resin (Unshielded)	1	0.5	0.001	1	1	3.9

The maximum capacity of LLW packages on the truck were assumed to ignite and burn together in a confined fire state where the exposed burning surface was assumed to be from the top (lid) of the packages only.

Note that an ARF of 1 was used to calculate source terms for volatile elements C-14, H-3, mercury, and selenium.

Dispersion and Consequence

The concentrations of radionuclides and non-radiological species in air were calculated using Equation 7-11. The source term emission rate into air was estimated using Equation 7-19. The concentration of radionuclides and non-radiological species in the vicinity of the public was calculated with Equation 7-15 based on 1 hour of exposure time at the nearest Bruce nuclear site boundary.

Worker exposure through inhalation and immersion of radioactivity released to the air was assessed; significant additional external dose was deemed unlikely for the 5 minute worker exposure and, therefore, was not assessed for this accident.

Results

Potential Impact of Radionuclides

Table A-2 shows that total radionuclide doses to workers over a 5 minute period (through inhalation, immersion and external radiation) are less than the dose limit for workers (50 mSv) for any of the assessed waste categories. Similarly, Table A-3 shows that the total dose to the public (through inhalation and immersion) over a 1 hour exposure duration is much less than the 1 mSv public dose limit for any of the assessed waste categories.

Table A-2: Dose to Worker – Outdoor Waste Package Fire

Selected Waste Category	Inhalation (mSv)	Immersion (mSv)	External Radiation (mSv)	Total (mSv)
Box Compacted Waste	1.7E-01	6.2E-07	N/A	1.7E-01
Non-Processible Boxed	1.5E-01	8.0E-06	N/A	1.5E-01
Non-Processible Drummed	1.3E+00	1.3E-06	N/A	1.3E+00
Moderator Resin (Unshielded)	1.5E+00	8.4E-04	N/A	1.5E+00

Table A-3: Dose to Public - Outdoor Waste Package Fire

Selected Waste Category	Inhalation (mSv)	Immersion (mSv)	Total (mSv)
Box Compacted	1.8E-05	1.1E-10	1.8E-05
Non-Processible Boxed	1.5E-05	1.4E-09	1.5E-05
Non-Processible Drummed	1.4E-04	2.2E-10	1.4E-04
Moderator Resin (Unshielded)	1.5E-04	1.5E-07	1.5E-04

Potential Impact of Non-Radiological Species

Table A-4 and Table A-5 show the ratios of air concentration to worker IDLH and public PAC 1 criteria, respectively. The air concentrations are less than non-radiological criteria for workers and the public.

The potential impacts on workers for some species are close to, but below, criteria. However, the conservatisms in the analysis, and in particular in the air volume (see Section 7.5.3.3), should be noted, especially given that this scenario considers 8 LLW packages burning.

A.2.1.2 Indoor Waste Package Fire

Scenario Description

An external fire was assumed to occur due to causes such as an electric malfunction in a package handling forklift, or maintenance activities in the WPRB. If the fire was close enough and not extinguished, the fire could ignite a fire in combustible LLW or unshielded ILW resin liners.

A maximum of 24 LLW packages and 2 ILW packages was conservatively assumed to be temporarily staged in the WPRB at a time; therefore, a maximum of 24 LLW packages and 2 ILW packages may catch fire. Unshielded resin liners with moderator resin were identified as the combustible ILWs to be assessed for this scenario.

Source Term

Parameters used for the calculation of the source term amounts (Equation 7-5) and release rates (Equation 7-6) due to this fire accident scenario are summarized in Table A-6 for the waste categories assessed. Only the combustible fraction of the respective waste category was assumed to burn.

The maximum number of LLW packages staged in the WPRB was assumed to ignite and burn together in a confined fire state. 24 packages are to be placed in 2 layers of 12 packages. Therefore, the exposed burning surface in a confined fire would be from the top lids of the 12 packages located on the top layer.

Table A-4: Ratios of Air Concentration to Worker IDLH - Outdoor Waste Package Fire

Non-Radiological Species	Box Compacted	Non-Processible Boxed	Non-Processible Drummed	Moderator Resin (Unshielded)
Antimony	2.2E-04	1.7E-04	7.8E-05	2.4E-07
Arsenic	2.8E-04	2.2E-04	9.8E-05	1.5E-06
Barium	7.1E-04	5.4E-04	2.4E-04	3.2E-06
Beryllium	0.0E+00	4.9E-04	2.2E-04	3.4E-06
Cadmium	8.3E-03	6.4E-03	2.9E-03	2.2E-05
Chromium	3.2E-03	2.8E-01	1.3E-01	1.4E-05
Cobalt	7.0E-05	5.2E-05	2.4E-05	8.3E-07
Copper	2.6E-03	3.4E-01	1.5E-01	1.2E-04
Lead	1.5E-03	2.7E-01	1.2E-01	1.5E-05
Manganese	8.2E-04	9.0E-04	4.1E-04	8.3E-07
Mercury	3.2E-02	3.9E-02	1.8E-02	1.0E-04
Nickel	2.3E-03	4.5E-02	2.0E-02	7.7E-03
Selenium	5.3E-01	4.1E-01	1.9E-01	1.4E-02
Strontium	N/D	N/D	N/D	N/D
Uranium	0.0E+00	6.0E-04	2.7E-04	1.5E-07
Zinc	9.5E-04	1.6E-03	7.2E-04	1.4E-05
Zirconium	1.2E-04	1.0E-04	4.6E-05	1.5E-07
Asbestos	0.0E+00	N/D	N/D	0.0E+00
Dioxins & Furans	0.0E+00	0.0E+00	0.0E+00	0.0E+00

Table A-5: Ratios of Air Concentration to Public PAC 1 - Outdoor Waste Package Fire

Non-Radiological Species	Box Compacted	Non-Processible Boxed	Non-Processible Drummed	Moderator Resin (Unshielded)
Antimony	3.2E-07	2.5E-07	1.1E-07	3.4E-10
Arsenic	6.7E-08	5.1E-08	2.3E-08	3.7E-10
Barium	4.1E-07	3.2E-07	1.4E-07	1.9E-09
Beryllium	0.0E+00	8.1E-06	3.7E-06	5.6E-08
Cadmium	3.6E-05	2.7E-05	1.2E-05	9.6E-08
Chromium	4.5E-05	4.0E-03	1.8E-03	2.0E-07
Cobalt	3.3E-07	2.5E-07	1.1E-07	4.0E-09
Copper	1.7E-05	2.2E-03	1.0E-03	8.0E-07
Lead	1.4E-05	2.6E-03	1.2E-03	1.5E-07
Manganese	2.0E-06	2.1E-06	9.7E-07	2.0E-09
Mercury	1.8E-05	2.3E-05	1.0E-05	5.9E-08
Nickel	5.6E-07	1.1E-05	4.9E-06	1.8E-06
Selenium	1.3E-05	9.9E-06	4.5E-06	3.3E-07
Strontium	1.3E-09	1.2E-09	5.6E-10	1.0E-12
Uranium	0.0E+00	1.4E-07	6.5E-08	3.7E-11
Zinc	2.3E-06	3.8E-06	1.7E-06	3.3E-08
Zirconium	4.4E-09	3.6E-09	1.6E-09	5.5E-12
Asbestos	0.0E+00	1.5E-03	7.0E-04	0.0E+00
Dioxins & Furans	0.0E+00	0.0E+00	0.0E+00	0.0E+00

Table A-6: Source Term Parameters - Indoor Waste Package Fire

Selected Waste Category	# of Packages	DR	ARF	RF	LPF	T_{FD} (hr)
Box Compacted	24	0.5	0.001	1	1	10.1
Non-Processible Boxed	24	0.5	0.001	1	1	1.4
Non-Processible Drummed	24	0.5	0.001	1	1	1.5
Moderator Resin (Unshielded)	1	0.5	0.001	1	1	3.9
Combined LLW and ILW Packages	24 [a]	0.5	0.001	1	1	1.5
	2 [b]	0.5	0.001	1	1	3.9

Notes:
a. Non-processible drummed.
b. Moderator resin (unshielded).

The nominal single unshielded ILW packages staged in the WPRB was assumed to ignite and burn together in a confined fire state where the exposed burning surface would be only from the top lid of the package.

As an exception, an ARF of 1 was used to calculate source terms for the volatile elements C-14, H-3, mercury, and selenium.

Dispersion and Consequence

The concentrations of radionuclides and non-radiological species in air in the WPRB were calculated using Equation 7-13 based on the assumption that the source term would be mixing within the WPRB due to the buoyant plume rise and building ventilation rate over the time of exposure to the worker. The source term emission rate into air was estimated using Equation 7-20. The concentration of radionuclides and non-radiological species in the vicinity of the public was calculated with Equation 7-15 based on a 1 hour public time of exposure at the nearest Bruce nuclear site boundary.

Significant worker exposure through external radiation was deemed unlikely for the 5 minute fire exposure and, therefore, was not assessed for this accident.

Results

Potential Impact of Radionuclides

Table A-7 shows that total radionuclide doses to workers over a 5 minute period (through inhalation, immersion and external radiation) are much less than the dose limit for workers (50 mSv) for any of the assessed waste categories. Similarly, Table A-8 shows that the total dose to the public (through inhalation and immersion) over a 1 hour exposure duration is much less than the 1 mSv public dose limit for any of the assessed waste categories.

Potential Impact of Non-Radiological Species

Table A-9 and Table A-10 show the ratios of air concentration to worker IDLH and public PAC 1 criteria, respectively. The air concentrations are less than non-radiological criteria near workers or the public.

Table A-7: Dose to Workers – Indoor Waste Package Fire

Selected Waste Category	Inhalation (mSv)	Immersion (mSv)	External Radiation (mSv)	Total (mSv)
Box Compacted	6.4E-03	2.3E-08	N/A	6.4E-03
Non-Processible Boxed	5.5E-03	3.0E-07	N/A	5.5E-03
Non-Processible Drummed	4.9E-02	4.7E-08	N/A	4.9E-02
Moderator Resin (Unshielded)	3.7E-02	2.1E-05	N/A	3.7E-02
Combined LLW and ILW Package	1.2E-01	4.2E-05	N/A	1.2E-01

Table A-8: Dose to Public – Indoor Waste Package Fire

Selected Waste Category	Inhalation (mSv)	Immersion (mSv)	Total (mSv)
Box Compacted	5.7E-06	3.4E-11	5.7E-06
Non-Processible Boxed	4.9E-06	4.5E-10	4.9E-06
Non-Processible Drummed	4.4E-05	7.1E-11	4.4E-05
Moderator Resin (Unshielded)	3.3E-05	3.1E-08	3.3E-05
Combined LLW and ILW Package	1.1E-04	6.2E-08	1.1E-04

Table A-9: Ratios of Air Concentration to Worker IDLH - Indoor Waste Package Fire

Non-Radiological Species	Box Compacted	Non-Processible Boxed	Non-Processible Drummed	Moderator Resin (Unshielded)	Combined LLW & ILW Packages
Antimony	8.3E-06	6.4E-06	2.9E-06	5.8E-09	2.9E-06
Arsenic	1.0E-05	8.0E-06	3.6E-06	3.8E-08	3.7E-06
Barium	2.6E-05	2.0E-05	9.0E-06	8.0E-08	9.2E-06
Beryllium	0.0E+00	1.8E-05	8.3E-06	8.4E-08	8.5E-06
Cadmium	3.1E-04	2.4E-04	1.1E-04	5.5E-07	1.1E-04
Chromium	1.2E-04	1.0E-02	4.7E-03	3.5E-07	4.7E-03
Cobalt	2.6E-06	1.9E-06	8.7E-07	2.1E-08	9.1E-07
Copper	9.6E-05	1.3E-02	5.7E-03	3.0E-06	5.7E-03
Lead	5.5E-05	1.0E-02	4.5E-03	3.8E-07	4.5E-03
Manganese	3.0E-05	3.3E-05	1.5E-05	2.1E-08	1.5E-05
Mercury	1.2E-03	1.5E-03	6.6E-04	2.5E-06	6.7E-04
Nickel	8.6E-05	1.7E-03	7.5E-04	1.9E-04	1.1E-03
Selenium	2.0E-02	1.5E-02	6.9E-03	3.4E-04	7.6E-03
Strontium	N/D	N/D	N/D	N/D	N/D
Uranium	0.0E+00	2.2E-05	1.0E-05	3.8E-09	1.0E-05
Zinc	3.5E-05	5.8E-05	2.7E-05	3.4E-07	2.7E-05
Zirconium	4.5E-06	3.7E-06	1.7E-06	3.8E-09	1.7E-06
Asbestos	0.0E+00	N/D	N/D	0.0E+00	N/D
Dioxins & Furans	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00

Table A-10: Ratios of Air Concentration on Public PAC 1 - Indoor Waste Package Fire

Non-Radiological Species	Box Compacted	Non-Processible Boxed	Non-Processible Drummed	Moderator Resin (Unshielded)	Combined LLW & ILW Packages
Antimony	1.0E-07	7.9E-08	3.6E-08	7.3E-11	3.6E-08
Arsenic	2.1E-08	1.7E-08	7.5E-09	7.9E-11	7.7E-09
Barium	1.3E-07	1.0E-07	4.6E-08	4.1E-10	4.7E-08
Beryllium	0.0E+00	2.6E-06	1.2E-06	1.2E-08	1.2E-06
Cadmium	1.1E-05	8.8E-06	4.0E-06	2.1E-08	4.0E-06
Chromium	1.5E-05	1.3E-03	5.9E-04	4.3E-08	5.9E-04
Cobalt	1.1E-07	8.0E-08	3.6E-08	8.5E-10	3.8E-08
Copper	5.5E-06	7.1E-04	3.2E-04	1.7E-07	3.2E-04
Lead	4.6E-06	8.3E-04	3.8E-04	3.1E-08	3.8E-04
Manganese	6.3E-07	6.9E-07	3.1E-07	4.3E-10	3.1E-07
Mercury	5.8E-06	7.3E-06	3.3E-06	1.3E-08	3.3E-06
Nickel	1.8E-07	3.4E-06	1.6E-06	3.9E-07	2.4E-06
Selenium	4.1E-06	3.2E-06	1.4E-06	7.0E-08	1.6E-06
Strontium	4.1E-10	4.0E-10	1.8E-10	2.2E-13	1.8E-10
Uranium	0.0E+00	4.6E-08	2.1E-08	7.9E-12	2.1E-08
Zinc	7.3E-07	1.2E-06	5.5E-07	7.0E-09	5.6E-07
Zirconium	1.4E-09	1.2E-09	5.3E-10	1.2E-12	5.3E-10
Asbestos	0.0E+00	5.0E-04	2.3E-04	0.0E+00	2.3E-04
Dioxins & Furans	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00

A.2.1.3 Above Ground Shielded ILW Package Steam Release

Scenario Description

A vehicle accident or equipment failure (e.g., electric malfunction) may ignite a fire in the equipment that may in turn propagate, if persistent, and cause a shielded moderator resin waste package (containing 2 resin liners) to heat and release volatile radionuclides (i.e., C-14, H-3) or non-radiological species (mercury, selenium) in its steam. A fire in the WPRB may also heat these waste packages and cause release of volatile species in its steam.

Source Term

Parameters used for the calculation of the source term amounts (Equation 7-5) and steam release rates (Equation 7-6) due to this steam release scenario are summarized in Table A-11 for the waste categories assessed. The steam release rate was assumed to be similar to the burning rate based on the total fire duration required to burn the entire waste.

Table A-11: Source Term Parameters – Above Ground Shielded ILW Steam Release

Indoor or Outdoor	Selected Waste Category	# of Packages [a]	DR	ARF [b]	RF	LPF	T_{FD} (hr)
Outdoor	Moderator Resin	1	0.1	1	1	1	1.6
Indoor	Moderator Resin	1	0.1	1	1	1	1.6

Notes:
a. Single shielded ILW package with 2 resin liners.
b. Conservatively assumed to be 1 for volatiles.

Dispersion and Consequence

For outdoor situations, the concentrations of radionuclides and non-radiological species in air were calculated using Equation 7-11. The source term emission rate into air was estimated using Equation 7-19.

For indoor situations, the concentrations of radionuclides and non-radiological species in air in the WPRB were calculated using Equation 7-13 based on the assumption that the source term would be mixed within the WPRB due to the buoyant plume and ventilation rate over the time of exposure to the worker. The source term emission rate into air was estimated using Equation 7-20.

The concentration of radionuclides and non-radiological species in the vicinity of the public was calculated with Equation 7-15 based on a 1 hour public time of exposure at the nearest Bruce nuclear site boundary.

Worker exposure through inhalation and immersion of radioactivity released to the air was assessed; significant additional external dose was deemed unlikely for the 5 minute worker exposure and, therefore, was not assessed for this accident.

Results

Potential Impact of Radionuclides

Table A-12 shows that total radionuclide doses to workers over a 5 minute period (through inhalation, immersion and external radiation) are much less than the dose limit for workers (50 mSv) for any of the assessed waste categories. Similarly, Table A-13 shows that the total dose to the public (through inhalation and immersion) over a 1 hour exposure duration is much less than the 1 mSv public dose limit for any of the assessed waste categories.

Table A-12: Dose to Workers – Above Ground Shielded ILW Steam Release

Indoor or Outdoor	Selected Waste Category	Inhalation (mSv)	Immersion (mSv)	External Radiation (mSv)	Total (mSv)
Outdoor	Moderator Resin	1.6E+00	6.8E-04	N/A	1.6E+00
Indoor	Moderator Resin	3.9E-02	1.7E-05	N/A	3.9E-02

Table A-13: Dose to Public – Above Ground Shielded ILW Steam Release

Indoor or Outdoor	Selected Waste Category	Inhalation (mSv)	Immersion (mSv)	Total (mSv)
Outdoor	Moderator Resin	1.6E-04	1.2E-07	1.6E-04
Indoor	Moderator Resin	3.5E-05	2.5E-08	3.5E-05

Potential Impact of Non-Radiological Species

Table A-14 and Table A-15 show the ratios of air concentration to worker IDLH and public PAC 1 criteria, respectively. The air concentrations are less than non-radiological criteria for workers and the public.

Table A-14: Ratios of Air Concentration to Worker IDLH – Above Ground Shielded ILW Steam Release

Non-Radiological Species	Moderator Resin	
	Outdoor	Indoor
Mercury	1.0E-04	2.5E-06
Selenium	1.4E-02	3.4E-04

Table A-15: Ratios of Air Concentration to Public PAC 1 – Above Ground Shielded ILW Steam Release

Non-Radiological Species	Moderator Resin	
	Outdoor	Indoor
Mercury	5.9E-08	1.3E-08
Selenium	3.3E-07	7.1E-08

A.2.2 Container Breach (Low Energy)

A.2.2.1 Outdoor Waste Package Breach

Scenario Description

This scenario considers accidents, caused by a vehicle accident or human error in handling, in which waste packages are dropped and breached outdoors.

Since a maximum of 8 LLW packages was conservatively assumed to be transported each time, a maximum of 8 LLW packages may also be involved in a low energy breach. Unshielded moderator resin liners and shielded moderator resin packages were considered for this scenario.

Source Term

Parameters used for the calculation of the source term amounts (Equation 7-5) due to this low energy breach scenario are summarized in Table A-16 for the waste categories assessed.

It should be noted that volatiles such as C-14, H-3, mercury and selenium were considered completely respirable (i.e., RF = 1).

Table A-16: Source Term Parameters - Outdoor Waste Package Breach

Selected Waste Category	# of Packages	DR	ARF	RF	LPF
Bottom Ash (Old)	8	0.25	0.002	0.3	1
Box Compacted	8	0.10	0.001	0.1	1
Non-Processible Boxed	8	0.10	0.001	0.1	1
Non-Processible Drummed	8	0.05	0.001	0.1	1
Moderator Resin (Unshielded)	1	0.1	0.001	0.1	1
Moderator Resin (Shielded) [a]	1	0.05	0.001	0.1	1

Note:
a. 1 shielded ILW package with 2 resin liners.

Dispersion and Consequence

The concentrations of radionuclides and non-radiological species released into the outdoor air were calculated using Equation 7-10. The source term emission rate into air was estimated using Equation 7-16, while the concentration of radionuclides and non-radiological species in the vicinity of the public was calculated with Equation 7-15 at the nearest Bruce nuclear site boundary.

In addition to the consequences from airborne releases, the external radiation dose to workers was also modelled using MicroShield based on the assumptions listed below.

- The package breach was modelled as a slice based on the fraction of the height of the packages breached (proportional to DR in Table A-16) with no shielding material. Effectively, the worker is exposed to an unshielded fraction DR of the waste.

- The configuration of the LLW packages was 2 by 4. The worker was assumed to be facing the side with 4 packages.

- The dose point was assumed to be along the centreline, 1 m from the ground (i.e., centre of the body of a person, and thus a good estimate of the whole body dose), and at a distance of 2 m from the source.

- The total external dose is the sum of the dose from intact packages and the dose from package breach.

Results

Potential Impact of Radionuclides

Table A-17 shows that total radionuclide doses to workers over a 5 minute period (through inhalation, immersion and external radiation) are much less than the dose limit for workers (50 mSv) for any of the assessed waste categories. Similarly, Table A-18 shows that the total dose to the public (through inhalation and immersion) over a 1 hour exposure duration is much less than the 1 mSv public dose limit for any of the assessed waste categories.

Potential Impact of Non-Radiological Species

Table A-19 and Table A-20 show the ratios of air concentration to worker IDLH and public PAC 1 criteria, respectively. The air concentrations are less than non-radiological criteria for workers and the public.

For bottom ash, the potential impacts on workers for some species are close to, but below, IDLH criteria. However the conservatisms in the analysis, and in particular in the air volume (see Section 7.5.3.3), should be noted, especially given that this scenario considers 8 LLW packages being transferred.

Table A-17: Dose to Workers - Outdoor Waste Package Breach

Selected Waste Category	Inhalation (mSv)	Immersion (mSv)	External Radiation (mSv)	Total (mSv)
Bottom Ash (Old)	2.5E-02	2.1E-04	7.1E-03	3.2E-02
Box Compacted	3.1E-03	7.4E-06	9.1E-04	4.1E-03
Non-Processible Boxed	1.1E-03	1.3E-05	3.1E-03	4.2E-03
Non-Processible Drummed	1.2E-03	1.1E-06	1.3E-03	2.5E-03
Moderator Resin (Unshielded)	8.3E-02	1.5E-03	5.5E-01	6.3E-01
Moderator Resin (Shielded)	1.9E-01	7.3E-03	2.3E-01	4.3E-01

Table A-18: Dose to Public - Outdoor Waste Package Breach

Selected Waste Category	Inhalation (mSv)	Immersion (mSv)	Total (mSv)
Bottom Ash (Old)	5.8E-06	1.1E-07	5.9E-06
Box Compacted	9.9E-07	4.0E-09	9.9E-07
Non-Processible Boxed	3.2E-07	7.2E-09	3.3E-07
Non-Processible Drummed	3.8E-07	6.0E-10	3.8E-07
Moderator Resin (Unshielded)	2.3E-05	8.2E-07	2.4E-05
Moderator Resin (Shielded)	4.5E-05	3.9E-06	4.9E-05

Table A-19: Ratios of Air Concentration to Worker IDLH - Outdoor Waste Package Breach

Non-Radiological Species	Bottom Ash (Old)	Box Compacted Waste	Non-Processible Boxed	Non-Processible Drummed	Moderator Resin (Unshielded)	Moderator Resin (Shielded)
Antimony	5.6E-02	2.7E-04	2.9E-05	6.9E-06	2.2E-07	2.2E-07
Arsenic	6.7E-02	3.4E-04	3.6E-05	8.6E-06	1.4E-06	1.4E-06
Barium	1.6E-01	8.5E-04	9.0E-05	2.2E-05	3.0E-06	3.0E-06
Beryllium	0.0E+00	0.0E+00	8.3E-05	2.0E-05	3.2E-06	3.2E-06
Cadmium	1.9E-03	1.0E-02	1.1E-03	2.6E-04	2.1E-05	2.1E-05
Chromium	7.7E-01	3.8E-03	4.7E-02	1.1E-02	1.3E-05	1.3E-05
Cobalt	1.7E-02	8.4E-05	8.7E-06	2.1E-06	7.8E-07	7.8E-07
Copper	6.6E-01	3.1E-03	5.7E-02	1.4E-02	1.2E-04	1.2E-04
Lead	3.6E-01	1.8E-03	4.5E-02	1.1E-02	1.4E-05	1.4E-05
Manganese	2.2E-02	9.9E-04	1.5E-04	3.6E-05	7.8E-07	7.8E-07
Mercury	1.9E-02	3.8E-04	6.6E-05	1.6E-05	9.6E-07	9.6E-07
Nickel	5.6E-01	2.8E-03	7.5E-03	1.8E-03	7.2E-03	7.2E-03
Selenium	0.0E+00	6.4E-03	6.9E-04	1.7E-04	1.3E-04	1.3E-04
Strontium	N/D	N/D	N/D	N/D	N/D	N/D
Uranium	0.0E+00	0.0E+00	1.0E-04	2.4E-05	1.4E-07	1.4E-07
Zinc	2.2E-01	1.1E-03	2.6E-04	6.3E-05	1.3E-05	1.3E-05
Zirconium	3.0E-02	1.5E-04	1.7E-05	4.0E-06	1.4E-07	1.4E-07
Asbestos	0.0E+00	0.0E+00	N/D	N/D	0.0E+00	0.0E+00
Dioxins & Furans	N/D	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00

Table A-20: Ratios of Air Concentration to Public PAC 1 - Outdoor Waste Package Breach

Non-Radiological Species	Bottom Ash (Old)	Box Compacted	Non-Processible Boxed	Non-Processible (Drummed)	Moderator Resin (Unshielded)	Moderator Resin (Shielded)
Antimony	2.5E-04	1.2E-06	1.3E-07	3.1E-08	9.8E-10	9.8E-10
Arsenic	4.9E-05	2.5E-07	2.7E-08	6.4E-09	1.1E-09	1.1E-09
Barium	3.0E-04	1.5E-06	1.6E-07	3.9E-08	5.5E-09	5.5E-09
Beryllium	0.0E+00	0.0E+00	4.2E-06	1.0E-06	1.6E-07	1.6E-07
Cadmium	2.5E-05	1.3E-04	1.4E-05	3.4E-06	2.8E-07	2.8E-07
Chromium	3.4E-02	1.7E-04	2.1E-03	5.0E-04	5.8E-07	5.8E-07
Cobalt	2.5E-04	1.2E-06	1.3E-07	3.1E-08	1.2E-08	1.2E-08
Copper	1.3E-02	6.4E-05	1.2E-03	2.8E-04	2.3E-06	2.3E-06
Lead	1.1E-02	5.3E-05	1.3E-03	3.2E-04	4.2E-07	4.2E-07
Manganese	1.6E-04	7.3E-06	1.1E-06	2.7E-07	5.8E-09	5.8E-09
Mercury	3.3E-05	6.8E-07	1.2E-07	2.8E-08	1.7E-09	1.7E-09
Nickel	4.1E-04	2.1E-06	5.6E-06	1.3E-06	5.3E-06	5.3E-06
Selenium	0.0E+00	4.7E-07	5.2E-08	1.2E-08	9.4E-09	9.4E-09
Strontium	9.9E-07	4.8E-09	6.4E-10	1.5E-10	3.0E-12	3.0E-12
Uranium	0.0E+00	0.0E+00	7.4E-08	1.8E-08	1.1E-10	1.1E-10
Zinc	1.6E-03	8.5E-06	2.0E-06	4.7E-07	9.4E-08	9.4E-08
Zirconium	3.4E-06	1.6E-08	1.9E-09	4.5E-10	1.6E-11	1.6E-11
Asbestos	0.0E+00	0.0E+00	8.0E-04	1.9E-04	0.0E+00	0.0E+00
Dioxins & Furans	6.3E-06	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00

A.2.2.2 Indoor Waste Package Breach

Scenario Description

Accidents resulting in waste package breach can potentially also occur indoors, within the WPRB. The most likely potential accidents involve handling or transfer equipment accidents (e.g., vehicle crashes, forklift impacts).

- A single LLW package may drop and cause a low energy breach to occur during handling and staging operations potentially impacting both workers and the public.
- In addition, a worst-case "what if" scenario may occur through severe natural disasters (e.g., earthquakes and extreme wind/snowfall) or other events resulting in the collapse of the roof or walls of the building leading to high energy breaches. In this "what if" scenario, the impact to workers is more likely to be incurred through injury due to conventional hazard rather than release of harmful radionuclides or non-radiological species. Thus, only the impact to the public will be considered for this scenario.

Source Term

Parameters used for the calculation of the source term amounts (Equation 7-5) are summarized in Table A-21 for the waste categories assessed.

For the "what if" roof collapse scenario, a maximum of 24 LLW packages and 2 ILW packages that were conservatively assumed to be temporarily staged in the WPRB at a time might be breached. In addition, a LPF of 0.1 was assigned, since the packages were assumed to be partially covered by building debris.

For single package breach, a LPF of 1 was used.

It should be noted that volatiles such as C-14, H-3, mercury, and selenium were considered completely respirable (i.e., RF = 1).

Dispersion and Consequence

The concentrations of radionuclides and non-radiological species in air in the WPRB were calculated using Equation 7-10 with V_{AIR} ~ 1000 m^3 also for indoor breach. The source term emission rate into air was estimated using Equation 7-17 based on the assumption that the contaminants would be well mixed in the WPRB over 1 hour of public exposure. The concentration of radionuclides and non-radiological species in the vicinity of the public was calculated with Equation 7-15 at the nearest Bruce nuclear site boundary. It should be noted that the ventilation rate in the WPRB was assumed to be intact during this case.

Table A-21: Source Term Parameters - Indoor Waste Package Breach

Selected Waste Category	# of Packages	DR	ARF	RF	LPF
Roof Collapse:					
Bottom Ash (Old)	24	1	0.01	0.3	0.1
Box Compacted	24	1	0.01	0.2	0.1
Non-Processible Boxed	24	1	0.01	0.2	0.1
Non-Processible Drummed	24	1	0.01	0.2	0.1
Mixed LLW and ILW Packages	24 [a]+2 [b]	1	0.01	0.2	0.1
Single LLW Package Breach:					
Bottom Ash (Old)	1	0.25	0.002	0.3	1
Box Compacted	1	0.1	0.001	0.1	1
Non-Processible Boxed	1	0.1	0.001	0.1	1
Non-Processible Drummed	1	0.05	0.001	0.1	1
Single ILW Package Breach:					
Moderator Resin (Unshielded)	1	0.1	0.001	0.1	1
Moderator Resin (Shielded) [c]	1	0.05	0.001	0.1	1

Notes:
a. Non-processible drummed.
b. Moderator resin (unshielded).
c. 1 shielded ILW package with 2 resin liners.

In addition to consequences from airborne releases, the external radiation dose from a single waste package to workers was modelled using MicroShield based on the assumptions described below.

- The package breach was modelled as a slice based on the fraction of the height of the packages breached (proportional to DR in Table A-21) with no shielding material. Effectively, the worker is exposed to an unshielded fraction DR of the waste.

- The external dose value for ILW package was based on the external radiation dose calculations for above ground outdoor ILW package breach (Section A.2.2.1).

- The dose point was assumed to be along the centreline, 1 m from the ground (i.e., centre of the body of a person, and thus a good estimate of the whole body dose), and at a distance of 2 m from the source.

- The total external dose is the sum of the dose from intact packages and the dose from package breach.

Results

Potential Impact of Radionuclides

Table A-22 summarizes the radionuclide dose to a worker due to breach of a single waste package. Table A-23 gives the radionuclide dose to the public due to the roof collapse scenario and due to the breach of a single waste package.

Table A-22 shows that for single waste package breach, total radionuclide doses to workers over a 5 minute period (through inhalation, immersion and external radiation) are much less than the dose limit for workers (50 mSv) for any of the assessed waste categories. Similarly, Table A-23 shows that the total dose to the public (through inhalation and immersion) over a 1 hour exposure duration is much less than the 1 mSv public dose limit for any of the assessed waste categories.

Potential Impact of Non-Radiological Species

Table A-24 gives the ratios of air concentration to worker IDLH for the breach of a single waste package. Table A-25 and Table A-26 give the ratios of air concentration to public PAC 1 for roof collapse scenario and for breach of a single waste package, respectively. The air concentrations are less than non-radiological criteria for workers and the public.

Table A-22: Dose to Workers - Indoor Single Waste Package Breach

Selected Waste Category	Inhalation (mSv)	Immersion (mSv)	External Radiation (mSv)	Total (mSv)
Bottom Ash (Old)	3.1E-02	2.7E-04	3.4E-02	6.6E-02
Box Compacted	3.9E-03	9.3E-06	4.0E-03	7.9E-03
Non-Processible Boxed	1.4E-03	1.7E-05	9.3E-03	1.1E-02
Non-Processible Drummed	1.5E-03	1.4E-06	4.1E-03	5.6E-03
Moderator Resin (Unshielded)	8.3E-02	1.5E-03	5.5E-01	6.3E-01
Moderator Resin (Shielded)	1.9E-01	7.3E-03	3.2E-01	5.2E-01

Table A-23: Dose to Public - Indoor Waste Package Breach

Selected Waste Category	Inhalation (mSv)	Immersion (mSv)	Total (mSv)
Roof Collapse:			
Bottom Ash (Old)	1.4E-05	2.7E-07	1.4E-05
Box Compacted	1.6E-05	4.7E-08	1.6E-05
Non-Processible Boxed	6.7E-06	8.5E-08	6.8E-06
Non-Processible Drummed	9.1E-06	1.4E-08	9.1E-06
Mixed LLW and ILW Packages	3.5E-04	6.5E-06	3.5E-04
Single Package Breach:			
Bottom Ash (Old)	2.9E-06	5.7E-08	2.9E-06
Box Compacted	4.9E-07	2.0E-09	4.9E-07
Non-Processible Boxed	1.6E-07	3.6E-09	1.6E-07
Non-Processible Drummed	1.9E-07	3.0E-10	1.9E-07
Moderator Resin (Unshielded)	9.1E-06	3.3E-07	9.5E-06
Moderator Resin (Shielded)	1.8E-05	1.5E-06	1.9E-05

Table A-24: Ratios of Air Concentration to Worker IDLH - Indoor Single Waste Package Breach

Non-Radiological Species	Bottom Ash (Old)	Box Compacted	Non-Processible Boxed	Non-Processible Drummed	Moderator Resin (Unshielded)	Moderator Resin (Shielded)
Antimony	7.0E-03	3.4E-05	3.6E-06	8.6E-07	2.2E-07	2.2E-07
Arsenic	8.4E-03	4.2E-05	4.5E-06	1.1E-06	1.4E-06	1.4E-06
Barium	2.0E-02	1.1E-04	1.1E-05	2.7E-06	3.0E-06	3.0E-06
Beryllium	0.0E+00	0.0E+00	1.0E-05	2.5E-06	3.2E-06	3.2E-06
Cadmium	2.4E-04	1.2E-03	1.3E-04	3.2E-05	2.1E-05	2.1E-05
Chromium	9.6E-02	4.8E-04	5.9E-03	1.4E-03	1.3E-05	1.3E-05
Cobalt	2.1E-03	1.1E-05	1.1E-06	2.6E-07	7.8E-07	7.8E-07
Copper	8.2E-02	3.9E-04	7.1E-03	1.7E-03	1.2E-04	1.2E-04
Lead	4.5E-02	2.2E-04	5.6E-03	1.4E-03	1.4E-05	1.4E-05
Manganese	2.8E-03	1.2E-04	1.9E-05	4.5E-06	7.8E-07	7.8E-07
Mercury	2.3E-03	4.8E-05	8.3E-06	2.0E-06	9.6E-07	9.6E-07
Nickel	7.0E-02	3.5E-04	9.4E-04	2.3E-04	7.2E-03	7.2E-03
Selenium	0.0E+00	8.0E-04	8.6E-05	2.1E-05	1.3E-04	1.3E-04
Strontium	N/D	N/D	N/D	N/D	N/D	N/D
Uranium	0.0E+00	0.0E+00	1.3E-05	3.0E-06	1.4E-07	1.4E-07
Zinc	2.8E-02	1.4E-04	3.3E-05	7.9E-06	1.3E-05	1.3E-05
Zirconium	3.8E-03	1.9E-05	2.1E-06	5.0E-07	1.4E-07	1.4E-07
Asbestos	0.0E+00	0.0E+00	N/D	N/D	0.0E+00	0.0E+00
Dioxins & Furans	N/D	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00

Table A-25: Ratios of Air Concentration to Public PAC 1 - Indoor Waste Package Breach (Roof Collapse)

Non-Radiological Species	Bottom Ash (Old)	Box Compacted	Non-Processible Boxed	Non-Processible Drummed	Mixed LLW & ILW Packages
Antimony	5.9E-04	2.8E-05	3.0E-06	1.5E-06	1.5E-06
Arsenic	1.2E-04	5.9E-06	6.3E-07	3.0E-07	3.2E-07
Barium	7.0E-04	3.7E-05	3.9E-06	1.9E-06	2.0E-06
Beryllium	0.0E+00	0.0E+00	1.0E-04	4.8E-05	5.0E-05
Cadmium	6.1E-05	3.2E-03	3.4E-04	1.6E-04	1.7E-04
Chromium	8.1E-02	4.0E-03	4.9E-02	2.4E-02	2.4E-02
Cobalt	5.9E-04	3.0E-05	3.1E-06	1.5E-06	1.7E-06
Copper	3.2E-02	1.5E-03	2.7E-02	1.3E-02	1.3E-02
Lead	2.5E-02	1.3E-03	3.2E-02	1.5E-02	1.5E-02
Manganese	3.9E-04	1.7E-04	2.6E-05	1.3E-05	1.3E-05
Mercury	7.8E-05	8.1E-06	1.4E-06	6.7E-07	6.8E-07
Nickel	9.8E-04	4.9E-05	1.3E-04	6.3E-05	1.5E-04
Selenium	0.0E+00	5.6E-06	6.1E-07	2.9E-07	3.7E-07
Strontium	2.4E-06	1.1E-07	1.5E-08	7.3E-09	7.3E-09
Uranium	0.0E+00	0.0E+00	1.8E-06	8.4E-07	8.5E-07
Zinc	3.9E-03	2.0E-04	4.6E-05	2.2E-05	2.4E-05
Zirconium	8.0E-06	3.9E-07	4.4E-08	2.1E-08	2.2E-08
Asbestos	0.0E+00	0.0E+00	1.9E-02	9.1E-03	9.1E-03
Dioxins & Furans	1.5E-05	0.0E+00	0.0E+00	0.0E+00	0.0E+00

Table A-26: Ratios of Air Concentration to Public PAC 1 - Indoor Single Waste Package Breach

Non-Radiological Species	Bottom Ash (Old)	Box Compacted	Non-Processible Boxed	Non-Processible Boxed	Moderator Resin (Unshielded)	Moderator Resin (Shielded)
Antimony	1.2E-05	5.9E-08	6.3E-09	1.5E-09	3.9E-10	3.9E-10
Arsenic	2.4E-06	1.2E-08	1.3E-09	3.2E-10	4.2E-10	4.2E-10
Barium	1.5E-05	7.7E-08	8.1E-09	1.9E-09	2.2E-09	2.2E-09
Beryllium	0.0E+00	0.0E+00	2.1E-07	5.0E-08	6.4E-08	6.4E-08
Cadmium	1.3E-06	6.6E-06	7.0E-07	1.7E-07	1.1E-07	1.1E-07
Chromium	1.7E-03	8.4E-06	1.0E-04	2.5E-05	2.3E-07	2.3E-07
Cobalt	1.2E-05	6.2E-08	6.4E-09	1.5E-09	4.6E-09	4.6E-09
Copper	6.6E-04	3.1E-06	5.7E-05	1.4E-05	9.2E-07	9.2E-07
Lead	5.3E-04	2.6E-06	6.6E-05	1.6E-05	1.7E-07	1.7E-07
Manganese	8.2E-06	3.6E-07	5.5E-08	1.3E-08	2.3E-09	2.3E-09
Mercury	1.6E-06	3.4E-08	5.8E-09	1.4E-09	6.8E-10	6.8E-10
Nickel	2.0E-05	1.0E-07	2.8E-07	6.6E-08	2.1E-06	2.1E-06
Selenium	0.0E+00	2.3E-08	2.5E-09	6.1E-10	3.7E-09	3.7E-09
Strontium	4.9E-08	2.4E-10	3.2E-11	7.6E-12	1.2E-12	1.2E-12
Uranium	0.0E+00	0.0E+00	3.7E-09	8.8E-10	4.2E-11	4.2E-11
Zinc	8.2E-05	4.2E-07	9.7E-08	2.3E-08	3.7E-08	3.7E-08
Zirconium	1.7E-07	8.1E-10	9.2E-11	2.2E-11	6.3E-12	6.3E-12
Asbestos	0.0E+00	0.0E+00	4.0E-05	9.5E-06	0.0E+00	0.0E+00
Dioxins & Furans	3.1E-07	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00

A.2.2.3 Inadequate Package Shielding

In this accident, waste packages were assumed to be shipped to the DGR accidentally without appropriate shielding. This would be due to an error in measuring the dose rate at WWMF or in placing a package in an inappropriate shield. The consequence is that workers would be exposed to higher dose rates than would have been expected.

However, the normal WWMF practice, which will be continued at the DGR, is for the workers to have their EPDs set to alarm at significantly higher dose rates relative to their expected activity. In event of inadequate shielding of a container containing any waste category, this EPD alarm setting would provide independent assurance that the worker does not receive more than 10 mSv per year. In addition, there will be multiple workers in the area, each with their own alarming EPD. In the event of failure of one EPD, the others would alarm, thereby providing a degree of redundancy.

A.3 Underground Accident Consequence

A.3.1 Fire

A.3.1.1 Waste Package Fire During Underground Transfer

Scenario Description

Accidents may occur in the underground staging area and/or tunnels during the handling and transfer of waste packages from the shaft to emplacement rooms for permanent storage. During transfer, a vehicle accident or equipment failure (e.g., electric malfunction) in a forklift may ignite a fire in the equipment that may in turn propagate to transported LLW or unshielded ILW resin liners (moderator resin). Shielded ILW packages are not expected to catch on fire, because the thick concrete shield provides a significant thermal barrier, in addition to the low combustibility due to the bound water content of the resins (40-50%), and the high ignition temperature of the resins.

Source Term

Parameters used for the calculation of the source term amounts (Equation 7-5) and release rates (Equation 7-6) due to this fire accident scenario are summarized in Table A-27 for the waste categories assessed.

A single LLW or unshielded ILW package during handling and transfer was assumed to ignite and burn in a confined fire state. The exposed burning surface in a confined fire would be from the top (lid) of the package. Only the combustible fraction of the respective waste category was assumed to burn.

As an exception, an ARF of 1 was used to calculate source terms for volatile elements C-14, H-3, mercury, and selenium.

Table A-27: Source Term Parameters – Waste Package Fire During Underground Transfer

Selected Waste Category	# of Packages	DR	ARF	RF	LPF	T_{FD} (hr)
Box Compacted	1	0.5	0.001	1	1	5.0
Non-Processible Boxed	1	0.5	0.001	1	1	0.7
Non-Processible Drummed	1	0.5	0.001	1	1	0.7
Moderator Resin (Unshielded)	1	0.5	0.001	1	1	3.9

Dispersion and Consequence

The concentrations of radionuclides and non-radiological species in air in the underground tunnel or staging area were calculated using Equation 7-14 based on the assumption that the accident occurred in the staging area or an active room, with a ventilation rate of about 18 m^3/s (ventilation rates are higher in the access tunnels). The exposed worker was assumed to be downstream from the package fire for the 5 minute period. The source term emission rate into air was estimated using Equation 7-19. Equation 7-19 is used to calculate the average rate of emission of material from an outdoor fire accident scenario, but is also used for underground fires, as it is assumed to be directly vented for the duration of the exposure period. The concentration of radionuclides and non-radiological species in the vicinity of the public was calculated with Equation 7-15 based on a 1 hour of exposure at the nearest Bruce nuclear site boundary.

Significant worker exposure through external radiation was deemed unlikely over the 5 minute exposure and, therefore, was not assessed for this accident.

Results

Potential Impact of Radionuclides

Table A-28 shows that total radionuclide doses to workers over a 5 minute period (through inhalation, immersion and external radiation) are much less than the dose limit for workers (50 mSv) for any of the assessed waste categories. Similarly, Table A-29 shows that the total dose to the public (through inhalation and immersion) over a 1 hour exposure duration is much less than the 1 mSv public dose limit for any of the assessed waste categories.

Potential Impact of Non-Radiological Species

Table A-30 and Table A-31 show the ratios of air concentration to worker IDLH and public PAC 1 criteria, respectively. The air concentrations are less than non-radiological criteria for workers and the public.

Table A-28: Dose to Workers - Waste Package Fire During Underground Transfer

Selected Waste Category	Inhalation (mSv)	Immersion (mSv)	External Radiation (mSv)	Total (mSv)
Box Compacted	4.0E-02	1.4E-07	N/A	4.0E-02
Non-Processible Boxed	3.4E-02	1.9E-06	N/A	3.4E-02
Non-Processible Drummed	3.1E-01	2.9E-07	N/A	3.1E-01
Moderator Resin (Unshielded)	2.8E-01	1.6E-04	N/A	2.8E-01

Table A-29: Dose to Public - Waste Package Fire During Underground Transfer

Selected Waste Category	Inhalation (mSv)	Immersion (mSv)	Total (mSv)
Box Compacted	8.2E-04	4.9E-09	8.2E-04
Non-Processible Boxed	7.1E-04	6.4E-08	7.1E-04
Non-Processible Drummed	6.3E-03	1.0E-08	6.3E-03
Moderator Resin (Unshielded)	5.8E-03	5.4E-06	5.8E-03

Table A-30: Ratios of Air Concentration to Worker IDLH - Waste Package Fire During Underground Transfer

Non-Radiological Species	Box Compacted	Non-Processible Boxed	Non-Processible Drummed	Moderator Resin (Unshielded)
Antimony	5.2E-06	4.0E-06	1.8E-06	4.4E-08
Arsenic	6.5E-06	5.0E-06	2.3E-06	2.8E-07
Barium	1.6E-05	1.2E-05	5.7E-06	6.0E-07
Beryllium	0.0E+00	1.1E-05	5.2E-06	6.3E-07
Cadmium	1.9E-04	1.5E-04	6.7E-05	4.1E-06
Chromium	7.3E-05	6.5E-03	2.9E-03	2.6E-06
Cobalt	1.6E-06	1.2E-06	5.5E-07	1.5E-07
Copper	6.0E-05	7.9E-03	3.6E-03	2.3E-05
Lead	3.4E-05	6.2E-03	2.8E-03	2.8E-06
Manganese	1.9E-05	2.1E-05	9.4E-06	1.5E-07
Mercury	7.3E-04	9.1E-04	4.1E-04	1.9E-05
Nickel	5.4E-05	1.0E-03	4.7E-04	1.4E-03
Selenium	1.2E-02	9.6E-03	4.3E-03	2.5E-03
Strontium	N/D	N/D	N/D	N/D
Uranium	0.0E+00	1.4E-05	6.3E-06	2.8E-08
Zinc	2.2E-05	3.7E-05	1.7E-05	2.5E-06
Zirconium	2.8E-06	2.3E-06	1.1E-06	2.8E-08
Asbestos	0.0E+00	N/D	N/D	0.0E+00
Dioxins & Furans	0.0E+00	0.0E+00	0.0E+00	0.0E+00

Table A-31: Ratios of Air Concentration to Public PAC 1 - Waste Package Fire During Underground Transfer

Non-Radiological Species	Box Compacted	Non-Processible Boxed	Non-Processible Drummed	Moderator Resin (Unshielded)
Antimony	1.5E-06	1.1E-06	5.2E-07	1.3E-08
Arsenic	3.1E-07	2.4E-07	1.1E-07	1.4E-08
Barium	1.9E-06	1.5E-06	6.7E-07	7.1E-08
Beryllium	0.0E+00	3.8E-05	1.7E-05	2.1E-06
Cadmium	1.7E-04	1.3E-04	5.8E-05	3.6E-06
Chromium	2.1E-04	1.9E-02	8.5E-03	7.5E-06
Cobalt	1.6E-06	1.2E-06	5.2E-07	1.5E-07
Copper	7.9E-05	1.0E-02	4.7E-03	3.0E-05
Lead	6.6E-05	1.2E-02	5.4E-03	5.5E-06
Manganese	9.1E-06	1.0E-05	4.5E-06	7.4E-08
Mercury	8.4E-05	1.1E-04	4.8E-05	2.2E-06
Nickel	2.6E-06	5.0E-05	2.3E-05	6.8E-05
Selenium	5.9E-05	4.6E-05	2.1E-05	1.2E-05
Strontium	6.0E-09	5.7E-09	2.6E-09	3.8E-11
Uranium	0.0E+00	6.6E-07	3.0E-07	1.4E-09
Zinc	1.1E-05	1.8E-05	7.8E-06	1.2E-06
Zirconium	2.0E-08	1.7E-08	7.6E-09	2.1E-10
Asbestos	0.0E+00	7.1E-03	3.3E-03	0.0E+00
Dioxins & Furans	0.0E+00	0.0E+00	0.0E+00	0.0E+00

A.3.1.2 In Room Unshielded Waste Package Fire

Scenario Description

In this bounding scenario, equipment/mechanical failures may result in igniting a fire involving tires and/or diesel fuel in an emplacement room. If the equipment fire were not quickly extinguished, the fire was assumed to spread to the stored combustible LLW packages or unshielded ILW resin liners.

A maximum of 2400 LLW containers may be stored in an underground emplacement room, and therefore, may be conservatively exposed to an unconfined full room fire. In addition, a maximum of 1200 unshielded ILW (resin liner) packages were assumed to be exposed to an unconfined full room fire.

Source Term

Parameters used for the calculation of the source term amounts (Equation 7-5) and release rates (Equation 7-6) due to this fire accident scenario are summarized in Table A-32 for the waste categories assessed.

Table A-32: Source Term Parameters - In Room Unshielded Waste Package Fire

Selected Waste Category	# of Packages	DR	ARF	RF	LPF	T_{FD} (hr)
Box Compacted	2400	1	0.01	1	1	743
Non-Processible Boxed	2400	1	0.001/0.01[a]	1	1	129
Non-Processible Drummed	2400	1	0.001/0.01[a]	1	1	137
Moderator Resin (Unshielded)	1200	1	0.01	1	1	291

Note:
a. ARF=0.001 for bulk metals and asbestos, and 0.01 otherwise.

In order to calculate the total fire duration to burn all waste in the room, the configuration described below was assumed.

- Box compacted wastes are stacked 4 high and 4 across in a single vertical layer, with 150 layers deep into the emplacement room.
- Non-processible boxed and drummed wastes are stacked 5 high and 4 across in a single vertical layer, with 120 layers deep into the emplacement room.
- Unshielded resin liners are stacked 2 high and 4 across in a single vertical layer, with 150 layers deep into the emplacement room.

The exposed burning surface for the room fire was approximated by the exterior five exposed surfaces (top and four sides) of the stacked configuration. The package bottom surface in contact with the room floor was excluded, as were gaps between boxes. Only the combustible fraction of the respective waste category was assumed to burn, while oxygen availability in the room due to ventilation rates was also considered as a limiting factor. This corresponds to a 70 MW fire, which is in the range of a large truck fire in a tunnel.

The ARF of 0.01 is applicable for surface or dispersed or combustible elements. It was assumed applicable to all radionuclides, and by default to all elements. An ARF of 1 was used for volatile elements - C-14, H-3, mercury, and selenium. An ARF of 0.001 was applied for bulk non-combustible metals and non-metallics, which in this case was limited to copper, chromium, lead and asbestos in non-processible wastes.

Dispersion and Consequence

Such a fire would take many minutes to develop from an initial small fire. During this period, the worker risk is addressed by the single waste package fire scenario (Section A.3.1.1). By the time the fire has developed into a room fire, any workers would have left the area and gone to refuge stations or safe locations. Therefore, only the potential impact on public was evaluated.

For public exposure, the source term emission rate into air was estimated using Equation 7-19. Equation 7-19 was used to calculate the average rate of emission of material from an outdoor fire accident scenario, but is also used for underground fires, as it is assumed to be directly vented for the duration of the exposure period. It was assumed that the room ventilation rate continues at the operating value of 18 m^3/s.

The concentration of radionuclides and non-radiological species in the vicinity of the public was calculated with Equation 7-15 based on a 1 hour time of exposure at the nearest Bruce nuclear site boundary. Although the fire could burn or smoulder for a long time, the public impact at site boundary was limited to 1 hour based on the assumption that for longer times, the fire would have been put out or the room would have been isolated, for example, by firedoors or temporary walls or by turning off the local ventilation. The implications of an unmitigated underground room fire is also considered as a “what if” case.

Results

Potential Impact of Radionuclides

Table A-33 shows that the total dose to the public (through inhalation and immersion) over a 1 hour exposure duration is much less than the 1 mSv public dose limit for any of the assessed waste categories. The total dose to the public over the maximum fire duration is also shown in Table A-33 for the LLW and ILW cases with the longest fire durations – about 700 hours for box compacted waste and 300 hours for unshielded moderator resin, respectively. In these case, the ADF decreases with time (due to plume meander) as described in Table 7-36. The results for these long duration fires are also less than the 1 mSv public dose limit for both waste types assessed.

Table A-33: Dose to Public - In Room Unshielded Waste Package Fire

Selected Waste Category	Inhalation (mSv)	Immersion (mSv)	Total (mSv)
Public Exposure Time: 1 hour			
Box Compacted	2.8E-03	1.6E-07	2.8E-03
Non-Processible Boxed	2.3E-03	1.7E-06	2.3E-03
Non-Processible	1.6E-02	2.6E-07	1.6E-02
Moderator Resin (Unshielded)	1.9E-02	4.8E-05	1.9E-02
Public Exposure Time: Full Fire Duration			
Box Compacted	6.4E-02	3.7E-06	6.4E-02
Moderator Resin (Unshielded)	2.0E-01	5.0E-04	2.0E-01

Potential Impact of Non-Radiological Species

Table A-34 shows the ratios of air concentration to public PAC 1 criteria. The air concentrations are less than non-radiological criteria for the public.

Table A-34: Ratios of Air Concentration to Public PAC 1 - In Room Unshielded Waste Package Fire

Non-Radiological Species	Box Compacted	Non-Processible Boxed	Non-Processible Drummed	Moderator Resin
Antimony	4.8E-04	3.0E-04	1.3E-04	4.1E-06
Arsenic	1.0E-04	6.2E-05	2.8E-05	4.4E-06
Barium	6.3E-04	3.8E-04	1.7E-04	2.3E-05
Beryllium	0.0E+00	9.7E-03	4.4E-03	6.7E-04
Cadmium	5.4E-02	3.3E-02	1.5E-02	1.1E-03
Chromium	6.9E-02	4.8E-01	2.2E-01	2.4E-03
Cobalt	5.1E-04	3.0E-04	1.4E-04	4.8E-05
Copper	2.6E-02	2.7E-01	1.2E-01	9.6E-03
Lead	2.2E-02	3.1E-01	1.4E-01	1.7E-03
Manganese	3.0E-03	2.6E-03	1.2E-03	2.4E-05
Mercury	2.7E-03	2.7E-03	1.2E-03	7.1E-05
Nickel	8.4E-04	1.3E-02	5.9E-03	2.2E-02
Selenium	1.9E-03	1.2E-03	5.4E-04	3.9E-04
Strontium	1.9E-06	1.5E-06	6.7E-07	1.2E-08
Uranium	0.0E+00	1.7E-04	7.8E-05	4.4E-07
Zinc	3.4E-03	4.5E-03	2.1E-03	3.9E-04
Zirconium	6.7E-06	4.3E-06	2.0E-06	6.6E-08
Asbestos	0.0E+00	1.9E-01	8.4E-02	0.0E+00
Dioxins & Furans	0.0E+00	0.0E+00	0.0E+00	0.0E+00

For non-processible waste, potential impacts to the public are close to, but below, PAC 1 criteria. However, the conservatism in the analyses should be noted. In particular, these are from assumed release of particulate from bulk materials like steel. In addition, the deposition of particulate was not credited and underground fire scenario releases was modelled as non-thermal plume.

A.3.1.3 Underground Shielded ILW (Resin Liner) Package Steam Release

<u>Scenario Description</u>

Although the direct burning of shielded ILW in potential fire scenarios is not considered a credible event, potential steam and volatile release from an ILW resin waste package due to an external fire was considered. An external fire due to mechanical/equipment failure in the underground tunnel/staging area may cause a shielded moderator resin waste package (containing 2 resin liners) to heat and release volatile radionuclides (i.e., C-14, H-3) or non-radiological species (mercury, selenium) in its steam.

Although identified as a separate accident scenario, the in-room (resin liner) package steam release involving a nominal single waste package is essentially the same as the accident scenario of ILW package steam release during transfer.

Source Term

Parameters used for the calculation of the source term amounts (Equation 7-5) and steam release rates (Equation 7-6) due to this steam release scenario are summarized in Table A-35 for the waste categories assessed. The steam release rate was assumed to be similar to the burning rate based on the total fire duration required to burn the entire waste.

Table A-35: Source Term Parameters – Underground ILW Package Steam Release

Selected Waste Category [a]	# of Packages	DR	ARF [b]	RF	LPF	T_{FD} (hr)
Moderator Resin	1	0.1	1	1	1	1.6

Notes:
a. Single shielded ILW package with 2 resin liners.
b. Conservatively assumed to be 1 for volatiles.

Dispersion and Consequence

The concentrations of radionuclides and non-radiological species in air in the underground tunnel or staging area were calculated using Equation 7-14 based on the assumption that the accident occurred in staging area or an active room, with a ventilation rate of about 18 m^3/s (ventilation rates are higher in the access tunnels). The exposed worker was assumed to be downstream from the package fire for the 5 minute period. The source term emission rate into air was estimated using Equation 7-19. The concentration of radionuclides and non-radiological species in the vicinity of the public was calculated with Equation 7-15 based on a 1 hour time of exposure at the nearest Bruce nuclear site boundary.

Significant worker exposure through external radiation was deemed unlikely over the 5 minute exposure as the shielding remained intact and, therefore, was not assessed.

Results

Potential Impact of Radionuclides

Table A-36 shows that total radionuclide doses to workers over a 5 minute period (through inhalation, immersion and external radiation) are much less than the dose limit for workers (50 mSv) for any of the assessed waste categories. Similarly, Table A-37 shows that the total dose to the public (through inhalation and immersion) over a 1 hour exposure duration is much less than the 1 mSv public dose limit for any of the assessed waste categories.

Potential Impact of Non-Radiological Species

Table A-38 and Table A-39 below show the ratios of air concentration to worker IDLH and public PAC 1 criteria, respectively. The air concentrations are less than non-radiological criteria for workers and the public.

Table A-36: Dose to Workers - Underground ILW Package Steam Release

Selected Waste Category	Inhalation (mSv)	Immersion (mSv)	External Radiation (mSv)	Total (mSv)
Moderator Resin	2.9E-01	1.3E-04	N/A	2.9E-01

Table A-37: Dose to Public - Underground ILW Package Steam Release

Selected Waste Category	Inhalation (mSv)	Immersion (mSv)	Total (mSv)
Moderator Resin	6.0E-03	4.4E-06	6.0E-03

Table A-38: Ratios of Air Concentration to Worker IDLH - Underground ILW Package Steam Release

Non-Radiological Species	Moderator Resin
Mercury	1.9E-05
Selenium	2.5E-03

Table A-39: Ratios of Air Concentration to Public PAC 1 - Underground ILW Package Steam Release

Non-Radiological Species	Moderator Resin
Mercury	2.2E-06
Selenium	1.2E-05

A.3.2 Container Breach (Low Energy)

A.3.2.1 Waste Package Breach During Underground Transfer

Scenario Description

During the handling and transfer of waste packages within the underground tunnel/staging area, waste packages may be breached in accidents (e.g., vehicle crash, forklift impact, container drop), affecting a nominal single waste package. The accident was assumed to occur in the shaft station, because of the frequency of package handling there.

Source Term

Parameters used for the calculation of the source term amounts (Equation 7-5) due to this low energy breach scenario are summarized in Table A-40 for the waste categories assessed.

It should be noted that volatiles such as C-14, H-3, mercury, and selenium were considered completely respirable (i.e., RF = 1).

Table A-40: Source Term Parameters - Waste Package Breach During Underground Transfer

Selected Waste Category	# of Packages	DR	ARF	RF	LPF
Bottom Ash (Old)	1	0.25	0.002	0.3	1
Box Compacted	1	0.1	0.001	0.1	1
Non-Processible Boxed	1	0.1	0.001	0.1	1
Non-Processible Drummed	1	0.05	0.001	0.1	1
Moderator Resin (Unshielded)	1	0.1	0.001	0.1	1
Moderator Resin (Shielded) [a]	1	0.05	0.001	0.1	1

Note:
a. Single shielded ILW package with 2 resin liners.

Dispersion and Consequence

The concentrations of radionuclides and non-radiological species in air in the underground tunnel/staging area were calculated using Equation 7-12 based on the assumption that the accident occurred in the staging area or an active room, with a ventilation rate of about 18 m^3/s (ventilation rates are higher in the access tunnels). The exposed worker was assumed to be downstream from the package breach for the 5 minute period. The source term emission rate into air was estimated using Equation 7-18 over the 1 hour time of exposure to the public. The concentration of radionuclides and non-radiological species in the vicinity of the public was calculated with Equation 7-15 based on a 1 hour time of exposure at the nearest Bruce nuclear site boundary.

In addition to consequences from airborne releases, the external radiation dose to workers was also modelled using MicroShield based the assumptions described below.

- The package breach was modelled as a slice based on the fraction of the height of the packages breached (proportional to DR in Table A-40) with no shielding material. Effectively, the worker is exposed to an unshielded fraction DR of the waste.
- The dose point was assumed to be along the centreline, 1 m from the ground (i.e., centre of the body of a person, and thus a good estimate of the whole body dose), and at a distance of 2 m from the source.
- The total external dose is the sum of the dose from intact packages and the dose from package breach.

Results

Potential Impact of Radionuclides

Table A-41 shows that total radionuclide doses to workers over a 5 minute period (through inhalation, immersion and external radiation) are much less than the dose limit for workers (50 mSv) for any of the assessed waste categories. Similarly, Table A-42 shows that the total dose to the public (through inhalation and immersion) over a 1 hour exposure duration is much less than the 1 mSv public dose limit for any of the assessed waste categories.

Potential Impact of Non-Radiological Species

Table A-43 and Table A-44 show the ratios of air concentration to worker IDLH and public PAC 1 criteria, respectively. The air concentrations are less than non-radiological criteria for workers and the public.

Table A-41: Dose to Workers – Waste Package Breach During Underground Transfer

Selected Waste Category	Inhalation (mSv)	Immersion (mSv)	External Radiation (mSv)	Total (mSv)
Bottom Ash (Old)	5.8E-03	5.0E-05	3.4E-02	4.0E-02
Box Compacted	7.3E-04	1.7E-06	4.0E-03	4.7E-03
Non-Processible Boxed	2.5E-04	3.1E-06	9.3E-03	9.6E-03
Non-Processible Drummed	2.8E-04	2.6E-07	4.1E-03	4.4E-03
Moderator Resin (Unshielded)	1.5E-02	2.9E-04	5.5E-01	5.7E-01
Moderator Resin (Shielded)	3.5E-02	1.4E-03	3.2E-01	3.6E-01

Table A-42: Dose to Public - Waste Package Breach During Underground Transfer

Selected Waste Category	Inhalation (mSv)	Immersion (mSv)	Total (mSv)
Bottom Ash (Old)	7.3E-06	1.4E-07	7.4E-06
Box Compacted	1.2E-06	4.9E-09	1.2E-06
Non-Processible Boxed	4.0E-07	9.0E-09	4.1E-07
Non-Processible Drummed	4.7E-07	7.5E-10	4.7E-07
Moderator Resin (Unshielded)	2.3E-05	8.2E-07	2.4E-05
Moderator Resin (Shielded)	4.5E-05	3.9E-06	4.9E-05

Table A-43: Ratios of Air Concentration to Worker IDLH - Waste Package Breach During Underground Transfer

Non-Radiological Species	Bottom Ash (Old)	Box Compacted	Non-Processible Boxed	Non-Processible Drummed	Moderator Resin (Unshielded)	Moderator Resin (Shielded)
Antimony	1.3E-03	6.2E-06	6.7E-07	1.6E-07	4.1E-08	4.1E-08
Arsenic	1.5E-03	7.8E-06	8.3E-07	2.0E-07	2.6E-07	2.6E-07
Barium	3.7E-03	2.0E-05	2.1E-06	5.0E-07	5.6E-07	5.6E-07
Beryllium	0.0E+00	0.0E+00	1.9E-06	4.6E-07	5.9E-07	5.9E-07
Cadmium	4.4E-05	2.3E-04	2.5E-05	5.9E-06	3.8E-06	3.8E-06
Chromium	1.8E-02	8.8E-05	1.1E-03	2.6E-04	2.4E-06	2.4E-06
Cobalt	3.9E-04	2.0E-06	2.0E-07	4.8E-08	1.4E-07	1.4E-07
Copper	1.5E-02	7.3E-05	1.3E-03	3.2E-04	2.1E-05	2.1E-05
Lead	8.3E-03	4.2E-05	1.0E-03	2.5E-04	2.6E-06	2.6E-06
Manganese	5.2E-04	2.3E-05	3.5E-06	8.3E-07	1.4E-07	1.4E-07
Mercury	4.3E-04	8.8E-06	1.5E-06	3.7E-07	1.8E-07	1.8E-07
Nickel	1.3E-02	6.5E-05	1.7E-04	4.2E-05	1.3E-03	1.3E-03
Selenium	0.0E+00	1.5E-04	1.6E-05	3.8E-06	2.4E-05	2.4E-05
Strontium	N/D	N/D	N/D	N/D	N/D	N/A
Uranium	0.0E+00	0.0E+00	2.3E-06	5.6E-07	2.6E-08	2.6E-08
Zinc	5.2E-03	2.7E-05	6.1E-06	1.5E-06	2.4E-06	2.4E-06
Zirconium	7.0E-04	3.4E-06	3.9E-07	9.3E-08	2.6E-08	2.6E-08
Asbestos	0.0E+00	0.0E+00	N/D	N/D	0.0E+00	0.0E+00
Dioxins & Furans	N/D	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00

Table A-44: Ratios of Air Concentration to Public PAC 1 – Waste Package Breach During Underground Transfer

Non-Radiological Species	Bottom Ash (Old)	Box Compacted	Non-Processible Boxed	Non-Processible Drummed	Moderator Resin (Unshielded)	Moderator Resin (Shielded)
Antimony	3.1E-05	1.5E-07	1.6E-08	3.8E-09	9.8E-10	9.8E-10
Arsenic	6.2E-06	3.1E-08	3.3E-09	8.0E-10	1.1E-09	1.1E-09
Barium	3.7E-05	1.9E-07	2.1E-08	4.9E-09	5.5E-09	5.5E-09
Beryllium	0.0E+00	0.0E+00	5.2E-07	1.3E-07	1.6E-07	1.6E-07
Cadmium	3.2E-06	1.7E-05	1.8E-06	4.3E-07	2.8E-07	2.8E-07
Chromium	4.3E-03	2.1E-05	2.6E-04	6.2E-05	5.8E-07	5.8E-07
Cobalt	3.1E-05	1.6E-07	1.6E-08	3.9E-09	1.2E-08	1.2E-08
Copper	1.7E-03	7.9E-06	1.4E-04	3.5E-05	2.3E-06	2.3E-06
Lead	1.3E-03	6.7E-06	1.7E-04	4.0E-05	4.2E-07	4.3E-07
Manganese	2.1E-05	9.2E-07	1.4E-07	3.3E-08	5.8E-09	5.8E-09
Mercury	4.1E-06	8.5E-08	1.5E-08	3.5E-09	1.7E-09	1.7E-09
Nickel	5.2E-05	2.6E-07	6.9E-07	1.7E-07	5.3E-06	5.3E-06
Selenium	0.0E+00	5.9E-08	6.4E-09	1.5E-09	9.4E-09	9.4E-09
Strontium	1.2E-07	6.0E-10	8.0E-11	1.9E-11	3.0E-12	3.0E-12
Uranium	0.0E+00	0.0E+00	9.3E-09	2.2E-09	1.1E-10	1.1E-10
Zinc	2.1E-04	1.1E-06	2.4E-07	5.9E-08	9.4E-08	9.4E-08
Zirconium	4.2E-07	2.1E-09	2.3E-10	5.6E-11	1.6E-11	1.6E-11
Asbestos	0.0E+00	0.0E+00	1.0E-04	2.4E-05	0.0E+00	0.0E+00
Dioxins & Furans	7.9E-07	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00

A.3.2.2 In Room Waste Package Breach

Scenario Description

During the handling and transfer of waste packages in the emplacement room, packages may be breached in accidents (e.g., vehicle crash, forklift impact, container drop during stacking). An in-room breach was assumed to involve the greater number of packages involved in either: i) the failure of a stacked column due to corroded or damaged bottoms that collapse under full load; or ii) a row of waste packages that may drop and breach due to rock fall.

Source Term

Parameters used for the calculation of the source term amounts (Equation 7-5) due to this low energy breach scenario are summarized in Table A-45 for the waste categories assessed.

It should be noted that volatiles such as C-14, H-3, mercury, and selenium were considered completely respirable (i.e., RF = 1).

Table A-45: Source Term Parameters - In Room Waste Package Breach

Selected Waste Category	# of Packages	DR	ARF	RF	LPF
Bottom Ash (Old)	3	0.25	0.002	0.3	1
Box Compacted	4	0.1	0.001	0.1	1
Non-Processible Boxed	5	0.1	0.001	0.1	1
Non-Processible Drummed	5	0.05	0.001	0.1	1
Moderator Resin (Unshielded)	4	0.1	0.001	0.1	1
Moderator Resin (Shielded)	3	0.05	0.001	0.1	1

Dispersion and Consequence

The concentrations of radionuclides and non-radiological species in air in the underground emplacement room were calculated using Equation 7-12 based on the assumption that the source term would be driven by the ventilation rate (18 m^3/s) during the duration of exposure to the public. The source term emission rate into air was estimated using Equation 7-18 over the 1 hour time of exposure to the public. The concentration of radionuclides and non-radiological species in the vicinity of the public was calculated with Equation 7-15 based on a 1 hour time of exposure at the nearest Bruce nuclear site boundary.

In addition to consequences from airborne releases, the external radiation dose to workers was also modelled using MicroShield based on the assumptions described below.

- Only the first row of packages from the waste emplacement room entrance was modelled. The configuration of the waste packages was different for each selected waste category:

  - LLW

    Bottom ash: 3 per row and stacked 3 rows high.

    Box compacted: 4 per row and stacked 4 rows high.

    Non-processible boxed: 4 per row and stacked 5 rows high.

    Non-processible drummed: 4 per row and stacked 5 rows high.

    A minimum 50 mm gap between packages.

  - ILW

    Moderator resin (unshielded): 4 per row and stacked 2 rows high with a minimum 50 mm gap between packages.

    Moderator resin (shielded): 3 per row with a 300 mm gap between packages.

- The package breach was modelled as a slice based on the fraction of the height of the packages breached (proportional to DR in Table A-45) with no shielding material. Effectively, the worker is exposed to an unshielded fraction DR of the waste.

- The dose point was assumed to be along the centreline, 1 m from the ground (i.e., centre of the body of a person, and thus a good estimate of the whole body dose), and at a distance of 2 m from the source.

- The total external dose is the sum of the dose from intact packages and the dose from package breach.

Results

Potential Impact of Radionuclides

Table A-46 shows that total radionuclide doses to workers over a 5 minute period (through inhalation, immersion and external radiation) are much less than the dose limit for workers (50 mSv) for any of the assessed waste categories. Similarly, Table A-47 shows that the total dose to the public (through inhalation and immersion) over a 1 hour exposure duration is much less than the 1 mSv public dose limit for any of the assessed waste categories.

Table A-46: Dose to Workers - In Room Waste Package Breach

Selected Waste Category	Inhalation (mSv)	Immersion (mSv)	External Radiation (mSv)	Total (mSv)
Bottom Ash (Old)	1.7E-02	1.5E-04	1.3E-01	1.5E-01
Box Compacted	2.9E-03	6.9E-06	2.2E-02	2.5E-02
Non-Processible Boxed	1.3E-03	1.6E-05	8.2E-02	8.3E-02
Non-Processible Drummed	1.4E-03	1.3E-06	3.6E-02	3.7E-02
Moderator Resin (Unshielded)	6.1E-02	1.1E-03	2.7E+00	2.8E+00
Moderator Resin (Shielded)	1.1E-01	4.1E-03	6.9E-01	8.0E-01

Table A-47: Dose to Public - In Room Waste Package Breach

Selected Waste Category	Inhalation (mSv)	Immersion (mSv)	Total (mSv)
Bottom Ash (Old)	2.2E-05	4.3E-07	2.2E-05
Box Compacted	4.9E-06	2.0E-08	5.0E-06
Non-Processible Boxed	2.0E-06	4.5E-08	2.1E-06
Non-Processible Drummed	2.4E-06	3.7E-09	2.4E-06
Moderator Resin (Unshielded)	9.2E-05	3.3E-06	9.5E-05
Moderator Resin (Shielded)	1.4E-04	1.2E-05	1.5E-04

Potential Impact of Non-Radiological Species

Table A-48 and Table A-49 show the ratios of air concentration to worker IDLH and public PAC 1 criteria, respectively. The air concentrations are less than non-radiological criteria near workers or the public.

Table A-48: Ratios of Air Concentration to Worker IDLH - In Room Waste Package Breach

Non-Radiological Species	Bottom Ash (Old)	Box Compacted	Non-Processible Boxed	Non-Processible Drummed	Moderator Resin (Unshielded)	Moderator Resin (Shielded)
Antimony	3.9E-03	2.5E-05	3.3E-06	8.0E-07	1.6E-07	1.2E-07
Arsenic	4.6E-03	3.1E-05	4.2E-06	1.0E-06	1.1E-06	7.9E-07
Barium	1.1E-02	7.9E-05	1.0E-05	2.5E-06	2.2E-06	1.7E-06
Beryllium	0.0E+00	0.0E+00	9.6E-06	2.3E-06	2.4E-06	1.8E-06
Cadmium	1.3E-04	9.2E-04	1.2E-04	3.0E-05	1.5E-05	1.2E-05
Chromium	5.3E-02	3.5E-04	5.4E-03	1.3E-03	9.7E-06	7.3E-06
Cobalt	1.2E-03	7.8E-06	1.0E-06	2.4E-07	5.8E-07	4.3E-07
Copper	4.6E-02	2.9E-04	6.6E-03	1.6E-03	8.5E-05	6.4E-05
Lead	2.5E-02	1.7E-04	5.2E-03	1.3E-03	1.1E-05	7.9E-06
Manganese	1.5E-03	9.2E-05	1.7E-05	4.2E-06	5.8E-07	4.3E-07
Mercury	1.3E-03	3.5E-05	7.6E-06	1.8E-06	7.1E-07	5.4E-07
Nickel	3.9E-02	2.6E-04	8.7E-04	2.1E-04	5.3E-03	4.0E-03
Selenium	0.0E+00	5.9E-04	8.0E-05	1.9E-05	9.4E-05	7.1E-05
Strontium	N/D	N/D	N/D	N/D	N/D	N/D
Uranium	0.0E+00	0.0E+00	1.2E-05	2.8E-06	1.1E-07	7.9E-08
Zinc	1.5E-02	1.1E-04	3.1E-05	7.3E-06	9.4E-06	7.1E-06
Zirconium	2.1E-03	1.4E-05	1.9E-06	4.7E-07	1.1E-07	7.9E-08
Asbestos	0.0E+00	0.0E+00	N/D	N/D	0.0E+00	0.0E+00
Dioxins & Furans	N/D	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00

Table A-49: Ratios of Air Concentration to Public PAC 1 - In Room Waste Package Breach

Non-Radiological Species	Bottom Ash (Old)	Box Compacted	Non-Processible Boxed	Non-Processible Drummed	Moderator Resin (Unshielded)	Moderator Resin (Shielded)
Antimony	9.3E-05	6.0E-07	8.0E-08	1.9E-08	3.9E-09	2.9E-09
Arsenic	1.9E-05	1.2E-07	1.7E-08	4.0E-09	4.2E-09	3.2E-09
Barium	1.1E-04	7.8E-07	1.0E-07	2.5E-08	2.2E-08	1.6E-08
Beryllium	0.0E+00	0.0E+00	2.6E-06	6.3E-07	6.5E-07	4.9E-07
Cadmium	9.6E-06	6.7E-05	8.9E-06	2.1E-06	1.1E-06	8.3E-07
Chromium	1.3E-02	8.5E-05	1.3E-03	3.1E-04	2.3E-06	1.7E-06
Cobalt	9.3E-05	6.2E-07	8.1E-08	1.9E-08	4.6E-08	3.5E-08
Copper	5.0E-03	3.2E-05	7.2E-04	1.7E-04	9.3E-06	7.0E-06
Lead	4.0E-03	2.7E-05	8.3E-04	2.0E-04	1.7E-06	1.3E-06
Manganese	6.2E-05	3.7E-06	6.9E-07	1.7E-07	2.3E-08	1.7E-08
Mercury	1.2E-05	3.4E-07	7.3E-08	1.8E-08	6.9E-09	5.1E-09
Nickel	1.5E-04	1.0E-06	3.5E-06	8.3E-07	2.1E-05	1.6E-05
Selenium	0.0E+00	2.4E-07	3.2E-08	7.7E-09	3.8E-08	2.8E-08
Strontium	3.7E-07	2.4E-09	4.0E-10	9.6E-11	1.2E-11	9.0E-12
Uranium	0.0E+00	0.0E+00	4.6E-08	1.1E-08	4.2E-10	3.2E-10
Zinc	6.2E-04	4.2E-06	1.2E-06	2.9E-07	3.8E-07	2.8E-07
Zirconium	1.3E-06	8.2E-09	1.2E-09	2.8E-10	6.4E-11	4.8E-11
Asbestos	0.0E+00	0.0E+00	5.0E-04	1.2E-04	0.0E+00	0.0E+00
Dioxins & Furans	2.4E-06	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00

A.3.3 Container Breach (High Energy)

A.3.3.1 Cage Fall with Waste Package Breach

Scenario Description

In a highly unlikely "what if" scenario, due to mechanical failure of the hoisting system (i.e., cables) and the emergency brakes (e.g., safety dogs), the cage and the waste packages inside the cage may fall down the shaft into the shaft bottom located 30 m below the underground DGR working level. This high energy breach accident is expected to cause all LLW and non-robust ILW packages (e.g., resin liners) packages to release their entire content. Robust ILW packages including retube waste containers are assumed to breach and release contents.

The maximum number of waste packages involved in this accident was based on the maximum number of waste packages that can be accommodated in a cage.

Source Term

Parameters used for the calculation of the source terms released (Equation 7-5) due to this accident scenario are summarized in Table A-50 for the waste categories assessed. It should be noted that an LPF of 0.1 is assigned to the source term calculations, since the containers are likely to be partially covered by cage debris.

Table A-50: Source Term Parameters - Cage Fall with Waste Package Breach

Selected Waste Category	# of Packages	DR	ARF	RF	LPF
Bottom Ash (Old)	2	1	0.01	0.3	0.1
Box Compacted	2	1	0.01	0.2	0.1
Non-Processible Boxed	3	1	0.01	0.2	0.1
Non-Processible Drummed	3	1	0.01	0.2	0.1
Moderator Resin (Unshielded)	2	1	0.01	0.2	0.1
Moderator Resin (Shielded)	1	1	0.01	0.2	0.1
Retube- End Fittings	1	1	0.001	0.2	0.1

Dispersion and Consequence

The concentrations of radionuclides and non-radiological species in air in the shaft were calculated using Equation 7-12 based on the assumption that the source term would be driven by the ventilation rate in the shaft station area. The ventilation flow rate into shaft station area is much lower (18 m^3/s) than the 100 m^3/s rate in main shaft above the impact point. The source term emission rate into air was estimated using Equation 7-18 over the 1 hour time of exposure to the public. The maximum concentration of radionuclides and non-radiological species in the vicinity of the public was calculated with Equation 7-15 based on a 1 hour time of exposure at the nearest Bruce nuclear site boundary.

In addition to consequences from airborne releases, the external radiation dose to workers was also modelled using MicroShield based on the assumptions described below.

- The waste packages fall down the main shaft to the shaft bottom.
- For LLW and ILW, the package breach was modelled as a complete spill of the contents in the package into the shaft bottom.
- The space between the spilled waste and a worker at the shaft station level was treated approximately as an effective shield of 2 m of rock (assumed to be calcium carbonate) between the dose point and source.
- The spill was modelled as a cylindrical volume, with diameter equal to the diameter of the main shaft (6.5 m). The height of the spill was adjusted to keep the original volume constant.
- The dose point is along the centreline, 1 m from the ground (i.e., centre of the body of a person, and thus a good estimate of the whole body dose), and at a distance of 2 m from the source.
- The total external dose is the sum of the dose from package breach and the dose from an intact package.

Results

Potential Impact of Radionuclides

Table A-51 shows that total radionuclide doses to workers over a 5 minute period (through inhalation, immersion and external radiation) are much less than the dose limit for workers (50 mSv) for any of the assessed waste categories. Similarly, Table A-52 shows that the total dose to the public (through inhalation and immersion) over a 1 hour exposure duration is much less than the 1 mSv public dose limit for any of the assessed waste categories.

Potential Impact of Non-Radiological Species

Table A-53 and Table A-54 show the ratios of air concentration to worker IDLH and public PAC 1 criteria, respectively. The air concentrations are less than non-radiological criteria for workers and the public.

Table A-51: Dose to Workers - Cage Fall with Waste Package Breach

Selected Waste Category	Inhalation (mSv)	Immersion (mSv)	External Radiation (mSv)	Total (mSv)
Bottom Ash (Old)	2.3E-02	2.0E-04	< 1.E-06	2.4E-02
Box Compacted	1.9E-02	3.4E-05	< 1.E-06	2.0E-02
Non-Processible Boxed	1.4E-02	9.4E-05	< 1.E-06	1.4E-02
Non-Processible Drummed	1.7E-02	1.6E-05	< 1.E-06	1.7E-02
Moderator Resin (Unshielded)	5.7E-01	5.7E-03	< 1.E-06	5.8E-01
Moderator Resin (Shielded)	1.4E+00	2.7E-02	< 1.E-06	1.4E+00
Retube- End Fittings	5.6E+00	2.3E-01	< 1.E-06	5.8E+00

Table A-52: Dose to Public - Cage Fall with Waste Package Breach

Selected Waste Category	Inhalation (mSv)	Immersion (mSv)	Total (mSv)
Bottom Ash (Old)	2.9E-05	5.7E-07	3.0E-05
Box Compacted	3.3E-05	9.9E-08	3.3E-05
Non-Processible Boxed	2.1E-05	2.7E-07	2.1E-05
Non-Processible Drummed	2.9E-05	4.5E-08	2.9E-05
Moderator Resin (Unshielded)	8.5E-04	1.6E-05	8.7E-04
Moderator Resin (Shielded)	1.7E-03	7.8E-05	1.8E-03
Retube- End Fittings	3.4E-03	6.7E-04	4.1E-03

Table A-53: Ratios of Air Concentration to Worker IDLH - Cage Fall with Waste Package Breach

Non-Radiological Species	Bottom Ash (Old)	Box Compacted	Non-Processible Boxed	Non-Processible Drummed	Moderator Resin (Unshielded)	Moderator Resin (Shielded)	Retube-End Fittings
Antimony	5.2E-03	2.5E-04	4.0E-05	1.9E-05	1.6E-06	1.6E-06	1.7E-06
Arsenic	6.2E-03	3.1E-04	5.0E-05	2.4E-05	1.1E-05	1.1E-05	1.1E-04
Barium	1.5E-02	7.9E-04	1.3E-04	6.0E-05	2.2E-05	2.2E-05	7.8E-10
Beryllium	0.0E+00	0.0E+00	1.1E-04	5.5E-05	2.4E-05	2.4E-05	4.8E-09
Cadmium	1.8E-04	9.2E-03	1.5E-03	7.1E-04	1.5E-04	1.5E-04	1.1E-07
Chromium	7.1E-02	3.5E-03	6.5E-02	3.1E-02	9.7E-05	9.7E-05	2.8E-02
Cobalt	1.5E-03	7.8E-05	1.2E-05	5.8E-06	5.8E-06	5.8E-06	6.0E-05
Copper	6.1E-02	2.9E-03	7.9E-02	3.8E-02	8.5E-04	8.5E-04	1.3E-04
Lead	3.3E-02	1.7E-03	6.3E-02	3.0E-02	1.1E-04	1.1E-04	9.7E-08
Manganese	2.1E-03	9.2E-04	2.1E-04	1.0E-04	5.8E-06	5.8E-06	9.2E-05
Mercury	1.7E-03	1.8E-04	4.6E-05	2.2E-05	3.6E-06	3.6E-06	1.5E-07
Nickel	5.2E-02	2.6E-03	1.0E-02	5.0E-03	5.3E-02	5.3E-02	1.6E-03
Selenium	0.0E+00	3.0E-03	4.8E-04	2.3E-04	4.7E-04	4.7E-04	3.4E-06
Strontium	N/D	N/D	N/D	N/D	N/D	N/D	N/D
Uranium	0.0E+00	0.0E+00	1.4E-04	6.7E-05	1.1E-06	1.1E-06	2.9E-09
Zinc	2.1E-02	1.1E-03	3.7E-04	1.8E-04	9.4E-05	9.4E-05	1.4E-07
Zirconium	2.8E-03	1.4E-04	2.3E-05	1.1E-05	1.1E-06	1.1E-06	5.2E-06
Asbestos	0.0E+00	0.0E+00	N/D	N/D	0.0E+00	0.0E+00	0.0E+00
Dioxins & Furans	N/D	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	N/D

Table A-54: Ratios of Air Concentration to Public PAC 1 - Cage Fall with Waste Package Breach

Non-Radiological Species	Bottom Ash (Old)	Box Compacted	Non-Processible Boxed	Non-Processible Drummed	Moderator Resin (Unshielded)	Moderator Resin (Shielded)	Retube-End Fittings
Antimony	1.2E-04	6.0E-06	9.6E-07	4.6E-07	3.9E-08	3.9E-08	4.2E-08
Arsenic	2.5E-05	1.2E-06	2.0E-07	9.6E-08	4.2E-08	4.2E-08	4.3E-07
Barium	1.5E-04	7.8E-06	1.2E-06	5.9E-07	2.2E-07	2.2E-07	7.7E-12
Beryllium	0.0E+00	0.0E+00	3.1E-05	1.5E-05	6.5E-06	6.5E-06	1.3E-09
Cadmium	1.3E-05	6.7E-04	1.1E-04	5.1E-05	1.1E-05	1.1E-05	7.8E-09
Chromium	1.7E-02	8.5E-04	1.6E-02	7.5E-03	2.3E-05	2.3E-05	6.7E-03
Cobalt	1.2E-04	6.2E-06	9.7E-07	4.6E-07	4.6E-07	4.6E-07	4.8E-06
Copper	6.6E-03	3.2E-04	8.6E-03	4.1E-03	9.3E-05	9.3E-05	1.4E-05
Lead	5.3E-03	2.7E-04	1.0E-02	4.8E-03	1.7E-05	1.7E-05	1.6E-08
Manganese	8.2E-05	3.7E-05	8.3E-06	4.0E-06	2.3E-07	2.3E-07	3.7E-06
Mercury	1.6E-05	1.7E-06	4.4E-07	2.1E-07	3.4E-08	3.4E-08	1.4E-09
Nickel	2.1E-04	1.0E-05	4.2E-05	2.0E-05	2.1E-04	2.1E-04	6.4E-06
Selenium	0.0E+00	1.2E-06	1.9E-07	9.2E-08	1.9E-07	1.9E-07	1.4E-09
Strontium	4.9E-07	2.4E-08	4.8E-09	2.3E-09	1.2E-10	1.2E-10	1.1E-09
Uranium	0.0E+00	0.0E+00	5.6E-07	2.7E-07	4.2E-09	4.2E-09	1.2E-11
Zinc	8.2E-04	4.2E-05	1.5E-05	7.0E-06	3.8E-06	3.8E-06	5.4E-09
Zirconium	1.7E-06	8.2E-08	1.4E-08	6.7E-09	6.4E-10	6.4E-10	3.1E-09
Asbestos	0.0E+00	0.0E+00	6.0E-03	2.9E-03	0.0E+00	0.0E+00	0.0E+00
Dioxins & Furans	3.1E-06	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00

A.4 Others

A.4.1 Ventilation System Failure

A ventilation system failure scenario was modelled and was used to estimate the minimum time in which the airborne H-3 and C-14 concentrations in a full room will reach the DAC limits for workers. See Section 7.5.3.5 for the modelling. The ventilation failure model was developed based on the void volume of a completely filled room of 60% in Panel 1 and 40% in Panel 2. It is conservative to use this void fraction for concentration calculations, as the void fraction is larger and hence airborne concentration would be smaller during room filling.

For complete ventilation system failure, the shortest time in which the H-3 concentration in a room will reach the DAC limit was calculated to be 14 hours. The shortest time in which the C-14 concentration in a room will reach the DAC limit was calculated to 36 hours. These times were calculated using Equations 7-26 and 7-27, and information from Table A-55.

Thus, in the case of a ventilation system failure, workers exposed to H-3 and C-14 for a 5 minute time period would be subjected to air concentrations much less than the DAC.

Table A-55: Input Parameter Values for the Minimum Time Ventilation Calculations

Input Parameters	H-3	C-14
Maximum Ventilated Inventory [a] (Bq)	3.2E+14	1.8E+15
Fractional Release Rate [b] (/year)	4.2E-03	5.0E-04
Average Void Volume [c] (m^3)	6.0E+03	5.0E+03
Ventilation Rate [d] (m^3/s)	18	18
DAC Limit [e] (Bq/m^3)	3.7E+05	7.4E+05

Notes:
a. Section 7.4.3.1.
b. From Table 7-9.
c. Based on a profile 1 emplacement room in Panel 2, with a voidage of 40%.
d. From Table 7-35.
e. From Table 7-2.

A.5 Sample Calculation for LLW Outdoor Fire

A sample calculation is given below for the LLW outdoor fire scenario. Public exposure to box compacted waste was used as an example.

A.5.1 Source Term and Release Rate

The source term is calculated using the following equation:

$$Q = MAR \times DR \times ARF \times RF \times LPF \quad \text{(A-1)}$$

where:

Q = Source term (Bq or µg)

MAR = Material at risk (Bq or µg)

DR = Damage ratio (-)

ARF = Airborne release fraction (-)

RF = Respirable fraction (-)

LPF = Leakpath factor (-).

The rate of release of contaminants is calculated using the following equation:

$$QR = Q / T_{FD} \quad \text{(A-2)}$$

where:

QR = Source term release rate (Bq/s or µg/s)

T_{FD} = Fire duration (s), time taken to burn all affected waste

Table A-56 provides the source term parameter values and radionuclide concentrations for the box compacted waste. Table A-57 provides the non-radiological species concentrations in the box compacted waste. The calculated values for MAR, Q, and QR for the radionuclides and non-radiological species are also given in Table A-56 and Table A-57, respectively.

A.5.2 Consequence Calculations

The average rate of emission, ER_O of material from an outdoors fire accident scenario depends on the source term, QR, assumed to be constant over this time:

$$ER_O = QR \text{ (fire, outdoors)} \quad (A\text{-}3)$$

The airborne concentration at the public receptor is given by the following equation:

$$C_P = ER_O \times ADF \quad (A\text{-}4)$$

where:

C_P = Average air concentration near public receptor (Bq/m^3 or µg/m^3)

ADF = = Atmospheric dilution factor (s/m^3)

For an above ground fire, an ADF of 4.3×10^{-6} s/m^3 was assumed as a conservative estimate for public receptors located at 1.1 km from the point of emission, due to a buoyant fire plume rise (see Table 7-36).

The calculated ER_0 and C_p for radionuclides and non-radiological species are given in Table A-56 and Table A-57 respectively.

Public radionuclide dose exposure through inhalation and immersion are calculated using the following equations:

$$PD_{INH} = C_P \times INH_P \times T_{EXP\_P} \times DC_{PINH} \quad (A\text{-}5)$$

$$PD_{IMM} = (C_P / RF) \times T_{EXP\_P} \times DC_{IMM} \quad (A\text{-}6)$$

where:

PD_{INH} = Dose to the public through inhalation (mSv)

PD_{IMM} = Dose to public through immersion (mSv)

INH_P = Public inhalation rate (m^3/hr) – 0.96 m^3/hr

T_{EXP_P} = Time of exposure (hr)

DC_{PINH} = Inhalation dose coefficient for public (mSv/Bq)

DC_{IMM} = Immersion dose coefficient ((mSv/year)/(Bq/m^3)).

The public was assumed to be exposed for 1 hour to the accident release. The inhalation and immersion dose coefficients are given in Table A-56.

The calculated inhalation and immersion doses for each radionuclide are given in Table A-56. The total inhalation and immersion doses are calculated to be 1.8 $x10^{-5}$ mSv and 1.1 $x10^{-10}$ mSv, respectively.

Impacts of short-term exposure to non-radiological chemicals on members of the general public were assessed through comparing the estimated concentrations with the PAC 1 inhalation criteria. The airborne concentrations, PAC 1 values, and the ratio of airborne concentrations over PAC 1 values are given in Table A-57.

Table A-56: Source Terms and Results for Outdoor LLW Package Fire (Radiological Impacts to Public for Box Compacted Waste)

Nuclide	Conc. (Bq/m^3)	# Pkg[a]	MAR (Bq)	DR (-)	ARF (-)	RF (-)	LP (-)	T_{FD} (hr)	Q (Bq)	$QR=ER_0$ (Bq/s)	ADF (s/m^3)	C_P (Bq/m^3)	T_{EXP_P} (hr)	DC_{PINH} (mSv/Bq)	INH_p (m^3/hr)	PD_{INH} (mSv)	DC_{IMM} ((mSv/y)/(Bq/m^3))	PD_{IMM} (mSv)
Am-241	3.0E+05	8	5.5E+06	0.5	0.001	1	1	5.0	2.8E+03	1.5E-01	4.3E-06	6.6E-07	1	4.2E-02	0.96	2.7E-08	2.1E-05	1.6E-15
C-14	6.7E+06	8	1.2E+08	0.5	1	1	1	5.0	6.2E+07	3.4E+03	4.3E-06	1.5E-02	1	1.2E-08	0.96	1.7E-10	8.2E-08	1.4E-13
Cm-244	1.3E+05	8	2.4E+06	0.5	0.001	1	1	5.0	1.2E+03	6.6E-02	4.3E-06	2.9E-07	1	2.7E-02	0.96	7.4E-09	1.1E-07	3.6E-18
Co-60	8.3E+07	8	1.5E+09	0.5	0.001	1	1	5.0	7.6E+05	4.2E+01	4.3E-06	1.8E-04	1	1.0E-05	0.96	1.8E-09	3.8E-03	7.9E-11
Cs-134	6.8E+06	8	1.3E+08	0.5	0.001	1	1	5.0	6.3E+04	3.5E+00	4.3E-06	1.5E-05	1	6.6E-06	0.96	9.5E-11	2.2E-03	3.8E-12
Cs-137+ Ba-137m	7.6E+07	8	1.4E+09	0.5	0.001	1	1	5.0	7.0E+05	3.9E+01	4.3E-06	1.7E-04	1	4.6E-06	0.96	7.4E-10	8.1E-04	1.5E-11
Eu-154	2.3E+06	8	4.2E+07	0.5	0.001	1	1	5.0	2.1E+04	1.2E+00	4.3E-06	5.1E-06	1	5.3E-05	0.96	2.6E-10	1.8E-03	1.0E-12
Fe-55	3.2E+05	8	5.9E+06	0.5	0.001	1	1	5.0	2.9E+03	1.6E-01	4.3E-06	7.0E-07	1	3.8E-07	0.96	2.6E-13	0.0E+00	0.0E+00
H-3	2.8E+11	8	5.2E+12	0.5	1	1	1	5.0	2.6E+12	1.4E+08	4.3E-06	6.2E+02	1	3.0E-08	0.96	1.8E-05	0.0E+00	0.0E+00
Nb-94	1.0E+06	8	1.8E+07	0.5	0.001	1	1	5.0	9.2E+03	5.1E-01	4.3E-06	2.2E-06	1	1.1E-05	0.96	2.3E-11	2.3E-03	5.8E-13
Pu-238	6.1E+04	8	1.1E+06	0.5	0.001	1	1	5.0	5.6E+02	3.1E-02	4.3E-06	1.3E-07	1	4.6E-02	0.96	5.9E-09	1.1E-07	1.7E-18
Pu-239	1.3E+05	8	2.4E+06	0.5	0.001	1	1	5.0	1.2E+03	6.6E-02	4.3E-06	2.9E-07	1	5.0E-02	0.96	1.4E-08	1.1E-07	3.6E-18
Pu-240	1.8E+05	8	3.3E+06	0.5	0.001	1	1	5.0	1.7E+03	9.2E-02	4.3E-06	4.0E-07	1	5.0E-02	0.96	1.9E-08	1.1E-07	5.0E-18
Pu-241	4.9E+06	8	9.0E+07	0.5	0.001	1	1	5.0	4.5E+04	2.5E+00	4.3E-06	1.1E-05	1	9.0E-04	0.96	9.3E-09	2.0E-09	2.5E-18
Ru-106	6.1E+07	8	1.1E+09	0.5	0.001	1	1	5.0	5.6E+05	3.1E+01	4.3E-06	1.3E-04	1	2.8E-05	0.96	3.6E-09	3.3E-04	5.1E-12
Sb-125	1.4E+07	8	2.6E+08	0.5	0.001	1	1	5.0	1.3E+05	7.2E+00	4.3E-06	3.1E-05	1	4.8E-06	0.96	1.4E-10	5.9E-04	2.1E-12
Sr-90+ Y-90	3.6E+06	8	6.6E+07	0.5	0.001	1	1	5.0	3.3E+04	1.8E+00	4.3E-06	7.9E-06	1	3.6E-05	0.96	2.7E-10	2.8E-05	2.5E-14
Total																**1.8E-05**		**1.1E-10**

Note: Package volume is 2.3 m^3.

Table A-57: Source Terms and Results for Outdoor LLW Package Fire (Non-Radiological Impacts to Public for Box Compacted Waste)

Non-radiological Species	Concentration (kg/m^3)	# Pkg	Pkg Volume (m^3)	MAR (kg)	DR (-)	ARF (-)	RF (-)	LPF (-)	T_{FD} (hr)	Q (kg)	QR (kg/s)	ER_o (kg/s)	ADF (s/m^3)	C_p (kg/m^3)	PAC 1 ($\mu g/m^3$)	C_p/PAC 1 (-)
Antimony	7.3E-02	8	2.3	1.3E+00	0.5	0.001	1	1	5.0	6.7E-04	3.7E-08	3.7E-08	4.3E-06	1.6E-13	500	3.2E-07
Arsenic	9.2E-03	8	2.3	1.7E-01	0.5	0.001	1	1	5.0	8.5E-05	4.7E-09	4.7E-09	4.3E-06	2.0E-14	300	6.7E-08
Barium	2.3E-01	8	2.3	4.2E+00	0.5	0.001	1	1	5.0	2.1E-03	1.2E-07	1.2E-07	4.3E-06	5.1E-13	1,220	4.2E-07
Cadmium	4.9E-01	8	2.3	9.0E+00	0.5	0.001	1	1	5.0	4.5E-03	2.5E-07	2.5E-07	4.3E-06	1.1E-12	30	3.6E-05
Chromium	5.2E-01	8	2.3	9.6E+00	0.5	0.001	1	1	5.0	4.8E-03	2.7E-07	2.7E-07	4.3E-06	1.1E-12	25	4.5E-05
Cobalt	9.2E-03	8	2.3	1.7E-01	0.5	0.001	1	1	5.0	8.5E-05	4.7E-09	4.7E-09	4.3E-06	2.0E-14	60	3.3E-07
Copper	1.7E+00	8	2.3	3.1E+01	0.5	0.001	1	1	5.0	1.6E-02	8.7E-07	8.7E-07	4.3E-06	3.7E-12	220	1.7E-05
Lead	9.8E-01	8	2.3	1.8E+01	0.5	0.001	1	1	5.0	9.0E-03	5.0E-07	5.0E-07	4.3E-06	2.2E-12	150	1.4E-05
Manganese	2.7E+00	8	2.3	5.0E+01	0.5	0.001	1	1	5.0	2.5E-02	1.4E-06	1.4E-06	4.3E-06	5.9E-12	3,000	2.0E-06
Mercury	2.1E-03	8	2.3	3.9E-02	0.5	1	1	1	5.0	1.9E-02	1.1E-06	1.1E-06	4.3E-06	4.6E-12	250	1.8E-05
Nickel	1.5E-01	8	2.3	2.8E+00	0.5	0.001	1	1	5.0	1.4E-03	7.7E-08	7.7E-08	4.3E-06	3.3E-13	600	5.6E-07
Selenium	3.5E-03	8	2.3	6.4E-02	0.5	1	1	1	5.0	3.2E-02	1.8E-06	1.8E-06	4.3E-06	7.7E-12	600	1.3E-05
Strontium	7.3E-02	8	2.3	1.3E+00	0.5	0.001	1	1	5.0	6.7E-04	3.7E-08	3.7E-08	4.3E-06	1.6E-13	125,000	1.3E-09
Zinc	3.1E+00	8	2.3	5.7E+01	0.5	0.001	1	1	5.0	2.9E-02	1.6E-06	1.6E-06	4.3E-06	6.8E-12	3,000	2.3E-06
Zirconium	2.0E-02	8	2.3	3.7E-01	0.5	0.001	1	1	5.0	1.8E-04	1.0E-08	1.0E-08	4.3E-06	4.4E-14	10,000	4.4E-09

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