A Preliminary Performance Assessment for Salt Disposal of High-Level Nuclear Waste – 12173

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ABSTRACT

A salt repository is one of the four geologic media currently under study by the U.S. DOE Office of Nuclear Energy to support the development of a long-term strategy for geologic disposal of commercial used nuclear fuel (UNF) and high-level radioactive waste (HLW). The immediate goal of the generic salt repository study is to develop the necessary modeling tools to evaluate and improve the understanding of the repository system response and processes relevant to long-term disposal of UNF and HLW in a salt formation. The current phase of this study considers representative geologic settings and features adopted from previous studies for salt repository sites. For the reference scenario, the brine flow rates in the repository and underlying interbeds are very low, and transport of radionuclides in the transport pathways is dominated by diffusion and greatly retarded by sorption on the interbed filling materials. I-129 is the dominant annual dose contributor at the hypothetical accessible environment, but the calculated mean annual dose is negligibly small. For the human intrusion (or disturbed) scenario, the mean mass release rate and mean annual dose histories are very different from those for the reference scenario. Actinides including Pu-239, Pu-242 and Np-237 are major annual dose contributors, and the calculated peak mean annual dose is acceptably low.

INTRODUCTION

The U.S. is currently re-evaluating the policy on commercial used nuclear fuel (UNF) and high-level radioactive waste (HLW) management. As part of the Fuel Cycle Research and Development (FCRD) program supported by the U.S. DOE Office of Nuclear Energy, the Used Fuel Disposition (UFD) campaign has been studying generic disposal system environment (GDSE) concepts to support the development of a long-term strategy for geologic disposal of commercial UNF and HLW. The GDSE study focuses on the comparative analysis of different GDSE options, and a salt repository is one of the options currently under study.

The immediate goal of the generic salt repository study is to develop the necessary modeling tools to evaluate and improve the understanding of the generic disposal system (GDS) response and processes relevant to long-term disposal of UNF and HLW in a salt formation. The current phase of this study considers representative geologic settings and features adopted from previous studies for salt repository sites including the Waste Isolation Pilot Plant (WIPP) [1] and the Gorleben salt dome site in Germany [2-5]. These studies were also used to develop a conceptual model and scenario for radionuclide release and transport. The preliminary performance assessment model of a generic salt repository described in this paper represents significant improvements over an initial effort [6], especially for the brine flows in a representative generic salt repository.

MODEL DESCRIPTION

Conceptual Model

Figure 1 shows a schematic of the geologic setting and the conceptual model for radionuclide release and transport in a generic salt repository. The model assumes that repository is located in a bedded salt formation in a saturated, chemically reducing environment. The waste package is assumed to be placed horizontally in an emplacement alcove and backfilled with crushed salt. Over a period of time following the emplacement, the confined space around the waste disposal area would be slowly closed by creep deformation of salt rock, and the crushed salt backfill undergo consolidation. This would result in close contact of the waste package with the consolidated salt rock and potential encapsulation of the waste package by salt rock.

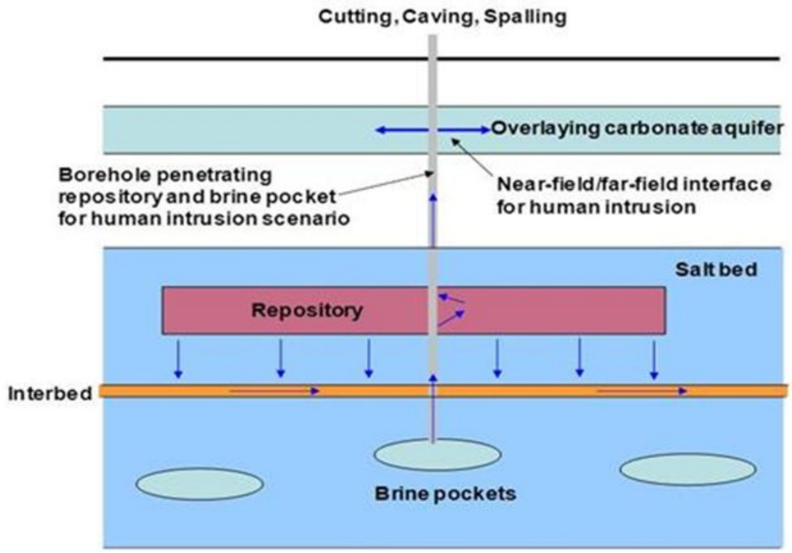

Fig 1. A Schematic Showing the Conceptual Model for Radionuclide Release and Transport from a Salt Generic Repository.

A horizontal interbed with a significant thickness (assume one meter thick) of relatively more permeable anhydrite is assumed to exist below the repository, and runs in parallel with the repository horizon to an extended distance; a carbonate aquifer is assumed to exist above the repository. Horizontal interbeds are commonly found in a bedded salt formation [1]. Two scenarios are considered for repository radionuclide release and transport: the reference (or nominal) case, and the disturbed case. The reference case releases radionuclides by a sequence of processes that could occur in a generic salt repository; the case assumes that the interbed provides the primary pathway for radionuclide release and transport from the repository; this model assumption is supported by the model results of a certain scenario from WIPP [1] and needs additional analysis for confirmation.

The above reference-case conceptual model for radionuclide transport pathway is different from that of the Gorleben salt dome repository as the salt dome site employs different conceptual models for brine flow and radionuclide transport pathway for the performance assessment of the repository [7-8]. The alternative conceptual models developed for the transport pathway of the salt dome repository include: 1) vertical upward movement of brine toward the overlying aquifer driven by pressurization from the reduction in the confined space as a result of salt rock creep deformation; and 2) lateral movement of brine caused by brine exchange driven by the brine density gradient in adjacent salt rock volumes, which could be enhanced by the variations in local heating by the waste decay heat [7-8].

The disturbed case considers a process that could result in a fast pathway for radionuclides to the far-field, and the case is represented by a "stylized" human intrusion scenario, which assumes that a single borehole penetrates waste package. The case assumes that a large pressurized brine reservoir exists below the repository and is also penetrated by the borehole.

The pressurized brine moves dissolved radionuclides from the breached waste package up through the borehole, resulting in the direct release of radionuclides into the overlying aquifer. Pressurized brine reservoirs are found in a bedded salt formation, and the feature is consistent with a human intrusion scenario developed for the performance assessment of the WIPP [1]. The current human intrusion scenario does not consider the potential dose impacts of the waste that could be brought up directly to the surface as a result of the drilling activities. The modeling assumption for the disturbed case will be updated as the study progresses.

In the postclosure repository, the waste decay heat would cause near-field brines (present in small quantities in undisturbed bedded salt rock as pore water) to boil during the peak thermal perturbation period, driving the water away from the waste disposal area leaving behind salt minerals in the pore space. This would create a dry-out zone around the waste disposal area, and its duration would depend mostly on the waste heat output characteristics, repository thermal loading, and thermal characteristics of the salt rock. The thermal perturbation and its associated moisture movement would also enhance creep deformation of salt rock and closure of the open space of the waste disposal area [13]. As the temperatures decrease following the peak, brines could start moving toward and into the waste disposal area driven by higher (i.e., near lithostatic) pore pressure in the far field.

Corrosion of waste package and other engineered materials in the disposal area could be enhanced when in contact with concentrated brines at elevated temperatures, and gases generated as a result of the corrosion under chemically reducing conditions. Subsequent to waste package corrosion failure, corrosion of the waste form canister, waste form, and waste package internal structure materials would occur, releasing radionuclides and generating additional corrosion gases. Combined actions of the corrosion gas generation and decreasing confined space in the disposal area by salt creep deformation would pressurize the disposal area; this could result in brine movements and potential transport of dissolved radionuclides away from the disposal areas to some distance.

Since a predictive model for thermal perturbation and the associated repository processes (such as brine movement, salt rock creep, waste package and waste form corrosion, corrosion gas generation, etc.) for a representative generic salt repository is not available, the current salt GDS model assumes an isothermal condition at 25°C for the generic repository. Due to the lack of information at this early stage of model development, the current GDS model does not consider conservatively the barrier performance of the waste package and waste form canisters. They are assumed to fail immediately (at time zero), and waste form degradation occurs from the beginning of the analysis. The performance of these barriers will be included in future salt GDS model efforts as their degradation model becomes available. Results from these analyses are only meant for demonstration purposes and should not be used to support any decisions.

PERFORMANCE ASSESSMENT MODEL

The current salt GDS model consists of four major model components: source-term, near-field, far-field, and biosphere. The source-term and near-field model include the following components: (1) waste package configurations, (2) inventory for different waste types (commercial UNF, existing DOE HLW (DHLW), and hypothetical commercial reprocessing HLW (CHLW)), (3) repository layout, (4) waste form degradation, (5) solubility of key radionuclides, (6) near-field volume, (7) repository waste inventory scenarios, and (8) repository radionuclide release scenarios. Detailed descriptions of the above model components and analyses are found in a recent report [9]. Analyses were conducted to estimate the brine flow rate out of the waste disposal area and in the underlying interbed. The brine flow rates resulting from this analysis are abstracted and input to the salt repository performance analysis. Radiation dose to a receptor in the biosphere is used as a performance metric for the generic salt repository. The model includes a hypothetical reference biosphere that is assumed to be located at 5-km down-gradient from the salt repository boundary [9].

The salt GDS model was implemented with the GoldSim software code [10]. A probabilistic repository performance analysis was conducted, with 100 realizations for each case and for a time period of one million years.

BRINE FLOW ANALYSIS

The BRAGFLO software [11] models two phase flow through porous media and includes the effects of many other processes such as gas generation from iron corrosion and rock compressibility. The software was used to evaluate the persistence of the dry-out region and to model the brine and gas flow in the waste disposal area of the generic salt repository. The NUTS software [12] draws on the flow fields determined by BRAGFLO and utilizes a tracer to track the flow of the fluid of interest. The software was used to determine the brine flow from the waste package to the surroundings and to determine the portion of the brine that has contacted waste.

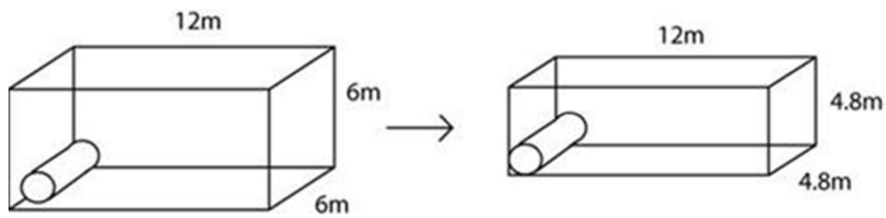

Fig 2. Illustration of Initial and Final Alcove Geometry.

The initial geometry for the brine flow analysis is a 6 m by 6 m by 12 m alcove completely filled with 38% porosity crushed salt and a 1.6 m diameter, 5.5 m length waste package placed against the back wall (Figure 2). After the placement of the waste package in the alcove, it is assumed that the crushed salt dries out due to the elevated temperatures and subsequently reconsolidates to a final porosity of 1%. This is reasonable given the rapid consolidation time relative to repository timeframes. Elevated temperatures would increase the consolidation process [13]. Taking into account the reconsolidation of the crushed salt and the creep of the surrounding salt rock, the final alcove dimensions were determined to be 4.8 m by 4.8 m by 12 m, assuming that the length of the alcove remained the same (Figure 2). To convert the waste package dimension to rectangular coordinates, the 1.6-m diameter waste package was approximated as a 1.4-m by 1.4-m rectangular waste package to preserve the volume of the waste package.

As the alcove is mined out, the surrounding rock is disturbed. The depth of the disturbed zone was estimated by approximating the stress trajectories around the alcove and determining the maximum distance between the stress trajectories and the alcove walls. The moisture from the disturbed zone is also assumed to be driven out by the elevated temperatures. The disturbed zone is assumed to be completely reconsolidated at the commencement of the calculation resulting in a dry intact salt material. This is reasonable given the rapid reconsolidation time relative to repository timeframes. For this analysis, the alcove was placed in the center of a salt bed with a 260 m vertical dimension and 1,035 m horizontal dimension, with impermeable layers both above and below the salt bed. A horizontal anhydrite layer (interbed) was connected to the lower disturbed zone. The boundaries were selected to be far enough away from the alcove so as not to influence the calculations near the alcove over the duration of the numerical simulation.

The primary objective in creating the modeling grid is to capture the effects of potential brine flow into the initial dry-out region. This can be accomplished by using a vertical two-dimensional (2D) grid, oriented along the length of the alcove (Figure 3). The grid is shown as a logical grid in Figure 3, with the length (Δx), width (Δz) and height (Δy) of each grid cell indicated (in meters). A technique of “radial flaring” was used to capture three-dimensional (3D) flow effects. The width of each grid cell increases with distance away from the center of the alcove, simulating the convergent or divergent flow centered on the alcove.

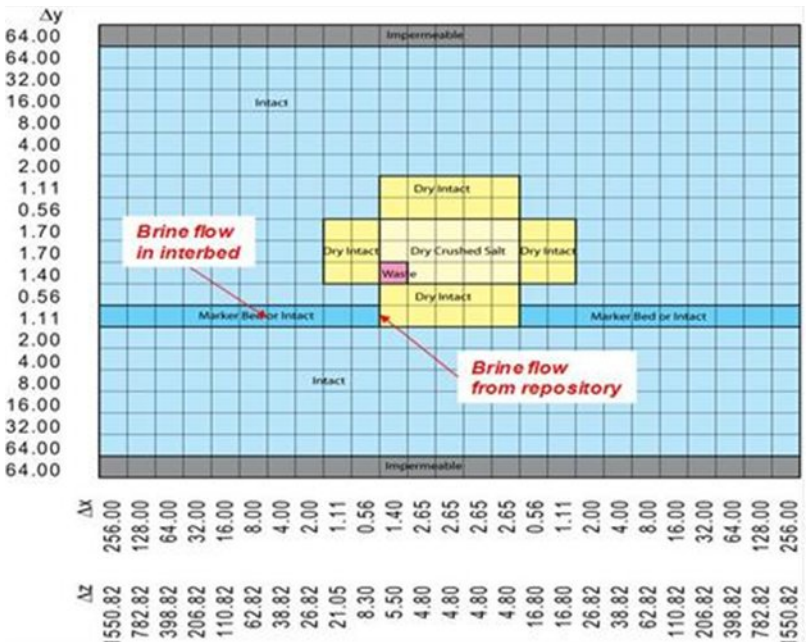

Fig 3. Long-term Brine Flow Grid Used in Analysis.

The resulting grid contains six different materials: (1) impermeable (gray); (2) saturated intact salt (light blue); (3) dry intact salt (yellow); (4) dry reconsolidated crushed salt (light yellow); (5) waste material (pink); and (6) interbed (or marker bed) (blue). The properties of an anhydrite layer were used for the interbed material.

Two locations were selected to obtain for the brine velocity information for use in developing the abstraction to be used in the salt GDS (Figure 3): (1) brine velocities from the repository chosen at the edge of the initial dry-out region right below the waste disposal area; and (2) brine velocities in the underlying interbed chosen at 8 m from the edge of initial dry-out region. The 8-m brine velocities were used for the far-field interbed brine velocities. This is conservative because brine velocities are expected to decrease with the distance from the repository due to increased spread-out of brines with distance and associated pressure drop.

The initial and boundary conditions, model parameters, and the analysis results are described in detail in Ref. [9]. The time-dependent velocity of the contaminated brine flowing out the repository and interbed was determined and used in the salt GDS performance analysis. Figure 4 shows the results of the brine velocity histories from the repository and underlying interbed that are abstracted into the salt GDS model. For each location, a set of 100 flow rate histories (or 100 realizations) were calculated to represent the uncertainty in the brine flow rate that is derived from the input parameter uncertainties [9]. The velocities are lower the further from the dry-out region due to the dilution of the surrounding uncontaminated brine. As the brine flow rates are very low for most of the realizations, especially in the interbed, radionuclide release and transport from a salt GDS would be dominated by diffusion as discussed in the model demonstration below. The flow rate histories are implemented in the salt GDS model as a look-up table. The histories are sampled randomly in the GDS performance analysis, having the histories at the two locations perfectly correlated.

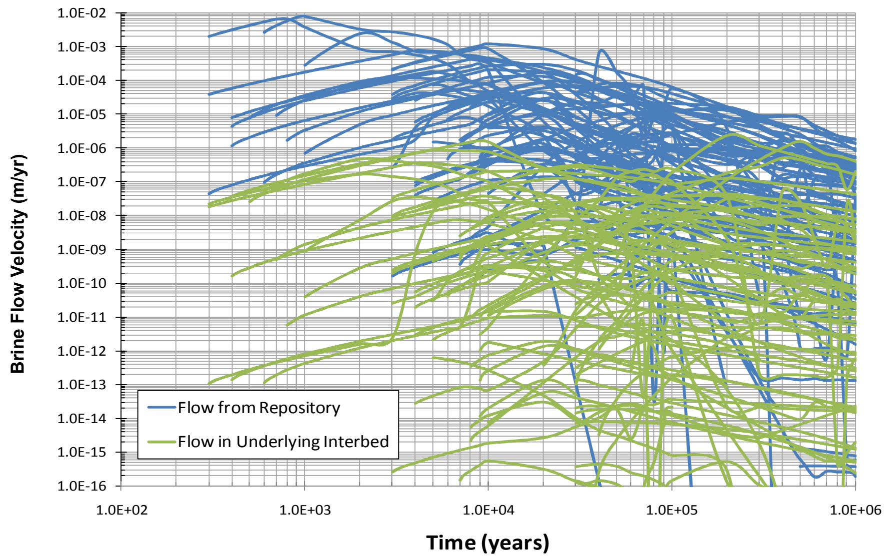

Fig 4. Abstraction Results for Brine Flow Rate Histories from the Repository and in Underlying Interbed.

MODEL DEMONSTRATION ANALYSIS

This section discusses analysis of the capability demonstration for the current version of the salt GDS model. The model results are presented in terms of the mean radionuclide mass release rate from the near field and far field as the intermediate performance metrics, and the mean annual dose (mrem/yr) by individual radionuclide at the hypothetical accessible environment. The current model is part of an on-going effort to develop the capability of modeling the performance of a repository for high-level nuclear waste located in a salt formation. The use of the mean annual dose is an arbitrary choice to present and discuss the analysis results in order to facilitate a consistent and useful comparison among GDS options. The scientific basis for the results presented is still under development, therefore *should not* be utilized for decision making at this time. The purpose remains a demonstration of modeling capability and as such represents a first look at viability.

The salt GDS model was implemented in GoldSim [11]. The model demonstration was performed probabilistically, with 100 realizations for each case and over a time period of 1,000,000 yr.

Reference Scenario Analysis

The waste inventory case considered for the reference scenario (undisturbed scenario) comprises the commercial UNF and DHLW (encapsulated in borosilicate glass). The analysis assumes a square repository footprint with a side of 3,270 m for disposal of a total of 37,157 waste packages (32,154 commercial UNF waste packages plus 5,003 DHLW waste packages). Details of the inventory analysis are discussed in Ref. [9].

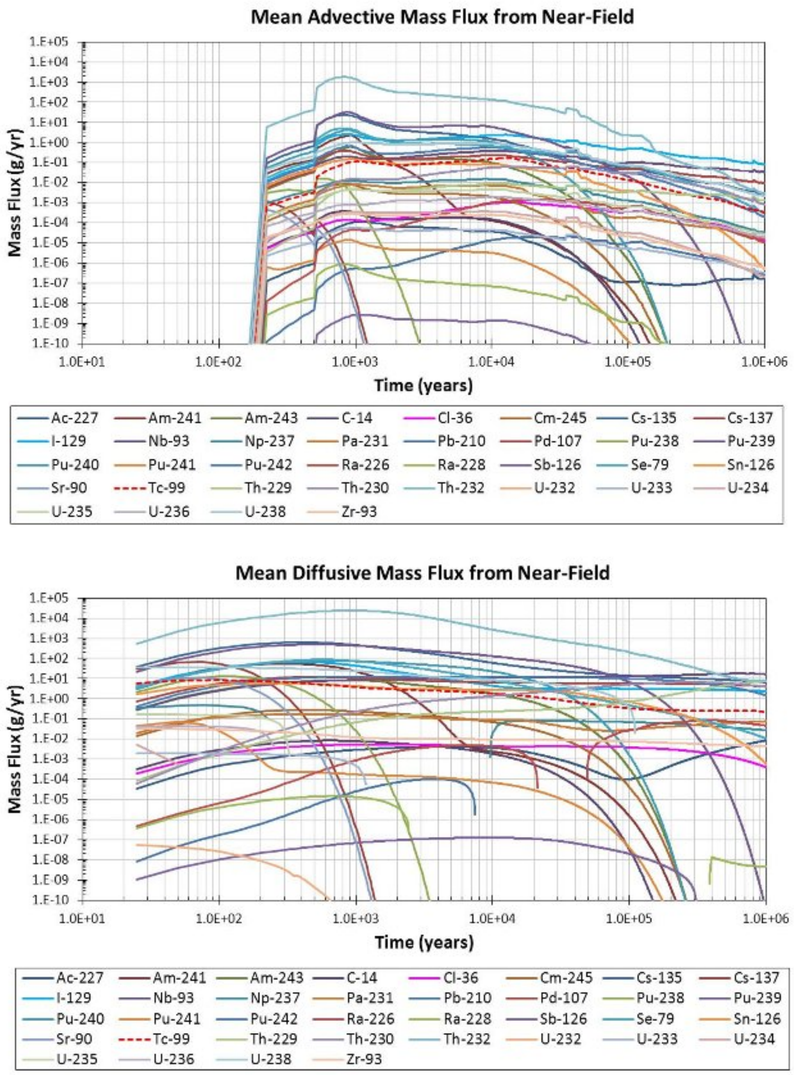

Fig 5. Model Results for the Reference Scenario: Mean Advective and Diffusive Release Rate from Repository.

Figure 5 shows the model results for the mean advective and diffusive radionuclide mass release rate from the repository. As expected from the very low calculated brine flow rates from the repository (Figure 4), the release rates are dominated by diffusion, and contribution from advection is negligible compared to the diffusive release. As discussed earlier for the conceptual model (Figure 1), the current salt GDS model does not consider sorption of radionuclides in the near field and interface rock below repository, so even these releases are over-estimated. Th-232 shows the highest mean release rate by both diffusion and advection, followed by Pu-239 and Cs-135. The mean release rate of I-129 (non-sorbing, mobile radionuclide with unlimited solubility and a very long half-life) becomes important at later times (after about 10^5 yr). The broken curves shown for some radionuclides (e.g., Np-237, Pb-210, Ra-226, Ra-228, etc.) in the diffusive release figure are due to the back-diffusion (negative mass flux) and the inability to present negative values on a log plot.

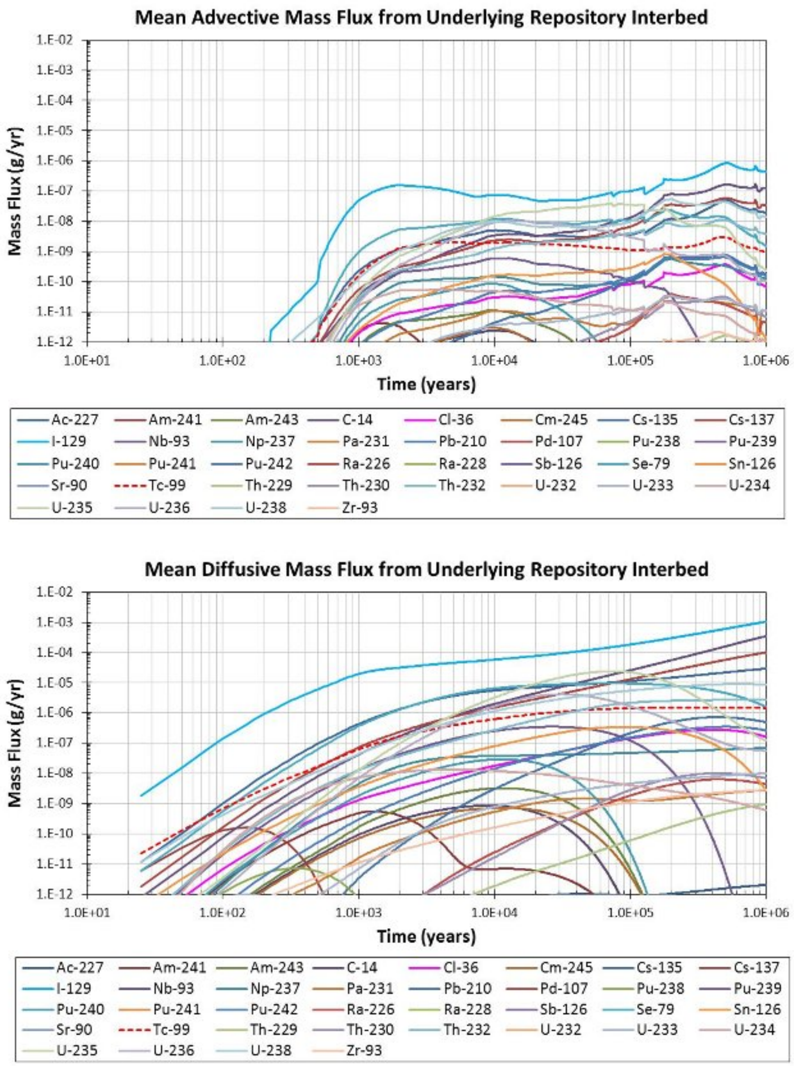

Fig 6. Model Results for the Reference Scenario: Mean Advective and Diffusive Release Rate from the Underlying Interbed at the Boundary of Repository Footprint.

Figure 6 shows the mean advective and diffusive release rates from the underlying interbed at the boundary of the repository footprint. Consistent with the conceptual model, radionuclides are transported into the underlying interbed over its entire area of the repository footprint (refer to the conceptual model discussion, Figure 1). As for the release from the repository, the mean diffusive release rate from the repository interbed is much greater than the mean advective release rate. Sorption of radionuclides on the interbed filling materials is considered in the interbed, and I-129 (non-sorbing and unlimited solubility) is the dominant radionuclide released at the repository boundary.

Figure 7 shows the mean advective and diffusive release rates from the far-field interbed at 5 km from the edge of the repository. Transport of radionuclides in the far-field interbed is similarly dominated by diffusion and is greatly retarded by sorption on the interbed filling materials. Again, I-129 is the dominant radionuclide released, and the calculated mean release rates are so low that there would be no meaningful consequence for the repository performance under this scenario and for this waste inventory case. This is demonstrated by the negligibly small mean annual dose (predominantly by I-129) at the hypothetical accessible environment shown in Figure 8.

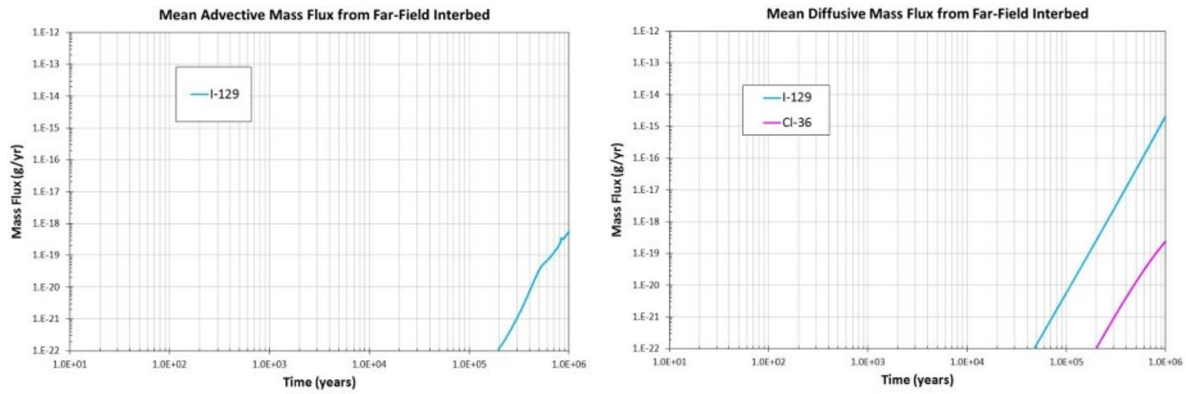

Fig 7. Model Results for the Reference Scenario: Mean Advective and Diffusive Release Rate from the far-Field Interbed at 5 km from the Boundary of Repository Footprint.

Human Intrusion Scenario Analysis

The same waste inventory case as the reference (or nominal) scenario analysis is analyzed for the disturbed (or human intrusion) scenario analysis. For simplification, the analysis considers the situation in which only the commercial UNF waste packages are affected by the human intrusion activity. Consistent with the conceptual model discussed above (Figure 1), the dissolved radionuclides that are released from the affected waste package(s) into the near-field water are transported upward through a borehole by pressurized brine from the underlying pressurized brine reservoir and released directly to the overlying aquifer. The aquifer water flow rate is several orders of magnitude greater than the brine flow rate in the interbed, and the radionuclides are transported advectively at much greater rates (see Ref. [9] for the discussion of the aquifer water flow rate used in the analysis). The model assumes that the location of the borehole penetration in the repository is uncertain and does not consider the distance from the penetration location to the repository boundary.

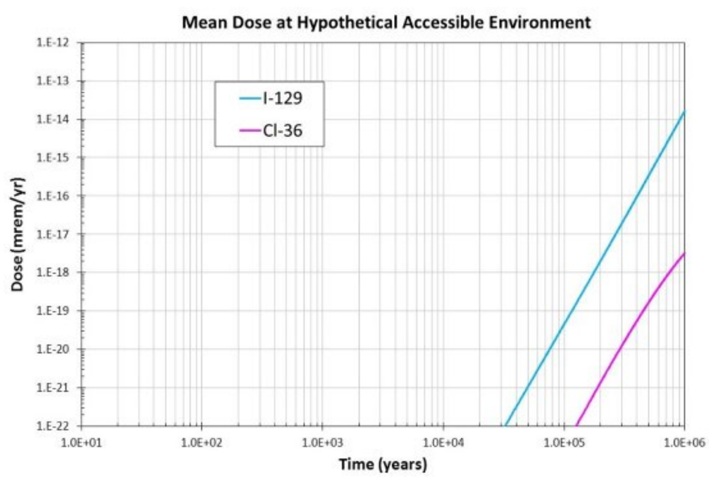

Fig 8. Model Results for the Reference Scenario: Mean Annual Dose at the Hypothetical Accessible Environment.

Figure 9 shows the model results of the mean mass release rate from the repository through a borehole that has penetrated the repository at 1,000 yr after repository closure. The release rate is at the location where the borehole has penetrated the repository. The number of waste packages affected (i.e., the amount of inventory that becomes available for release) by the human intrusion activity is sampled between one and five [9]. U-238 is the dominant radionuclide in terms of the mean release rate for the entire analysis time period. The dissolved U-238 concentration in the near-field water is limited by the solubility for the entire analysis time period. Pu-239 is the second dominant radionuclide for up to about 8×10^4 yr, then Nb-93, U-235, Np-237, Cs-135 and U-236 become important by about the same degree. Nb-93 is a stable isotope and does not have dose consequences.

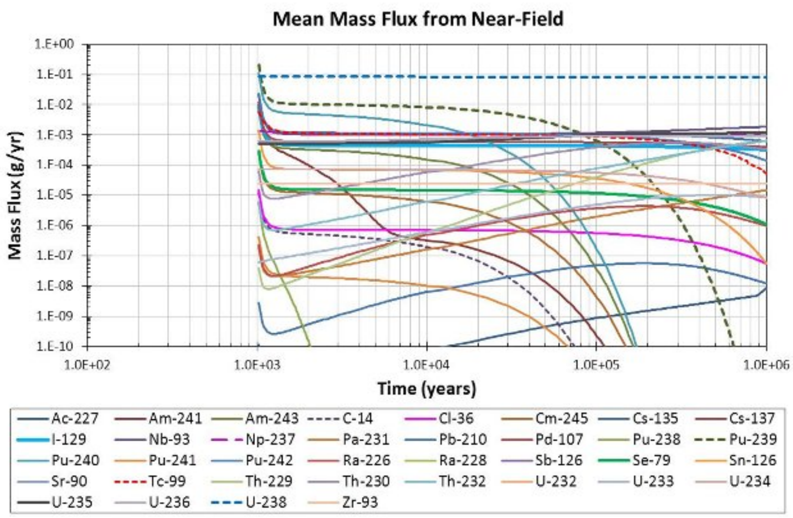

Fig 9. Model Results for the Human Intrusion Scenario: Mean Release Rate from the Repository.

Shown in Figure 10 is the mean mass release rate from the far-field overlying aquifer at 5 km from the boundary of repository footprint. For most radionuclides, the far-field mean release rates are substantially lower than the mean repository release rates due mainly to transport retardation by sorption on the aquifer materials and dilution in the aquifer. U-238 shows the highest mean release rate for the entire analysis time period, except the very early time period for up to about 1,400 yr, for which the highest mean release rate is by Cl-36.

The calculated mean annual doses by individual radionuclides at the hypothetical accessible environment are shown in Figure 11. Although the far-field mean mass release rate is dominated by U-238, other radionuclides dominate in terms of mean annual dose at different times: C-14 is the dominant mean annual dose contributor for about first 3×10^3 yr; Np-237 is the dominant mean annual dose contributor from about 3×10^3 yr to about 3.5×10^4 yr and again from about 2×10^5 yr to the end of analysis (1,000,000 yr); and Pu-239 is the dominant mean dose contributor from about 3.5×10^4 yr to about 2×10^5 yr. This is the result mainly of much higher specific activity of the radionuclides (C-14: 4.47 Ci/yr, Pu-239: 0.06 Ci/yr, Pu-242: 0.004 Ci/yr, and Np-237: 0.0007 Ci/yr) than U-238 (3.4×10^{-7} Ci/yr).

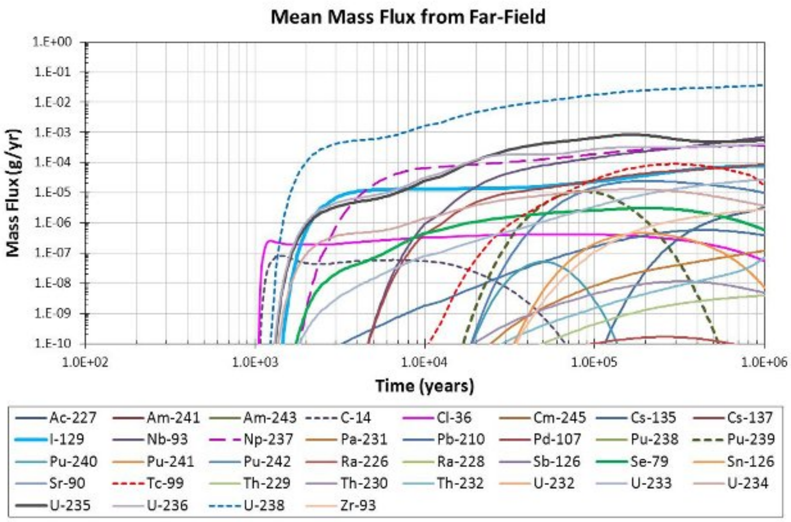

Fig 10. Model Results for the Human Intrusion Scenario: Mean Release Rate from the Far-Field Overlying Aquifer at 5 km from the Boundary of Repository Footprint.

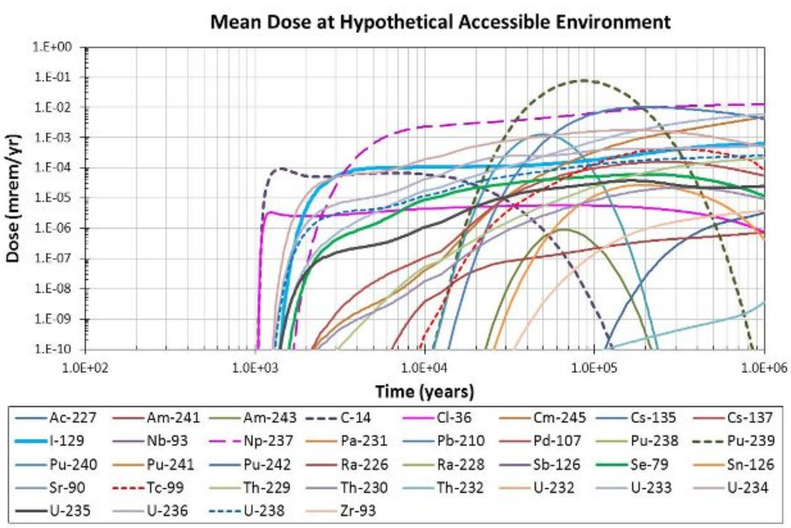

Fig 11. Model Results for the Human Intrusion Scenario: Mean Annual Dose at the Hypothetical Accessible Environment.

SUMMARY AND CONCLUSIONS

A performance assessment model for a generic salt repository has been developed incorporating, where applicable, representative geologic settings and features adopted from literature data for salt repository sites. The conceptual model and scenario for radionuclide release and transport from a salt repository were developed utilizing literature data. The salt GDS model was developed in a probabilistic analysis framework. The preliminary performance analysis for demonstration of model capability is for an isothermal condition at the ambient temperature for the near field. The capability demonstration emphasizes key attributes of a salt repository that are potentially important to the long-term safe disposal of UNF and HLW. The analysis presents and discusses the results showing repository responses to different radionuclide release scenarios (undisturbed and human intrusion).

For the reference (or nominal or undisturbed) scenario, the brine flow rates in the repository and underlying interbeds are very low, and transport of radionuclides in the transport pathways is dominated by diffusion and greatly retarded by sorption on the interbed filling materials. I-129 (non-sorbing and unlimited solubility with a very long half-life) is the dominant annual dose contributor at the hypothetical accessible environment, but the calculated mean annual dose is negligibly small that there is no meaningful consequence for the repository performance.

For the human intrusion (or disturbed) scenario analysis, the mean mass release rate and mean annual dose histories are very different from those for the reference scenario analysis. Compared to the reference scenario, the relative annual dose contributions by soluble, non-sorbing fission products, particularly I-129, are much lower than by actinides including Pu-239, Pu-242 and Np-237. The lower relative mean annual dose contributions by the fission product radionuclides are due to their lower total inventory available for release (i.e., up to five affected waste packages), and the higher mean annual doses by the actinides are the outcome of the direct release of the radionuclides into the overlying aquifer having high water flow rates, thereby resulting in an early arrival of higher concentrations of the radionuclides at the biosphere drinking water well prior to their significant decay.

The salt GDS model analysis has also identified the following future recommendations and/or knowledge gaps to improve and enhance the confidence of the future repository performance analysis.

- Repository thermal loading by UNF and HLW, and the effect on the engineered barrier and near-field performance.
- Closure and consolidation of salt rocks by creep deformation under the influence of thermal perturbation, and the effect on the engineered barrier and near-field performance.
- Brine migration and radionuclide transport under the influence of thermal perturbation in generic salt repository environment, and the effect on the engineered barrier and near-field performance and far-field performance.
- Near-field geochemistry and radionuclide mobility in generic salt repository environment (high ionic strength brines, elevated temperatures and chemically reducing condition).
- Degradation of engineer barrier components (waste package, waste canister, waste forms, etc.) in a generic salt repository environment (high ionic strength brines, elevated temperatures and chemically reducing condition).
- Waste stream types and inventory estimates, particularly for reprocessing high-level waste.

REFERENCES

1. Helton, J.C., et al (1998). *Uncertainty and Sensitivity Analysis Results Obtained in the 1996 Performance Assessment for the Waste Isolation Pilot Plant*. Technical Report SAND98-0365, Sandia National Laboratories, Albuquerque, NM.

2. Klinge, H., et al (2007). *Description of the Gorleben Site Part 1: Hydrology of the Overburden of the Gorleben Salt Dome*. Bundesanstalt für Geowissenschaften und Rohstoffe (BGR), Hannover, Germany.

3. Köthe, A., et al (2007). *Description of the Gorleben Site Part 2: Geology of the Overburden and Adjoining Rock of the Gorleben Salt Dome*. Bundesanstalt für Geowissenschaften und Rohstoffe (BGR), Hannover, Germany.

4. Bornemann, O, et al (2008). *Description of the Gorleben Site Part 3: Results of Geological surface and Underground Exploration of the Gorleben Salt Dome*. Bundesanstalt für Geowissenschaften und Rohstoffe (BGR), Hannover, Germany.

5. Bräuer, V, et al (2011). *Description of the Gorleben Site Part 4: Geotechnical Exploration of the Gorleben Salt Dome*. Bundesanstalt für Geowissenschaften und Rohstoffe (BGR), Hannover, Germany.

6. Lee, J.H., Siegel, M., Jove-Colon, C., and Wang, Y. (2011). *A Performance Assessment Model for Generic Repository in Salt Formation*. Proceedings of the 13th International High-Level Radioactive Waste Management Conference (IHLRWMC), April 10-14, 2011, Albuquerque, NM.

7. Buhmann, D., et al (2009). *PAMINA Performance Assessment Methodologies in Application to Guide the Development of the Safety Case — Report on the Benchmarks on Rock Salt*. Deliverable (D-N: D4.1.1), PAMINA Sixth Framework Programme, European Commission.

8. Rübel, A., et al (2009). *PAMINA Performance Assessment Methodologies in Application to Guide the Development of the Safety Case — PA Approaches Based on Different Geometric Complexity of Modeling for the Far-Field of a Repository in Salt*. Deliverable (D-N: D4.2.1), PAMINA Sixth Framework Programme, European Commission.

9. Clayton, D., et al (2011). *Generic Disposal System Modeling — Fiscal Year 2011 Progress Report*. Technical Report SAND2011-5828P, Sandia National Laboratories, Albuquerque, NM.

10. GoldSim Technology Group 2011. *GoldSim version 10.50*.

11. Nemer, M.B. (2007). *Software Installation and Checkout for BRAGFLO, Version 6.0*. ERMS 545019, Sandia National Laboratories, Carlsbad, NM.

12. Gilkey, A.P. (2006). *Software Installation and Checkout and Regression Testing Report of NUTS Version 2.05c on the Compaq ES40, ES45 and ES47 Platforms*. ERMS 543789, Sandia National Laboratories, Carlsbad, NM.

13. Hansen, F.D., and Leigh, C.D. (2011). *Salt Disposal of Heat-Generating Nuclear Waste*. Technical Report SAND2011-0161, Sandia National Laboratories, Albuquerque, NM.

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