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POSTCLOSURE SAFETY ASSESSMENT OF A DEEP GEOLOGICAL REPOSITORY FOR CANADA'S USED NUCLEAR FUEL

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ABSTRACT

This paper reports on elements of a postclosure safety assessment performed for a conceptual design and hypothetical site for a deep geological repository for Canada's used nuclear fuel. Key features are the assumption of a copper used fuel container with a steel inner vessel, container placement in vertical in-floor boreholes, a repository depth of 500 m, and a sparsely fractured crystalline rock geosphere.

The study considers a Normal Evolution Scenario together with a series of Disruptive Event Scenarios. The Normal Evolution Scenario is a reasonable extrapolation of present day site features and receptor lifestyles, while the Disruptive Event Scenarios examine abnormal and unlikely failures of the containment and isolation systems.

Both deterministic and probabilistic simulations were performed.

The results show the peak dose consequences occur far in the future and are well below the applicable regulatory acceptance criteria and the natural background levels.

1. INTRODUCTION

The purpose of the postclosure safety assessment is to determine potential effects of the repository on the health and safety of persons and the environment. Results are compared against acceptance criteria established for the protection of persons and the environment from potential radiological and non-radiological hazards.

The approach is consistent with that outlined in CNSC Regulatory Guide G-320 *Assessing the Long Term Safety of Radioactive Waste Management* [1].

The assessment timeframe extends from closure until the time at which the maximum impact is predicted, with a one million year baseline adopted based on the time needed for the used fuel radioactivity to decay to essentially the same level as that in an equivalent amount of natural uranium.

2. SCOPE OF THE SAFETY ASSESSMENT

The postclosure safety assessment is performed by applying a series of computer models to a range of analysis cases. The analysis cases include a Normal Evolution Scenario and a set of Disruptive Event Scenarios.

The Normal Evolution Scenario is based on a reasonable extrapolation of the site and repository. It accounts for the expected behaviour and addresses the effects of anticipated significant events, in particular glaciation.

The analysis is conducted for a Reference Case of the Normal Evolution Scenario, where the Reference Case has the following attributes:

- A conceptual repository design as outlined in Section 3,
- A hypothetical geosphere as outlined in Section 4,
- Two containers with undetected defects (area = 2 mm^2) placed in the repository at the position with the shortest groundwater transit time to the surface,
- No evolution of the defect with time,
- No other container failures occur,
- Groundwater fills the defective containers 100 years after the containers are placed in the repository,
- Constant temperate climate,
- Self-sufficient farming family growing crops and raising livestock on the surface above the repository,
- Drinking and irrigation water for the family obtained from a 100 m deep well located along the main pathway for contaminants released from the defective containers,
- The well is pumping at a rate of 979 m^3/a. This is sufficient for drinking water and irrigation of household crops, and
- Input parameters that are represented by probability distributions are set to either the most probable value (when there is one) or to the median value otherwise.

The following deterministic sensitivity cases were examined to determine the effect of variations in the engineered and natural barriers on the Reference Case predictions:

- Fuel dissolution rate increased by a factor of 10,
- Instant release fractions increased by a factor of 2. Radionuclides with no instant release have their fractions set to 0.01,
- Container defect area increased by a factor of 10,
- No solubility limits in the container,
- No sorption in the buffer,
- Distance from the repository to the nearest major fracture decreased from 129 m to 25 m,
- Geosphere permeability increased,
- No sorption in the geosphere,
- Permeability of the excavation damage zone (EDZ) and thermal damage zone (TDZ) increased by a factor of 10, and
- No sorption anywhere.

Monte Carlo methods were also used to simultaneously vary all input parameters for which probability distribution functions are available. In these simulations, the number of defective containers and their location is treated as a random variable.

The Disruptive Event Scenarios considered in the safety assessment are listed below. These Scenarios postulate the occurrence of unlikely events leading to possible penetration of barriers and abnormal loss of containment.

- All Containers Fail at 100,000 Years (with a sensitivity case that has the failure occurring at 10,000 years); and
- Inadvertent Human Intrusion.

3. WASTE FORM AND REPOSITORY DESCRIPTION

3.1 Waste form

The reference waste form is a standard CANDU®[1] 37-element fuel bundle with a burnup of 220 MWh/kgU and an average fuel power during operation of 455 kW.

Radionuclides within the UO_2 fuel are released by two mechanisms which operate on very different time scales:

- Initially, there would be a comparatively rapid release of a small fraction (typically a few percent) of the inventory of a selected group of radionuclides that are either very soluble (such as I-129, Cs-137, C-14 and Cl-36) or gaseous (such as Xe), and that are residing in the fuel sheath gap or at grain boundaries which are quickly accessed by water. This release process is referred to as "instant-release".
- The second and slower release process comprises release of radionuclides from the UO_2 fuel matrix as the matrix itself dissolves.

The ceramic used fuel matrix is durable and dissolves slowly in water. The most important factor in the rate of dissolution is the chemical oxidation and reduction conditions in the surrounding groundwater. The groundwater is anticipated to maintain a reducing environment in and around the container, with any residual oxygen available at the time of closure consumed by reactions with the copper and steel container materials or with the sealing materials.

Conditions at the used fuel surface are likely to remain oxidizing for a long time due to radiolysis of groundwater that contacts the fuel following failure of the container and fuel cladding. Radiolysis would be caused by alpha, beta, and gamma radiation from the used fuel, at rates that decrease with time. Note that although used fuel dissolution experiments indicate that the dissolution rate decreases by several orders of magnitude in the presence of significant hydrogen gas as would be produced by corrosion of the steel container [2], this effect is conservatively ignored.

Figure 1 shows the fuel dissolution as a function of time.

[1] CANDU® is a registered trade mark of Atomic Energy of Canada Limited.

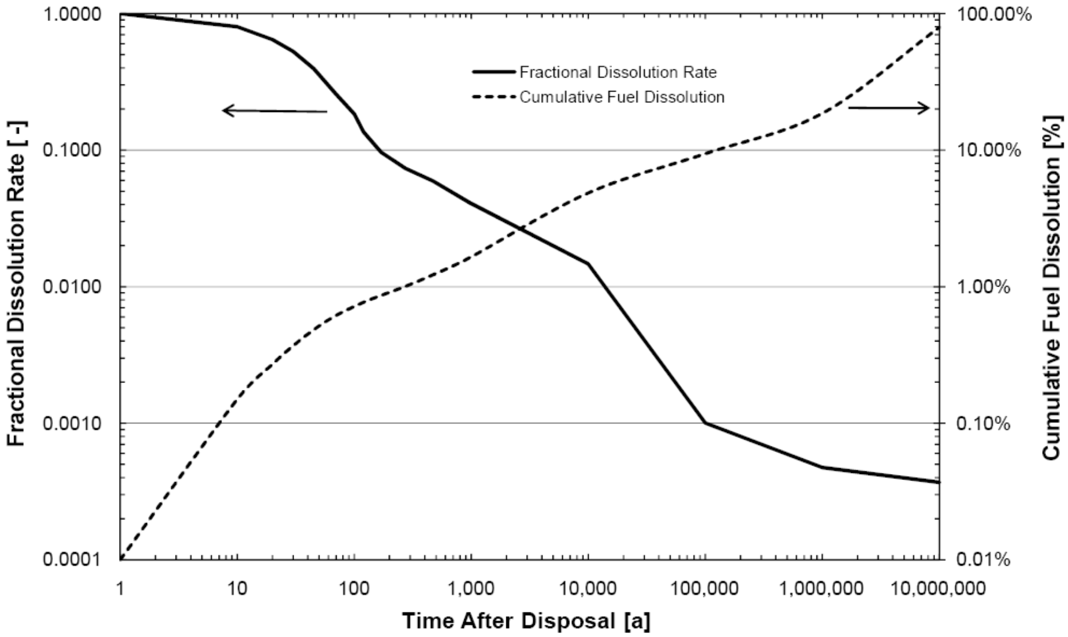


Figure 1: Fuel Dissolution

3.2 Repository

Figure 2 and Figure 3 illustrate the layout of the repository and placement rooms. A series of parallel, single-ended placement rooms are arranged in eight panels with used fuel containers placed in a single row within vertical boreholes and surrounded by engineered sealing materials. Spacing between the used fuel containers and placement rooms is chosen to limit the temperature of the engineered sealing materials. Used fuel containers are spaced 4.2 m centre-to-centre and placement rooms are spaced 40 m centre-to-centre.

The repository is assumed to contain 3.6 million used CANDU® fuel bundles held in 10,000 used fuel containers.

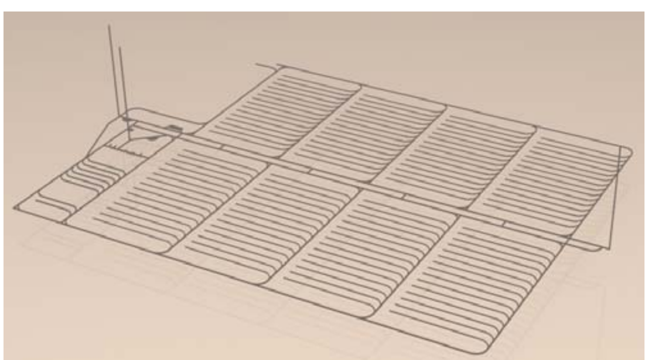


Figure 2: Underground Layout

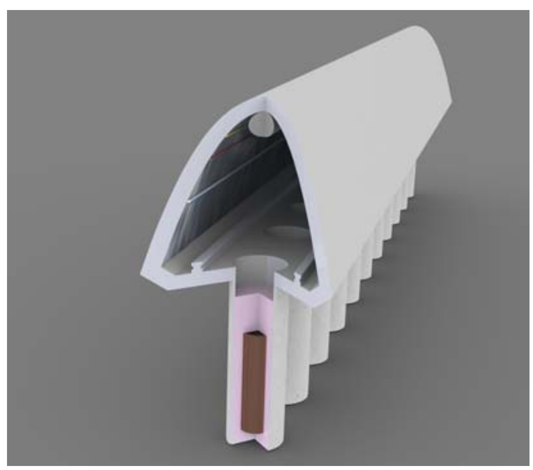


Figure 3: Illustration of a Placement Room

The used fuel container consists of a copper outer vessel, a steel inner vessel, and three steel baskets as shown in Figure 4. The purpose of the copper vessel is to provide a corrosion-resistant barrier in the repository environment while the inner vessel is designed to withstand mechanical stresses, including stresses due to glaciation. Each container holds 360 used fuel bundles, distributed in six layers of 60 bundles each held in three stacked baskets (two layers per basket).

Figure 4: Used Fuel Container

Used fuel containers are surrounded by compacted bentonite clay and all excavated spaces are filled with engineered sealing materials made from mixtures of clay, sand, and rock to minimize the movement of water. Placement rooms are sealed with bulkheads of low-heat high-performance concrete. All tunnels and shafts are filled with similar engineered sealing materials, isolating the repository from the biosphere.

The design of the repository is similar to that implemented by other used fuel management agencies internationally, and is within the practical application of known technologies.

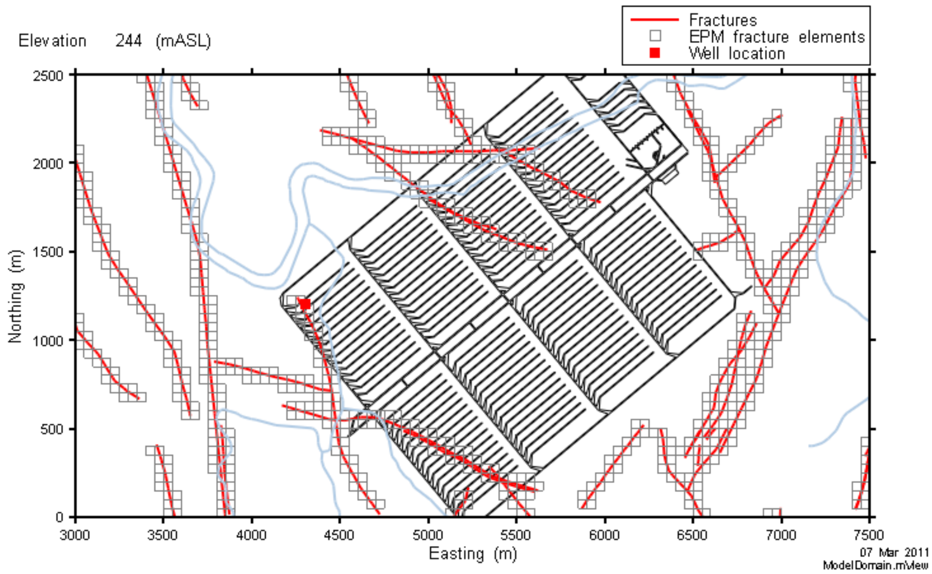
4. GEOSPHERE DESCRIPTION

The hypothetical geology is igneous rock, with granitic crystalline rock assumed to extend from the surface to several hundred meters below repository depth.

In crystalline rock, fractures are the main pathway for water movement and contaminant transport. The hydrogeology of the setting can be generally characterized by a model in which the flow system is decoupled into shallow and deep groundwater zones. In the shallow groundwater zone located near the surface, groundwater flow is driven by topographic gradients through the more permeable rock mass and fractures. Advection would be the dominant transport mechanism. The deep groundwater zone is stagnant and more isolated. Fractures are less common and less likely to be interconnected. The groundwater in this zone is old, slow-moving, and chemically distinct. Diffusion would be the dominant mechanism for contaminant transport until the contaminant plume intersects a transmissive fracture.

In this study, a set of discrete fractures was defined across a subregional area measuring about 11 km x 18 km down to a depth of 1500 m. The fracture network is a statistical representation of fractures on a Canadian Shield area, and consists of a large number of intersecting features within the first few hundred meters with fewer, larger, and more vertical features extending to greater depths. The repository is placed at a location where it fits between the assumed fractures at a defined depth of 500 m, and positioned to maximize distance from fractures at this depth. A well is placed at the fracture location with the shortest contaminant transit time from repository to surface.

Figure 5 shows the well location and near surface fracture system in the vicinity of the repository.



The repository is shown for context. It is at a much lower depth where there are no intersecting fractures.

Figure 5: Well Location and Near Surface Fracture System

Hydraulic conductivities are assumed to decrease with depth as shown in Table 1. Fractures are assumed to be uniformly and highly permeable, with a hydraulic conductivity of 4.0×10^{-7} m/s from repository depth to surface.

Table 1: Geosphere Hydraulic Conductivity

Depth (mBGS) or Elevation mASL)	Thickness (m)	Hydraulic Conductivity (m/s)*		
		Reference Case	Sensitivity 1	Sensitivity 2
Ground surface to 10 mBGS	10	1.3E-08	1.3E-08	1.3E-08
10 mBGS to 150 mBGS (210 mASL)	~140	1.7E-10	1.6E-09	1.6E-09
210 mASL to –340 mASL	550	8.2E-13	3.7E-12	3.7E-11
-340 mASL to -1250 mASL	910	2.1E-13	1.1E-12	1.1E-11

*Sensitivity 1 and Sensitivity 2 are cases that examine the effect of increase geosphere permeability

5. METHODOLOGY OVERVIEW

The general approach for conducting the postclosure safety assessment was as follows.

1) Perform Radionuclide Screening

Used fuel contains many hundreds of radionuclides. Screening is performed to identify the potentially radiologically significant radionuclides so that subsequent work need only consider this group.

In this study, 39 radionuclides are considered, with these being a mix of long-lived fission products and actinides.

2) Perform Detailed 3D Groundwater Flow and Radionuclide Transport Modelling

Detailed 3D steady-state hydrogeological modelling is performed with the FRAC3DVS-OPG code to determine the groundwater flow field near the repository. FRAC3DVS-OPG is the reference groundwater flow and groundwater transport code.

FRAC3DVS-OPG is described in Reference [3].

Once the flow field is determined, detailed 3D radionuclide transport calculations (accounting for diffusive and advective processes) are performed for a small group of radionuclides (i.e., I-129, Cl-36, Ca-41, U-234 and U-238) that represent a range of low-sorption to high-sorption species. These calculations aid in understanding the transport behaviour and provide data for subsequent use in SYVAC3-CC4 system modelling.

The FRAC3DVS-OPG code does not have a biosphere model so another code must be used to determine the dose consequences.

3) Perform System Modelling

The 3D groundwater advective flow field generated with FRAC3DVS-OPG is used to develop the geosphere network used in the SYVAC3-CC4 system model. SYVAC3-CC4 is a simple model used for radionuclide transport and the calculation of dose consequences. It simulates the used fuel, container, vault, geosphere, and biosphere, allowing feedback between these components - for example, the geosphere groundwater flows can be affected by the well demand established by the biosphere model. The code is very fast (in comparison to FRAC3DVS-OPG) and is used to perform both deterministic and probabilistic calculations.

SYVAC3-CC4 is described in Reference [4].

To provide confidence in the resulting model, radionuclide transport calculations are performed for I-129, Cl-36, Ca-41, U-234 and U-238 and compared to similar results obtained from the more detailed FRAC3DVS-OPG model.

The code is then used to determine the dose consequences to the critical group for the Reference Case and for the various deterministic sensitivity cases. In these simulations, the full suite of 39 radionuclides of potential interest is considered.

6. RESULTS

6.1 Ground water flow modelling

Figure 6 shows groundwater velocity magnitudes and directions together with hydraulic heads as determined with the FRAC3DVS-OPG code for a location 22 m above the repository elevation. Vectors are only shown in locations where velocities are greater than 10^{-4} m/s since transport would be diffusion-dominated at lower velocities.

Although the fracture system has a large effect on the head distribution and velocities, gradients across the repository footprint are relatively uniform. The impact of the well can be seen with the tightly spaced head contours present at the fracture located near the western corner of the repository.

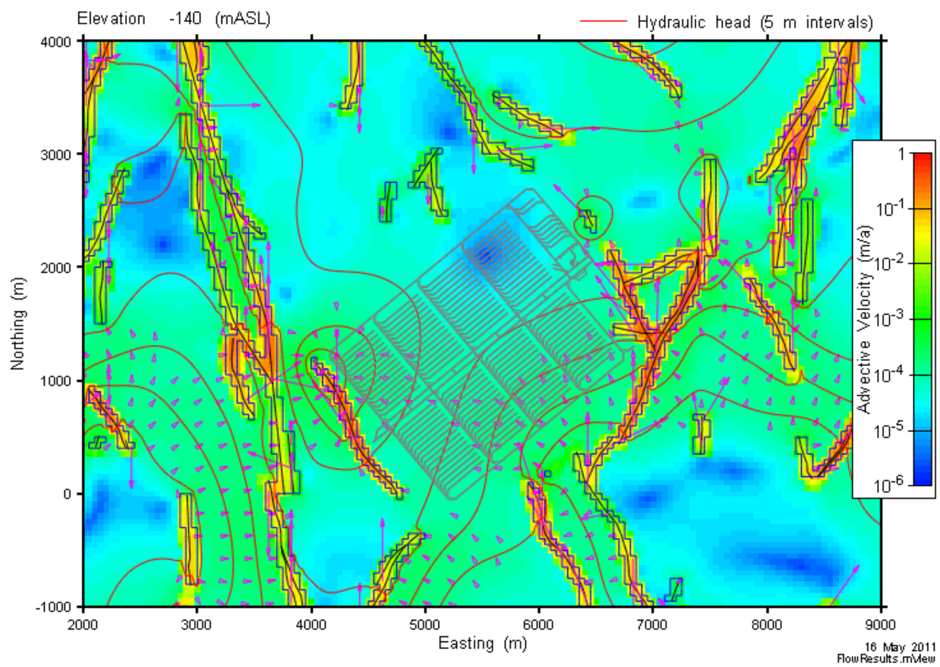


Figure 6: Advective Velocity and Hydraulic Head 22 m above Repository Elevation

Figure 7 shows the Mean Life Expectancy, which is a measure of radionuclide transport including diffusion and advection. In this context, "Life Expectancy" is the time for any subsurface location to discharge to the biosphere. Since transport disperses through the geosphere, life expectancy is represented by a probability density function, obtained by solving for transport at each nodal upstream point. The Mean Life Expectancy (MLE) represents the average time for any subsurface location to discharge to the biosphere.

The figure shows the repository is situated in rock in which the mean lifetimes are long (i.e., 10^6 to 10^7 years) indicating the transport regime is diffusion dominated. The well location is shown by the black dot.

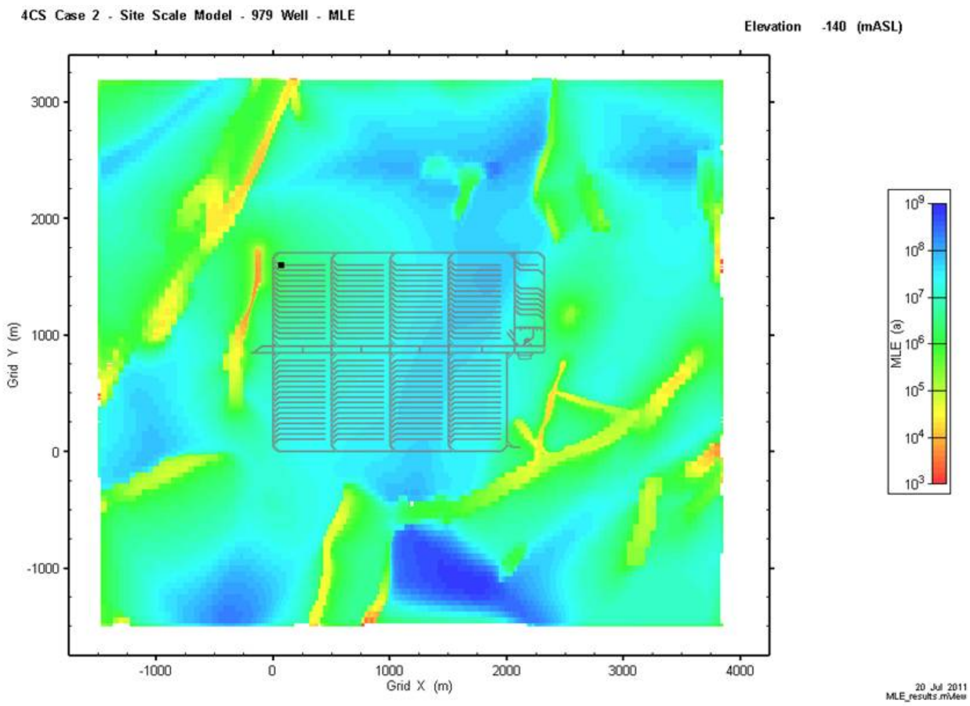


Figure 7: Mean Life Expectancy

6.2 Radionuclide Transport

The peak dose rate for the Reference Case is 9.71×10^{-8} Sv/a occurring at 1.1×10^{6} years. This is well below the average Canadian background dose rate of 1.8×10^{-3} Sv/a and is a factor of 3100 times less than the 3×10^{-4} Sv/a proposed interim acceptance criterion adopted for the radiological protection of persons.

The analysis shows that I-129 is the only significant contributor to dose in the timeframe of interest. This is because I-129 has a sizeable initial inventory, a non-zero instant release fraction, a very long half-life and is non-sorbing. Essentially all other fission products and actinides either decay away, or are released very slowly as the fuel dissolves and are thereafter sorbed in the engineered barriers and geosphere.

Table 2 shows the peak impacts for the sensitivity cases (expressed in Bq/a I-129 for the FRAC3DVS-OPG simulations and in Sv/a for SYVAC3-CC4 simulations), the time of the peak impact, the effect of the parameter variation on the Reference Case result and the factor by which the result is below the acceptance criterion for the radiological protection of persons. Results are shown in Bq/a I-129 for the FRAC3DVS-OPG simulations because the FRAC3DVS-OPG model does not have a biosphere. In lieu of this, I-129 release to the well / surface is used as a surrogate for dose because SYVAC3-CC4 simulations show essentially the entire dose consequence is due to I-129.

Table 2: Results of Radionuclide Transport Simulations

Case	Time of Peak Impact (a)	Peak Impact		Ratio to Reference Case	Factor to Dose Limit
		I-129 (Bq/a)	Dose (Sv/a)		
Reference Case	**1.1×10^6**	**529**	**9.71×10^{-8}**	**-**	**3100**
Geosphere Sensitivity Cases					
Geosphere Permeability (Sensitivity 1)	6.50×10^5	711	-	1.3	2300
Geosphere Permeability (Sensitivity 2)	1.59×10^5	927	-	1.8	1700
Fracture Distance (25 m)	5.70×10^5	682	-	1.3	1800
EDZ and TDZ Permeability 10 times Higher	9.75×10^5	568	-	1.1	2900
Degraded Physical Barrier Sensitivity Cases					
Fuel Dissolution Rate 10 times Higher	8.61×10^5	-	5.51×10^{-7}	5.7	540
Container Defect Area 10 times Higher	7.36×10^5	-	1.30×10^{-7}	1.3	2400
Instant Release Fraction Sensitivity	1.02×10^6	-	1.14×10^{-7}	1.2	2600
Degraded Chemical Barrier Sensitivity Cases					
No Sorption in the Geosphere	1.10×10^6	-	1.10×10^{-7}	1.1	2700
No Solubility Limits	1.10×10^6	-	9.71×10^{-8}	1.0	3100
No Sorption in the Buffer	1.10×10^6	-	9.71×10^{-8}	1.0	3100
No Sorption Anywhere	1.00×10^6	-	2.80×10^{-7}	2.9	1100
Probabilistic Simulations					
All parameters varied (95th percentile)	-	-	3.8×10^{-7}	3.9	790

The table shows that the Reference Case result is largely insensitive to the variations tested in the sensitivity cases. Some key items to note are:

- Despite the large increase in geosphere permeability in "Sensitivity 2" (i.e., 46 times higher than the Reference Case for the rock at repository level), the result is increased by only a factor of 1.8; and
- Despite the reduction in fracture respect distance (i.e., to 25 m from 129 m), the Reference Case result is increased by only a factor of 1.3.

  The lack of high sensitivity occurs because the time at which radionuclides are released to the environment is sufficiently long that short-lived fission products decay away. Actinides are sorbed in the engineered barriers and geosphere so that the total radionuclide release to the environment does not substantially change. I-129 remains the only significant contributor to dose.
- Taking no credit for sorption increases the result by a factor of 2.9.

  The increase is due to normally sorbing radionuclides now reaching the surface in small quantities. The dose rate remains low because the radionuclide release is still dependent on fuel dissolution rate (which is low as shown in Figure 1).
- The remaining sensitivity cases, examining the effects of degraded chemical and physical barriers, show very little difference from the Reference Case.

  This is due to the multi-barrier approach adopted in the repository design. Even if a barrier under performs, the other barriers remain to mitigate the consequences.

While detailed modelling of glaciation was not part of this postclosure safety assessment, previous work for a different repository and site [5] indicated glaciation could cause the peak dose for the Normal Evolution Scenario to increase by an order of magnitude (depending on the glacial cycle) compared to that for the constant temperate climate case.

6.3 Probabilistic Results

Many of the parameters that affect dose consequences are uncertain or have a natural degree of variability. With its fixed groundwater transport model, SYVAC3-CC4 can be used to perform probabilistic assessments of uncertainties in the other input parameters. A total of 120,000 simulations were performed with each set of input parameters selected using Monte Carlo sampling, taking correlated parameters into account. The resulting distribution of dose consequences is a direct reflection of parameter uncertainty.

Figure 8 shows the dose rate histogram for cases with at least one defective container together with the average and 95th percentile values as a function of time. The peak average dose rate is 3.9×10^{-7} Sv/a, while the 95th and 99th percentile values are 3.8×10^{-7} Sv/a and 1.2×10^{-6} Sv/a respectively. The 95th percentile value is lower than the average because the average is skewed by a small number of simulations with high dose consequence. Investigations subsequently indicated these higher dose consequence simulations arise due to overly conservative models. These conservatisms will be re-evaluated in subsequent postclosure studies.

The 99th percentile value is over two orders of magnitude below the $3x10^{-4}$ mSv/a acceptance criterion for the radiological protection of persons. I-129 remains the dominant dose contributor.

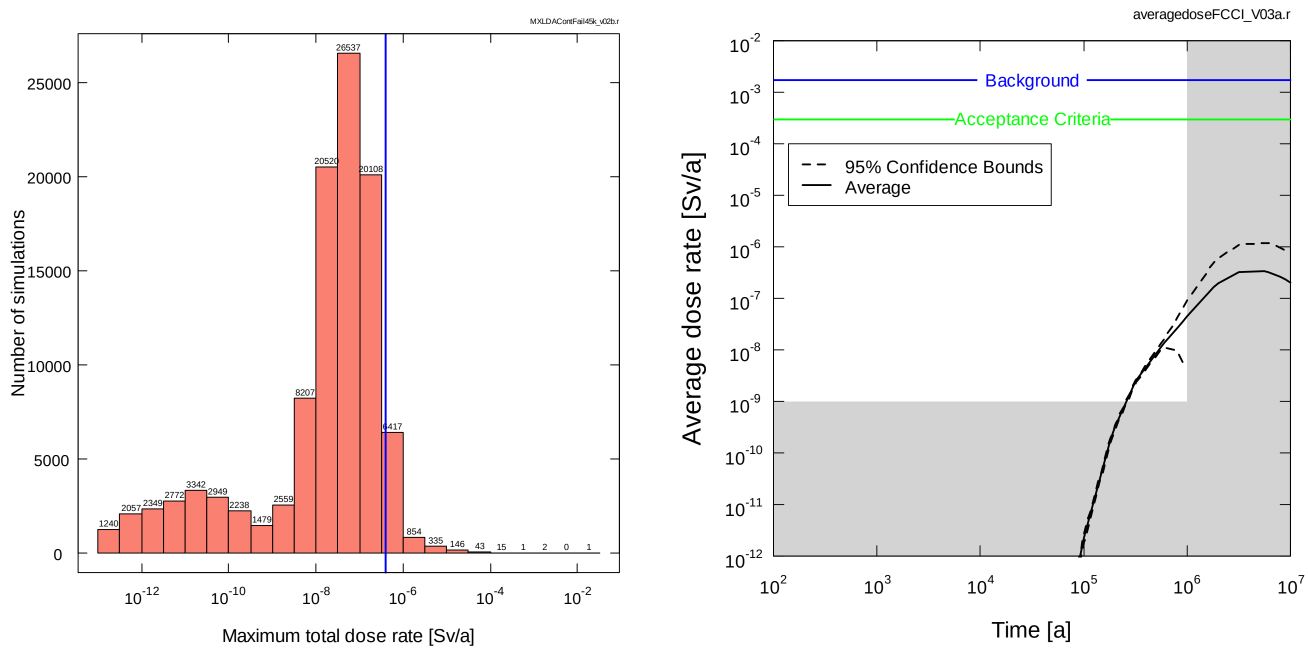


Figure 8: Results of Probabilistic Simulations

6.4 Disruptive Events

<u>All Containers Fail at 10,000 and 100,000 Years</u>

Figure 9 shows the dose rate for the two All Containers Fail Disruptive Event Scenarios. I-129 is the dominant dose contributor in both cases.

The leftmost figure shows that the peak dose rate assuming the failures occur at 100,000 years is $9.4x10^{-5}$ Sv/a. The increase is not linear with the number of failed containers because not all groundwater passing through the repository discharges to the well (i.e., the dominant exposure pathway for the Self-Sufficient Farmer critical group). The peak dose rate is about a factor of 10 below the acceptance criterion of $1x10^{-3}$ Sv/a.

The rightmost figure contrasts this result with that for a case in which the container failures occur at 10,000 years. The results are not substantially different (i.e., the peak dose rate is now $1.0x10^{-4}$ Sv/a). The lack of sensitivity occurs because the release time is still sufficiently long to permit essentially all other fission products to decay away. The actinides are not released in any significant amounts due to retention in the used fuel and sorption in the engineered and natural barriers.

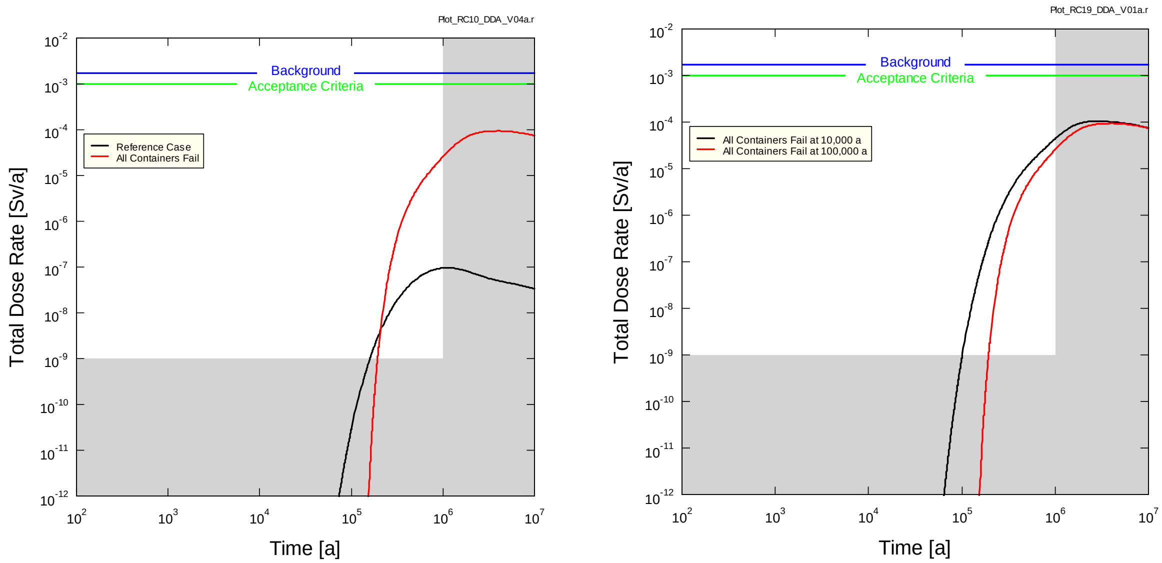


Figure 9: All Containers Fail

Inadvertent Human Intrusion

The repository siting minimizes the possibility of inadvertent intrusion by:

- selecting a site with no minerals of known economic potential;
- excavating the repository 500 m below the surface;
- selecting a site with no potable groundwater at repository depth;
- using markers and other means to preserve institutional memory for as long as possible.

If the markers or records become lost or misunderstood, then any intrusion into the repository would be inadvertent.

Two potential critical groups are considered:

- a drill crew exposed to contaminated drill slurry spread on the surface around the drill rig and to a core section containing used nuclear fuel, and
- a resident at the site, exposed by living nearby and growing food on soil contaminated by drill slurry.

The Human Intrusion Model was created using AMBER 5.3 [6]. The model solves for doses received from inhalation, ingestion, groundshine and external irradiation.

Acute doses are received by the drill crew technician upon intrusion while the resident receives a chronic dose that assumes no site remediation.

The calculated dose rates are shown in Figure 10. The dose tends to be dominated by Am-241 for the first 300 to 1000 years, by Pu-240 and Pu-239 from 10^3 to 10^5 years, and by the U-238 decay chain radionuclides for longer times. This is in contrast to the Normal Evolution Scenario in which actinides are slow to dissolve, sorb strongly in the repository and geosphere, and generally do not reach the surface.

Figure 10 also shows the dose to a resident that begins living on the site 100 years after the intrusion has occurred – credit for leaching is included in this sensitivity case.

This scenario is very unlikely and the number of people receiving these doses is small and limited to people having intimate contact with the used fuel debris.

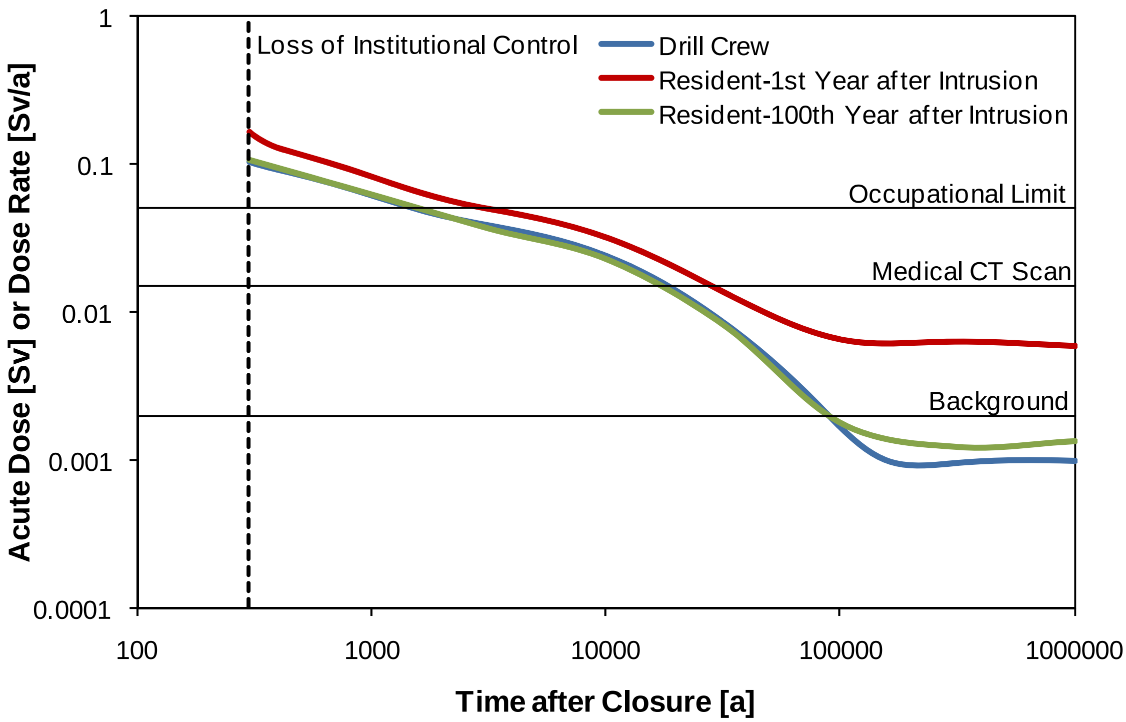


Figure 10: Dose as a Function of Time after Closure

7. CONCLUSIONS

The paper reports on a postclosure safety assessment performed for a conceptual repository in a hypothetical geosphere in crystalline rock. The study considers a Reference Case of the Normal Evolution Scenario, a range of sensitivity cases conducted about the Reference Case and a set of Disruptive Event Scenarios.

The overall conclusion, based on the cases considered, indicates that the deep geological repository design and hypothetical site provide effective containment, isolation and retention of radionuclides. The health and safety of persons and the environment is well protected.

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