ONTARIO POWER GENERATION

Laurie Swami
Vice President
Nuclear Services

889 Brock Road Pickering, ON L1W 3J2

Tel: 905.839.6746 ext. 5306
laurie.swami@opg.com

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Project ID: 10-60004

Dr. Stella Swanson
Chair, Joint Review Panel
Deep Geologic Repository Project

c/o Canadian Nuclear Safety Commission
280 Slater Street
Ottawa, Ontario
K1P 5S9

Dear Dr. Swanson:

Deep Geologic Repository Project for Low and Intermediate Level Waste – Submission of Responses to Information Requests in Package #13

The purpose of this letter is to provide OPG's responses to Information Requests (IRs) in IR Package #13 (Reference 1).

The responses to the Information Requests are provided in the Attachment.

OPG continues to monitor the recovery activities and examine the available root cause reports at the Waste Isolation Pilot Plant (WIPP) and will provide an update at the upcoming hearing session if significant new information becomes available.

An updated Tracking Table showing how all submissions to date, including those for IR Package #13, link to various sections in the documents submitted on April 14, 2011 (References 2 and 3) will be submitted with responses to IR Package #12b (Reference 4).

If you have questions on the above, please contact Mr. Allan Webster, Director, Nuclear Regulatory Affairs, at (905) 623-6670, ext. 3326.

Sincerely,

Laurie Swami
Vice President, Nuclear Services
Ontario Power Generation

Attach.

cc. Dr. J. Archibald – Joint Review Panel c/o CNSC (Ottawa)
Dr. G. Muecke – Joint Review Panel c/o CNSC (Ottawa)
P. Elder – CNSC (Ottawa)
D. Wilson – NWMO (Toronto)

References:

1. JRP letter from Dr. Stella Swanson to Laurie Swami, “Information Request Package #13 from the Joint Review Panel”, March 21, 2014, CD# 00216-CORR-00531-00231.

2. OPG letter from Albert Sweetnam to JRP Chair, “Submission of Information in Support of OPG’s Licence Application for a Deep Geologic Repository for Low and Intermediate Level Waste”, April 14, 2011, CD# 00216-CORR-00531-00090.

3. OPG letter from Albert Sweetnam to JRP Chair, “Submission of an Environmental Impact Statement for a Deep Geologic Repository for Low and Intermediate Level Waste”, April 14, 2011, CD# 00216-CORR-00531-00091.

4. JRP letter from Dr. Stella Swanson to Laurie Swami, "Information Request Package #12b from the Joint Review Panel”, April 15, 2014, CD# 00216-CORR-00531-00238.

ATTACHMENT

Attachment to OPG letter, Ms. Laurie Swami to Dr. Stella Swanson, "Deep Geologic Repository Project for Low and Intermediate Level Waste – Submission of Responses to Information Requests in Package #13"

May 9, 2014

CD#: 00216-CORR-00531-00235

OPG Responses to Information Requests in IR Package #13 from Joint Review Panel

OPG Responses to Information Requests in IR Package #13 from Joint Review Panel

IR#	EIS Guidelines Section	Information Request and Response
EIS 13-514	• Section 8.1, General Information and Design Description	***Information Request:*** *Provide the following:* • *The results and evaluations of the re-runs of postclosure safety assessment models at a similar level of detail and clarity as that provided in NWMO DGR-TR-2011-25 "Postclosure Safety Assessment";* • *An assessment of how the revised inventories will affect the pre-closure safety evaluation of the DGR, with special emphasis on the occupational health and safety of the workforce, as well as radiation protection requirements. This assessment should also address the impact of the revised inventories on the possible future expansion of the DGR;* • *An assessment of how the revised inventories would affect the environmental effects of accidents, malfunctions and malevolent acts, with emphasis on the pre-closure phase;* • *A Waste Inventory Verification Plan, similar to the Geoscientific Verification Plan, which provides clear objectives, activities, and time-lines of future endeavours to improve the accuracy of the Reference Waste Inventory. The response should also include any plans for an independent expert evaluation of the methodology and verification procedures; and* • *Clarification of the methodology used to determine radioisotope concentrations and activity levels in filter resins.* ***Context:*** *Recent correspondence between Dr. Frank Greening and the NWMO (See CEAR numbers 1777, 1808, 1809, 1810 and 1811) has raised questions regarding the accuracy of OPG's 2010 Reference Waste Inventory of L&ILW that would be emplaced into the proposed DGR. These questions concern radionuclide concentrations in CANDU pressure tubes and garter springs for which the concentrations of some radioisotopes appear to have been significantly underestimated or not estimated at all. The underestimates appear to be due to the use of calculated values and scaling factors, rather than measured values.* *In its February 20, 2014 response to Dr. Greening, the NWMO stated that:* • *the estimated tritium content of the PT waste is approximately 300 times higher than in the 2010 Reference Waste Inventory.* • *an inventory estimate of Cm-244 is not included in the 2010 Reference Waste Inventory. Cm-244 is the dominant PT transuranic radionuclide in terms of activity at reactor shutdown.* • *for the pressure tube wastes, the values for Cs-134 and Sb-125 are low by a factor of 3-4, and Cs-137 is significantly underestimated by a factor of 2300.*

IR#	EIS Guidelines Section	Information Request and Response
		• *the garter spring activity was not included in the 2010 Reference Waste Inventory. Although the garter spring mass is small, the total amounts of Co-60, Ni-63 and Ni-59 in the garter springs are significant compared to the amounts in the pressure tubes in part because the garter springs are primarily nickel. The ratio change in total DGR inventory at 2062 is 1.5 for Ni-59 and 2.2 for Ni-63.* *The NWMO has also noted that some of the radioisotopes that were underestimated (H-3, Cs-137, and Cm-244) have short half-lives and would not impact the long-term safety case. The NWMO also stated that it has re-run DGR postclosure assessment models using revised pressure tubes inventories for several key scenarios and calculation cases. It concluded that the changes in the waste inventories did not change the safety case conclusions for the DGR.* *While the waste inventory is a work in progress and cannot be finalized at this stage of the Project, additional quality assurance would be provided by a Waste Inventory Verification Plan.*
		OPG Response: The response is provided in three parts: postclosure safety, preclosure safety and inventory verification. <u>Postclosure Safety</u> The postclosure safety assessment models as presented in NWMO DGR-TR-2011-25 "Postclosure Safety Assessment" have been rerun with revised inventories. These revisions are as noted in the Information Request. The revised radionuclide inventories are primarily due to surface deposit from coolant on the pressure tubes. Therefore they are assumed to be located in the surface layer of the pressure tubes, and assumed to be quickly released on contact with water. Other than this change, the revisions have no effect on the other postclosure safety assessment assumptions or models. Attachment A provides the results of the revised postclosure safety assessment. It shows that the changes have no significant effect on the long-term safety. This is because these changes do not significantly affect the total repository inventory, the affected radionuclides have relatively short half-lives, and the DGR design and site provide a large safety margin. <u>Preclosure Safety</u> All waste packages are required to meet the DGR Waste Acceptance Criteria. This sets limits on the gamma dose rates outside of waste packages. Pressure tube wastes (including garter springs) are handled in steel-and-concrete containers, with sufficient shielding and/or decay time to ensure that dose rate waste acceptance criteria are met. Compliance with these dose rates is part of ensuring the health and safety of the workers, and in meeting radiation protection requirements. The change in inventory does not affect the ability to meet the waste acceptance criteria.

IR#	EIS Guidelines Section	Information Request and Response
		The revised inventories would affect the releases from pressure tube containers in the event of an accident, malfunction or malevolent act resulting in breach of a container. These are robust steel-and-concrete containers, so such a breach is very unlikely. The consequences of assumed breach accidents are described in Attachment B. The dose consequences for accidents and malfunctions remain well below 1 mSv. For most malevolent acts, the dose also remains below 1 mSv. In one specific malevolent scenario the dose increased from 2 mSv to 3 mSv, for a person at the nearest Bruce nuclear site boundary. In addition, OPG has security programs in place to guard against any malevolent acts at the Bruce Nuclear Site. These increases in dose do not affect the conclusions of the preclosure safety case. The revised inventories for pressure tube wastes from refurbishment would have no significant effect on the possible future expansion of the DGR. After these wastes have been emplaced in the DGR, the relevant panels would eventually be closed off, and therefore isolated from the expansion activities. And, as discussed in the Postclosure Safety section above, these revised inventories have no effect on the postclosure safety assessment so do not constrain the ability of the DGR to accept additional wastes. <u>Inventory Verification</u> Attachment C provides the requested Waste Inventory Verification Plan. The purpose of this document is to summarize the activities underway and planned at OPG to continue to measure and verify the properties of the L&ILW arising from operations and refurbishment of OPG-owned or operated nuclear generating facilities and intended for disposal in the proposed DGR. This document has been prepared in response to Information Request EIS-12-514. The work is implemented by more specific work programs and plans within the OPG management system. The plan includes an external third party review of the waste characterization program. The plan includes a description of the main methods used for waste characterization, including measurement of radioisotopes on ion exchange resins.
EIS 13-515	• Section 12, Accidents, Malfunctions and Malevolent Acts	***Information Request:*** *Provide a brief description of the recent incidents at the Waste Isolation Pilot Plant (WIPP) near Carlsbad, New Mexico. Include an explanation of the relevance of these incidents to worker and public health and safety (both occupational health and safety and radiation protection requirements) at the proposed DGR under normal and accident conditions.* *Describe how the consequences of such incidents might or might not fall within what OPG modeled for its analysis of accidents, malfunctions, and malevolent acts.* ***Context:*** *Recent events at the WIPP have received media attention and raised concerns with interested parties. The requested information will provide context for the Panel's review of the proposed DGR.*

IR#	EIS Guidelines Section	Information Request and Response
		OPG Response: There have been two recent incidents at the United States Department of Energy (US DOE) Waste Isolation Pilot Plant (WIPP) near Carlsbad, New Mexico: 1) the February 5, 2014, mine fire, and 2) the February 14, 2014, radiological release. The events are considered by the US DOE to be independent as they occurred in different sections of the facility. OPG and NWMO are carefully reviewing the information that the US DOE is publishing with respect to these two events on their website (http://www.wipp.energy.gov/wipprecovery/recovery.html). The WIPP facility is managing disposal of transuranic radioactive wastes arising from the nuclear weapons program in the United States. Some of the wastes that are being placed in the WIPP are therefore substantially different in character than the wastes that are proposed to be placed in the OPG DGR. OPG, as a nuclear facility operator, and NWMO both engage in the ongoing process of seeking operational experience from other nuclear facility operators, including the United States DOE and other radioactive waste facility operators worldwide. Consistent with our established management system, we carefully review the available information provided by these operators and consider its direct and indirect application to our facility designs and processes. To the extent information is available on the events at the WIPP, we have reported on it below. When additional information becomes available, it will also be assessed for applicable lessons for the DGR facility, in accordance with our management system. <u>Mine Fire Event</u> The US DOE Office of Environmental Management publicly released an Accident Investigation Report of the February 5, 2014, mine fire event on March 13, 2014 (DOE 2014a). In summary, at approximately 10:45 Mountain Standard Time (MST), a fire initiated in a salt haulage truck (EIMCO Haul Truck 74-U-006B) as a result of engine fluids (hydraulic oil or diesel fuel) coming into contact with hot surfaces on the truck and igniting. Upon noticing the flames, the operator attempted to extinguish the fire, both with a portable extinguisher and with the on-board fire suppression system, and when unsuccessful, notified maintenance and his supervisor of the fire. A series of activities were undertaken to notify underground personnel to evacuate to the surface via the waste hoist. By approximately 11:35 MST, all underground personnel had been accounted for and medical attention provided. The Accident Investigation Report provides a full description of the event and the results of the investigation. The investigators determined that the incident was preventable and a full listing of observations and conclusions is provided in the US DOE Accident Investigation Report. The key findings were related to: • Inadequate preventative and corrective maintenance of equipment, including safety related equipment; • Inadequate follow through of fire protection program standards into training, field procedures and re-

IR#	EIS Guidelines Section	Information Request and Response
		inforcement of acceptable field conditions by management; • Inadequate training and qualifications of operations staff for their documented emergency roles; • Elements of the emergency preparedness program were not maintained and/or tested for adequacy through simulated drills; and • Ineffectiveness of various oversight groups in identifying weaknesses and correcting identified deficiencies associated with the root cause. An underground fire such as occurred at WIPP is considered a credible event at the proposed OPG L&ILW DGR during both construction and operations phases and has been considered in the design and processes including: • Fire prevention (e.g. minimize use of combustible materials); • Fire detection equipment; • Fire suppression equipment (e.g. on-board automatic fire suppression equipment); • Communication equipment and notification systems (e.g. use of stench gas); • Use and location of portable refuge stations; and • Egress and emergency response. Although the WIPP fire event occurred on construction equipment, the potential for a fire on equipment transporting waste packages was considered in the design and safety assessment for the OPG DGR operations phase. The potential impacts to worker and public safety were assessed to be below criteria. The proposed management systems, and more specifically the project health, safety and environmental management and emergency response plans, have been described in detail in OPG’s response to IR LPSC-04-66 and discussed at the hearings October 30, 2013 (IRI 2013). A list of OPG’s responses addressing emergency response plans was provided in response to IR EIS-08-354. For completeness, OPG’s responses to the following information requests addressed various aspects of the emergency response plans: LPSC-01-09, LPSC-01-15, LPSC-01-37, LPSC-01-41, LPSC-01-45, LPSC-03-59, LPSC-03-60, LPSC-03-61, EIS-01-04, EIS-03-53, EIS-03-76, EIS-05-186, EIS-06-269, EIS-06-271. OPG’s responses to the following information requests addressed various aspects of the proposed fire detection and protection systems, as well as the assessment of fire events: LPSC-01-02, LPSC-01-10, LPSC-01-15, LPSC-01-15a, LPSC-01-16, LPSC-01-20, LPSC-01-21, LPSC-01-22, LPSC-01-26, LPSC-01-36, LPSC-01-41, LPSC-01-43, EIS-04-135, EIS-06-248, EIS-06-270, EIS-06-275, EIS-07-279, EIS-07-280, EIS-07-281, EIS-08-354, EIS-09-402, EIS-09-430, EIS-09-466, EIS-10-499. OPG has committed to the development of detailed Fire Protection Programs prior to the start of site preparation and construction, and future operations phase activities. This includes the development of Fire Hazard Analyses (FHA) which support specific fire protection plans for the DGR activities. Plans include required elements such as roles and responsibilities, fire response, fire assessments, managing changes that affect fire protection, work practice and

IR#	EIS Guidelines Section	Information Request and Response
		procedures, fire planning, inspection and maintenance of fire protection systems, quality assurance, housekeeping, storage and handling of hazardous goods, control of ignition sources, transient material, reporting and drills. The fire protection measures and processes developed for the DGR project will be subject to regulatory oversight by the CNSC and other regulating bodies. The Fire Protection Program and Emergency Response Plan are licensing requirements and identified by the CNSC in their response to Undertaking No. 67 (CEAA 1739) as hold points for regulatory review and acceptance prior to the start of site preparation and construction. OPG is confident that the measures and processes we have established will prevent or mitigate a similar event at the proposed OPG DGR. Documented programs will be translated thoroughly into training, field procedures and management expectations. Implementation of a common Project Management System to all staff and contractors, and continued monitoring and improvement (i.e. Plan-Do-Check-Act), will help to ensure common understanding and testing of processes. <u>Radiological Release Event</u> At approximately 23:14 Mountain Standard Time (MST) on February 14, 2014, there was an event/incident in the underground repository at WIPP that resulted in a radiological release (americium and plutonium). The release was detected by a continuous air monitor located underground and the exhaust was directed through a high-efficiency particulate air filter at the surface exhaust building. Some exhaust air by-passed the filters as a result of ineffective dampers and was discharged directly to the environment. The DOE has confirmed that there were no personnel underground at the time of the incident, no worker injuries resulting from the event, the monitoring system detected the release, and the mitigation systems responded to reduce surface emissions. The radioactivity concentrations measured at the surface were well below regulatory limits for public and worker exposure and they quickly decreased to around historic background levels. The US DOE Office of Environmental Management publicly released a Phase 1 Accident Investigation Report of the Radiological Release Event at the Waste Isolation Plant on April 24th (DOE 2014b). The report presents valuable insight and information surrounding the root and contributing causes specific to the surface release of radioactive material from underground. The findings of this report are quite similar to those from the vehicle fire event. There is a common theme that is largely related to a degraded safety culture, ineffective programs and program implementation as well as training. The following highlights the key aspects of the report and provides an OPG perspective of our practices in these same areas: • Effectiveness of the WIPP Nuclear Safety Program, specifically related to the reduction in conservatism in the Documented Safety Analysis and corresponding Technical Safety Requirements; • OPG has maintained an effective nuclear safety program which has ensured safe reactor operations for several decades. The program is well guarded against degradation by OPG's programmatic controls which not only monitor and measure its effectiveness, but seeks opportunities for improvement. OPG expects this

IR#	EIS Guidelines Section	Information Request and Response

		level of nuclear safety program rigour will continue into future DGR operations.
		• Implementation of the Emergency Management System related to adequately recognize, categorize and implement protective actions in a timely manner;
		  ◦ Prior to the DGR receiving its operating licence, OPG will have demonstrated to the CNSC that it has a strong and sustainable emergency management system. This program will not only be reflective of those developed for our safe operations, but will consider the unique potential hazards of being deep underground. OPG has a strong performance history in this area and is confident it will further improve with time as we enter into DGR operations.
		• The site Safety Culture and lack of a questioning attitude, reluctance to report issues to management, and an acceptance of degraded equipment and conditions;
		  ◦ This is an area where OPG’s overriding priority of safety is routinely monitored and measured for effectiveness, not only by ourselves but also by our industry peers (e.g., WANO/INPO) and the CNSC. This safety culture is company wide and not limited to the large nuclear fleet. For example, this was demonstrated through the safe construction of the recently competed Niagara Tunnel project. It is expected this will continue and be pervasive throughout all phases of the DGR project as well.
		• Implementation of the Conduct of Operations to DOE requirements;
		  ◦ OPG is well regarded for its strong conduct of operations program. The strength of this program is not only demonstrated through decades of safe reactor and waste facility operations but also by OPG’s history of regulatory compliance. OPG will transfer all relevant aspects of these programs into the DGR operations programs prior to receiving its operating licence. To further ensure a robust conduct of operations program, OPG will review and incorporate the appropriate lessons learned from WIPP operations as well as other key repository and mining related operating experience.
		• Ineffective maintenance program, specifically related to the critical equipment and components contributing to the radiological detection and release;
		  ◦ OPG has long recognized the value in proper maintenance of its critical equipment and components. Maintaining this equipment not only contributes to reliable generation of power, but more importantly to the safety of its workers and the public. OPG has an active maintenance program and will apply this to the DGR.
		• Ineffective Radiation Protection Program to Federal requirements, including training, qualifications, equipment, instruments, and auditing;
		  ◦ OPG recognizes that a robust radiation protection program is at the center of assuring worker and public safety. Similar to OPG’s programs noted above, OPG has a long history of maintaining an effective and regulatory compliant radiation protection program. This is accomplished through a commitment to regulatory compliance, well trained and qualified staff, staying current with advancements in technology and practices and by a continuous view to the industry to learn and improve from operating experience. Prior to placing the DGR into operations, OPG will have demonstrated to the CNSC that it has established an effective radiation protection program which meets all applicable regulatory requirements.

IR#	EIS Guidelines Section	Information Request and Response
		• Ineffective execution of DOE oversight, both from the Carlsbad Field Office and DOE Headquarters. • OPG's company wide operations depend heavily on the expertise and skill of a large number of contractors. OPG recognizes that any breakdown in its oversight of contractors can lead to risk of worker, environment and public safety. Therefore OPG builds strong terms and conditions into its contracts and provides the necessary level of oversight to ensure predictable, safe outcomes. Moreover, and unlike the DOE, OPG conducts its contracted work under regulatory oversight. Should a contractor fail in its duty to safety, the regulating body will hold OPG ultimately accountable for a failure in oversight. Conversely, the DOE has no regulatory oversight in its control of contractors. In summary, OPG is accountable for its oversight of contractors in the design, construction and operations of the DGR facility. This accountability will be managed through rigorous management of contracts and direct oversight and auditing of our contractors approved programs. OPG's culture of safety, in its many forms, values the experience of the industry and continually seeks to learn and improve from it. This has been fundamental to OPG's long history of high standards, performance and regulatory compliance in its nuclear operations. It is this deep rooted safety culture that OPG expects will continue to guide and develop the programs and processes for safe DGR construction and operations. There is still more to be learned from the experiences at WIPP and OPG remains committed under our current programs which assure they are evaluated and opportunities for improvement are sought. In summary, the DGR will be operated through a system of OPG governance including appropriate management systems, programs and plans, and subject to independent regulatory oversight. As demonstrated through its current reactor and waste facility operations, OPG has well developed programs in the areas of emergency management, safety culture, human performance, radiation protection, operations and maintenance. As many of the Phase 1 Report findings are directly related to radiological operations, future operating plans and procedures specific to the DGR will consider the WIPP findings in their development. OPG has conducted a preliminary review of the recently released Phase 1 report and has made an initial determination that no design changes, including to the ventilation system, are required at this time. Further, some of the findings related to emergency management processes are similar to those described above for the fire event. OPG will continue a detailed review of the Phase 1 report to identify opportunities to incorporate specific findings into the future planning for the DGR project consistent with our management system and the regulatory process. The L&ILW DGR preclosure safety analyses (Section 7.5, OPG 2011) included the evaluation of a number of credible underground accident scenarios, including waste package breach. The consequences were determined to be well below the regulatory criteria for worker and public protection. The US DOE Phase 2 investigation will address the specific root cause of the release from the waste package(s). As with the Phase 1 report, OPG will review it for potential lessons when it becomes available.

IR#	EIS Guidelines Section	Information Request and Response
		References: DOE. 2014a. U.S. Department of Energy, Office of Environmental Management – Accident Investigation Report, Underground Salt Haul Truck Fire at the Waste Isolation Pilot Plant February 5, 2014, March 2014. DOE. 2014b. U.S. Department of Energy, Office of Environmental Management – Accident Investigation Report, Phase 1 Radiological Release Event at the Waste Isolation Pilot Plant on February 14, 2014, April 2014. IRI. 2013. DGR Hearing Transcript Volume 25, October 30, 2013. International Reporting Inc. to the Joint Review Panel. (CEAA Registry Doc# 1741) OPG. 2011. OPG's Deep Geologic Repository for Low and Intermediate Level Waste - Preliminary Safety Report. Ontario Power Generation report 00216-SR-01320-00001 R000. Toronto, Canada. (CEAA Registry Doc# 300) CNSC. 2013. Deep Geologic Repository Project Joint Review Panel – Undertaking Response No. 67 (CEAA Registry Doc# 1739)

Attachment A to OPG Response to IR-EIS-13-514

POSTCLOSURE SAFETY IMPLICATIONS OF REVISED PRESSURE TUBE INVENTORIES

POSTCLOSURE SAFETY IMPLICATIONS OF REVISED PRESSURE TUBE INVENTORIES

CONTENTS

LIST OF TABLES

Page

LIST OF FIGURES

Page

1. INTRODUCTION

Ontario Power Generation (OPG) is proposing to build a Deep Geologic Repository (DGR) for Low and Intermediate Level Waste (L&ILW) near the existing Western Waste Management Facility at the Bruce nuclear site in the Municipality of Kincardine, Ontario. The Nuclear Waste Management Organization, on behalf of OPG, has prepared the Environmental Impact Statement (EIS) (OPG 2011a) and Preliminary Safety Report (PSR) (OPG 2011b) for the proposed repository. The EIS and PSR and supporting documentation were submitted for regulatory review in April 2011, as part of the application for a site preparation and construction licence.

The supporting documentation included a postclosure safety assessment (the “2011 PostSA”) of the long-term safety of the proposed facility undertaken by Quintessa and its subcontractors, Geofirma Engineering Limited and SENES Consultants Limited (Quintessa et al. 2011).

The safety assessment was conducted based on the projected final DGR inventory presented in OPG's 2010 Reference Inventory report (OPG 2010). Some of the inventory projections were based on estimates. An ongoing OPG waste characterization programme was underway to reduce uncertainties in these estimates. A revised inventory is planned to be prepared in support of the future application for an operating licence.

During the review process of the current Preliminary Safety Report, the estimates for the inventories of some radionuclides in pressure tube wastes have been specifically identified as significantly underestimated. The Joint Review Panel has issued an information request (EIS 13-514) asking for “the results and evaluations of the re-runs of postclosure safety assessment models at a similar level of detail and clarity as that provided in NWMO DGR-TR-2011-25 Postclosure Safety Assessment”.

The current Technical Memorandum addresses this request.

Rather than reproduce the entire content of the Postclosure Safety Assessment report (Quintessa et al. 2011), this memorandum focuses on the changes made, in particular:

- modifications to the inventory (Section 2);
- modifications to the models (Section 3);
- results and analysis (Section 4); and
- conclusions (Section 5).

All other aspects of the 2011 PostSA, such as the repository, geosphere and biosphere descriptions and the scenarios assessed, are not modified by the changes to the pressure tubes inventory and so are not replicated in the main body of this memorandum. This approach allows the impact of the revised pressure tubes inventory on the postclosure safety assessment to be evaluated in a clear and concise manner.

2. MODIFICATIONS TO THE INVENTORY OF RADIONUCLIDES IN THE PRESSURE TUBES

Pressure tubes from reactor mid-life retubing are one of more than 20 waste types considered in the reference inventory. The 2010 Reference Inventory report (OPG 2010) recognized that some radionuclides with pressure tubes would be present as a result of surface deposition from coolant, and included an estimate of this contribution.

Based on actual pressure tube data, and including the inventory of garter springs which are disposed along with pressure tubes, some radionuclides have been identified as being underestimated in the reference inventory. These are being addressed in the ongoing waste characterization programme, and revised inventory values will be used in future updates to the reference inventory. However, in this Technical Memorandum, the specific effects of these changes in radionuclide inventories on the 2011 postclosure safety assessment are presented.

The revised inventory data for the pressure tubes are provided in Table 1, including garter springs. These are interim values based on recent analysis. A more complete revision including current measurement programmes would be used as part of a future safety case update, in particular in support of an operating licence.

Table 1 also provides the revised total inventory for each radionuclide (summed over all waste streams) compared with that considered in the 2011 PostSA, and the ratio of the revised total inventory to the 2011 PostSA total inventory.

Note that the pressure tubes are not a key contributor to the total DGR inventory for many radionuclides; consequently Table 1 shows that the total DGR inventory is only increased by more than 10% for Ni-59, Ni-63, Cs-137 and Cm-244.

Table 1: Reference and Revised Inventory Data at 2062 (Assumed DGR Closure)

Radionuclide	Inventory in Pressure Tubes and Garter Springs (Bq)		Total Inventory in the DGR (Bq)		
	2011 PostSA Inventory	Revised Inventory	2011 PostSA Inventory	Revised Inventory	Revised to Reference Ratio
H-3	2.4E+11	7.8E+13	1.0E+15	1.1E+15	1.1
Cl-36	1.3E+12	1.3E+12	1.4E+12	1.4E+12	1.0
Mn-54	3.6E-01	3.1E-01	2.7E+02	2.7E+02	1.0
Fe-55	3.2E+11	3.6E+11	5.5E+13	5.5E+13	1.0
Co-60	9.3E+12	5.0E+13	9.0E+14	9.4E+14	1.0
Ni-59	2.7E+11	1.7E+13	3.6E+13	5.3E+13	1.5
Ni-63	7.5E+13	4.8E+15	3.9E+15	8.6E+15	2.2
Zr-93	1.5E+14	1.5E+14	2.1E+14	2.1E+14	1.0
Nb-94	4.6E+15	4.6E+15	4.6E+15	4.6E+15	1.0
Sb-125	1.2E+09	4.5E+09	5.7E+11	5.7E+11	1.0
Cs-134	1.5E+06	4.4E+06	3.1E+10	3.1E+10	1.0
Cs-137	6.6E+09	1.5E+13	1.1E+14	1.2E+14	1.1
U-235	2.1E+05	3.1E+05	2.3E+07	2.3E+07	1.0
U-238	1.7E+07	7.8E+07	6.0E+09	6.1E+09	1.0
Pu-238	4.6E+09	3.1E+10	5.0E+11	5.3E+11	1.1
Pu-239	8.3E+09	1.0E+10	9.2E+11	9.2E+11	1.0
Pu-240	1.1E+10	1.7E+10	1.3E+12	1.3E+12	1.0
Am-241	1.4E+10	6.9E+10	2.4E+12	2.4E+12	1.0
Cm-244	0.0E+00	1.9E+12	2.9E+11	2.2E+12	7.5
All others [a]	5.8E+14	5.8E+14	6.2E+15	6.3E+15	1

Notes:
[a] Inventory for all other radionuclides in the pressure tubes in unchanged from Table 3.16 of Quintessa and Geofirma (2011a).

Of the radionuclides listed, Mn-54, Fe-55, Co-60, Sb-125 and Cs-134 were screened out for consideration in the 2011 postclosure safety assessment by the calculations presented in Appendix A of Quintessa and Geofirma (2011a), primarily due to their short half-lives (all about 5 years or less). The revised total radionuclide inventory for these radionuclides changes very little (less than 5%) in comparison to the 2011 PostSA, so these radionuclides are also not included in the calculations presented below.

3. MODIFICATIONS TO THE MODELS

The 2011 PostSA models were implemented in three software codes.

- Assessment-level (system) models were implemented in AMBER, which is a compartment-model code that represents radioactive decay, package degradation, radionuclide transport through the repository, geosphere and surface environment, and evaluates the associated potential impacts such as dose. AMBER calculations were undertaken for both the Normal Evolution Scenario (Quintessa 2011) and Disruptive Scenarios (Quintessa and SENES 2011). They drew directly on detailed groundwater and gas modelling calculations undertaken with the following codes.

- Detailed groundwater flow and transport calculations were implemented in the 3-D finite element/finite-difference code FRAC3DVS-OPG (Geofirma 2011).

- Detailed gas generation and transport calculations were implemented in T2GGM (Geofirma and Quintessa 2011), a code that couples the Gas Generation Model (GGM) and TOUGH2 (Quintessa and Geofirma 2011b). GGM is a project-specific code that was used to model the generation of gas within the DGR due to corrosion and microbial degradation of the metals and organics present. TOUGH2 modelled the subsequent two phase transport of gas through the repository and geosphere.

The FRAC3DVS-OPG and T2GGM calculations are unaffected by the modifications to the inventory information. FRAC3DVS-OPG was primarily used for groundwater flow calculations, and radionuclide transport calculations were only undertaken for Cl-36 as this is an important radionuclide for groundwater transport but which is unaffected by the revised pressure tubes inventory. T2GGM calculations did not consider radionuclide transport; their focus was on the calculation of repository gas pressures and bulk gas transport. There is therefore no need to re-run any of the detailed groundwater and gas model calculations that support the 2011 PostSA.

The revised pressure tubes inventory has been introduced into the AMBER model for the 2011 PostSA so that results can be compared for all the AMBER calculation cases considered in the 2011 PostSA.

The pressure tubes inventory is represented in the AMBER model as being present within the metal of the pressure tubes. In the 2011 PostSA, radioactivity in the pressure tubes was released based on the corrosion rate for zirconium alloys of 10^{-8} m/a under anaerobic saline conditions[1] and a metal thickness of 5 mm (described as a “congruent release” model).

The revised inventory for some radionuclides associated with the pressure tubes will be present as surface (or near-surface) contamination. Other radionuclides will be present as activation products within the matrix of the metal. To reflect potential for surface contamination to be more-readily released, the model for pressure tubes has been adapted to use an instant release model for the following radionuclides: Ni-59, Ni-63, Cs-137, U-235, U-238, Cm-244 and for the plutonium and americium radioisotopes. The release model for all other radionuclides in the pressure tubes remains the same as used in the 2011 PostSA.

[1] Table 3.20 of Quintessa and Geofirma (2011a).

These changes have been made to the inventory and waste thickness parameters in the AMBER model for the near field and geosphere.[2] The calculations have been undertaken in AMBER 5.7.1[3].

[2] AMBER case files AMBER_V2_NF&GEOv1.01_pt.cse and AMBER_V2_BIOv1.01.cse

[3] AMBER 5.7.1 is the latest version of the AMBER software, released December 2013. AMBER 5.3 was used for the original 2011 PostSA calculations in 2011. To demonstrate that the software developments in the intervening time have not significantly affected the calculated results, the 2011 PostSA case files have been re-run with AMBER 5.7.1 and the results are in agreement.

4. RESULTS AND ANALYSIS

A summary of the calculation cases and a comparison of revised calculations against the 2011 PostSA results are presented in Section 4.1. Further information and a greater degree of detail is then presented for a sub-set of key calculation cases for the Normal Evolution Scenario, Human Intrusion Scenario and Severe Shaft Seal Failure Scenario in Sections 4.2, 4.3 and 4.4, respectively.

4.1 OVERVIEW OF RESULTS FOR ALL CALCULATION CASES

All of the AMBER cases for radionuclide release and migration, and potential exposure[4] considered in the 2011 PostSA have been re-run with the revised pressure tubes inventory and release model. For convenience, Table 2 and Table 3 provide summaries of each of the calculation cases for the Normal Evolution Scenario and for the Disruptive Scenarios, respectively.

Table 2: Assessment Modelling Cases for the Normal Evolution Scenario

Case ID	Case Description
NE-RC-A*	Reference Case parameters based on inventory, original preliminary design and site characterization data. Based on detailed groundwater and gas modelling reference cases. Considers: • instantaneous and congruent contaminant release; • source terms with release for certain radionuclides (e.g., C-14) partitioned between gas and groundwater; • no sorption or solubility limitation in repository (except for carbon solubility limitation); • gas generation and gradual repository resaturation; • no consumption (or production) of water by corrosion and degradation reactions; • 10 m rockfall at closure; • sorption of limited number of contaminants in shaft and geosphere; • steady state Cambrian overpressure (+165 m); • initial Ordovician underpressures with subsequent transient evolution towards equilibrium; • initial gas saturations of 10% in the Ordovician; • no salinity profile in the geosphere; • no horizontal groundwater flow in the Cambrian, Guelph or Salina A1 upper carbonate; • no explicit representation of glacial cycling; • self-sufficient farming family.

4 The inventory of non-radioactive contaminants in the DGR is unchanged by the revised pressure tube inventory, therefore calculations for non-radioactive contaminants did not need to be re-run. Note that probabilistic calculations were undertaken for Cl-36 and I-129 in the PostSA (NE-PC-A); however, the inventory for these radionuclides is unaffected by the changes to the pressure tube inventory. Therefore the case did not need to be re-run.

Case ID	Case Description
NE-PD-RC-A	As NE-RC-A but adopting the final preliminary design, including: • additional ventilation drifts; and • ILW filters & elements, irradiated core components, and IX columns disposed to ILW shield containers rather than concrete arrays.
NE-SBC-A*	As NE-RC-A but with: • no underpressures in the Ordovician; and • no initial gas saturation in the Ordovician.
NE-RS-A	As NE-RC-A but with: • immediate water resaturation of repository (including shaft); and • no gas generation in repository.
NE-EDZ1-A	As NE-SBC-A but with excavation damaged zone (EDZ) hydraulic conductivities increased to maximum values in the Data report, i.e.: • shaft inner EDZ increased by two orders of magnitude (i.e., four orders of magnitude greater than rock mass); • shaft outer EDZ increased by an order of magnitude (i.e., two orders of magnitude greater than rock mass); and • repository EDZ increased by an order of magnitude, (i.e., four orders of magnitude greater than rock mass).
NE-HG-A	As NE-SBC-A but with: • horizontal groundwater flow in the Guelph (gradient of 0. 0026) and Salina A1 upper carbonate formations (gradient of 0.0077); and • 1.25 km travel path along Guelph and Salina A1 upper carbonate to lake.
NE-GT5-A	As NE-GG1-A but with: • asphalt seal in shaft replaced by bentonite/sand; • gas entry pressure for shaft materials reduced by factor of two to 5 x 10^6 Pa; and • bentonite/sand hydraulic conductivity increased by an order of magnitude to 10^{-10} m/s.
NE-PD-GT5-A	As NE-GT5-A but with final preliminary design (as for NE-PD-RC-A).
NE-BF-A	As NE-SBC-A but with repository backfilled with coarse aggregate material with a porosity of 0.3.
NE-GG1-A	As NE-SBC-A but with: • increased metal inventory (~ 25% increase); and • corrosion and organic degradation rates increased to maximum rates in the Data report (up to an order of magnitude increase).
NE-GG2-A	As NE-SBC-A but with organic degradation rates decreased to minimum rates in the Data report (by up to an order of magnitude decrease)
NE-NM-A	As NE-SBC-A but with no methanogenic reactions, which includes both methane generation from organic degradation and also the conversion of H_2 and CO_2 to CH_4.

Case ID	Case Description
NE-RT1-A	As NE-RC-A but with: • immediate water resaturation of repository; • no gas generation in repository; • instantaneous release of radionuclides to repository water; and • no radionuclides sorbed or solubility limited in repository or geosphere.
NE-RT2-A	As NE-SBC-A but with: • immediate water resaturation of repository; • no gas generation in repository; • instantaneous release of radionuclides to repository water; and • no radionuclides sorbed or solubility limited in repository or geosphere.
NE-IV-A	As NE-RC-A but with radionuclide inventory increased by a factor of ten.
NE-ER-A	As NE-RC-A but with removal of 100 m of geosphere due to erosion over 1 million years.
NE-CC-A	As NE-RC-A but with alternative constant state biosphere (i.e., tundra rather than temperate).
NE-CG-A	As NE-HG-A but with dose to a Site Shore Resident Group and a Downstream Resident Group exposed via consumption of lake fish and water from the near shore and the South Basin of Lake Huron, respectively.

Notes: Based on Table B.1 of Quintessa et al. (2011). ‘A’ in the case ID indicates that an AMBER calculation was included in the 2011 PostSA. Refer to the original report for further details.
* A version of this case was also run using gas flow information from the T2GGM water-limited version that accounts for the effect of the consumption (or production) of water by corrosion and degradation reactions.

Table 3: Assessment Modelling Cases for the Disruptive Scenarios

Case ID	Case Description
HI-BC-A	As NE-RC-A but with: • exploration borehole drilled from surface down into Panel 1 at some time after controls are no longer effective (i.e., 300 years); • borehole terminated at repository depth; • repository largely unsaturated; • short-term surface release of contaminated gas immediately following intrusion; and • retrieval of contaminated drill core.
HI-GR2-A	As NE-RC-A but with: • exploration borehole drilled from surface down into Panel 1 at some time after controls are no longer effective (i.e., 300 years); • borehole penetrates down to the pressurized Cambrian; • repository rapidly resaturated; • borehole poorly sealed resulting in a hydraulic conductivity of 10^{-4} m/s and porosity of 0.25; and • long-term release of radionuclides in water from the repository to the Shallow Bedrock Groundwater Zone.
SF-BC-A	As NE-RC-A but with: • hydraulic conductivity of 10^{-9} m/s for bentonite/sand, asphalt and concrete in shafts; • porosity of 0.3 for bentonite/sand, asphalt and concrete in shafts; • effective diffusion coefficient of 3×10^{-10} m^2/s for bentonite/sand, asphalt and concrete in shafts; • sorption values for bentonite/sand given in the Data report reduced by an order of magnitude; • zero capillary pressure for shaft sealing material; and • repository and shaft EDZ hydraulic conductivity increased to maximum values in the Data report.
SF-ED-A	As SF-BC-A but increased bentonite/sand, asphalt and concrete hydraulic conductivity (10^{-7} m/s) in order to understand the sensitivity of 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.
BH-BC-A	As NE-RS-A but with: • poorly sealed site investigation/monitoring borehole from surface down to Precambrian located 100 m from the southeast edge of Panel 2; • hydraulic conductivity of 10^{-4} m/s for borehole seal; • porosity of 0.25 for borehole seal; and • no sorption on borehole seal.
VF-BC-A	As NE-RS-A but with a hypothetical transmissive vertical fault: • 500 m northwest of the repository; • from Cambrian to Guelph; • width of 1 m; • hydraulic conductivity of 10^{-8} m/s;

Case ID	Case Description
	• porosity of 0.1; and • no sorption in fault. In addition: • horizontal groundwater flow in the Cambrian (gradient of 0.0031), the Guelph (gradient of 0.0026) and Salina A1 upper carbonate formations (gradient of 0.0077); and • ~1 km travel path along Guelph from fault to lake.
VF-AL-A	As for the VF-BC-A case but with hypothetical transmissive vertical fault 100 m southeast of the repository.

Note: Based on Table B.2 of Quintessa et al. (2011). 'A' in the case ID indicates that an AMBER calculation was included in the 2011 PostSA. Refer to the original report for further details.

The maximum calculated effective doses for all of the calculation cases that have been re-run are summarized in Table 4 and Table 5 and compared against the 2011 PostSA results. The results for the Normal Evolution Scenario calculation cases are presented in Figure 1 and those for the base case Disruptive Scenarios in Figure 2. Note that the calculated doses within the shaded range of Figure 1 and Figure 2 are negligible and the magnitude of the values within this area is illustrative.

Table 4 and Table 5 demonstrate that the maximum calculated effective dose is relatively unaffected by the revised inventory for all of the calculation cases. This is principally because the inventories for key radionuclides contributing to the maximum calculated dose are unchanged. The tables show the key contributing radionuclides to the maximum calculated effective dose in each calculation case. The tables demonstrate that none of the radionuclides with increased inventories in Table 1 are the main contributors to the maximum calculated dose in any of the calculation cases. Ra-226 and its progeny are important to three calculation cases; these are from in-growth from U-238 and Pu-238 which are listed in Table 1, however, the total inventories for these radionuclides are increased by less than 10%.

In addition to the increased inventory for some radionuclides, the model for radionuclide release from the pressure tubes has also been changed; the revised model explicitly represents some of the radionuclides as being present as surface contamination on the pressure tubes, such that the inventory for these radionuclides is released more quickly (Section 3). Table 4 and Table 5 demonstrate that these changes, combined with the changes in the inventory for some radionuclides in the pressure tube waste stream, only have a small effect on the maximum calculated doses; the changes are small enough to not be discernible in Figure 1 and Figure 2.

Table 4: Maximum Calculated Dose to Adults for the Normal Evolution Scenario Calculation Cases

Basis	Case ID	Brief Description	Max. Calculated Dose (mSv/a)		Difference	Key radionuclide(s)
			2011	Revised		
Reference Case	NE-RC	Reference Case	1.5E-15	1.5E-15	+0.28%	I-129
	NE-PD-RC	Reference Case, final preliminary design	1.8E-15	1.8E-15	+0.22%	I-129
	NE-RC (WL)	Reference Case, water limited	4.1E-16	4.1E-16	+0.31%	I-129
	NE-CC	Tundra climate state	7.1E-15	7.1E-15	+0.32%	I-129
	NE-ER	100 m surface erosion	1.3E-13	1.3E-13	+0.28%	I-129
	NE-IV	Increased inventory	1.5E-14	1.5E-14	+0.29%	I-129
	NE-RS	Instant resaturation, no gas generation	4.0E-14	4.3E-14	+7.5%	I-129
	NE-RT1	Instant resat. & release, no sorption, no gas gen.	4.2E-09	4.3E-09	+1.1%	Ra-226 chain
Simplified Base Case	NE-SBC	Simplified Base Case	9.8E-14	9.9E-14	+0.64%	I-129
	NE-SBC (WL)	Simplified Base Case, water limited	6.2E-14	6.3E-14	+0.50%	I-129
	NE-EDZ1	Increased permeability of shaft and repository EDZs	1.9E-11	2.0E-11	+3.1%	Cl-36
	NE-EG	Alternative critical groups	6.1E-16	6.3E-16	+3.8%	I-129/Ra-226 chain
	NE-HG	Horizontal g/w flow in Guelph and Salina A1 upper carbonate	4.5E-16	4.6E-16	+1.2%	I-129
	NE-RT2	Instant resat. & release, no sorption, no gas gen.	4.5E-09	4.7E-09	+2.7%	Ra-226 chain
	NE-NM	No methanogenic gas reactions	5.1E-14	5.1E-14	+0.0%	C-14
	NE-GG1	Increased gas generation rates	9.3E-11	9.3E-11	+0.0%	C-14
	NE-GG2	Decreased degradation rates	9.5E-14	9.5E-14	+0.52%	I-129
	NE-GT5	Increased gas gen. & reduced shaft seal performance	4.9E-07	4.9E-07	+0.0%	C-14
	NE-PD-GT5	Increased gas gen. & reduced shaft seal perf., final prelim. design	2.7E-07	2.7E-07	+0.0%	C-14

Note that the results are presented to two significant figures to aid comparison. The degree of uncertainty associated with postclosure safety assessments of this nature means that it is ordinarily appropriate to present results rounded to one significant figure to avoid an undue implication of precision.

Table 5: Maximum Calculated Dose to Adults for the Disruptive Scenario Calculation Cases

Case ID	Brief Description	Max. Calculated Dose (mSv/a)		Difference	Key radionuclide(s)
		2011	Revised		
HI-BC	Human Intrusion, base case	1.0E+00	1.0E+00	+0.13%	Nb-94
HI-GR2	Human Intrusion, shallow groundwater release	3.4E+01	3.4E+01	+0.0%	C-14
SF-BC	Severe Shaft Failure, base case	1.1E+00	1.1E+00	+0.0%	C-14
SF-ED	Severe Shaft Failure, extreme degradation	7.5E+01	7.5E+01	+0.0%	C-14
BH-BC	Poorly Sealed Borehole, base case	4.3E-08	4.2E-08	-0.60%	Zr-93/Nb-93m
VF-BC	Vertical Fault, base case	4.6E-10	4.6E-10	+0.70%	Zr-93/Nb-93m
VF-AL	Vertical Fault, alternative location	4.6E-10	4.6E-10	+0.63%	Zr-93/Nb-93m

Note that the results are presented to two significant figures to aid comparison. The degree of uncertainty associated with postclosure safety assessments of this nature means that it is ordinarily appropriate to present results rounded to one significant figure.

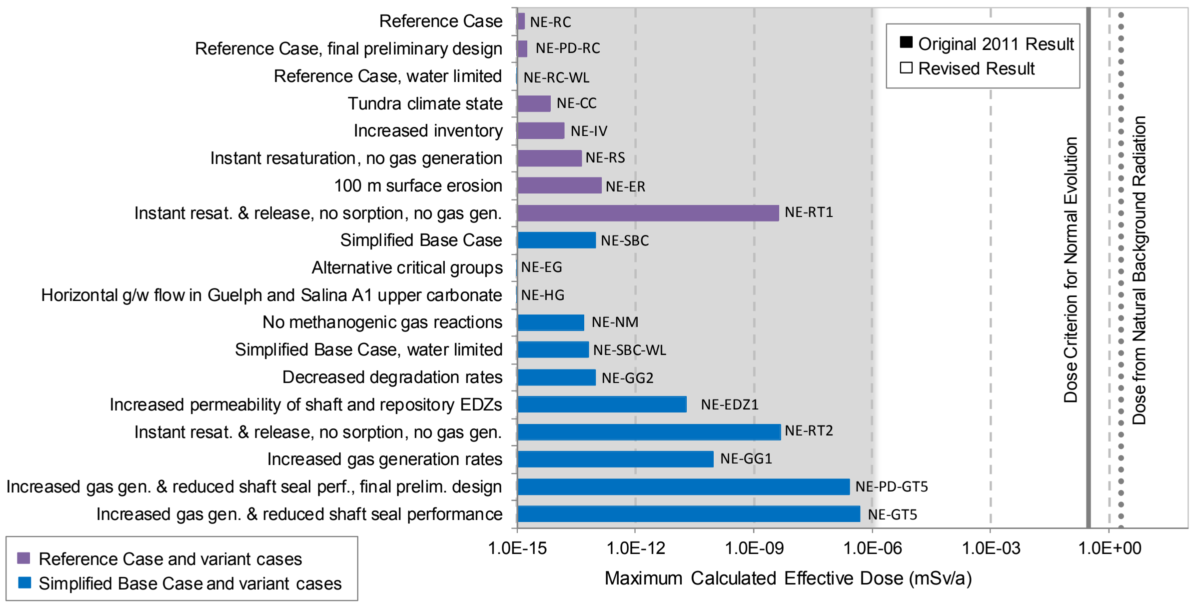

Figure 1: Maximum Calculated Doses for the Normal Evolution Scenario Calculation Cases showing 2011 PostSA and Revised Results (Note that differences in results are too small to be visible on this scale.)

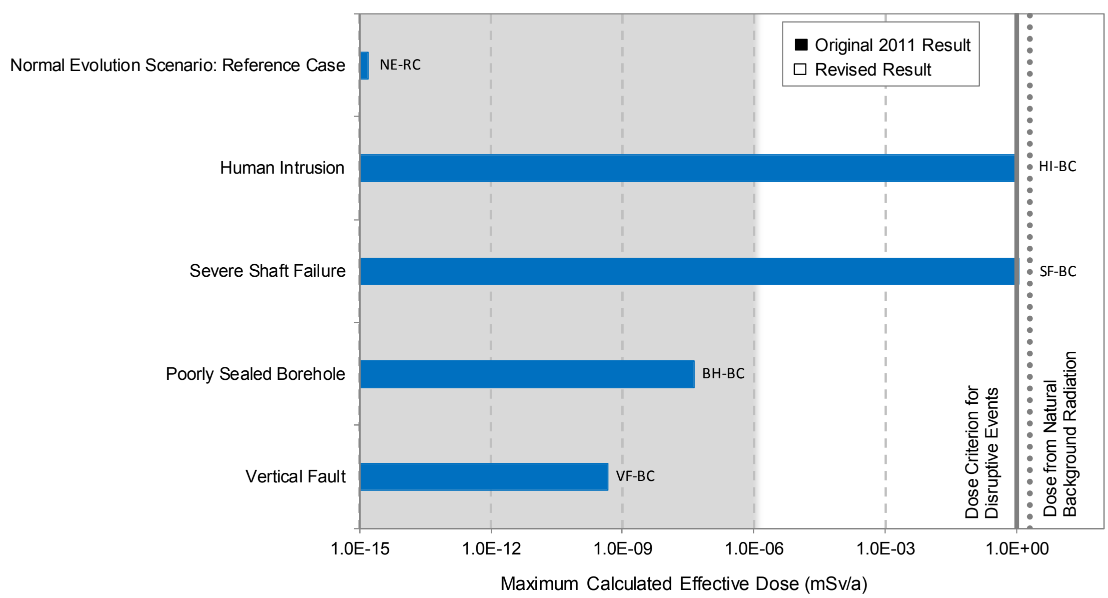

Figure 2: Maximum Calculated Doses for the Base Case Disruptive Scenarios showing 2011 PostSA and Revised Results (Note that differences in results are too small to be visible on this scale.)

In addition to presenting maximum doses for all calculation cases, the results for a sub-set of calculation cases are explored in more detail in the following sub-section. Key base cases are explored for the Normal Evolution Scenario (NE-RC and NE-SBC), together with two of the higher impact variants (NE-RT1 and NE-GT5) in Section 4.2. The highest impact disruptive scenarios (Human Intrusion and Severe Shaft Seal Failure) are then explored in Sections 4.3 and 4.4.

Potential impacts on non-human biota were assessed in the 2011 PostSA by comparison of maximum calculated radionuclide concentrations against ‘no effect concentrations’ for non-human biota for eleven radionuclides[5]. Of the eleven radionuclides for which no effect concentrations are available, Pb-210[6], Po-210[7], Ra-226[7], U-238 and Np-237[7] are affected by the changes to the inventory in the pressure tubes. The inventory of the associated parent radionuclides is changed by less than 10% in the revised analysis, so the conclusions of the 2011 PostSA remain valid and are not revisited in detail in this Technical Memorandum.

4.2 NORMAL EVOLUTION SCENARIO CALCULATION CASES

The results for the Reference Case are explored in Section 4.2.1, the Simplified Base Case in Section 4.2.2 and three variant cases in Section 4.2.3.

4.2.1 The Normal Evolution Scenario Reference Case (NE-RC)

The Reference Case (NE-RC) assumes instant rockfall at closure and draws on detailed groundwater and gas flow calculations that represent the limited degree of repository resaturation and explicitly represent the underpressures observed in the Ordovician formations. All waste packages are assumed to fail at closure and the release of radionuclides from the waste is explicitly modelled, although solubility limitation of releases is ignored. The detailed gas modelling shows that no free gas reaches the shafts, so radionuclides can only migrate from the repository into the host rock and up the shafts by diffusion and advection in groundwater. Potential exposure is considered of a site resident group, who live over the shafts and draw water for domestic and agricultural use from potable shallow groundwater via a well drilled into the flow path between the shafts and Lake Huron. The group has a self-sufficient lifestyle and consumes fish taken from local water courses and the lake. The case is summarized in Table 2 and in Figure 3.

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[5] The ‘no effect concentrations’ are given for C-14, Cl-36, Zr-93, Nb-94, Tc-99, I-129, Pb-210, Po-210, Ra-226, Np-237 and U-238 in Table 7.11 of Quintessa and Geofirma (2011a).

[6] Potential to in-grow from U-238 and Pu-238.

[7] Potential to in-grow from Am-241.

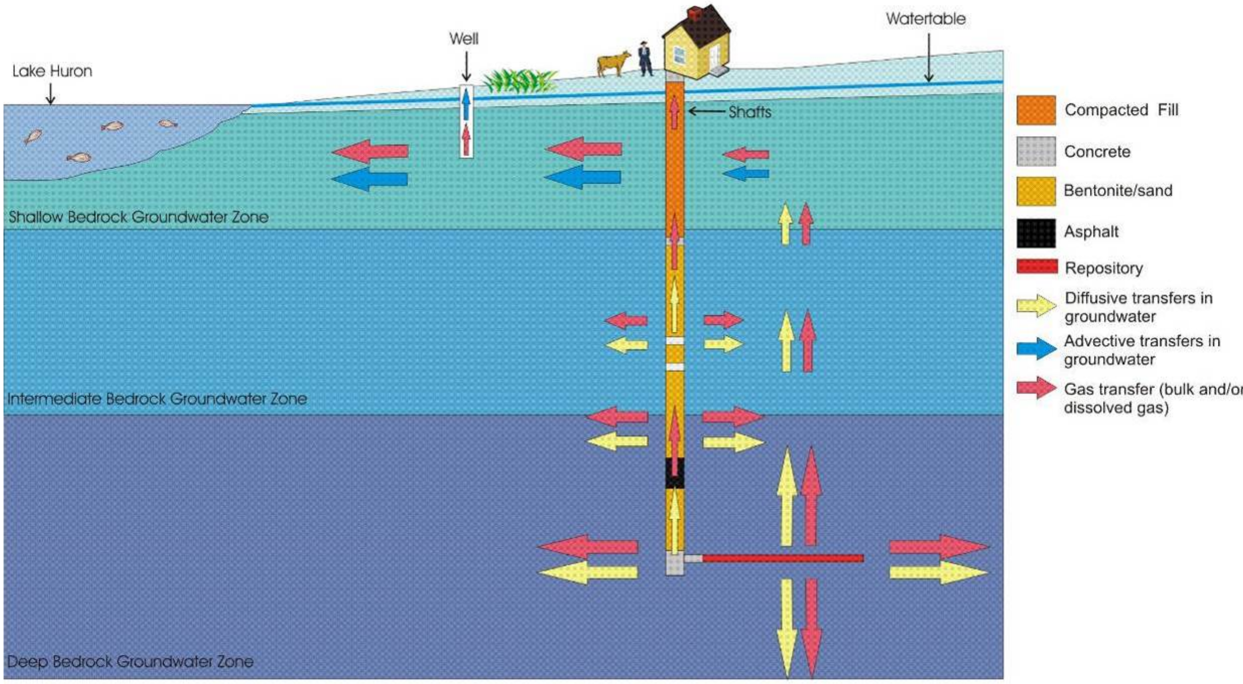

Figure 3: Schematic Representation of Potential Transport Pathways for the Normal Evolution Scenario

The results for the Reference Case including the changes to the pressure tubes inventory and release model are described in the sub-sections below and compared against the 2011 PostSA results[8].

Containment of Contaminants in the Repository

Radionuclides are initially present in the wastes within the waste packages. It is assumed in the safety assessment that all waste packages fail at closure. Radionuclides may be released either as gas (mainly C-14 and H-3) or after contact of the wastes with repository water. The release to repository water is either instant on contact with water, or determined by the corrosion/degradation rate of the associated wasteform.

The water level in the DGR determines the degree to which the wastes are contacted by water and, therefore, their potential to release radionuclides into the repository water. The detailed gas modelling indicates that the repository would not resaturate completely with the Reference Case assumptions and that the water level would remain very low (not exceeding about 10 cm over several million years). These results are unchanged by the change to the pressure tube inventory and radionuclide release model.

H-3 is assumed released instantly to the gas phase in the DGR and C-14 is released relatively rapidly to the gas phase. However, the small degree of repository resaturation means that other radionuclides remain within the wastes as they are only released on contact with water. Most of the total radioactivity decays without being released. This is illustrated in Figure 4, which shows

8 The format for the results reflects that used in Section 7.1 of Quintessa et al. (2011).

the amount of radioactivity that is released from the waste but remaining within the DGR, and that released from the DGR to the host rock and shafts. For comparison, the figure also shows the natural radioactivity in the rocks 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 4 shows that the revised pressure tubes inventory results in a small increase in the total inventory in the DGR. The figure also shows that the changes to the pressure tubes results in a higher release of radionuclides from the DGR on a timescale of up to about 1000 years. This reflects the more rapid release of surface contamination from the pressure tubes. However, the maximum amount of radioactivity released from the DGR in comparison to the initial inventory remains very small at 0.2%, which compares to 0.03% in the 2011 PostSA.

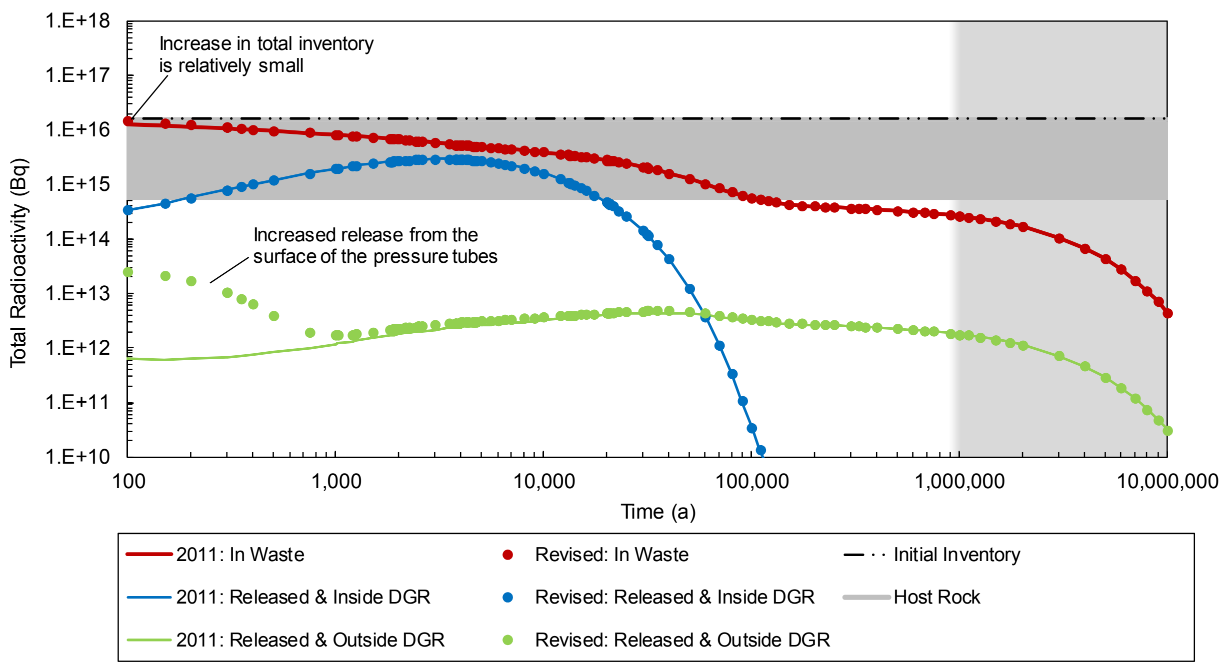

Figure 4: Total Radioactivity in the Reference Case (NE-RC)

Radionuclides in the DGR water can be released to the host rock via diffusion from the repository floor, and can be released to the shafts (and their EDZs) via diffusion and flow through the concrete monolith and its associated damaged zones. The detailed T2GGM modelling shows that free gas is not released from the DGR; this is unaffected by the change to the pressure tubes inventory and release model.

Figure 5 provides a summary of the radionuclide transfer fluxes from the DGR and shows that diffusion into the host rock dominates over contaminant migration to the shafts by more than three orders of magnitude due to the relatively large interface with the host rock compared to the small interface with the shafts via the monolith and its damaged zones together with low rates of groundwater advection. The perturbations in the radionuclide transfer flux from the repository to the monolith reflect fluctuations in groundwater flow rates. The increased release from the surface of the pressure tubes is evident in an increased transfer flux to the host rock on a timescale of up to about 1000 years.

Radionuclide transfer fluxes via the monolith to the shafts increase when groundwater flow away from the repository commences after 25,000 years for the Reference Case, indicating that groundwater advection dominates over diffusion as a process for contaminant migration to the shafts (see Figure 5).

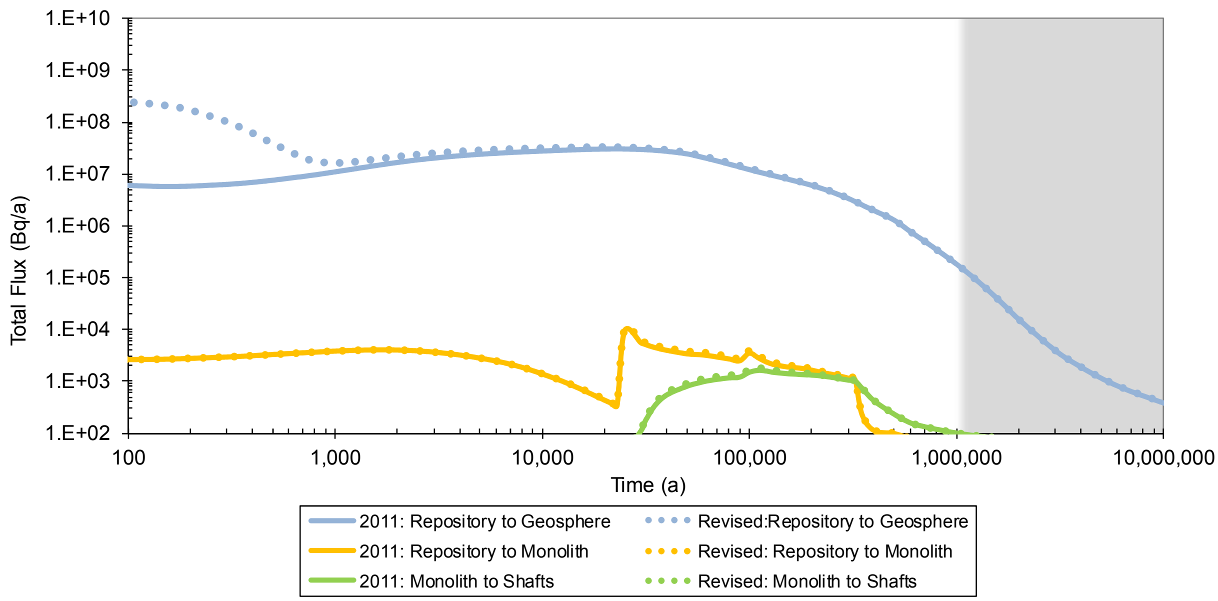

Figure 5: Radionuclide Transfer Fluxes from the DGR for the Reference Case (NE-RC)

The transfer flux from the DGR into the host rock is shown by radionuclide in Figure 6. The figure shows the diffusive flux via groundwater into the repository highly damaged zone (HDZ) and is indicative of the radionuclides present in the repository water. The figure shows that C-14, Zr-93, Nb-93m and Nb-94 are dominant radionuclides on a timescale beyond 1000 years. The amounts of these radionuclides in the pressure tubes are unaffected by the change in the inventory (see Table 1). However, Figure 6 shows that the increase in the Ni-63 inventory within pressure tubes, combined with its more rapid release as surface contamination, makes it the dominant radionuclide on timescales up to about 1000 years in the revised calculations. Ni-63 has a half-life of 100 years and has decayed on longer timescales.

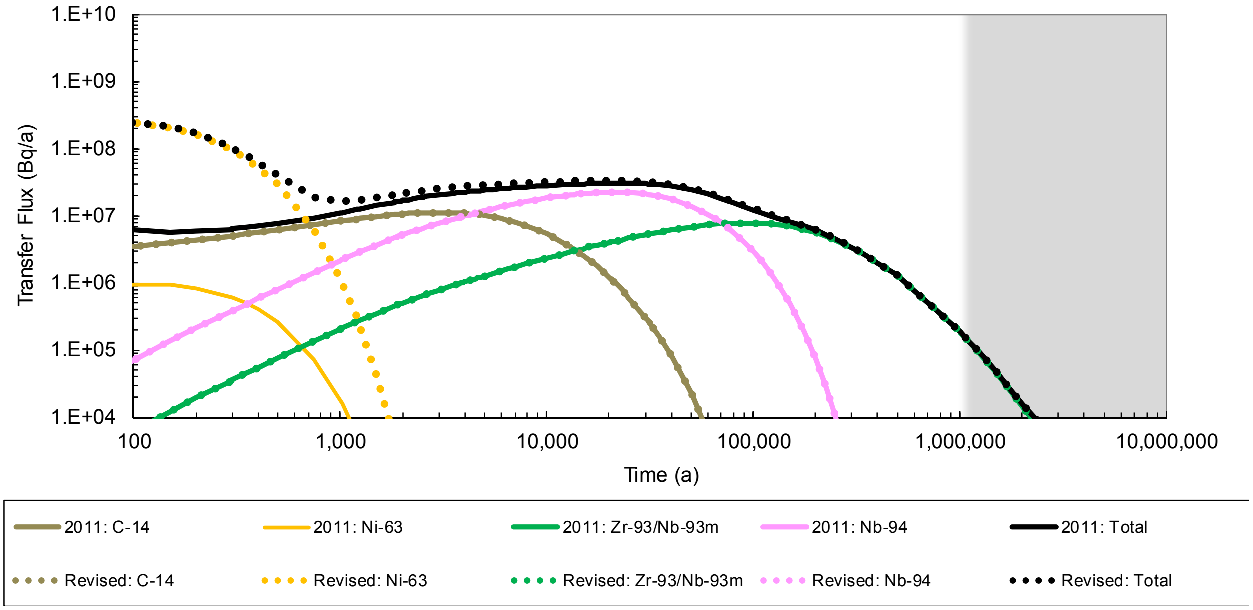

Figure 6: Radionuclide Transfer Flux from the DGR to the Host Rock Due to Diffusion in Groundwater for the Reference Case (NE-RC)

Containment of Contaminants in the Geosphere and Shafts

The host rock surrounding the DGR has very low permeability, such that transport of contaminants away from the repository is diffusion dominated. Figure 7 shows the total calculated concentrations in host rock above the DGR. The figure illustrates the decline in calculated concentrations with distance from the DGR. Calculated concentrations decline further with greater distance from the DGR and do not exceed 1 Bq/m^3 of rock beyond the Queenston formation at the top of the Deep Bedrock Groundwater Zone.

Nb-94 and Zr-93 (and its decay product Nb-93m) dominate the releases from the DGR on timescales beyond a few thousand years. Their greater sorption on the shales rather than limestone means that concentrations in the Collingwood formation exceed those in the Cobourg formation, which is closer to the DGR, after about 100,000 years. The increased flux of Ni-63 on a timescale up to about 1000 years is evident in Figure 7. The increased inventory of other radionuclides in the pressure tubes is evident in some visibly higher concentrations at longer times.

The shales in the vicinity of the DGR contain about 3 x 10^6 Bq/m^3 of natural radioactivity (mostly K-40 and U-238). This is also illustrated in Figure 7, which shows that the calculated concentrations in the host rock above the DGR, arising from radionuclides released from the DGR, do not exceed the natural background concentration for the Reference Case.

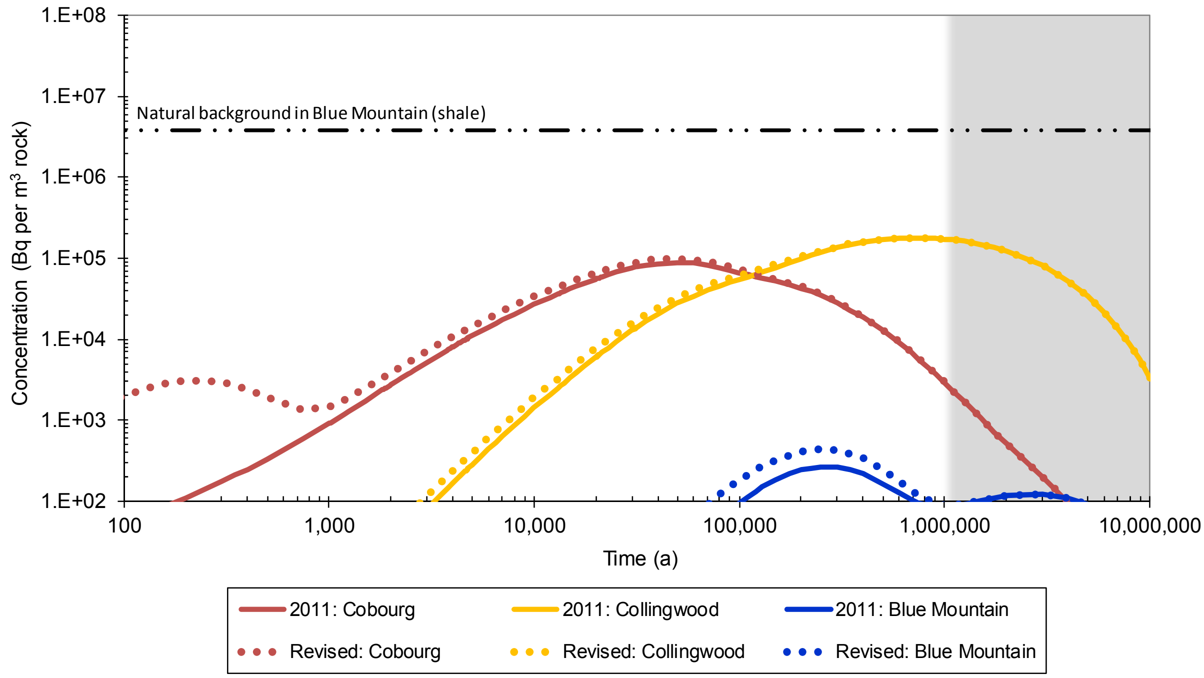

Figure 7: Radionuclide Concentration in the Deep Bedrock Groundwater Zone above the DGR for the Reference Case (NE-RC)

The shafts are also not a pathway for contaminants. Figure 5 indicates that a relatively small amount of radionuclides (up to 1 x 10^4 Bq/a) reaches the base of the shafts. Figure 8 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 as the distance from the DGR increases. No concentrations greater than 1 Bq/m^3 are calculated above the top of the Asphalt seals between the Georgian Bay and Queenston formations for the Reference Case, in-spite of a small increase in concentrations resulting from the revised pressure tube release model. Figure 8 also shows that calculated concentrations in the shafts continue to remain below natural background concentrations at the points shown for the Reference Case.

The concentrations in the shafts are low because contaminant transport via the shafts is dominated by diffusion in the Reference Case. Groundwater flow via the shafts in the upper regions of the Ordovician remains downwards throughout the assessment period due to the underpressure in the Ordovician rocks. Therefore, contaminant transport up through the shafts towards the Shallow Bedrock Groundwater Zone needs to be both diffusive and to operate against the direction of groundwater flow.

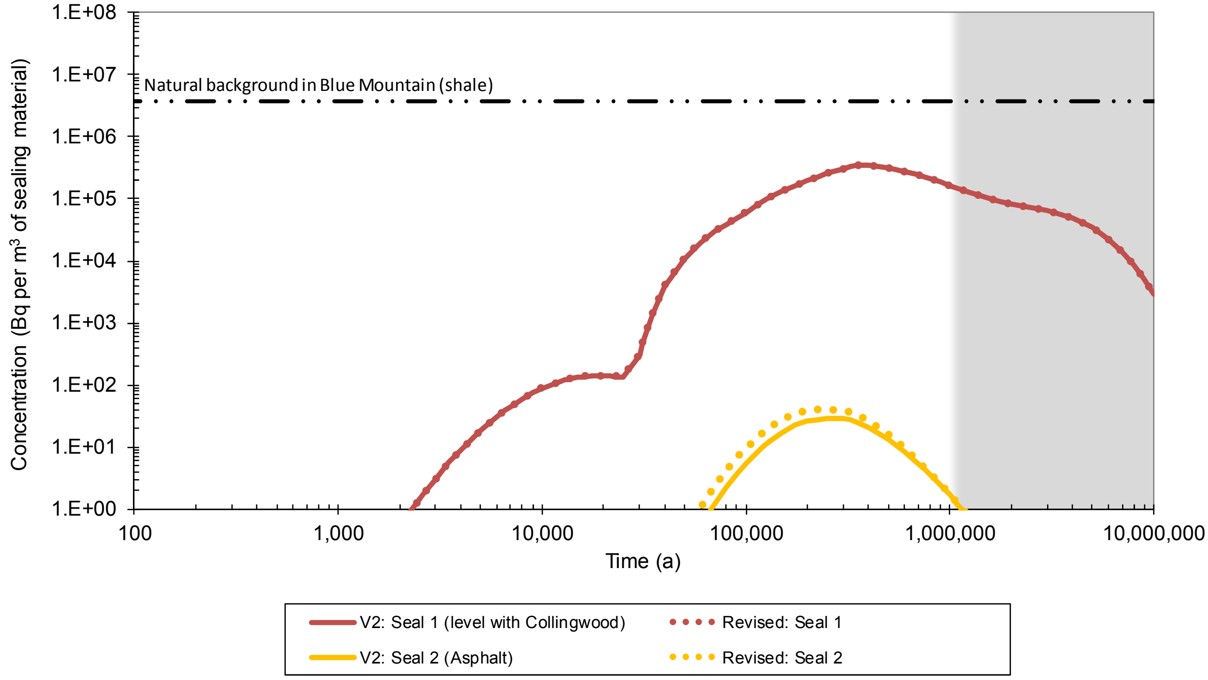

Figure 8: Radionuclide Concentration in Shafts for the Reference Case (NE-RC)

The low and slow level of repository resaturation, combined with the very low permeability of the host rock and the effectiveness of the shaft seals means that effectively no contamination enters the Shallow Bedrock Groundwater Zone (see Table 6). I-129 and Cl-36 dominate the small calculated radionuclide flux due to the sorption of other radionuclides to the bentonite/sand seals in the shafts (notably radioisotopes of Zr and Nb). The very small fluxes given in Table 6 can be compared against an estimated present-day flux of around 4 MBq/a in the flowing groundwater within the shallow system.

Neither I-129 nor Cl-36 are affected by the change in the inventory in the pressure tubes (see Table 1) and neither is associated with surface contamination on the pressure tubes. Therefore, the maximum calculated radionuclide flux to the shallow groundwater is unaffected and remains negligible (see Table 6).

Table 6: Maximum Calculated Flux to the Shallow Bedrock Groundwater Zone for the Reference Case (NE-RC)

Calculation Case	Maximum Calculated Flux	Time of Maximum Calculated Flux	Main Contaminant Contributing to the Peak
2011 PostSA	3 x 10^{-6} Bq/a	> 1 Ma	I-129
Revised Result	3 x 10^{-6} Bq/a	> 1 Ma	I-129

After any contaminants enter the Shallow Bedrock Groundwater Zone via the shafts and their EDZs, at 144 m below ground surface, horizontal groundwater flow takes the contaminants towards the lake (see Figure 3).

Vertical dispersion and the draw resulting from groundwater extraction will enable contaminants to reach the groundwater well, which is drilled to a depth of 80 m below ground surface. The well depth is typical of wells in the region. It is consistent with the more permeable near-surface formations, and avoids the higher salinity groundwater at greater depths. The well demand is consistent with the needs for a self-sufficient farm. The well is placed downstream from the shaft, so as to intercept the plume, but not so far downstream that there is much dilution. The detailed groundwater modelling results show that the well captures a small fraction of the contaminant plume in the Shallow Bedrock Groundwater Zone (see Section 5.2.2.2 of Geofirma 2011).

Consistent with the small calculated fluxes to the Shallow Bedrock Groundwater Zone listed in Table 6, Table 7 shows the small calculated fluxes to the biosphere for the Reference Case. Two biosphere discharge points are considered – the well and the lake. Consistent with the observations above, the maximum calculated fluxes to the biosphere are unaffected by the changes to pressure tubes and remain negligible.

Table 7: Maximum Calculated Flux to the Biosphere for the Reference Case (NE-RC)

Calculation Case	Biosphere Receptor	Max. Calculated Flux	Time of Max. Calculated Flux	Main Contaminant Contributing to the Max.
2011 PostSA	Well	4×10^{-8} Bq/a	> 1 Ma	I-129
	Lake	3×10^{-6} Bq/a		
Revised Result	Well	4×10^{-8} Bq/a	> 1 Ma	I-129
	Lake	3×10^{-6} Bq/a		

Impact of Contaminants

The very small release of contaminants to the biosphere results in very small calculated concentrations. Maximum calculated total concentrations in biosphere media are shown in Table 8 for the Reference Case. The table shows that the maximum calculated concentrations are unaffected by the changes to the pressure tubes and remain negligible.

For comparison, surface waters have provincial background concentrations ranging from 0.02 to 0.19 Bq/L gross-beta (Section 5.6 of AMEC NSS 2011). 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 (Section 5.7.1 of AMEC NSS 2011). Soils have concentrations of K-40 and Cs-137 ranging from 446 to 500 Bq/kg and 2.7 to 3.9 Bq/kg (respectively) at provincial background locations (Section 5.8.4 of AMEC NSS 2011).

Table 8: Summary of Maximum Calculated Biosphere Concentrations for the Reference Case (NE-RC) and the Main Contributing Radionuclide (in brackets)

Calculation Case	Well Water (Bq/L)	Soil (Bq/kg)	Surface Water (Bq/L)	Sediment (Bq/kg)
2011 PostSA	6×10^{-15} (I-129)	5×10^{-15} (Cl-36)	1×10^{-17} (I-129)	1×10^{-14} (I-129)
Revised Model	6×10^{-15} (TBD)	5×10^{-15} (Cl-36)	1×10^{-17} (I-129)	1×10^{-14} (I-129)

The extremely small calculated concentrations result in equivalently negligible calculated doses to the Site Resident group, which are shown in Table 4. I-129 and Cl-36 dominant the small calculated doses, which are therefore unaffected by the changes to the pressure tubes and remain negligible.

4.2.2 The Normal Evolution Scenario Simplified Base Case (NE-SBC)

The Simplified Base Case for the Normal Evolution Scenario (NE-SBC) is the same as the Reference Case, except in that the initial underpressures observed in the Ordovician formations are assumed to not be present (see Table 2). The absence of the underpressures increases the potential for groundwater to flow up the shafts and their EDZs.

Containment of Contaminants in the Repository

The water level in the DGR determines the degree to which the wastes are contacted by water and, therefore, their potential to release radionuclides into the repository water. Figure 9 compares the water level for the Simplified Base Case with that of the Reference Case, based directly on the results of the detailed T2GGM calculations (these are unaffected by the changes to the pressure tubes inventory and release model). The figure shows that the water level in the DGR continues to increase in the Simplified Base Case, whereas it peaks after a few thousand years in the Reference Case before declining. However, the water level remains well below the top of the DGR (the emplacement rooms are 7 m high plus assumed 10 m of rockfall) throughout the calculations in both cases.

Figure 10 shows the calculated radionuclide release rates from the DGR for the Simplified Base Case. As was the case for the Reference Case, the figure shows an increase in the radionuclide flux from the DGR to the rock on timescales up to about 1000 years, due to the increased inventory and rate of release of Ni-63 from pressure tubes. Beyond 1000 years calculated radionuclide fluxes from the DGR are similar in both the revised and 2011 PostSA calculations.

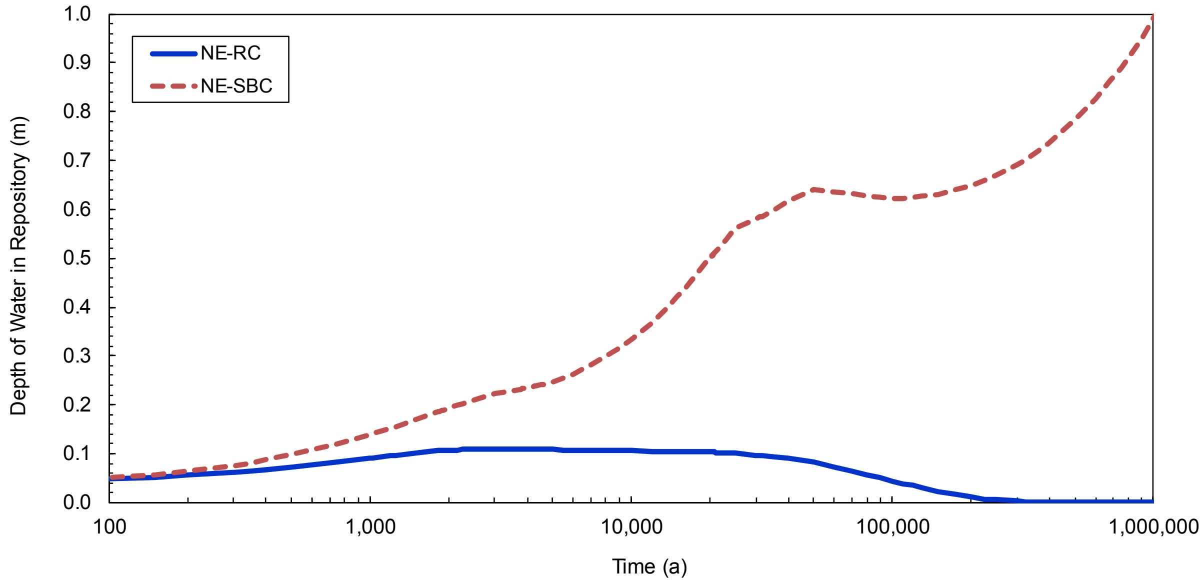

Figure 9: Depth of Water in the Repository for the Reference Case (NE-RC) and Simplified Base Case (NE-SBC)

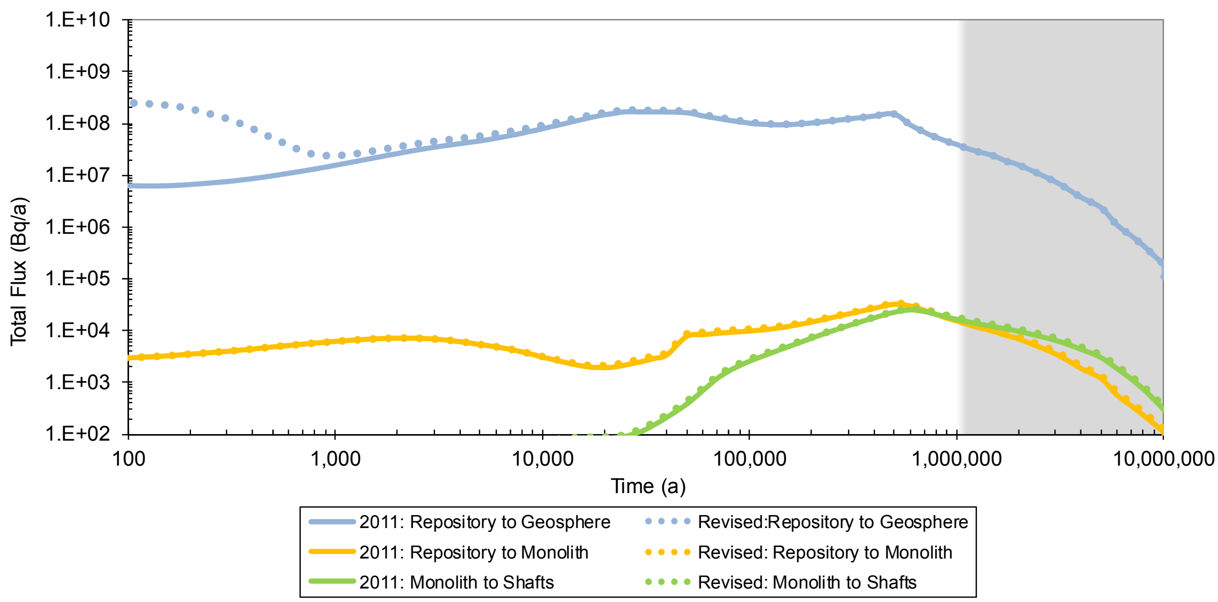

Figure 10: Radionuclide Transfer Fluxes from the DGR for the Simplified Base Case (NE-SBC)

Containment of Contaminants in the Geosphere and Shafts

The host rock and shafts remain extremely effective in isolating the DGR from the shallow groundwater for the Simplified Base Case. Table 9 shows the maximum calculated radionuclide flux to the shallow groundwater, which remains less than 1 Bq/a throughout. The table shows that the revised pressure tube inventory and release model has no effect on the peak calculated flux to the shallow groundwater. This is because the radionuclide flux to the shallow groundwater is dominated by I-129 and Cl-36, which are unaffected by the changes to the pressure tubes.

Table 9: Maximum Calculated Flux to the Shallow Bedrock Groundwater Zone for the Simplified Base Case (NE-SBC)

Calculation Case	Maximum Calculated Flux	Time of Maximum Calculated Flux	Main Contaminant Contributing to the Peak
2011 PostSA	2×10^{-3} Bq/a	> 1 Ma	Cl-36
Revised Result	2×10^{-3} Bq/a	> 1 Ma	Cl-36

The small calculated radionuclide fluxes to the shallow groundwater are reflected in the calculated fluxes to the biosphere for the Simplified Base Case (see Table 10). Most of any contamination reaching the Shallow Bedrock Groundwater Zone discharges to Lake Huron, while about 1.15% is intercepted by the well and used for domestic and agricultural purposes by the Site Resident group.

Table 10: Maximum Calculated Flux to the Biosphere for the Simplified Base Case (NE-SBC)

Calculation Case	Biosphere Receptor	Max. Calculated Flux	Time of Max. Calculated Flux	Main Contaminant Contributing to the Max.
2011 PostSA	Well	2×10^{-5} Bq/a	> 1 Ma	Cl-36
	Lake	2×10^{-3} Bq/a		
Revised Result	Well	2×10^{-5} Bq/a	> 1 Ma	Cl-36
	Lake	2×10^{-3} Bq/a		

Impact of Contaminants

The extremely small calculated radionuclide fluxes to the biosphere are reflected in the negligible calculated concentrations in biosphere media (see Table 11). The changes to the pressure tubes have very little effect on the calculated concentrations in the biosphere; the only difference evident in Table 11 results from a small change to the maximum calculated concentration in well water, which is sufficient to cause a rounding difference.

Table 11: Summary of Maximum Calculated Biosphere Concentrations for the Simplified Base Case (NE-SBC) and the Main Contributing Radionuclide (in brackets)

Calculation Case	Well Water (Bq/L)	Soil (Bq/kg)	Surface Water (Bq/L)	Sediment (Bq/kg)
2011 PostSA	3×10^{-12} (Cl-36)	4×10^{-12} (Cl-36)	6×10^{-15} (Cl-36)	3×10^{-13} (Cl-36)
Revised Model	4×10^{-12} (Cl-36)	4×10^{-12} (Cl-36)	6×10^{-15} (Cl-36)	3×10^{-13} (Cl-36)

The maximum calculated doses for the Simplified Base Case are shown in Table 4. The table shows that I-129 is the main contributing radionuclide to the calculated dose, which remains many orders of magnitude below the dose criterion. I-129 is relatively unaffected by the change in the pressure tube inventory and release model, so the maximum calculated dose to the Site Resident increases very little (by less than 1%).

4.2.3 Other Normal Evolution Scenario Cases

Results for two further Normal Evolution Scenario calculation cases are explored further below. The cases are (see Table 4 and Figure 1):

- the NE-RT1 case, which results in the highest calculated effective doses for variants based on the Reference Case; and
- the NE-GT5 case, which results in the highest calculated effective doses for variants based on the Simplified Base Case.

Instant Resaturation, Instant Release, No Sorption and No Gas Generation (NE-RT1)

This variant is based on the Reference Case (i.e., it includes the observed underpressures in Ordovician formations). In addition to the conservatisms inherent in the Reference Case (including rockfall at closure), the case assumes that the repository is fully resaturated, the full inventory is released to the groundwater at closure and the case ignores sorption on engineering materials and the host rock. Although unfeasible, the case was used in the 2011 PostSA to maximise the potential impact of the groundwater pathway for contaminants.

The instantaneous release of all radionuclides to groundwater in the DGR at closure means that the inventory is the only difference between the original and revised NE-RT1 calculations (i.e., the change in the release model for pressure tubes has no effect due to the instantaneous release). There is therefore little difference in the total radionuclide releases from the DGR (see Figure 11). The relatively small differences are propagated through to the dose assessment, which shows a small (1.1%) increase with the revised pressure tube inventory (see Table 4), which remain many orders of magnitude below the dose criterion.

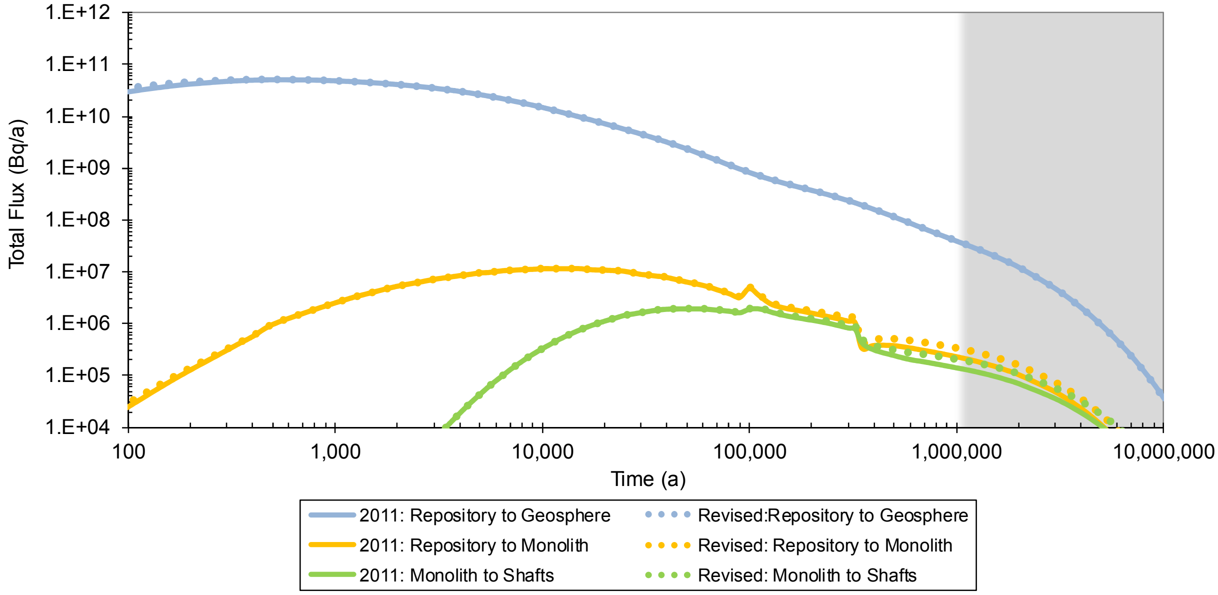

Figure 11: Radionuclide Transfer Fluxes from the DGR for the NE-RT1 Case

Increased Gas Generation & Reduced Shaft Seal Performance (NE-GT5)

This variant to the Simplified Base Case (i.e., ignoring the observed underpressures in Ordovician formations) explores the gas pathway through increased gas generation rates and amounts, along with lower gas entry pressures for the shaft sealing materials. The detailed gas modelling for the 2011 PostSA showed that free gas could migrate part of the way up the shaft in this case. The key radionuclide is C-14, which is carried part of the way up the shafts with the free gas before dissolving in groundwater.

The amount of C-14 in the DGR is unaffected by the changes to the pressure tube inventory. C-14 is not present as surface contamination, so it is also unaffected by the changes to the release model for the pressure tubes. The changes to the pressure tubes therefore have no discernible effect on the Normal Evolution Scenario case that has the highest calculated doses and they remain more than five orders of magnitude below the dose criterion, as shown in Table 4 and Figure 1.

4.3 HUMAN INTRUSION SCENARIO CALCULATION CASES

Deep geologic disposal isolates the wastes from the surface environment and minimises the potential for human intrusion during the period that the wastes remain hazardous. The Human Intrusion Scenario allows a “what if” style assessment of potential consequence of direct intrusion into the DGR, bypassing the significant geological barrier. Large scale excavation is extremely unlikely; therefore, the scenario addresses the potential for borehole intrusion (see Figure 12).

For the Human Intrusion Base Case (HI-BC), potential exposures are assessed from: gas releases as the borehole intrudes into the pressurised repository; retrieval of contaminated raw waste with the drill core; and release of contaminated slurry to the surface. Exposures to the drill crew, a technician examining the core, a nearby resident and potential future residents on an area contaminated by the material brought up by the borehole are considered.

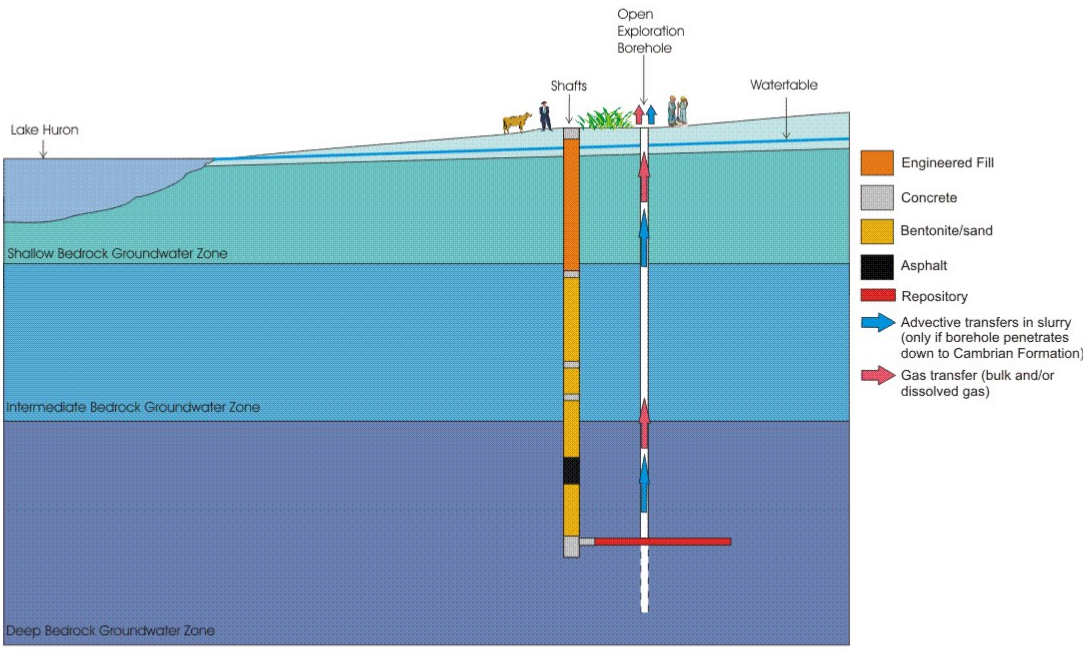

Figure 12: Schematic Representation of the Human Intrusion Base Case

The calculated dose rates for the Human Intrusion Base Case are shown in Figure 13. External irradiation from Nb-94 dominates the peak calculated dose rates to those groups directly exposed to raw waste, which are the drill crew, laboratory technician and future resident. C-14 dominates the peak calculated dose to the nearby resident group, which is only exposed via release of gas from the intruding borehole and its dispersion 100 m downwind. Neither Nb-94 nor C-14 are affected by the changes to the inventory in the pressure tubes. Neither C-14 nor Nb-94 are primarily present as surface contamination on the pressure tubes; therefore they are unaffected by the changes to the release model for the pressure tubes.

Although some of the calculated doses for the Human Intrusion Base Case are close to the dose criterion shown in Figure 13, the scenario has a low probability of occurrence at about 10^{-5}/a (see Section 2.5.3 of Quintessa and SENES 2011). Based on a health risk of 0.057/Sv (ICRP 2007), the associated risk of serious health effects for the future resident is around 6×10^{-10}/a, well below the reference health risk value of 10^{-5}/a given in Section 3.4.2 of Quintessa et al. (2011). This conclusion is therefore unaffected by the changes to the pressure tubes inventory and release model.

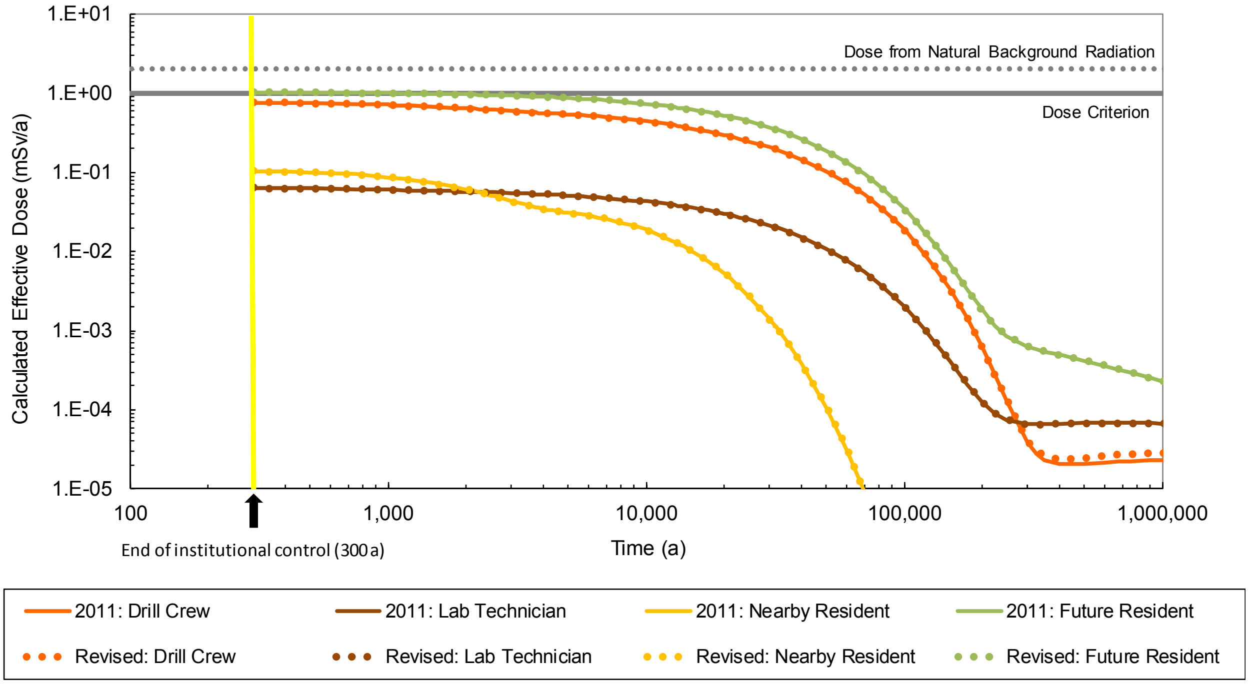

Figure 13: Calculated Effective Doses from Surface Release of Gas and Drill Core Resulting from Human Intrusion, as a Function of the Time of Intrusion, for the Human Intrusion Base Case (HI-BC)

A variant calculation case was also considered in the 2011 PostSA in which the intruding borehole extended beyond the DGR to the pressurised Cambrian formation and was not properly sealed, so that it provided a direct groundwater pathway to the shallow groundwater (HI-GR2). The case is based on a repository that is fully resaturated at the time of the borehole intrusion. For this case, C-14 in groundwater is the key radionuclide contributing to the calculated doses (see Table 5). C-14 is not effected by the change to the inventory in the pressure tubes and it is not present as surface contamination, so it is also unaffected by the changes to the release model. The maximum calculated dose for the HI-GR2 case is therefore unchanged from the 2011 PostSA.

Assuming the same probability of occurrence for the HI-GR2 case as for intrusion into the repository (thereby conservatively assuming the probability of continuing into the Cambrian and poorly sealing the borehole is unity), the peak dose equates to a risk of serious health effects that remains around 2 x 10^{-8}/a, more than two orders of magnitude below the reference health risk value of 10^{-5}/a. This remains unchanged from the conclusions of the 2011 PostSA.

4.4 SEVERE SHAFT SEAL FAILURE SCENARIO CALCULATION CASES

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 and the repository/shaft EDZs; it includes some expected degree of degradation of the seals with time. The Severe Shaft Seal Failure Scenario considers the same evolution of the DGR system and the same exposure pathways as the Normal Evolution Scenario, the difference being that there is rapid and extensive shaft seal degradation and the repository/shaft EDZs have significantly degraded properties (see Figure 14). Like the other Disruptive Scenarios, the scenario is a bounding "what if" scenario that is designed to investigate the robustness of the DGR system.

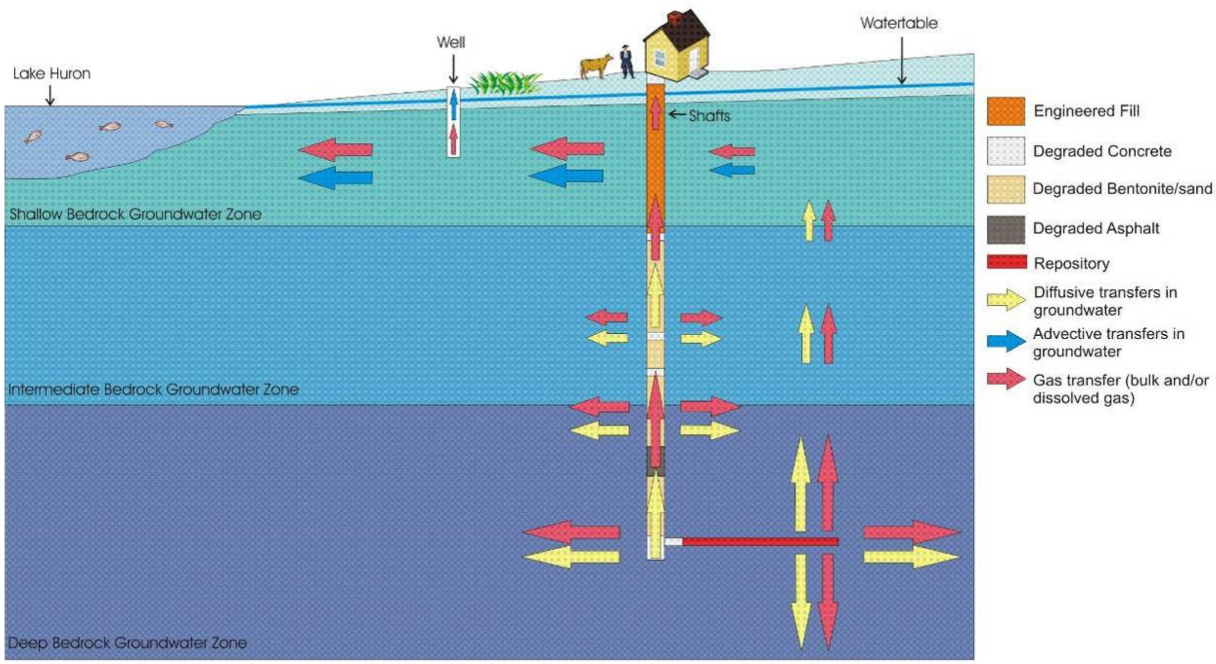

Figure 14: Schematic Representation of Severe Shaft Seal Failure Scenario

The calculated doses to the Site Resident Group for the Severe Shaft Seal Failure Scenario's Base Case are shown in Figure 15. C-14 dominates the calculated exposures due to a direct gas pathway being created from the DGR via the severely degraded shafts to the shallow groundwater and surface after about 20,000 years. C-14 is not affected by the change to the inventory in the pressure tubes and it is not present as surface contamination, so it is also unaffected by the changes to the release model. Therefore, the peak calculated dose for the SF-BC case is unaffected, as shown in Figure 15.

It is noted that, consistent with the 2011 PostSA, 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 health risk value of 10^{-5}/a. The probability of 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.

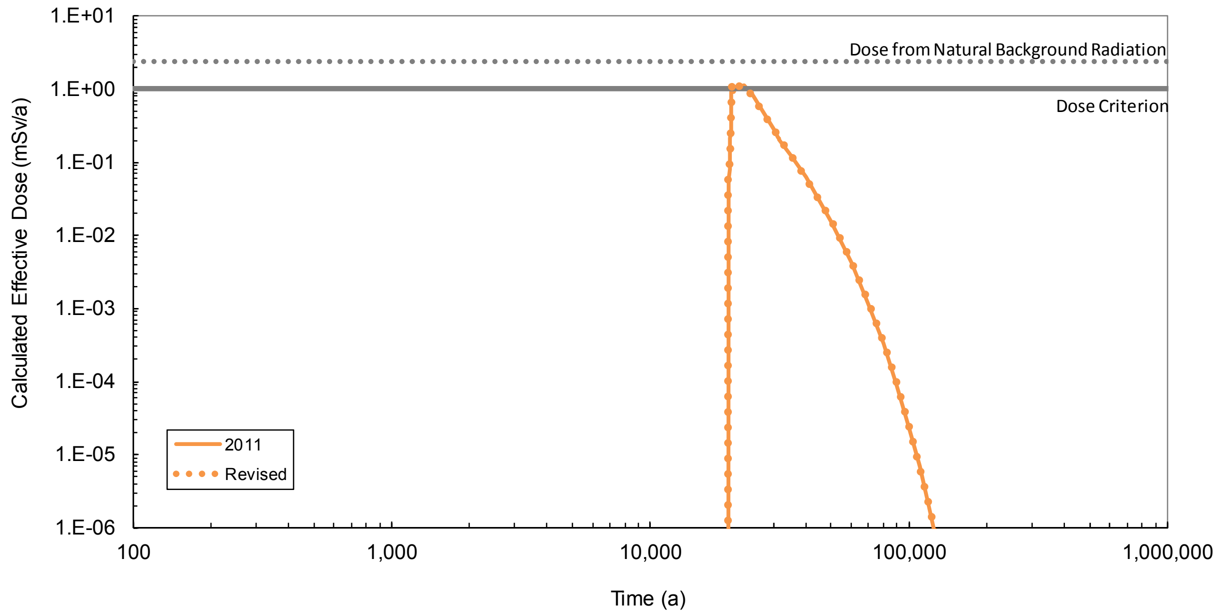

Figure 15: Calculated Effective Dose to Adults for the Site Resident Group for the Severe Shaft Seal Failure Scenario Base Case (SF-BC)

A variant to the Severe Shaft Seal Failure Base Case was also considered in the 2011 PostSA, representative of a bounding case with an even greater degree of shaft seal degradation (SF-ED). The case is representative of potential impacts should the shaft seals perform like fine silt and sand from the point of closure. Table 5 shows that the changes to the pressure tube release model result in the calculated effective dose to someone living directly above the shafts remains unchanged at 75 mSv/a. The peak calculated impact occurs due to C-14 after about 3800 years, coincident with peak calculated gas flows via the shafts of close to 10,000 kg/a. 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 around 500 m of low-permeable shaft seals to degrade so as 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.

5. CONCLUSIONS

All of the assessment-level cases for the 2011 PostSA have been recalculated to explore the effect of changes to the radionuclide inventory in pressure tubes and a change in the release model for pressure tubes permitting more rapid release of some of the associated radionuclides. The detailed gas and groundwater calculations that support the assessment-level modelling are unaffected by the changes and therefore have not been revised.

The calculations demonstrate that the revised inventory has very little effect on the calculated effective doses. This is because the inventories of only four radionuclides in the DGR are increased by more than 10% in comparison to the 2011 PostSA (Ni-59, Ni-63, Cs-137 and Cm-244) and none of these nor their progeny are important contributors to maximum calculated effective doses.

Since the revised radionuclides are largely present as surface-deposit or as thin garter springs, the radionuclide release model for the pressure tubes has been changed to enable radionuclides present as surface contamination to be released more quickly than activation products present within the metal itself. The pressure tubes are an important source for some of the radionuclides present as surface contamination, notably representing more than 50% of the total inventory for Ni-63. The more rapid availability of these radionuclides is evident when exploring calculated radionuclide fluxes in the DGR system, although they do not contribute significantly to calculated effective doses.

For the Normal Evolution Scenario cases, in all calculation cases considered, the maximum calculated effective dose remains more than five orders of magnitude below the public dose criterion of 0.3 mSv/a for the revised inventory.

Some small differences in the Normal Evolution Scenario results between the revised inventory and the 2011 PostSA are given below:

- maximum calculated effective doses increase by less than 1% for the Reference Case and Simplified Base Case; and
- maximum calculated effective doses increase by less than 10% in all 19 calculation cases.

For the “what if” calculations exploring potential disruptive scenarios:

- maximum calculated effective doses for the Human Intrusion Scenario Base Case increase by less than 1%, due to the relative unimportance of the release model for this case;
- maximum calculated effective doses for the Severe Shaft Seal Failure cases are unaffected, due to the dominance of C-14 and its being unchanged by the revisions to the pressure tube inventory and release model;
- maximum calculated effective doses for the Poorly Sealed Borehole and Vertical Fault cases change by less than 1%, due to the dominance of Zr-93/Nb-93m, which are unaffected by the revisions to the pressure tube inventory and release model; and
- potential risks associated with all of the “what if” disruptive cases remain substantially less than the reference health risk value of 10^{-5}/a, once the low likelihood of occurrence is taken into account.

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Attachment B to OPG Response to IR-EIS-13-514

PRECLOSURE SAFETY IMPLICATIONS OF REVISED PRESSURE TUBE INVENTORIES

PRECLOSURE SAFETY IMPLICATIONS OF REVISED PRESSURE TUBE INVENTORIES

CONTENTS

LIST OF TABLES

Page

1. INTRODUCTION

Ontario Power Generation (OPG) is proposing to build a Deep Geologic Repository (DGR) for Low and Intermediate Level Waste (L&ILW) near the existing Western Waste Management Facility at the Bruce nuclear site in the Municipality of Kincardine, Ontario. The Nuclear Waste Management Organization, on behalf of OPG, has prepared the Environmental Impact Statement (EIS) (OPG 2011a) and Preliminary Safety Report (PSR) (OPG 2011b) for the proposed repository. The EIS, PSR and supporting documentation were submitted for regulatory review in April 2011, as part of the application for a site preparation and construction licence.

The preclosure safety assessment is documented in Chapter 7 of the Preliminary Safety Report, which deals with normal operations and accidents. Dose estimates for the Malevolent Acts Scenarios are provided in OPG's response to the Information Request EIS-06-248 (OPG 2012).

The safety assessment was conducted based on the projected final DGR inventory presented in OPG's 2010 Reference Inventory report (OPG 2010). Some of the inventory projections were based on estimates. An ongoing OPG waste characterization program is underway to reduce uncertainties in these estimates.

During the review process of the current Preliminary Safety Report, the estimates for the inventories of some radionuclides in pressure tube wastes have been identified as significantly underestimated. Pressure tubes from reactor mid-life retubing are one of over 20 waste types considered in the reference inventory. The Joint Review Panel has issued an Information Request EIS-13-514 asking for an assessment of how the revised inventory will affect the preclosure safety assessment of the DGR. This Technical Memorandum addresses this request.

2. MODIFICATIONS TO THE RADIONUCLIDE CONCENTRATIONS IN THE PRESSURE TUBES

The revised radionuclide concentration data for the pressure tubes containers, including garter springs, are provided in Table 1. This table shows the source term for normal operations shielding calculations and for accident assessment. Radionuclide concentrations for normal operations shielding calculations are given in the 2011 Preliminary Safety Report (PSR) (Table 7-18 of OPG 2011b). The net volume of a retube container for pressure tube wastes (RWC-PT) is 0.8 m^3.

For normal operations, the main concern is the external dose to workers during handling. As a result, the list of radionuclides focussed on gamma emitters and was not comprehensive for alpha and beta emitters (Table 7-18 of OPG 2011b). The original values were based on a plausible 10-year decay period at surface before emplacement in the DGR. The modelled retube waste package was compliant with the DGR Waste Acceptance Criteria (WAC) for package dose rate.

The revised pressure tube container concentrations in Table 1 simply reflect the recent changes to the inventories, decayed for 10 years. This results in a package dose rate that is higher than WAC. *Such a waste package would not be accepted at the DGR without further shielding or decay. However for conservatism and as a direct comparison, it is assumed to be accepted as-is at the DGR.*

For the accident assessment, as in the PSR Report, the radionuclide concentrations in Table 1 have been increased by a factor of 10 to represent an accident involving a small number of packages in which, conservatively, the radionuclide concentrations are higher than typical values. This higher-inventory package is assumed to have decayed for 15 years, which helps bring the implied package dose rates within the WAC. *As with the normal operation example above, for the revised concentrations, it was conservatively ignored whether the changes would result in the package dose rate exceeding WAC.*

Table 1: Radionuclide Concentrations in Pressure Tube Packages

Radionuclide	Half-Life (Years)	Normal Operations (Shielding Calculations)	Accident Assessment
		Revised Concentrations (Bq/m^3) [a]	Revised Concentrations (Bq/m^3) [b]
Am-241	4.3E+02	3.5E+08	3.4E+09
C-14	5.7E+03	2.6E+12	2.6E+13
Ce-144	7.8E-01	2.5E+00	2.9E-01
Cm-244	1.8E+01	8.2E+09	6.7E+10
Co-60	5.3E+00	2.1E+13	1.1E+14
Cs-134	2.1E+00	3.1E+09	5.9E+09
Cs-137 [c]	3.0E+01	1.5E+11	1.4E+12
Eu-152	1.3E+01	3.1E-01	2.3E+00
Eu-154	8.8E+00	4.2E+03	2.8E+04
Fe-55	2.7E+00	1.2E+13	3.2E+13

Radionuclide	Half-Life (Years)	Normal Operations (Shielding Calculations)	Accident Assessment
		Revised Concentrations (Bq/m^3) [a]	**Revised Concentrations (Bq/m^3) [b]**
Fe-59	1.2E-01	3.7E-12	1.1E-23
H-3	1.2E+01	2.7E+12	2.0E+13
Mn-54	8.6E-01	2.7E+08	4.7E+07
Ni-59	7.5E+04	8.9E+10	8.9E+11
Ni-63	9.6E+01	3.3E+13	3.2E+14
Nb-94	2.0E+04	2.3E+13	2.3E+14
Nb-95	9.5E-02	1.4E-08	2.1E-23
Pu-238	8.8E+01	1.5E+08	1.5E+09
Pu-239	2.4E+04	5.2E+07	5.2E+08
Pu-240	6.5E+03	9.3E+07	9.3E+08
Pu-241	1.4E+01	7.5E+08	5.9E+09
Ru-106	1.0E+00	2.7E-07	8.3E-08
Sb-124	1.7E-01	2.4E-06	3.4E-14
Sb-125	2.8E+00	1.2E+11	3.5E+11
Sn-119m	8.0E-01	3.8E+08	5.0E+07
Sr-90 [c]	2.9E+01	1.2E+10	1.0E+11
Te-125m	1.6E-01	1.0E+01	4.0E-08
U-235	7.0E+08	1.6E+03	1.6E+04
U-238	4.5E+09	4.1E+05	4.1E+06
Zr-93	1.5E+06	6.9E+11	6.9E+12
Zr-95	1.8E-01	2.0E-01	8.8E-09

Notes:

a. Revised pressure tube concentrations, including garter springs. 10 years decay before transfer to DGR.

b. Revised pressure tube concentrations, including garter springs. 15 years decay before transfer to DGR. The concentrations have also been increased by a factor of 10 to represent a maximum package inventory for accidents involving a small number of packages.

c. Cs-137 and Sr-90 assumed in secular equilibrium with their short-lived daughters.

3. PRECLOSURE SAFETY ASSESSMENT METHODOLOGY

The preclosure safety assessment methodology is described in Section 7.4 of the PSR for normal operations and Section 7.5 for accidents, specifically:

- Section 7.4.4.1 for assessment of external radiation on workers and public during normal operations; and
- Section 7.5.3 for methodology for consequence assessment for accidents.

Malevolent Acts scenarios are described in Section 6 of the Malfunctions, Accidents and Malevolent Acts Technical Support Document (AMEC NSS 2011). The methodology used in calculating dose to a member of the public was described in OPG's response to the Information Request EIS-06-248 (OPG 2012).

A member of the public is conservatively assumed to be at the nearest Bruce nuclear site boundary from the DGR.

4. RESULTS

4.1 RADIOLOGICAL SAFETY DURING NORMAL OPERATIONS

4.1.1 Radiological Assessment of Air and Water Emission from DGR on Workers and Public

During normal operations, the retube waste package arriving at the DGR is sealed tight (Section 8.3.3.1 of OPG 2006). Therefore, radioactive release to air and water and potential exposure to public during normal operations is not expected. In addition, there is no inhalation dose to the workers as the package is air tight.

4.1.2 Assessment of External Radiation on Workers and Public

Shielding calculations were carried out for workers handling representative low level and intermediate level containers during normal operations (Section 7.4.4.1 of OPG 2011b).

This scenario considers the handling of a single pressure tube waste container (RWC-PT) in the Waste Package Receipt Building (WPRB) (Scenario 2). Figure 7-6 of OPG (2011b) illustrates the receptor locations. The worker external dose results are given in Table 7-22 of OPG (2011b).

Table 2~~Table 2~~ shows the estimated external worker dose due to the handling of a RWC-PT for the revised pressure tube radionuclide concentrations. As discussed in Section 7.4.4.2 of OPG (2011b), the calculations indicate potentially high dose rate in the WPRB for the RWC-PT, and show that a wall around the WPRB staging area similar to WWMF Low Level Storage Building 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/h and that the dose rate in the office/main control room is below 10 mSv/year. These will be addressed during the detailed design.

Furthermore, the waste packages would be required to meet DGR WAC for package dose rate. These packages with the revised concentrations would not be consistent with the WAC, and therefore either further shielding or further decay would be included before the packages were accepted at the DGR. *That is, the results presented in Table 2 are conservative.*

Since RWC-PT containers are not stored in the WPRB staging area, the dose rate to the member of the public at the Bruce site boundary (about 1 km distant) due to handling of RWC-PT would be very low even with the revised inventories.

Table 2: Worker External Dose Rates for Retube- Pressure Tube (Scenario 2)

Receptor Location [a]	Location Description	Estimated Worker Dose Rate - PSR Results (Table 7-22 of OPG 2011b) (mSv/h)	Estimated Worker Dose Rate due to Revised Radionuclide Concentrations (mSv/h) [b]	Allowable Occupancy - PSR Results [c] (Table 7-22 of OPG 2011b) (h/year)	Allowable Occupancy at Estimated Dose Rate due to Revised Radionuclide Concentrations [b,c] (h/year)
R1	In the adjacent offices and control room	(d)	(d)	(d)	(d)
R2	Standing outside the WPRB [e]	5.7E-03	2.2E-02	1800	440
R3	Inside the package loading area (forklift driver moving waste packages, 2 m away)	4.8E-02	1.9E-01	210	53
R4	On the roof directly above the source	7.7E-04	3.1E-03	>2000	>2000

Notes:

a. Receptor location is shown in Figure 7-6 of OPG (2011b).
b. Modelled waste package exceeds DGR WAC for dose rate, and would require additional shielding or decay to be accepted at DGR. But this is conservatively ignored in this analysis.
c. Allowable occupancy without other mitigating measures, based on OPG occupational dose target of <10 mSv/year (footnote in Table 7-22 of OPG 2011b).
d. Detailed design of WPRB building/wall will ensure that workers in this location are below 10 mSv/year dose target (footnote in Table 7-22 of OPG 2011b).
e. 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/h (footnote in Table 7-22 of OPG 2011b).

4.2 ACCIDENT ASSESSMENT

The accident assessment considered the potential consequences of bounding scenarios for fire, container breach (low and high energy), and inadequate package shielding (Section 7.5.1.5 of OPG 2011b). Retube waste packages are robust and designed not to fail under accident conditions including drop from stacking height (Section 8.3.6.1 of OPG 2006). In Section 7.5.1.5 of OPG 2011b, the retube- end fittings container (RWC- EF) was considered as representative retube waste for analysis of consequences of a high energy breach due to cage fall in the underground repository.

Breach of RWC-PT is considered here to study the implication of the revised inventory in pressure tubes. Both high energy breach due to cage fall and low energy breach in the emplacement room are analysed and reported below. Fire scenario is not considered as retube waste and containers are sealed and not combustible. Inadequate package shielding is discussed in Appendix A.2.2.3 of OPG 2011b).

The assumed radionuclide concentrations for pressure tube package with revised inventory are given in Table 1. The radionuclide concentrations for pressure tube package for the PSR inventory are based on Table B.3 of OPG (2010), further decayed by 10 years and also increased by a factor of 10.

The acute accident dose limit is 1 mSv for the public and 50 mSv for the workers (Section 7.1.2.1 of OPG (2011b)).

4.2.1 Cage Fall with Retube Waste Package Breach

In a highly unlikely "what if" scenario, due to mechanical failure of the hoisting system (i.e., failure of multiple cables or the redundant braking system), the cage and a RWC-PT inside the cage are assumed to fall down the shaft into the shaft bottom located 30 m below the underground DGR working level (Appendix A.3.3.1 of OPG 2011b). The retube waste package is assumed to breach and release its entire contents.

The accident release factors for a pressure tube package are assumed to be the same as for an end fitting package, and are given in Table A-50 of OPG (2011b). The inhalation, immersion and external radiation pathways are considered, with the assumptions given in Appendix A.3.3.1 of OPG (2011b). The dose results for RWC-PT are given in Table 3 for workers and in Table 4 for the public for both the revised and PSR inventory cases. The results for RWC-EF, listed in Tables A-51 and A-52 of OPG (2011b), are also given in the following tables.

Table 3 shows that the total radionuclide doses to workers over a 5 minute period (through inhalation, immersion and external radiation) are less than the acute accident dose limit for workers (50 mSv) for both the revised and PSR retube- pressure tube inventory cases and retube- end fittings case. Similarly, Table 4~~Table 4~~ shows that the total dose to the public (through inhalation and immersion) at the nearest Bruce nuclear site boundary from the DGR over 1 hour exposure duration is much less than the acute accident dose limit for public (1 mSv) for both the revised and PSR RWC-PT inventory cases and RWC-EF case. The key dose contributors are Nb-94, Co-60 and, for the revised pressure tube inventory, Cm-244.

Table 3: Dose to Workers - Cage Fall with Retube Waste Package Breach

Selected Waste Category	Inhalation (mSv)	Immersion (mSv)	External Radiation (mSv)	Total (mSv)
Retube- Pressure Tubes (Revised Inventory)	6.3E+00	1.3E-01	< 1.0E-06	6.4E+00
Retube- Pressure Tubes (PSR Inventory)	4.5E+00	8.5E-02	< 1.0E-06	4.5E+00
Retube- End Fittings	5.6E+00	2.3E-01	< 1.0E-06	5.8E+00

Table 4: Dose to Public - Cage Fall with Retube Waste Package Breach

Selected Waste Category	Inhalation (mSv)	Immersion (mSv)	Total (mSv)
Retube- Pressure Tubes (Revised Inventory)	4.2E-03	3.8E-04	4.6E-03
Retube- Pressure Tubes (PSR Inventory)	2.1E-03	2.4E-04	2.3E-03
Retube- End Fittings	3.4E-03	6.7E-04	4.1E-03

4.2.2 In Room Retube Waste Package Breach

The retube waste package is robust and designed not to fail under accident conditions, including a drop from stacking height (Section 8.3.6.1 of OPG 2006). Therefore, releases from breached package are expected to be minimal. Because the package remains intact, potentially only gaseous radionuclides and very fine particulate might be released. Therefore potential impacts due to release of radioactive particulates/volatile species (through inhalation and immersion) are considered only.

In this scenario, a row of RWC-PTs (3) is assumed to be breached. The RWC-PT can be stacked two high and three wide in the emplacement room, so this is equivalent to the top front row of containers falling.

The accident release factor parameters for the pressure tube package, which are the same for end fittings and for pressure tubes, are given in Table 5. They are based on Tables 7-32, 7-33 and 7-34 of OPG (2011) and a leakpath factor (LPF) of 1.

Table 5: Accident Release Factor Parameters - In-Room Retube Package Breach

Selected Waste Category	# of Packages	Damage Ratio (DR)	Airborne Release Fraction (ARF)	Respirable Fraction (RF)	LPF
Retube - Pressure Tubes	3	0.05	0.0001	0.1 [a]	1

Note:

a. RF for volatile species such as gaseous C-14 as CO_2 and H-3 as tritiated water is taken to be 1. In the breach scenario, 25% of the released C-14 is considered as particulate, while 75% is considered as CO_2 (Section 7.5.3.4 of OPG 2011b). 100% of the released H-3 is volatile.

The dose results for RWC-PT are given in Table 6 for workers and Table 7 for the public for both the revised and PSR inventory cases. Table 6 shows that the total radionuclide doses to workers over a 5 minute period (through inhalation and immersion) are less than the acute accident dose limit for workers (50 mSv) for the pressure tube package. Similarly, Table 7 shows that the total dose to the public (through inhalation and immersion) at the nearest Bruce nuclear site boundary over 1 hour exposure duration is much less than the acute accident dose limit for public (1 mSv). The key dose contributors are Nb-94, Co-60 and, for the revised inventory, Cm-244.

Table 6: Dose to Workers - In-Room Breach of Pressure Tube Package

Selected Waste Category	Inhalation (mSv)	Immersion (mSv)	Total (mSv)
Retube- Pressure Tubes (Revised Inventory)	4.7E-01	2.0E-02	4.9E-01
Retube- Pressure Tubes (PSR Inventory)	3.3E-01	1.3E-02	3.5E-01

Table 7: Dose to Public - In-Room Breach of Pressure Tube Package

Selected Waste Category	Inhalation (mSv)	Immersion (mSv)	Total (mSv)
Retube- Pressure Tubes (Revised Inventory)	3.1E-04	5.7E-05	3.7E-04
Retube- Pressure Tubes (PSR Inventory)	1.6E-04	3.7E-05	1.9E-04

4.3 MALEVOLENT ACTS

The Malevolent Acts considered the following scenarios as provided in OPG's response to the Information Request EIS-06-248 (OPG 2012):

a) Deliberately driving a forklift into a package or dropping a package during handling

b) Pushing a package or vehicle into the shaft

c) Setting waste packages on fire

d) A person using an explosive or incendiary device

e) Remote military-style attack from the site boundary

f) Aircraft crash.

The public dose estimate for each of the above scenario is given in OPG's response to the Information Request EIS-06-248. The radionuclide concentrations for retube- pressure tube package are given in Table 1. The potential radiological consequences from retube- pressure tube containers to a member of the public are discussed below:

- For Scenario (a), the radiological consequence of the malevolent act would be limited to the breach of the retube waste packages directly affected. Since the retube waste packages are robust, the radiological consequence to a member of the public is limited. The public dose is estimated to be $\leq$ 0.0004 mSv for the revised inventory case and $\leq$ 0.0002 mSv for the PSR inventory case due to the breach of three RWC-PTs (Table 7).

- For Scenario (b), the radiological consequence from pushing a RWC-PT into the shaft is estimated to be about 0.005 mSv to a member of the public for the revised inventory case and about 0.002 mSv for the PSR inventory case (Table 4~~Table 4~~).

- Retube waste is not combustible, and therefore Scenario (c) is not applicable.

- For Scenario (d), the consequences would be limited by the amount of explosives that an employee could smuggle into the DGR and place near a retube waste package, e.g., during transfer to underground. As discussed in OPG's response to the Information Request EIS-06-248, the consequence of an explosion may be estimated based on experimental data on the fragmentation of metal from a pressure impulse directed outward through the material (Section 3.3.1.3 of NRC 1998). The experimental data correlates the airborne release fraction (ARF) and respirable fraction (RF) to the ratio of inert mass to the mass of high explosive, specifically, the TNT-equivalent mass, referred to as the mass ratio. The reference data provides estimates for mass ratios up to 24.

  For the purpose of this calculation, the mass of explosive is taken to be 100 kg (equivalent to about 160 kg of TNT). The retube waste package is robust and heavy (a pressure tube package weighs about 29,100 kg (pg. 122 of OPG 2010). Therefore, the mass ratio for one package is about 180. Consequently, using data for a mass ratio of 24 effectively assumes much more explosives.

  At a mass ratio of 24, the ARF and RF are found to be 0.366 and 0.0242 respectively (Table 3-6 of NRC 1998). The damage ratio (DR) and the leakpath factor (LPF) are conservatively set to 1. The public was assumed to be exposed for one hour at the nearest Bruce site boundary; the atmospheric dispersion factor (ADF) for the public is given in Table 7-36 of OPG (2011b). The public dose due to breaching of one RWC-PT resulting from detonation of explosives is estimated to be 3 mSv for the revised inventory case and 2 mSv for the PSR inventory case. This ~~exceeds the criterion for public for accidents, but it~~ is around the annual natural background dose level of 2 mSv.

- For Scenarios (e) and (f), the radiological consequence of this malevolent act would be limited by the number of waste packages in the WPRB. Since retube waste packages are not stored in the WPRB staging area, the retube waste package is not affected by remote military-style attack from the site boundary and aircraft crash. Instead, because the retube waste packages are emplaced underground, they are protected by several hundred metres of rock.

In summary, the estimated public dose for each of the Malevolent Acts scenarios except Scenario (d) is much less than ~~the acute accident dose criterion of~~ 1 mSv ~~for public (Section 7.1.2.1 of OPG 2011b)~~ for both the revised and PSR pressure tube inventory cases. For Scenario (d), the estimated ~~public~~ dose to a person at the nearest site boundary from detonation of explosives is around the annual natural background dose of 2 mSv.

5. CONCLUSIONS

Analyses were performed to show how the revised retube-pressure tube inventory affects the preclosure safety assessment of the DGR and to estimate the potential public dose of malevolent acts.

- For normal operations, there are no impacts to the public from airborne and waterborne release from the retube waste package and no inhalation dose to the workers due to air-tightness of the package. However, worker doses need to be monitored when handling the retube waste. The potential external dose rates to the workers have increased as a result of the revised retube-pressure tube concentrations, mostly due to an increase in Co-60 concentration. However, the radiation fields on all waste packages are monitored, and must meet the DGR WAC, either through additional shielding or longer decay time before transfer to the DGR. If necessary, the dose rates would be further reduced by limiting worker exposure time, use of shielded forklifts, and/or use of greater stand-off distances. This would be considered further within the context of the final ALARA assessment.

- The total doses to workers for both accident assessments (i.e., cage fall and low energy breach of retube-pressure tube packages) over a 5 minute period are less than the acute accident dose limit for workers (50 mSv) for both the revised and PSR pressure tube inventory cases. In addition, the total doses to the public at the nearest Bruce nuclear site boundary over 1 hour exposure duration are much less than the acute accident dose limit for public (1 mSv) for both the revised and PSR pressure tube inventory cases.

- For the Malevolent Acts scenarios, the estimated public doses are much less than 1 mSv for both the revised and PSR pressure tube inventory cases. The exception is the scenario involving using explosives on a retube waste package, where the estimated public dose is around the annual natural background dose of 2 mSv for both the revised and PSR pressure tube inventory cases.

For normal operations, the estimated worker external dose rates for the revised pressure tube inventory case are about 4 times higher than those for the PSR pressure tube inventory case with the same package and decay times. However, such a waste package in reality would not be accepted at the DGR without further shielding or decay if it did not meet the DGR WAC, so the actual difference would be smaller.

For accidents, malfunctions and malevolent acts, the estimated doses for the revised pressure tube inventory case are up to about twice of those for the PSR pressure tube inventory case.

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Attachment C to OPG Response to IR-EIS-13-514

WASTE INVENTORY VERIFICATION PLAN

WASTE INVENTORY VERIFICATION PLAN

CONTENTS

LIST OF TABLES

Page

LIST OF FIGURES

Page

1. PURPOSE

The purpose of this document is to summarize the activities underway and planned at Ontario Power Generation (OPG) to continue to measure and verify the properties of the Low & Intermediate Level Wastes (L&ILW) arising from operations and refurbishment of OPG-owned or operated nuclear generating facilities and intended for disposal in the proposed Deep Geologic Repository (DGR). This document has been prepared in response to Information Request EIS-13-514. The work is implemented by formal Plans within the OPG management system.

2. OBJECTIVE

The objective of this waste inventory verification plan is to determine with reasonable confidence the radionuclide activity to be placed in the DGR. It covers the next several years leading to application for an Operating Licence.

3. BACKGROUND

OPG is proposing the development of a Deep Geologic Repository (DGR) for the long-term management of L&ILW from OPG-owned or operated nuclear generating facilities. The DGR would be located on the Bruce nuclear site.

L&ILW has been stored at the Bruce nuclear site since the start of the OPG (then Ontario Hydro) nuclear power program in the early 1970's. These wastes are currently largely stored at OPG's Western Waste Management Facility (WWMF) in interim at-surface and in-ground storage structures. Approximately 95,000 m^3 of waste packages are presently at WWMF.

L&ILW consists of a variety of waste types which are generated from activities in support of CANDU power stations, and in particular from the operations of the Pickering, Bruce and Darlington stations. It does not include high-level waste or used fuel.

All waste packages received at WWMF are characterized by dose rate measurements and other physical properties required to ensure that they meet the Waste Acceptance Criteria for either processing or storage at WWMF. Other waste characterization activities have also been underway since the 1970's to obtain more detailed information on waste streams and packages. Starting around 1999, OPG initiated a formal waste characterization program to provide consistent data on a range of alpha, beta and gamma emitting radionuclides in the L&ILW. The initial focus of this program was on operational wastes.

Operational wastes are those generated during the routine operations of the reactors, and include cleaning materials, tools, equipment and filters that are lightly contaminated during use in the reactor buildings, as well as filters, ion exchange resins and replaceable core components that are exposed to Primary Heat Transport (PHT) or moderator or other higher activity sources.

In addition to operational wastes, refurbishment wastes include components that are replaced only as part of a major reactor mid-life refurbishment. This is primarily steam generators and fuel channel components. The initial design basis for the DGR included refurbishment components from Pickering B, Bruce A and B, and Darlington. (Pickering A refurbishment wastes are stored at the Pickering site and will be disposed of during station decommissioning.)

Currently only Bruce A refurbishment components are in storage at WWMF as refurbishment of the other stations are still in the planning stages. It has also been decided not to refurbish Pickering B, so that retube waste will therefore not be generated from this station.

The preliminary design of the DGR and its supporting preliminary safety case have been based on assumptions and data that are derived primarily from these waste characterization investigations. The status of the waste information was documented in the DGR Reference Inventory report. The first comprehensive summary of information relevant to the proposed DGR was prepared as the Revision 0 (R0) Reference Inventory report in 2006. This information was used in the draft preliminary safety assessment, and served in part to identify what were the likely important radionuclides and waste types. This in turn identified priorities for future work in waste characterization.

The next Revision 1 (R1) Reference Inventory report was released in 2008, and was posted to the OPG website for public information. R2 was an internal update, and R3 was released publicly in 2010. R3 (OPG 2010) was used as the basis for the DGR Preliminary Safety Report, submitted in 2011.

The Reference Inventory report provides a comprehensive summary of the information available on the operational and refurbishment L&ILW intended for placement in the OPG DGR. It includes waste volume projections based on an assumed operating lifetime for the current OPG-owned or operated reactors. It includes radionuclides relevant to long-term safety, waste physical characteristics, and a description of the main container types. R3 (OPG 2010) was released as a public record to provide a detailed technical summary of the basis for the DGR inventory.

The R3 Reference Inventory report was based on a combination of measurements, models and estimates. Models and estimates are used in part because some of the inventories had not yet been measured. In particular, there was less information on radionuclides considered to be less important to the safety case. Also about half of the estimated 200,000 m^3 DGR capacity has been received at the WWMF. The remainder of the wastes will be generated over the next approximately 40 years of planned operation of the existing reactors. This includes most of the refurbishment wastes, which have not yet been generated. A description of the uncertainties was provided in the Reference Inventory report.

Accordingly, OPG has an ongoing waste characterization program that is improving the information and reducing uncertainties. The waste characterization program also includes characterizing the physical composition of the wastes, including the presence of hazardous elements, and the amounts of metals and organic materials.

This document presents the waste inventory verification plan. It covers the next several years leading to application for an Operating licence. It is implemented within the OPG governance system through a formal set of documents, which include the Waste Characterization Work Program, the multi-year Waste Characterization Plan, and annual work plans.

Section 4 describes key aspects of the Reference Inventory, Sections 5 to 8 describe the methods used for operational LLW, operational ILW and refurbishment L&ILW. The verification plan and timelines are described in Section 9.

4. REFERENCE INVENTORY

Table 4.1 provides an overview perspective on the waste characteristics. It groups the various waste types into eight main categories. For each category, it provides the estimate of total radioactivity and the number of containers that would be in the DGR at 2062, the earliest assumed time of closure. This information is from the 2010 Reference Inventory report.

Table 4.1 provides some perspective on the importance of the different wastes types. It indicates for example that the bulk of the radioactivity is in the Retube Waste, and that the bulk of the waste volume (represented approximately by the number of containers) is Non-Processible Waste.

The *relevant radionuclides* for the DGR are identified in the Reference Inventory report. This is based on the following considerations:

- Radionuclides that are measured to contribute significantly to the total activity in wastes as-received at WWMF.
- Radionuclides that were identified as potentially important to safety assessment using preclosure and postclosure screening analyses.
- Comparison with radionuclides tracked in similar inventory reports for other nuclear reactor waste management organizations.
- Short-lived daughter radionuclides are generally included with their parent, assuming secular equilibrium.

The *key radionuclides* for the DGR are those that dominate the dose consequences in normal or abnormal scenarios at the DGR. Based primarily on the results documented in the Preliminary Safety Report (OPG 2011) and its supporting analyses, the following are key radionuclides for preclosure and/or postclosure safety: H-3, C-14, Cl-36, Fe-55, Co-60, Ni-59, Ni-63, Zr-93, Nb-94, I-129, Cs-137, U-238, Pu-239, Cm-244.

Table 4.1: Summary Parameters

Waste Category	Total Activity at DGR in 2062 (TBq)	Number of Containers in DGR at 2062
LLW		
Incinerator ash	0.3	1,100
Compacted wastes	280	7,500
Non-processible wastes	580	32,200
Low level resins and sludges	1.6	3,900
Steam generators (from refurbishment)	17	500
Sub-total LLW	**880**	**45,200**
ILW		
Ion exchange (IX) resins	5,600	1,600
Filters, core components, miscellaneous	130	4,500
Retube (from refurbishment)	10,000	1,400
Sub-total ILW	**16,000**	**7,400**
Total	**17,000**	**52,600**

5. METHODS

The characteristics of the wastes are determined using direct measurement, scaling factors and activation calculations.

5.1 Direct Measurement

Direct measurement of radionuclides is typically by gamma spectrometry for gamma emitters, and by radiochemical analysis for alpha and beta emitters. The typical radiochemical processes involve liquid scintillation and alpha/beta proportional flow counting. The standard radiochemical methods used for some important radionuclides are listed in Table 5.1.

However for some radionuclides, especially those that are present at low levels, special method development may be needed. Of particular relevance to the post-closure safety assessment for the L&ILW DGR is Zr-93. This is a dominant radionuclide in the long-term and is mostly in retube waste. As it is long-lived and a beta-emitter, it is not readily measured, especially with the large background of stable zirconium isotopes present in the wastes. As it is not particularly important for operational safety, it has not been widely studied within prior OPG radionuclide characterization studies. Therefore, over the past two years, OPG has supported work to develop a standard radiochemical measurement approach (Wu et al 2013).

5.2 Scaling Factors

Scaling factors may be used for difficult-to-measure (DTM) radionuclides, typically alpha or beta emitters. In this approach, the amount of a DTM nuclide is estimated based on measurement of an easy-to-measure (ETM) nuclide and a scaling factor. The method is applicable when there is a correlation between the concentration of the DTM and the ETM nuclides.

Typically ETM nuclides are gamma-emitters like Co-60, Cs-137 and Nb-94.

Scaling factors may be developed based on direct measurements of DTM and ETM radionuclides (using methods as outlined in Section 5.1), or through models or calculations.

Scaling factors are semi-empirical. They have been found to be useful for a variety of radionuclides, and they are widely used internationally (ISO 2007, IAEA 2009). As they are semi-empirical, however, the specific applications need to be verified with measurements.

5.3 Activation Calculations

For in-core components, neutron activation calculations can be used to determine the radionuclide concentrations. Within Canada, ORIGEN-S is the industry standard code (ORNL 2014).

Neutron activation calculations are particularly useful for activation of primary alloying elements. For activation of trace elements, the calculation accuracy depends on knowledge of the trace element composition, or at least of bounding values from material specifications. Activation calculations are not applicable for radionuclides present from other mechanisms, such as from sorption from coolant.

Neutron activation calculations are useful for projecting the end-of-life inventory in components not presently available as a waste and in particular retube wastes. They are also suitable for estimating inventories of radionuclides that may be present in smaller amounts and not easily measured. Finally activation calculations can account for variations in inventory due to flux profiles across the reactor core.

Table 5.1: Methods Used to Measure Radioactivity of Several Radionuclides

Radionuclide	Type of Matrix	Principles of Determination	Methodology of Chemical Separation
Co-60 Cs-137 Nb-94 (gamma emitters)	• Solid • Aqueous	• Gamma spectrometry	• Chemical separation is not generally needed for important gamma emitters for purposes of waste inventory characterization. • Samples may be allowed to decay so that dominant but shorter-lived gamma emitters decay, so that less intense but longer-lived gamma emitters can be measured.
Zr-93	• Solid	• Liquid scintillation counting.	• Challenge in Zr-93 analysis for pressure tubes is that stable Zr is also the dominant constituent of the matrix. • Sample dissolved in acid and further processed using wet chemistry procedures, including cleanup with IX resins, to remove interfering species.
Sr-90	• Solid • Sample dissolved in acid; insoluble material is heat fused and combined with acid digested sample.	• Liquid scintillation or beta counting is used to measure Y-90, the daughter product in equilibrium with Sr-90. • Beta counting is the preferred technique; it is particularly useful for lower activity samples.	• Solvent extraction is performed to separate Sr-90 from other radionuclides including the existing daughter product Y-90. • Y-90 is allowed to re-equilibrate with Sr-90. The time period for this is 7-10 days. • Sr-90 is estimated from the measured Y-90 activity.
Pu-238 Pu-239/40 Am-241 Cm-242 Cm-244 (alpha emitters)	• Solid • Sample dissolved in acid; insoluble material is heat fused and combined with acid digested sample.	• Alpha spectrometry	• Precipitations are performed to remove undesired elements and radionuclides (including Ni-63 if present). The aqueous acidic phase contains the desired alpha emitting radionuclides along with Fe-55 (if present). • The aqueous phase radionuclides are transferred on to ion exchange media and then sequentially eluted. All Pu species are thus separated from Am-241 and Cm species. Fe-55 (if present) is also separated out. • The radionuclides are then precipitated, filtered and counted.

Radionuclide	Type of Matrix	Principles of Determination	Methodology of Chemical Separation
Pu-241	• Solid • Sample dissolved in acid; insoluble material is heat fused and combined with acid digested sample.	• Liquid Scintillation Counting or Induction Coupled Plasma	• See method for determining alpha emitters. • Filtered precipitates containing Pu species are re-dissolved and prepared for Pu-241 analysis.
Fe-55	• Solid • Sample dissolved in acid; insoluble material is heat fused and combined with acid digested sample.	• Liquid scintillation counting	• See method for determining alpha emitters. • Eluant from ion exchange resin column is prepared for analysis.
Ni-63	• Solid • Sample dissolved in acid; insoluble material is heat fused and combined with acid digested sample.	• Liquid scintillation counting	• See method for determining alpha emitters. • N-63 is extracted from the precipitates (see first bullet under alpha emitters).
C-14	• Solid • IX Resins	• Liquid scintillation counting	• Generally digestion in acide is required to dissolve the matrix and free up C-14 present in CO_2 form. • C-14 is stripped from IX resins using acid, where the carbonate or bicarbonate species attached to resin is released as CO_2. • If C-14 is in non-CO_2 form, then sample must be combusted to convert all C-14 into CO_2 form.
Cl-36	• IX Resins	• Liquid scintillation counting	• Chlorine in the chloride form is stripped from the resin. • Series of precipitations and extractions are performed to remove interferences. • A large sample (~50 g) is typically required to obtain an appropriate Minimum Detection Limit (MDL).
I-129	• IX Resins	• Liquid scintillation counting	• Iodine is stripped using a basic solution. • Series of extractions are performed to remove interferences; also chemical treatments are performed to convert all the iodine to the highest oxidation state. • A large sample (~50 g) is typically required to obtain an appropriate MDL.

6. OPERATIONAL LLW

Table 6.1 presents the main types of Operational LLW, and the key activities to characterize and verify the inventory in these wastes.

Figure 6.1 shows a simple gamma spectrometry configuration for a side measurement of an LLW bin.

Table 6.1: Key Verification Activities for Operational LLW

Waste Category	Waste Types	Verification
Incinerator Ash	• Baghouse and bottom ash from current and previous incinerators	• Not expected to be significant waste type with respect to radionuclide inventory. • Complete sampling and analysis in particular, DTM nuclides in old ash and new bottom ash.
Compacted Wastes	• Baled wastes (older process) • Compacted wastes with current high-force compactor.	• Further sampling and analysis, in particular, DTM nuclides in old baled wastes.
Non-Processible Wastes	• Non-processible containers • Non-processible drums • Feeder pipes • Auxiliary heat exchangers • Other wastes (sealed sources, magnetite)	• Further sampling and analysis to ensure data representativeness due to waste heterogeneity. • Further waste composition data through container sampling, and review of waste receipt records and station/WWMF data. • Gamma spectrometry of containers to extend the marker nuclide information.
Low Level Resins and Sludge	• Low Level / Active Liquid Wastes resins and sludge	• Not expected to be significant waste type. • Complete sampling and analysis for DTM nuclides.

Figure 6.1: Gamma Spectrometry of LLW Container

7. OPERATIONAL ILW

Table 7.1 presents the main types of Operational ILW, and the key activities to characterize and verify the inventory in these wastes.

Figure 7.1 shows a typical sample probe used to collect an array of resin specimens from a resin waste tank or container.

Figure 7.2 shows the equipment that can be used to obtain a gamma spectrometry profile of high-activity core components as they are being unloaded from a transport package into an in-ground container.

Table 7.1: Key Verification Activities for Operational ILW

Waste Category	Waste Types	Verification
IX Resins - primary systems	• PrimaryHeat Transport (PHT) resins • Moderator resins • IX columns (Pickering)	• Key waste types for DGR, notably C-14. • Additional sampling and analysis to ensure C-14 inventory has low uncertainty, including ensuring data representativeness (station/unit differences). • Sampling and analysis also needed to provide complete coverage for all DTM nuclides.
IX Resins - auxiliary systems	• Fuel Bay filters • Tritium Removal Facility resins • Heavy Water Upgrader resins • CANDECON resins	• Additional sampling and analysis needed to ensure sufficient coverage of range of resins, and of DTM nuclides. • Future plans for use of CANDECON to be reviewed to determine importance of further CANDECON data.
Filters	• PHT and Fuelling Machine filters • Heavy Water Upgrader filters • Moderator purification system filters • Other miscellaneous filters	• Gamma spectrometry using equipment in Figure 6.1. • Additional sampling and analysis needed to ensure sufficient coverage of DTM nuclides, especially for Fuelling Machine filters. Due to high radiation fields, it may be most feasible to characterize DTM nuclides through crud samples from various systems.
Core components	• Flux detectors	• Sampling and analysis needed to validate activation analyses.

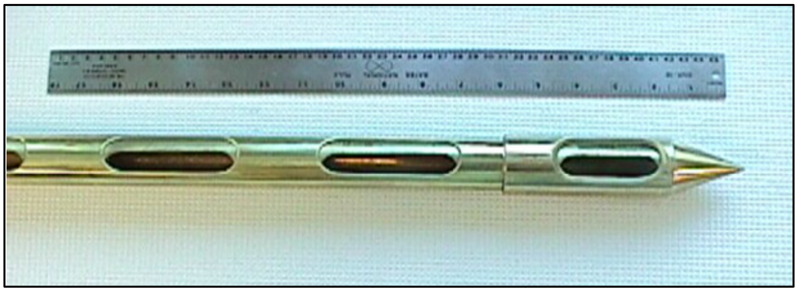
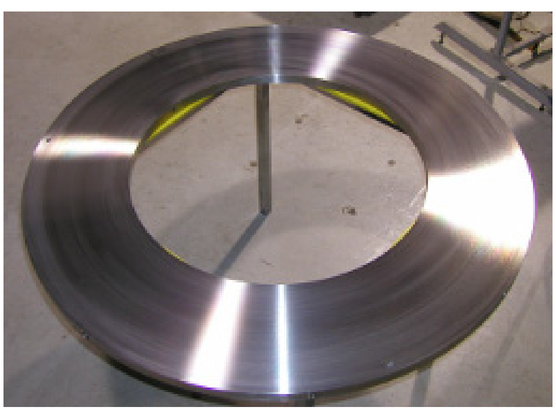


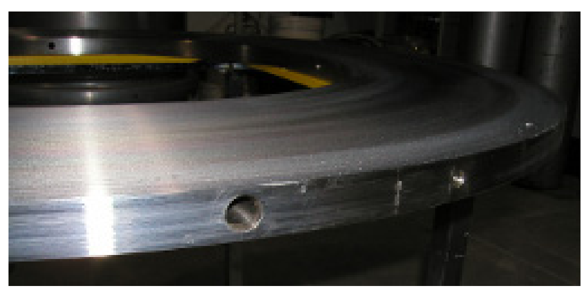
Figure 7.1: IX Resin Sample Probe

(a)

(b)



(c)



(d)

Figure 7.2: Equipment for Gamma Spectrometry of High Activity ILW Filters or Core Components during Placement into In-ground Containers

(a) Transfer Package; (b) Open In-ground Container; (c) Detector shielding disk placed between Transfer Package and In-ground Container; (d) holes in shielding disk where gamma spectrometers are placed. Measurements are obtained while the contents of the transfer packages are lowered into the in-ground containers.

8. REFURBISHMENT L&ILW

Table 8.1 presents the main types of Refurbishment L&ILW, and the key activities to characterize and verify the inventory in these wastes.

Figure 8.1 shows a photo of the oxiprobe delivery system, currently in use for obtaining axial profiles of activity along steam generator tubes.

Table 8.1: Key Verification Activities for Refurbishment L&ILW

Waste Category	Waste Types	Verification
Steam Generators (SGs)	• Shell and tubes	• SG gamma scanning data. • Complete sampling and analysis to ensure data representativeness (profile, end-of-life, data for other Bruce SGs planned for DGR).
Retube components	• Pressure tubes • Garter springs/ Girdle wires • End fittings and Shield Plugs • Calandria tubes • Calandria tube inserts	• Pressure tubes are a key DGR waste type, especially for Nb-94 and Zr-93. • Sampling and analysis to validate activation calculations for activation radionuclides and scaling factors for surface deposit radionuclides. • Data representativeness to consider all stations and axial profile. • Archived (older) samples may be used to validate the models at first, with verification against actual retube wastes after they are generated.

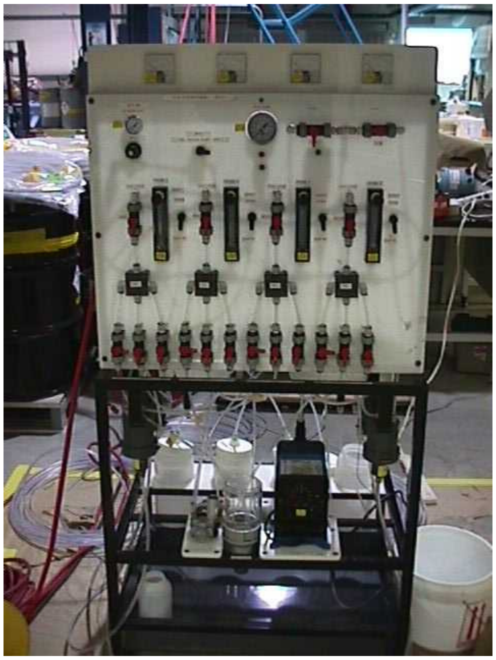
(a)  (b) 

Figure 8.1: Oxiprobe System for Measuring Gamma Activity in Steam Generators

(a) Delivery system controls, (b) Mockup showing probe insertion from steam generator bottom manway access into tubesheet.

9. PLAN

The main elements of the waste characterization plan are:

- Work Program Definition
- Data Quality
- Verification.

These are discussed below. This plan is compliant with international guidelines (IAEA 2007).

9.1 Work Program Definition

The Work Program defines the corporate responsibilities and business planning authority for undertaking waste characterization activities. This ensure that waste characterization is included within the business planning cycles for the stations, WWMF and other supporting groups. The waste characterization program will incorporate this Waste Inventory Verification Plan.

Status: OPG is presently defining a new Waste Characterization Work Program. The governance is expected to be completed by end-2014. The Work Program in turn is implemented through a multi-year Waste Characterization Plan.

9.2 Data Quality

Data quality define the targets for data from waste sampling. This is a graded program that includes minimum number of sampling, and expectations regarding more sampling for more important waste types. The current waste types are listed in Tables 5.1, 6.1 and 7.1.

Status: The waste characterization program is proceeding under the following guidelines:

- Screening. Acquire at least 3 data/nuclide/waste type for radionuclides identified in the Reference Inventory report. This would primarily serve as a screening test, i.e., it could provide positive confirmation that some nuclides in some waste types were sufficiently low that they were insignificant to the overall safety case and further data was a low priority. Or conversely that they were potentially significant enough to warrant further data. It would also provide a basic validation for activation calculation models. In general, the results would require evaluation to document that this data was sufficient for specific nuclide/waste types.

- Uncertainty basis. Acquire sufficient data for radionuclides identified as potentially important to the safety case to support the calculation of statistical quantities, notably the 95^{th} percentile upper confidence limit in mean value, using statistical software such as U.S. Environmental Protection Agency ProUCL (US EPA 2013). This upper confidence limit can be used to determine, for each nuclide/waste type, whether the uncertainty is important to the safety case and more data is desirable in order to reduce the upper confidence limit value. As a general rule, a minimum 10 data/nuclide/waste type are needed for statistical analysis. These data points should include at least 2 from each station where appropriate, and cover an extended timeframe, in order to provide basic information on variability between stations and over time.

- Data Representativeness. Additional waste sampling is conducted as needed to ensure data representativeness. This is a waste-type-specific judgement. In particular, more data would be needed if reactor-specific differences are significant (e.g. due to different fuel defect history) or if the inventory in the wastes is not uniformly distributed (e.g. steam generators, non-processible wastes). Testing for reactor differences would be guided by records of reactor operations and by monitoring for trending across the waste sampling program (e.g. whether one reactor has consistently higher inventories). This would also include checking whether waste activities are stable over time.

- Key radionuclides. For those radionuclides that are either important to the safety case, or are ETM nuclides that are widely used as a scaling factor basis, additional samples would be undertaken. This would be guided by the importance of the radionuclide, and by the uncertainty as indicated by the 95th percentile upper confidence limit. See Section 4 for a current list of key radionuclides.

9.3 Verification

Verification activities provide assurance that the inventory basis is correct.

Status:

- OPG waste characterization analyses are carried out in accredited laboratories under appropriate quality assurance programs.

- Waste characterization results are compared with those from other relevant programs, notably other CANDU reactors where available. Information that may be comparable includes scaling factors and key radionuclides.

- Reference waste characterizations are compared with measured package dose rate distribution (this provides a validation of gamma-emitting radionuclides).

- Conduct periodic interlaboratory comparisons about every 3 years, each time testing a different waste characterization aspect and guided by the importance to the DGR safety case.

In addition, in the near-term it is planned to conduct a review of the waste characterization program by an independent third party.

9.4 Analysis and Integration

The results of the waste characterization program are analyzed and integrated with prior data and models to provide, for each waste type, a best-estimate inventory, an uncertainty analysis supporting an upper bound inventory value, and.an estimate for the total projected DGR inventory.

9.5 Timeframe

Some waste characterization activities are presently underway. Other activities need to be scheduled for completion to support the application for an operating licence. Based on the current projected construction schedule, this means that the activities need to be complete by end-2021.

The approximate timeframe for these activities is as follows. The specific activities are defined as part of the five-year business planning cycle, which is updated annually.

2014

- Continue annual sampling and gamma spectrometry of various LLW containers.
- Characterization of current pressure tube samples.
- Sample and analysis of Pickering B IX resins.
- Complete development of assay method for Zr-93 in pressure tubes.

2015 - 2017

- Continue annual sampling and gamma spectrometry of various LLW containers.
- Sample and analysis of additional resin and filter specimens.
- Characterization of at least two samples of all retube waste types.
- Opportunistic analysis of other samples as become available.
- One interlaboratory comparison of measurement methods.
- Third party review of the waste characterization program.

2018-2021

- Continue annual sampling and gamma spectrometry of various LLW containers.
- Sample and analysis of additional ILW samples, including validation of the radionuclide characterization of refurbishment wastes.
- One interlaboratory comparison of measurement methods.
- Characterization to meet data quality objectives complete.

These results of the waste characterization program would be documented in technical reports, and in the container-specific information in the OPG Integrated Waste Tracking System, and summarized in an updated Reference Inventory report.

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