PMD 13-P1.171

File / dossier : 8.01.07
Date: 2013-08-13
Edocs: 4186577

Oral intervention from

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In the Matter of

À l'égard de

Ontario Power Generation Inc.

Ontario Power Generation Inc.

Proposed Environmental Impact Statement for OPG's Deep Geological Repository (DGR) Project for Low and Intermediate Level Waste

Étude proposée pour l'énoncé des incidences environnementales pour l'Installation de stockage de déchets radioactifs à faible et moyenne activité dans des couches géologiques profondes

Joint Review Panel

Commission d'examen conjoint

September 16 to October 12, 2013

16 septembre au 12 octobre 2013

Canada

Joint Federal Review of Ontario Power Generation's Proposed Deep Geologic Repository for Low and Intermediate Level Radioactive Wastes

DGR Joint Review Panel Hearing Written Submission in Support of an Oral Intervention

International perspectives and comparisons of standards for post-closure safety case assessments

Laura Bowman, J.D.

August, 2013

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Nuclear Waste – Post-Closure Safety Case for Deep Geologic Repository

If you don't know where you are going, how will you know when you get there?

International perspectives and comparisons of standards for post-closure safety case assessments

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August, 2013

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Executive Summary

What follows is a review of international standards and practices for the review of the post-closure safety assessment of disposal facilities and deep geologic repositories for low and intermediate level radioactive waste (L&ILW) and where relevant, other deep geologic repositories. This review is intended to assist the Joint Review Panel in understanding appropriate conditions and approval considerations for reviewing the Deep Geologic Repository DGR proposal of Ontario Power Generation.

The Joint Review Panel is tasked with determining whether the post-closure assessment and safety case is adequate, and what is the appropriate methodology for conducting a post-closure safety assessment. Canadian standards set under the *Nuclear Safety and Control Act* are very general and do not provide sufficient detail to guide the Joint Panel's review of the adequacy of a post-closure safety assessment or safety case. This submission contains an overview of regulatory requirements used for long-term management and disposal of a nuclear wastes in other states and those developed by the International Atomic Energy Agency (**IAEA**).

In comparison, other countries have adopted more specific regulatory requirements for disposal of radioactive wastes. These set out in some detail the expectations that licensees would have to meet in preparing a post-closure safety assessment. This is consistent with the approach recommended by the IAEA.

The Joint Review Panel should use the regulatory approaches of other countries and the IAEA as a guide to assess appropriate conditions and to assess the robustness of the models employed by Ontario Power Generation (**OPG**). In lieu of specific regulations to govern the conditions for licensing, the Panel should review the proposal in light of IAEA recommendations and by comparison to international standards for other geologic repositories and low level waste disposal facilities.

Background

The IAEA is authorized by its statute to develop standards for safety for the protection of health and the minimization of danger to life and property and to provide for their application. Further to this, the IAEA regulates nuclear waste repositories through Safety Fundamentals, Safety Requirements, Safety Guides and Safety Standards. The IAEA produces these guides to support the implementation of the *Convention on Nuclear Safety*[1] and the *Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management*.[2] Generally, the IAEA defines waste management in terms of storage, disposal and predisposal stages. Deep Geologic Repositories are normally evaluated as disposal facilities.[3] The post-closure assessment is part of the assessment of a disposal facility for radioactive waste, by implication if a facility is closed, it is intended to be used for permanent disposal.

The IAEA has produced *Safety Requirements for the Pre-disposal Management of Radioactive Waste* as part of its *General Safety Requirements* (**GSR-5**).[4] These requirements outline methodology for preparation of a safety case for pre-disposal stages of waste management. The *General Safety Requirements* also provide Safety Standards for conducting safety assessments but these are not specific to waste facilities.[5] More detailed guidance exists for Deep Geologic Repositories in IAEA technical documents such as the technical document for the Siting, Design and Construction of a Deep Geological Repository for the Disposal of High Level Waste and Alpha Bearing Waste.[6] However, this guidance does not necessarily apply to all L&ILW.

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[1] *Convention on Nuclear Safety*, 17 June 1994, 33 I.L.M. 1514 (entered into force on 24 October 1996).

[2] *Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management*,29 September 1997, 36 I.L.M. 1431 (entered into force 18 June 2001). Also see *The Convention on the Physical Protection of Nuclear Material*, 3 March 1980, online: International Atomic Energy Agency <http://www.iaea.org/Publications/Documents/Infcircs/Others/inf274r1.shtml>

[3] Also see ICRP 1998, ICRP Publication 81: Radiation Protection Recommendations as Applied to the Disposal of Long-lived Solid Radioactive Waste, International Commission on Radiological Protection, 1998

[4] IAEA, General Safety Requirements 5, (SSR-5) “Predisposal Management of Radioactive Waste” (Vienna: IAEA, 2009) online: <http://www-pub.iaea.org/MTCD/publications/PDF/Pub1368_web.pdf

[5] IAEA, General Safety Requirements 4, (GSR-4) “Standards Safety Assessment for Facilities and Activities” (Vienna: IAEA, 2007), online: < http://www-pub.iaea.org/MTCD/publications/PDF/Pub1375_web.pdf>

[6] IAEA, TECHDOC “Siting, Design and Construction of a Deep Geological Repository for the Disposal of High Level Waste” (Vienna: IAEA, 1990) online: <http://www-pub.iaea.org/MTCD/Publications/PDF/te_563_web.pdf>

The IAEA has produced *Specific Safety Guides* (**SSG**) that are applicable to disposal of Low and Intermediate level wastes (**L&ILW**) as well as high-level wastes. For example, SSG-14, *Specific Safety Guides on Geological Disposal Facilities for Radioactive Waste*.[7] and SSG-15, *Specific Safety Guide for the Storage of Spent Nuclear Fuel*.[8] Responsibilities of the national regulatory body for multiple waste types are set out in Specific Safety Requirements (**SSR**) document SSR-5 *Disposal of Radioactive Waste*.[9]

SSR-5 provides that a safety case shall describe all safety relevant aspects of the site including:

- design of the facility
- managerial control measures
- regulatory controls
- demonstrate the level of protection of people and the environment provided
- shall provide assurance to the regulatory body and other interested parties that safety requirements will be met.

With respect to the requirements of a safety case and safety assessment, there is also SSG-23 *The Safety Case and Safety Assessment for the Disposal of Radioactive Waste*.[10] SSG-23 is not specific to deep geologic repositories or a specific type of radioactive waste, but instead provides guidance for safety case and safety assessments for all types of radioactive waste disposal facilities.[11] The latter is intended as guidance for the implementation of the *Joint Convention* as well as the International Commission for Radiological Protection (**ICRP**) recommendations.

SSG-14 defines geological disposal broadly, including disposal facilities for all types of solid radioactive waste.[12] Notably, some aspects of SSG-14 may not be applicable to waste that is suitable

[7] IAEA, Specific Safety Guide 14, (SSG-14) "Geological Disposal Facilities for Radioactive Waste" (Vienna: IAEA, 2011), online: <http://www-pub.iaea.org/MTCD/Publications/PDF/Pub1483_web.pdf

[8] IAEA, Specific Safety Guide 15, (SSG-15) "Storage of Spent Nuclear Fuel" (Vienna: IAEA , 2011), online: <http://www-pub.iaea.org/MTCD/publications/PDF/Pub1503_web.pdf>

[9] IAEA, Specific Safety Requirements 5, (SSR-5) "Disposal of Radioactive Waste" (Vienna: IAEA, 2011), online: <http://www-pub.iaea.org/MTCD/publications/PDF/Pub1449_web.pdf >

[10] IAEA, Specific Safety Guide 23, (SSG-23) "The Safety Case and Safety Assessment for the Disposal of Radioactive Waste" (Vienna: IAEA, 2012).

[11] *Ibid*. at 2.

[12] SSG-14, *supra* note 7 at 1.

for disposal in a landfill or near-surface facility.[13] However, where waste is to be disposed of in a deep geologic repository, SSG-14 provides guidance for the appropriate regulatory standards states should apply to comply with SSR-5, and is not limited in its application to high-level waste.

SSG-14 starts from the premise that operational control will not be effective in a disposal facility for radioactive waste and this differentiates the safety approach for deep geologic repositories from other types of facilities.[14] SSR-5 requires the containment of radioactive waste and isolation from the biosphere.[15] The SSG-14 refers to pre-operational, operational and post-closure periods, for the purpose of this report, comparative international standards applicable to the post-closure period will be reviewed. SSR-5 requires that safety be achieved by passive means.[16]

Safety Case assessment is dealt with in even more detail in the IAEA SSG-23, *The Safety Case and Safety Assessment for the Disposal of Radioactive Waste.* This guide covers low, intermediate and high level waste types of radioactive waste and disposal facilities.[17]

The IAEA and states have developed and promoted the concept of a safety case as follows:

- An integrated collection of arguments and evidence to demonstrate the safety of a *facility.*
- includes a *safety assessment*, and
- the robustness and reliability of the *safety assessment* and the assumptions made therein.[18]

In addition to the overall requirements for deep geologic repositories set out by the IAEA, it is important to recognize that the IAEA has also provided guidance on how to assess disposal approaches for low and intermediate waste.[19] This confirms that post-closure safety case assessment features such as timescales, human intrusion, seismic events and robust multiple lines of argument apply in the context of low and intermediate level waste.

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[13] *Ibid*. at 3.
[14] *Ibid*. at 2.
[15] Requirement 8-9, see SSG-14, *supra* note 7 at 14-15.
[16] SSG-14, *supra* note 7 at 18.
[17] SSG-23, *supra* note 10.
[18] CNSC G-320, IAEA glossary
[19] IAEA, Nuclear Energy Series, (No. NW-T-1.20), “Disposal Approaches for Long Lived Low and Intermediate Level Radioactive Waste” (IAEA: Vienna, 2009).

SSG-14 provides that the regulatory body should provide regulations and guidance setting out requirements for the safety case, requirements for closure, and for the waste, waste form, disposal container backfill and sealing material.[20]

SSG-14 particularizes the content of a safety case and safety assessment pursuant to Requirements 12-13 of SSR-5 which requires that the safety case and safety assessment must be sufficiently detailed and comprehensive to provide assurance that Safety Requirements *will be met*.[21]

Doses

In combination, SSG-14 and SSG-23 set out particularized expectations for both safety case and safety assessments. SSG-23 indicates that both quantitative and qualitative criteria should be used by the regulatory body.[22] SSG-23 also discusses dose criteria in the post-closure context, and highlights the requirement set out in SSR-5 for quantitative regulatory criteria stating that the calculated dose or risk to the representative person who might be exposed in the future… does not exceed a dose constraint of 0.3 msv/year.[23] It also provides optimization criteria for human intrusion of 1 msv/year.[24] SSG-23 provides that the timescales on which these criteria must be met "need to be specified in the context" by the regulator.[25]

Timescales.

With respect to timescales, SSG-14 provides that "the regulatory body should stipulate or provide guidance concerning timescales for safety assessments" and that dose analysis should be required for at least several thousand years and may be extended in support of analysis of peak dose.[26] SSG-14 further provides that regulatory criteria will typically specify characteristics of exposed groups or individuals to be used in dose calculations (the concepts of critical group). SSG-14 states that identifying and

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[20] SSG-14, *supra* note 7 at 10.
[21] SSG-14, *supra* note 7., at 20-21.
[22] SSG-23, *supra* note 10 at 23.
[23] *Ibid*. at 49.
[24] *Ibid*. at 49.
[25] *Ibid*. at 50.
[26] *Ibid*. at 22.

addressing uncertainties is a major part of the post-closure assessment.

A range of techniques should be used to evaluate uncertainties in post-closure performance of the facility. Detailed models of particular parts of the disposal system and particular events and processes should be used to investigate behaviour and to decide how to handle system components and features, events and processes in the overall safety assessment.

Moreover, SSG-23 provides that clear end points for the assessment should be used, for example clear radiological impacts on humans and non-humans and on what timescale.[27] Timescales should cover a time period long enough to determine the maximum, or peak dose. The landscape and hydrological regime, climate, geology and glaciation are all relevant to the selection of a timescale.[28] Specifically SSG-23 provides that deep geologic facilities or buried facilities ought to be assessed for periods of tens of thousands of years or beyond.[29] SSG-23 recognizes that as timescales increase, the focus may shift from quantitative to qualitative assessments.[30] The rationale for the timescale should be described clearly.[31] Notably SSR-5 prohibits reliance on institutional control in the long-term and SSG-23 supports the need to refrain from relying from ongoing institutional control for new disposal facilities.[32]

Specifically, SSG-14 provides that the safety case for the post-closure period in a deep geologic facility should address scenarios for more likely evolutions of the facility and its regional setting over very long time periods, for example a time period over which the waste remains hazardous and the less likely events that might affect performance of the facility. Three items are specified for these assessments:

a) Present evidence that the key features, events and processes that might significantly affect geological disposal are sufficiently well understood that scenarios of possible evolutions are properly generated;
b) Provide estimates of the performance of the geological disposal system regarding compliance

[27] *Ibid*. at 50.
[28] *Ibid.* at 76.
[29] *Ibid*. at 76.
[30] *Ibid.* at 78.
[31] *Ibid*. at 78
[32] *Ibid*. at 84-85.

with all the relevant safety requirements;

c) Identify and present an analysis of the associated uncertainties.[33]

The timescales used for the evaluation of nuclear waste management or disposal are often controversial. International authorities are consistent that the nature of the hazards posed by the waste should guide the timescales that are used. SSG-14 applies to any radioactive waste disposal facility for solid waste.[34] SSG-14 provides that the safety case for the period after facility closure should address scenarios over very long time periods, for example the period comparable to that over which the waste remains hazardous, including for non-fuel waste.[35] SSR-5 requires that regulators develop requirements for disposal facilities and licensing, as well as conditions for development, operation and closure and the use of passive means for safety and a requirement that the operator explain the features of the facility and its host environment over "suitably long time periods" so that "a sufficient level of confidence in safety scan be achieved". SSG-14 also provides that there should be national regulations or guidance concerning the timescales for safety assessments in deep geologic repositories and that regulatory requirements should be compared to doses or risks for at least several thousand years, or beyond to estimate peak dose.[36] The SSG-15 (covering only fuel waste) does not cover final disposal but addresses long term storage of spent fuel.

SSG-14 provides that "a distinctive feature of geological disposal is that post-closure safety of the facility is provided, in part, by passive means inherent in the characteristics of the geological formation.[37] Requirement 16 of SSR-5 and the design requirements for DGR facilities in SSG-14 provide that the disposal facility shall be designed to be compatible with the physical and chemical environment, and to provide safety features after closure and that engineered barriers shall provide safety during the operational period. It also provides that the design of the facility for safety in the period after closure should meet the precepts of robustness, simplicity, technical feasibility and passivity.[38] Geological Repository facilities are designed to be passively safe in the post-closure period (i.e. not requiring intervention to ensure safety) and "the long-term safety of a disposal facility for

[33] SSG-14 *supra* note 7 at 22.
[34] *Ibid*. at 3.
35 *Ibid*., at 22.
36 *Ibid*., at 22.
[37] *Ibid.* at 4-5.
[38] SSG-14, *surpa* note 7 at 37.

radioactive waste is not to be dependent on active institutional control."[39]

Risk Assessment principles

SSG-14 provides that it is crucial to develop confidence in modeling, in particular for the significance of possible migration pathways for radionuclides using scenarios. Scenarios should be robust, in that they are tolerant of uncertainties and should be compared to regulatory criteria.[40] Acceptable waste inventories, forms and packages are usually dependent on the analysis of scenarios of radionuclide release to the environment and transfer along environmental pathways.[41]

SSG-14 recognizes that there will be uncertainty, but requires it to be defined, and addressed in the safety assessment. One relates to model inputs/data limitations and this should be reduced by improving the quality of site characterization and waste data until it is "deemed acceptable" or shown to be unimportant to performance. The second relates to future predictions, and this should be addressed so that there is reasonable assurance that the disposal system will be safe even if the model outcomes are uncertain, for example by using conservative models. Uncertainty analysis should focus on those parameters that are shown to be important to facility performance and safety outcomes.[42]

Canadian regulatory approach

The *Nuclear Safety and Control Act* (**NSCA**) governs the licensing of nuclear waste facilities.[43] The objects of the Canadian Nuclear Safety Commission (**CNSC**) and the Commission's licensing authority governing nuclear waste facilities are set out in sections 9 and 24. The objects of the commission include prevention of unreasonable risk, to the environment and to the health and safety of persons, associated with that development, production, possession or use. This is the regulatory standard applied to licensing decisions under s.24 of the Act.

Paragraph 12(1)(c) of the *General Nuclear Safety and Control Regulations* requires that a licensee

[39] *Ibid*. at 46.
[40] *Ibid*. at 71.
[41] *Ibid*. at 72.
[42] *Ibid.* at 84-87.
[43] *Nuclear Safety and Control Act*, S.C., 1997, c. 9.

“take all reasonable precautions to protect the environment and the health and safety of persons and to maintain security” The *Class I Nuclear Facilities Regulations* require a license applicant to include a preliminary safety analysis and evaluation of environmental effects. Effects on humans from radioactive substances are regulated by the dose limits set out in the *Radiation Protection Regulations*.[44]

Canada has no specific design standards, operational limits, methodological approaches or other standards for applications to build nuclear waste facilities or for the post-closure safety assessment of those facilities. The Canadian Nuclear Safety Commission (has established Regulatory Guide G-320, *Assessing the Long Term Safety of Radioactive Waste Management*[45] to be applied by the licence applicant, and Regulatory Policy P-290, *Managing Radioactive Waste*,[46] an internal statement of philosophy and principle to guide CNSC staff in evaluating applications. The content of this document is less than one page. These documents are applicable to the long-term management of waste, but do not comprehensively review disposal concepts or provide clear guidance for post-closure disposal assessments. G-320 instead describes approaches for assessing the potential long term impact that radioactive waste storage and disposal methods may have on the environment and on the health and safety of people.

This project is also subject to the transition provisions under the *Canadian Environmental Assessment Act, 2012* and the related Environmental Impact Statement (**EIS**) Guidelines dating from January 2009.[47] These EIS Guidelines apply to the Review Panel process pursuant to s.126 of the *Canadian Environmental Assessment Act,* 2012. Notably with respect to a safety case, the EIS guidelines reference the predecessor to IAEA safety requirements in SSR-5, a document which has been superseded by SSR-5.[48] The EIS guidelines outline the minimum information requirements while

[44] S.O.R. /2000-203

45 CNSC, Regulatory Guide 320, (G-320) “Assessing the Long Term Safety of Radioactive Waste Management” (CNSC, 2006), online: <http://nuclearsafety.gc.ca/pubs_catalogue/uploads/G-320_Final_e.pdf >.

46 CNSC, Regulatory Policy 290,(P-290) “Managing Radioactive Waste” (CNSC, 2004), online: <http://nuclearsafety.gc.ca/pubs_catalogue/uploads/P290_e.pdf >

[47] CNSC, *Guidelines for the Preparation of the Environmental Impact Statement for the Deep Geolgic Repository for Low and Intermediate level Radioactive Wastes,* (January 2009) e-doc 3808854 online: <http://www.ceaa.gc.ca/050/documents/39323/39323E.pdf>

[48] See footnote 45 [reference11] to note 48. IAEA, 2006. *IAEA Safety Standards for Protecting People and the Environment: Geological Disposal of Radioactive Waste – Safety Requirements No. WS-R-4*. Online: < http://www-pub.iaea.org/mtcd/publications/pdf/pub1231_web.pdf>

providing the proponent with flexibility in selecting methods to compile data for the EIS, and are not binding.

Environmental assessment boundaries are described in part 9 of the EIS Guidelines. It leaves the timescale of the assessment open-ended, and references the standards set out in G-320 providing:

> At a minimum, the assessment is expected to include the period of time during which the maximum impact is predicted to occur. The approach taken to determine the temporal boundary of assessment should take into account the following elements:
> • Hazardous lifetime of the contaminants associated with waste or with releases to the environment during both normal operation and postulated accidents and malfunctions;
> • Duration of the operational period (before the facility reaches its end state);
> • Design life of engineered barriers;
> • Duration of both active and passive institutional controls; and
> • Frequency and duration of natural events and human-induced environmental changes (e.g., seismic occurrence, flood, drought, glaciation, climate change, etc).[49]

The requirements of the EIS guideline reflect the CNSC Regulatory Guide P-390 described below, with the addition of the timeframe during which the contaminants *associated with waste* during *normal operation* remain hazardous. The EIS guideline requires that the frequency and duration of natural events be assessed but it does not specify how or require that certain kinds of natural events be included. Part 13 of the EIS guidelines deal with long-term safety requirements.

Part 13 of the EIS guidelines does not outline all of the types of scenarios that are mentioned in other safety case documents internationally. Instead, it focuses on the normal evolution scenario and mentions that other scenarios should also be developed. The normal evolution scenario is also mentioned in CNSC Regulatory Guide G-320. However G-320 also mentions the need to include disruptive event scenarios including human intrusion.[50]

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[49] CNSC EIS Guidelines, *supra* note 48 at 28-29.
[50] G-320 at 27.

The OPG post-closure safety assessment for the DGR uses different radiological criteria for different scenarios. The concept of "as low as reasonably achievable" (**ALARA**) is addressed by way of a dose criterion of 0.3 mSv/a. ALARA is applied to the normal evolution scenario. For disruptive scenarios, the assessment first uses 1 mSv/a and if that is not met, a probalistic assessment is used to determine whether the scenario is likely and it is compared to a reference health risk of 10^{-5}.[51]

The content of a safety case, and elements related to the content of a safety case are found in part 5 of G-320. G-320 is a simplified version of the same content that is found in IAEA safety guide SSG-15, *Specific Safety Guide for the Storage of Spent Nuclear Fuel*.[52] The Safety Guide does not provide comprehensive and detailed recommendations on physical protection of nuclear material and nuclear facilities in the disposal context and neither does G-320. More specific physical standards for deep geologic repositories and disposal facilities for nuclear waste do exist in other IAEA documents, such as SSR-5 and SSG-14 but these have no equivalent in the Canadian regulatory system.

In addition to federal regulations and guidance documents for nuclear waste management, the Canadian Standards Association (**CSA**) has developed draft standards for management of radioactive waste that are currently under consideration: N292.3 for the *Management of low and intermediate level radioactive waste* and N292.0 *General principles for the management of radioactive waste.* These are intended to be applied to management of low and intermediate level waste. A separate standard is under development with respect to fuel waste (N292.2). These documents do not refer to a safety case, or specifically to geologic repositories, but 292.0 does propose requirements for assessing hazards as well as proposing to provide some limited design standards for waste management facility design generally. The CSA standards cover the management of waste more broadly, and are not focused on specific standards for post-closure assessments of disposal.[53]

The requirements of the *Canadian Environmental Assessment Act, 2012* and where applicable, its predecessor under the transitional provisions (**CEAA**) and CNSC licensing will be applied to both high level waste and low and intermediate level waste. CNSC licensing requirements are articulated in the

[51] OPG 2011, *infra* note 55 at xi, 15, 149, 190, 192, 193.
[52] See note 8.
[53] Canadian Standards Association, Draft: CSA-N292.3-08 "Management of Low- and Intermediate-Level Radioactive Waste" (CSA, 2008).

NSCA, the *General Nuclear Safety and Control Regulations*, and the *Class I Nuclear Facilities Regulations*. Under the NSCA, a licence must be obtained from the CNSC to prepare a site to, construct, operate, decommission or abandon a nuclear waste facility. Accordingly, once a site has been identified, OPG and the NWMO would seek a site preparation licence and construction licence from the CNSC before proceeding to construct a deep geologic repository.

It is important to understand that there are many nuances to decisions and approaches to timescales for addressing pre-closure items as well as post-closure safety at a deep geologic repository. The length of the hazards posed by the waste will guide timescales, but this alone cannot guide how the timescales are applied. For example, what doses are acceptable on what timescales? What scenarios are reviewed on what timescales? In some cases, such as in the United States regulations for Yucca Mountain, doses higher than the regulatory limit are considered acceptable at timeframes beyond one million years. While some geologic events are required to be evaluated for a timescale of up to one million years, dose assessment methods are prescribed only for a ten thousand year period and unlikely events are excluded. Then there is a further question of what physical, institutional and operational issues are assessed and whether they are subject to distinct timeframes.

CNSC G-320 provides that assessment timescales shall include "the period of time during which the maximum impact is predicted to occur." The CNSC regulatory guide does not specify what items have to be assessed on what timescales. For example, the period during which institutional controls need to be assessed is not specified, the period during which geological stability must be assessed is also not specified, the period for containment, shaft seal viability and related items is not specified. The decision to have no regulations and to limit guidance periods to periods of maximum impact, as opposed to the period during which waste remains hazardous/peak dose does not appear to be fully consistent with SSG-14.

The post-closure safety assessment completed for OPG's deep geologic repository that is before the JRP includes institutional control for up to 300 years post decommissioning, and predicts that impacts become small after 60,000-100,000 years.[54] The period of predicted maximum impact is

[54] Ontario Power Generation, OPG's Deep Geologic Repository Project for Low & Intermediate Level Waste, at.23, 25 Online: <http://www.opg.com/power/nuclear/waste/pdf/nwmo228-DGRLakeHuronReport.pdf>.

approximately the first 10,000 years post-closure.[55] However peak impacts associated with releases in groundwater might occur after more than one million years due to isolation and containment and therefore some items are evaluated up and beyond to 1,000,00 years.[56] OPG includes a shaft seal failure, borehole failure and vertical fault scenario in its post-closure assessment. However there was no guidance for OPG on the appropriate timescale on which to run these scenarios, nor what the acceptable outcome of the scenarios should be.

To demonstrate why it is a potential problem to leave the methods of evaluation for the safety case as well as acceptable doses under various scenarios and timescales open-ended in this way a further examination of one of these scenarios is instructive. For example, OPG runs the severe shaft failure scenario at the time of closure.[57] This may be prior to other events such as container failure, or geologic events. It is therefore not clear if this timescale for running the scenario is appropriate as it does not reflect the period during which the waste remains hazardous. OPG's assessment is that the scenario would have little consequence if the degradation occurred after about 60,000 years. Be that as it may, the scenario "base case" results in a dose of 1.1 mSv/a after about 23,000 years.[58] The assumptions for the degradation of the shaft seals in the SF-ED case result in a calculated dose to an adult member of the Site Resident Group that reaches about 80 mSv/a after around 3800 years.[59] Obviously these would exceed the limits to public doses under the *Radiation Protection Regulations.* In summary OPG notes that for the Disruptive Scenarios, the maximum calculated doses for the Human Intrusion and Severe Shaft Seal Failure cases are at or just below the dose criterion of 1 mSv/a for times up to about 30,000 years. However, when the low likelihood of such scenarios is taken into account, the health risk criterion of 10^{-5} is not exceeded.[60]

G-320 provides no real guidance regarding whether these are appropriate outcomes for a disruptive scenario, or whether the assumption that the event occurs early in the post-closure period is

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[55] Quintessa, OPG's Deep Geologic Repository for Low and Intermediate Level Radioactive Waste: Postclosure Safety Assessment, (OPG, 2011) NWMO DGR-TR-2011-25 at 24 online: <http://www.nwmo.ca/uploads/DGR%20PDF/Licensing/Postclosure-Safety-Assessment.pdf>
[56] *Ibid*. at 24.
[57] *Ibid.* at 186.
[58] *Ibid.* at 188.
[59] OPG (2011), *supra* note 55 at 189.
[60] OPG (2011) *supra* note 55 at 193.

appropriate. Neither Canadian regulations, nor G-320 assist in evaluating what role the probability of the scenario should play either. As a result, it is this panel and the CNSC who will have to decide.

In light of this, it is incumbent on the JRP to turn its mind to what the role of these disruptive assessments and timescales should be in the approval process. Are they simply to speculate at large or do they play a meaningful role in assessing reasonable risk to health, safety and the environment and in determining significant adverse environmental effects? What role does the dose criterion play: must it be met deterministically or once the dose is exceeded, is it appropriate to evaluate it based on the low probability of the scenario occurring? G-320 is particularly unclear on the role of probabilistic assessment.

Notably, the IAEA provides that the regulatory body should stipulate or provide guidance concerning timescales for safety assessments at deep geologic repositories.[61] SSG-14 recommends that scenarios should address "very long" periods. SSG-23 in setting out the expectations for a safety case contains specific dose constraints for natural evolution and for human intrusion scenarios.[62] Namely SSG-23 provides that the safety criteria should be specified by the regulatory framework, and references a 0.3 mSv/a criteria for normal evolution and 1 mSv/a for human intrusion, suggesting that disruptive scenarios should be meeting identifiable dose criteria. SSG-23 provides that at doses greater than 20 mSv/a in a human intrusion scenario the disposal method should be reconsidered.[63]

SSG-23 provides helpful commentary on the purpose of the scenarios and how it relates to design, the purpose being to "illustrate the properties of one or more of the natural or engineered barriers."[64] In other words, running a scenario where there is shaft seal failure should not be done for the purpose of determining the likelihood of shaft seal failure, but to examine how reliant the system is on the shaft seal for safety by assessing the performance of the other barriers in that scenario. This relates back to defence-in-depth. Therefore, having no standards by which to evaluate when the system is sufficiently

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[61] SSG-14, *supra* note 7 at 22.
[62] SSG-23, *supra* note 10 at 49.
[63] OPG's human intrusion scenario includes one scenario in which the peak calculated dose to an adult member of the Site Resident Group would be around 30 mSv/a, occurring after 400 years, decreasing to 0.003 mSv/a after 60,000 years., OPG (2011), *supra* note 55 at 185. This is then dismissed as having low probability without assessing whether measures to prevent human intrusion or its effects during the higher dose post-closure period would be appropriate.
[64] SSG-23, *supra* note 10 at 53.

robust when disruptive scenarios are examined is liable to create confusion in evaluating the assessment. SSG-23 is instructive in this regard. SSG-23 is also instructive on the appropriate role of probabilistic assessments in that it speaks to avoiding "risk dilution" in scenarios and the assessment of uncertainty:

> *Probabilistic assessments should also be conducted so as to avoid undue 'risk dilution', i.e. masking of the impact of a very significant event at some point in the lifetime of the facility by rendering its consequences of little significance in the overall assessment of risk when multiplied by the probability of occurrence of the event*.[65]

When one considers that the purpose of running a scenario is, in part, to identify weaknesses in the remainder of the system and assess if or how they can be improved, it is obvious why risk dilution or heavy reliance on probabilistic assessment, particularly in long-term post-closure scenarios is counter-productive. The JRP and the CNSC are urged to keep the commentary in SSG-23 in mind when crafting conditions on approval, if any.

If one reviews items like containment and defence-in-depth, the challenges arising from the lack of clear guidance on timescales and standards for assessment methods become readily apparent. The IAEA requirements in SSG-23 that provide standards for the safety case recommends that "the safety strategy should identify the intended safety functions, the time frames over which they will be available and how degraded performance of one barrier will be compensated by another mechanism or by components of the disposal system."[66] For example a safety case should assess whether, if a waste container fails on a given timescale, how the other barriers will function to ensure containment. The OPG post-closure safety assessment addresses this by indicating that "during the years following closure" waste containers will degrade and that the waste packaging "is not long-lived".[67] Although OPG does not rely on the containers for safety, it is not clear if the containers are expected to maintain integrity for any period or if so, for how long, nor how the other layers of defence come into play or during what timescales.[68]

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[65] SSG-23, *supra* note 10 at 61.
[66] *Ibid*. at 25.
[67] OPG, 2011, *supra* note 55, at 76.
[68] *Ibid.* at 33.

Even for migration of contaminants outside containers SSR-5 and SSG-14 require that "containment shall be provided until radioactive decay has significantly reduced the hazard."[69] Depending on the type of radioactive waste, containment may be required for several hundred years.[70] OPG's post-closure assessment does not articulate whether these goals are achieved or achievable nor does CNSC's regulatory guidance clarify what is expected in terms of safety outcomes for containment or on what timescales they should be achieved or with what level of certainty.

While the degree to which these items should be prescribed in detail by regulation or handled by way of guidance documentation is the subject of debate; it is clear that Canadian regulations provide very limited guidance to the proponent and to licensing authorities. Minimum standards for safety evaluation timescales would assist in determining whether the safety case meets minimum standards sufficient to have a level of confidence in a safety case. By way of comparison, the approaches taken for evaluating DGR and other radioactive waste facilities in other countries is helpful.

United States

In the United States nuclear waste is regulated by the Environmental Protection Agency (**EPA**) and the Nuclear Regulatory Commission (**NRC**). The United States regulatory authorities typically do not employ the concept of "intermediate" level waste. Instead wastes are classified by various means and are typically referred to as either low level or high level wastes. The *Low-level Radioactive Waste Policy Amendments Act* of 1985 gave the states responsibility for the disposal of their low-level radioactive waste. The Act encouraged the states to enter into agreements that would allow them to dispose of waste at a common disposal facility. Most states have entered into these agreements; however, no new disposal facilities have been built since the Act was passed. There are three existing low-level waste disposal facilities in the United States that accept various types of low-level waste.[71]

10 CFR 61 of the Nuclear Regulatory Commission regulations *Licensing Requirements for Land Disposal of Radioactive Waste* are specific to disposal but would not be directly applicable to a

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69 SSG-14, *supra* note 7 at 15.
70 *Ibid*. at 15.
71 NRC, Advisory Committee on Nuclear Waste, "History and Framework of Commercial Low-Level Radioactive Waste Management in the United States," online: <http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1853/sr1853.pdf>

geologic repository.[72] The regulatory requirements for licensing a land disposal facility in 10 CFR Part 61 establish a system for classifying radioactive waste for disposal, based on the concentrations of specific radionuclides that the waste contains. In particular, 10 CFR 61.55(a)(8) allows the concentration of a radionuclide to be determined indirectly, or averaged over the volume of the waste, or weight of the waste, provided that the concentration is expressed in nanocuries per gram.

Internationally, the SSG-14 provides that “regulatory criteria will typically specify characteristics of exposed groups to be used in dose calculations” including for long timescales.[73] However Canada does not specify methods for dose calculation in nuclear waste facilities as a regulatory requirement, although all facilities must meet the standards set out in the *Radiation Protection Regulations*.[74] In contrast, in the United States the NRC applies the dose limits set forth in 10 CFR 20. The limits in 10 CFR Part 20 specify that the total effective dose equivalent to individual members of the public from the licensed operation must not exceed 0.1 rem (1 mSv) in a year.

10 CFR 60 *Disposal of High-Level Radioactive Wastes in Geologic Repositories* contains specific NRC regulations for the disposal of high-level waste in geologic repositories. According to 10 CFR 60 the site geology must be such that pre-waste-emplacement groundwater travel time along the fastest path of likely radionuclide travel from the disturbed zone to the accessible environment that substantially exceeds 1,000 years.[75] The regulation requires that pre-closure radioactive releases will be maintained within regulatory limits. The geologic repository operations area shall be designed to preserve the option of waste retrieval throughout the period during which wastes are being emplaced and, thereafter, until the completion of a performance confirmation program and Commission review of the information obtained from the program. To satisfy this objective, the geologic repository operations area shall be designed so that any or all of the emplaced waste could be retrieved on a reasonable schedule starting at any time up to 50 years after the waste emplacement operations are initiated, unless a different time period is approved or specified by the Commission. This different timescale may be established on a case-by-case basis.[76]

[72] 10 CFR 61 § 61.42
[73] SSG-14, *supra* note 7 at 23.
[74] S.O.R. /2000-203
[75] 10 CFR 60 § 60.122(7)

[76] 10 CFR 60 § 60.111 see: Performance of the geologic repository operations area through permanent closure.

Under Section 121(a) of the *Nuclear Waste Policy Act*, the Environmental Protection Agency is required to promulgate generally applicable standards for protection of the general environment from offsite releases from radioactive materials in repositories. The containment requirements of 40 CFR 191 *Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes* apply to the disposal of High-Level wastes. These requirements are not specific to geologic repositories, and apply only to high level waste. Subpart B contains specific disposal requirements, as distinct from storage and regulates both doses and timescales. Pursuant to these regulations, passive controls are applied along with performance assessments that assess isolation of the wastes from the accessible environment shall not consider any contributions from active institutional controls for more than 100 years after disposal. Disposal systems for waste and any associated radioactive material shall be designed to provide a reasonable expectation that, for 10,000 years after disposal, undisturbed performance of the disposal system shall not cause the annual committed effective dose, received through all potential pathways from the disposal system, to any member of the public in the accessible environment, to exceed 150 microsieverts.[77]

In the United States, each deep geologic repository facility may also have its own site-specific regulations. For example, the EPA issued final regulations regarding the disposal of spent nuclear fuel, high-level radioactive waste, and transuranic waste for the Waste Isolation Pilot Plant (WIPP) for high-level waste. The regulations in 40 CFR Part 194 contains the EPA requirements governing the WIPP. The specific release limits are based on the amount of waste in the repository at the time of closure.[78] The WIPP review process by the EPA involves a series of computer simulations that captures the behaviors and interactions among its various components. The computer simulations require the use of conceptual models that represent physical attributes of the repository based on features, events, and processes that may impact the disposal system. The results of the simulations are intended to show estimated releases of radioactive materials from the disposal system to the accessible environment over the 10,000-year regulatory time frame. [79] The proponent must consider both natural and man-made

[77] 40 CFR 191, § 191.13
[78] 40 CFR 194 §194.31
[79] *Ibid.*

processes and events which could have an effect on the disposal system.[80] All reasonably probable release mechanisms from the disposal system must be considered and the proponent must demonstrate an adequate understanding of the physical conditions in the disposal system.[81] The licence application must also evaluate potential releases from both human-initiated activities (e.g., via drilling intrusions) and natural processes (e.g., dissolution) that may occur independently of human activities.[82] DOE must justify the omission of events and processes that could occur but are not included in the final PA calculations. [83]

Meanwhile, 40 CFR 197 contains regulations specific to the proposed Yucca Mountain repository for high level waste. For the Yucca mountain repository, dose compliance is judged against a standard of 150 μSv/yr up to 10,000 years after disposal and against a standard of 1 mSv/yr at times after 10,000 years and up to 1 million years after disposal. The effects of climate change, earthquakes, volcanoes, and corrosion during the 1 million-year period must be considered.

Sweden

Swedish radiation authority regulations for final management of spent nuclear fuel waste (SSMFS 2008:37) require that a safety analysis shall comprise the requisite duration of barrier functions, though a minimum of ten thousand years.[84] In the case of a repository intended for long-lived waste, the safety analysis may need to include scenarios taking greater expected climate changes into account, primarily in the form of future glaciations. For example, the next complete glacial cycle, currently estimated to be in the order of 100,000 years, should be taken into account. Long lived fuel or spent nuclear fuel risk analysis should cover at least one hundred thousand years or the period for the next glaciation cycle for up to one million years. These regulations apply to the disposal of high-level waste. In the case of periods up to 1,000 years after closure, in accordance with the provisions of SSMFS 2008:37, the dose and risk calculated for current conditions in the biosphere constitute the basis for assessing repository safety and the repository's protective capabilities.[85]

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[80] *Ibid.*, §194.32 and § 194.33
[81] *Ibid.*
[82] *Ibid.*
[83] *Ibid.*
[84] Regulations concerning safety in connection with the disposal of nuclear material and nuclear waste - General advice on application of regulations (SSMFS 2008:21) online: <http://www.stralsakerhetsmyndigheten.se/Global/Publikationer/Forfattning/Engelska/SSMFS-2008-21E.pdf>
[85] The Swedish Radiation Safety Authority's general advice on the application of the regulations (SSMFS

Sweden has also enacted *Regulations concerning safety in connection with the disposal of nuclear material and nuclear waste* (SSMFS 200:21).[86] These regulations contain specific provisions for nuclear waste barriers, containment, safety analysis and the content of safety analysis reports. In particular these regulations emphasize the need to use passive barriers and properly analyse the disposal system in terms of passive barriers. The guidance document for this regulation sets out the types of specific post-closure scenarios that have to be modeled. The guidance notes that "The time period for which safety needs to be maintained and demonstrated should be a starting point for the safety analysis." In the case of a repository intended for long-lived waste, the safety analysis may need to include scenarios taking greater expected climate changes into account, primarily in the form of future glaciations. For example, the next complete glacial cycle, currently estimated to be in the order of 100,000 years, should be particularly taken into account. [87]

In the case of periods up to 1,000 years after closure, in accordance with the provisions of SSMFS 2008:37, the dose and risk calculated for current conditions in the biosphere constitute the basis for assessing repository safety and the repository's protective capabilities. Furthermore, in the case of more extended periods of time, the assessment can be made using dose as one of several safety indicators. This should be taken into account in connection with calculations as well as presentation of analysis results. Examples of these supplementary safety indicators include the concentrations of radioactive substances from the repository which can build up in soils and near-surface groundwater as well as the calculated flow of radioactive substances to the biosphere.[88]

2008:21) concerning safety in connection with the disposal of nuclear material and nuclear waste; SSMFS 2008: 21 Section 10, at 8 online: <http://www.stralsakerhetsmyndigheten.se/Global/Publikationer/Forfattning/Engelska/SSMFS-2008-21E.pdf>.

[86] The Swedish Radiation Safety Authority's regulations and general advice concerning safety in connection with the disposal of nuclear material and nuclear waste, online: <http://www.stralsakerhetsmyndigheten.se/Global/Publikationer/Forfattning/Engelska/SSMFS-2008-21E.pdf>

[87] For more information see Harold Feiveson, et al., "Managing Spent Fuel from Nuclear Power Reactors Experience and Lessons from Around the World" (presented at the International Panel on Fissile Materials, 2011); Junker, Berit et al., NSSI, Paper 46, "Description of the Safety Case for Long-term Disposal of Radioactive Waste - the Iterative Safety Analysis Approach as Utilized in Switzerland" (NSSI, 2008); and NEARWMC, "The Post-Closure Radiological Safety Case for a Spent Fuel Repository in Sweden; An International Peer Review of the SKB License-Application Study of March 2011 (Final Report)" (Organisation for Economic Co-operation and Development, 2012).

[88] Guidance SSMFS 2008:21, *supra* note 84 at 8.

Conclusion

Each country applies the IAEA's guidance on post-closure safety evaluations in accordance with its own regulatory framework. The practices of other countries are instructive and should guide the Panel in evaluation of OPG's assessments. IAEA guidance on safety case preparation and deep geologic repository evaluation in SSG-14 and SSG-23 also provides helpful benchmarks against which the OPG post-closure safety assessment can be evaluated. In general, Canadian guidance lacks detail and lacks clear benchmarks for determining the adequacy of a safety case.

I recommend that the JRP seek further clarification from OPG on the following items in order to have a clearer picture of the appropriateness of the methodology used by OPG when compared to SSG-23 and SSG-14 as well as the methods required in other countries:

1. How the remaining barriers function in disruptive scenarios where one or more barriers do not function as expected;
2. What are the design objectives for the performance of engineered and passive barriers for containment of liquids and gases on what timescales;
3. How the dose criterion and risk criterion are applied in each scenario and on what timescales;
4. How any differentiation in deterministic dose criterion on different timescales is justified;
5. How the safety case treats uncertainties in the evaluation of risk criterion of 10^{-5} on different timescales;
6. How periods of maximum impact are determined and employed in modeling, particularly of post-closure scenarios;
7. How the appropriate period of institutional control was determined and how it is used in the safety case;
8. What dose criterion and other endpoints are used for the proposal on what different timescales and justification for this.