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Independent Post-closure Performance Estimates of the Proposed Repository at Yucca Mountain, Nevada

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ABSTRACT - *This paper summarizes the key findings from a suite of independent analyses of the performance of the proposed repository at Yucca Mountain Nevada, USA conducted by the Center for Nuclear Waste Regulatory Analyses (CNWRA) and the U.S. Nuclear Regulatory Commission (NRC). The analyses are geared toward obtaining risk insights from deterministic and probabilistic analyses of potential exposure to people in a down-gradient community; the determination of the capability of barriers to reduce flow of water and to prevent or delay radionuclide transport; and the identification of models, parameters and subsystems that have the most influence on repository performance through sensitivity and uncertainty analyses. The analyses by the staffs of the CNWRA and NRC have allowed them to focus attention on the most critical parts of the analysis of post-closure repository performance.*

I. INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC), with technical assistance from the Center for Nuclear Waste Regulatory Analyses (CNWRA), has developed performance assessment tools to review and quantitatively evaluate the safety analyses in a potential license application by the U.S. Department of Energy (DOE) for the proposed repository at Yucca Mountain, Nevada. To date, the NRC staff has reported four independent preliminary performance assessments for the proposed Yucca Mountain repository, showing the evolution and maturing of the NRC performance assessment approach and the assessment tools[1,2,3,4]. The most current revision reflects implementation of the DOE Enhanced Design Alternative II concept[5], new and revised conceptual models in the NRC/CNWRA Total-system Performance Assessment (TPA) code Version 4.1 code[4], and new risk assessment methods. Similar analyses have been performed by the DOE[5] and EPRI[6].

The paper summarizes the key results from this suite of analyses documented in a soon-to-be published report[4] that includes estimation of (i) risk to potentially exposed individuals in the critical group, (ii) capability of barriers to reduce flow of water and to prevent or delay radionuclide transport; and (iii) system-level sensitivity and uncertainty analyses to identify models, parameters and subsystems that have the most influence on repository performance. The analyses are preliminary and are performed solely for improving our understanding of the system. They are not intended to determine compliance with the NRC regulations[7].

II. COMPUTATIONAL MODELS

Conceptual models that describe the interactions and couplings of the physical and chemical processes can be grouped into the following categories: (1) precipitation, infiltration, and deep percolation, (2) near-field environment, (3) engineered barrier system, (4) disruptive events, (5) radionuclide release from the engineered barrier system, (6) aqueous-phase radionuclide transport in unsaturated and saturated zones, (7) airborne transport from possible extrusive volcanism, and (8) exposure to the biosphere from groundwater and ground surface releases. Since TPA uses Monte Carlo techniques requiring hundreds to thousands of computations, models are highly simplified and abstracted from more complex models.

The model representing *precipitation, infiltration, and deep percolation* assumes percolation of meteoric water at the land surface vertically downward through the repository, and ultimately to the water table. The deep percolation flux is calculated from knowledge of present-day percolation at the site[8], taking into consideration potential climate changes, elevation, and soil depth on the mountain. The effects of site-specific soil cover thickness and elevation are used to reflect the spatial variability of infiltration and percolation.

The *near-field environment model* calculates the physical and chemical processes in the near field, which are affected by repository heat and chemistry and hydrology close to the waste. The model calculates drift wall and waste package surface temperature, relative humidity, water chemistry, and water reflux. The temperature model considers conduction, thermal radiation, convection and latent heat transfer (in special cases). Estimates of pH and chloride concentration are calculated externally using a geochemical code run externally to the TPA model[9].

Engineered Barrier System - The Engineered Barrier System consists mainly of the emplacement drifts, drip shield, waste packages and invert. The drip shield consists of a "mailbox" shaped cover fabricated of titanium alloy 7 over all waste packages. The main purpose of the drip shield is to protect the waste packages from dripping water and falling rocks. The main failure mode of the drip shield is expected to be from corrosion, especially from fluoride contained in dripping water, and damage from large falling rocks. Version 4.1 of the TPA code does not explicitly model drip shield failure, rather it treats externally computed drip shield failure time simply as a sampled parameter.

The waste package consists of an outer shell of corrosion-resistant, nickel-based Alloy 22 and an inner shell of stainless steel. The waste package failure model considers general and pitting corrosion, stress-corrosion cracking, undetected manufacturing defects, and failure from disruptive events. Corrosion of the outer shell is assumed to be possible under conditions of high relative humidity in the presence of minerals deposited on the surface by dust or dripping water. The inner shell's main function is mechanical strength, and is not expected to provide significant corrosion resistance. The model assumes that radionuclides cannot escape the waste package until its drip shield has failed, the waste package has been penetrated by corrosion, and water can enter and leave. A small number (less than 0.1 %) of waste packages is specified to have failed at the time of repository closure, as a result of fabrication defects and damage.

Disruptive events - There are three classes of disruptive events that could lead to radionuclide releases: seismic activity, fault displacement, and igneous activity. Seismicity can cause the waste packages to fail mainly by inducing large rocks to fall into the excavated tunnels onto the DSs and waste packages[10]. Fault displacement could cause failure by shearing of waste packages. Igneous intrusions are assumed to fail the waste packages by weakening the materials at high temperatures and through the forces exerted by the magma. Extrusive igneous activity (i.e., volcanism) is assumed to fail the waste packages and also carry their contents to the surface and into the air.

Water-borne radionuclide release from engineered barrier system - The waste form, either UO_2, uranium metal, or glass, will degrade in the presence of air and water. Failed cladding can partially protect the waste form but because of the large uncertainty, no cladding credit is taken in the basecase. Commercial spent nuclear fuel (CSNF) in the form of UO_2 constitutes the bulk of the waste. Fuel is specified to dissolve only in the presence of water, which comes into contact either by immersion (bathtub model), or dripping (flowthrough model). For the bathtub model, water must fill the failed waste package to an assumed overflow height before radionuclides can leave the waste package. In the flow-through model, the fraction of fuel wetted is the same as the fraction immersed in water in the bathtub model, but upon failure, radionuclides can leave without water first filling the waste package.

Most of the radionuclides are assumed to be released from waste at the rate it degrades or dissolves in water, or at a rate determined by the water flow and elemental solubility of the radionuclides released from the waste form. Volatile elements (e.g., iodine) are assumed to be partially available as soon as the waste package fails. There are several alternative models for CSNF dissolution in TPA Models 1 and 2 are based on assumptions about water chemistry in contact with the waste. Model 3 allows a release rate to be set to a prescribed value (e.g., from empirical observations of natural analogs)[11]. Model 4 assumes equilibrium with the uranium mineral schoepite[11]. Overall dissolution rate is computed from specific dissolution rate by factoring in the surface area of the exposed fuel, fraction of fuel wetted, and flow through the waste package.

Once released from the waste package, the radionuclides would first pass through the invert (the material under the waste packages), which allows for radionuclide decay, diffusion and retardation. If the infiltration rate exceeds the saturated hydraulic conductivity of the invert, then rapid fracture flow is specified to bypass the invert model, and direct release to the unsaturated flow zone (UZ) model.

Unsaturated and saturated zone flow and transport - Transport through the UZ below the repository is assumed to be in parallel, one-dimensional flow paths with non-steady, vertical flow. The model allows for advection, longitudinal dispersion, matrix diffusion for fractured-porous media, radioactive decay and radionuclide ingrowth. Transport through the saturated zone is assumed to be in four parallel, steady flowing tubes with advection, longitudinal dispersion, matrix diffusion and radioactive decay. Radionuclides travel through several zones characterized as fracture-matrix and porous flow before reaching the assumed points of groundwater use. The one-dimensional streamtubes were derived from an external two-dimensional modeling study of sub-regional flow[12].

Airborne transport from extrusive volcanism - Doses from extrusive volcanism are calculated by modeling releases of radionuclides in the airborne plume. The volcanism model assumes that magma intercepts and breaches waste packages, moves upward to the surface, and then ejects the ash and SF mixture to the atmosphere. Three primary factors determine the ash plume transport: (1) power and duration, (2) wind speed and direction (although we considered wind only blowing in the direction of the exposed group) and (3) SF and ash particle sizes. The ash transport model of Suzuki[13] was modified to take into account the ash blanket thickness, leaching and erosion rates and radioactive decay. Doses are most influenced by the timing of the event, with early events expected to result in larger doses.

Exposure to the reference biosphere - The exposed individual in this study is a person in a farming community of 100 families located 20 km downgradient from the site. The average member of the group is assumed to be exposed to radionuclides transported through the groundwater pathway, air pathway, or both. Dose results from ingestion, inhalation, and direct exposure. We assume that all radionuclides released from the repository to the groundwater (except for the fraction decayed) will eventually be taken up in user wells. Doses are based on the amount of radionuclides dissolved in groundwater reaching the wells, mixed into the total quantity of water used by the community. Groundwater dose pathways include drinking water, irrigation, and stock watering.

III. ANALYSIS METHODS

Analyses involved estimation of overall risk (Section III.A–B), barrier capability analysis (section III.C), (ii) sensitivity analysis to identify parameters and subsystems driving performance and associated uncertainties (section III.D–G), and several supplemental analyses such as criticality and human intrusion (Section III.H). The following is a brief description of the specifics of the analyses.

III.A. Deterministic Analysis

The first set of runs with the TPA 4.1 code were deterministic, using the mean value data set (i.e., a single run with all input variables represented as constants, chosen to be the mean value of each of the sampled parameter ranges). Restricting the code to mean value input data allowed the code to be analyzed in detail and many intermediate data streams to be checked from one module that are fed into the next.

III.B. Probabilistic Analyses

Most of the calculations with TPA Version 4.1 code were Monte Carlo, for which the values of as many as 330 parameters were sampled from input distributions using the Latin Hypercube Sampling method. The remaining 620 model parameters were specified constant. Some of the sampled parameters were correlated to other sampled parameters. Typically, a set consisted of 350 runs or vectors. The main purpose of the Monte Carlo calculations was to study the performance of the repository for the full range of uncertainty in parameters. Monte Carlo results were also used in many of the sensitivity analyses and to look at the ranges of the intermediate outputs. The output of the TPA code is presented in terms of the "peak-of-the-mean" dose to an average member of the critical group, which is the peak of the mean dose curve generated by averaging over all realizations for each time interval. The peak-of-the-mean dose is specified in NRC's rule for the Yucca Mountain repository[10].

III.C. Barrier Capability Analysis

Multiple barriers can be evaluated by assessing barrier capacity to substantially delay movement of water or radionuclides. The analysis involved the use of the system-level and intermediate-level performance assessment results.

III.D. Parametric Sensitivity Analysis

Parametric sensitivity analysis methods used in this study build on previous NRC total-system performance assessments. Several new methods were added for this study, including sensitivity based on the mean dose[14], fractional factorial design, and the Cumulative Distribution Function-Based Sensitivity Method[15]. Parametric sensitivity analyses were used to identify parameters for which a change to an input parameter has a relatively large effect on estimated repository performance. In most cases, sensitivity analyses were based on peak dose from each realization, which differs from the peak-of-the-mean approach for reporting the dose results. This procedure was generally acceptable; one exception was for parameters that affect the timing of release (e.g., drip shield failure time), for which the use of peak doses from each realization led to an overestimate of sensitivity for the peak-of-the-mean dose. The peak-of-the-mean dose was used for the comparison of alternative conceptual models, however.

Most of the statistical analyses relied on a 4,000-vector Monte Carlo set calculated for the base case scenario. Sensitivities for the igneous activity scenario relied on multiple smaller (350) Monte Carlo run sets. Data were also scaled or standardized to take into account change in a variable relative to its allowed range. Variable transformations such as logarithmic and ranks were used to improve the regressions, but tended to distort the meaning of the results by giving too much weight to small doses.

All nonstatistical sensitivity analysis techniques required sets of runs specified by the method itself. These include the Morris method, fractional factorial design, Fourier Analysis Sensitivity Test, and differential method.

Rankings for parametric sensitivity took a "consensus" approach[16] that determined the most sensitive variables according to their relative ranks for all sensitivity analyses. In the consensus approach, the conclusion on relative importance of the parameters was reached by examining the number of times each of the parameters appeared in the top group identified by the various sensitivity measures.

III.E. Distributional Sensitivity Analysis

This technique identified parameters for which choice of distribution function significantly affects the dose responses. The input distributions were changed either by shifting the mean of a distribution by 10 percent or changing the shape of the distribution while keeping the minimum and maximum fixed. This study used the 10 most influential parameters identified by the parametric sensitivity analyses.

III.F. Barrier Component Sensitivity Analysis

This technique looks at the whole barrier at once, either performing or not performing. The technique was useful for several reasons. First, it is an easily understood method that shows the direct effect of a physical barrier component. Second, it provides useful information on the importance of a barrier component when parametric sensitivity analysis fails to do so. In the Monte Carlo uncertainty analysis, the performance of barriers often could not be seen (e.g., there were never any corrosion failures of the waste packages within 10,000 years). Barrier component sensitivity analysis, which assumes failure of specific barriers, allows the exploration of barrier performance by reducing the overlapping capabilities of multiple barriers. The six barrier components of the engineered and natural barriers are drip shield, waste package, spent nuclear fuel, invert, unsaturated zone, and saturated zone. Barrier component failure or suppression was simulated by changing input parameters to degrade the performance severely (e.g., setting the alluvium distance to zero to suppress the saturated zone barrier). There was no attempt to define a probability associated with the suppressed barrier, and the technique was never used to calculate risk. The analyses considered several possibilities: (i) one-off analysis, for which the performance was calculated with a single barrier component suppressed; (ii) one-on analysis, for which only a single component barrier was active at a time; and (iii) multiple barrier suppression.

III.G. Alternative Conceptual Model Sensitivity Analysis

This technique replaces one of the conceptual models in the system model for one of the systems with an alternative conceptual model, and compares the results of the change. Alternative conceptual models considered were various spent fuel dissolution mechanisms, fuel wetting models (bathtub versus flowthrough), and transport through the geosphere. These are not the preferred models but in some cases represent possible alternatives that could be supported by available information. In other cases (e.g., no retardation, no solubility limits), the alternative conceptual models represent conservative, bounding analyses that are not necessarily supported by available data. Conceptual models may be activated in the code by changing the equations describing the model abstraction (e.g., Models 1 through 4 for the spent nuclear fuel-dissolution model) or changing parameter values (e.g., changing retardation coefficients to simulate no retardation).

III.H. Supplemental Analyses

Supplemental analyses involved the use of the system model to determine the significance of features, events and processes.

Criticality - This analysis assumes that waste packages that were initially failed (juvenile failures) degrade to the point that the fuel assemblies can collapse to a more compact configuration, the waste packages can fill with water, and the neutron poisons and other criticality measures degrade. The waste package may undergo either a gradual, steady state criticality limited by water entering the waste package, or a sudden transition to criticality leading to a steam explosion and collateral damage to other waste packages.

Human intrusion - This is a stylized analysis for which it was assumed that drilling for mineral exploration or water occurred directly above the repository. A single drill hole passes through one waste package, and water infiltrating the drill hole carries contaminated water to the waste package, which subsequently migrates to the water table. No probability is associated with this scenario.

IV. RESULTS

Results were produced for two simulation periods: 10,000 years, corresponding to the period of regulatory concern; and 100,000 years, looking at long-term processes where most of the waste packages would be expected to fail by corrosion.

IV.A. Deterministic Analysis

For brevity, results from deterministic analysis will not be presented in this paper.

IV.B. Probabilistic Analyses

Monte Carlo analyses produced a peak expected dose of 2.1×10^{-4} mSv/yr [0.021 mrem/yr] for the 10,000-year simulation period and 9.9×10^{-2} mSv/yr [9.9 mrem/yr] for the 100,000-year simulation period. Figure 1 shows the expected dose, the 75th, and the 95th percentile doses from 350 realizations for the 10,000-year simulation period. Igneous activity causes the largest increase in dose conditionally (i.e., assuming that the event has occurred) from both groundwater and airborne pathways for the 10,000-year period. The probability weighted dose from igneous activity is approximately 0.0035 mSv/yr [0.35 mrem/yr], which is greater than the base case groundwater dose of 0.00021 mSv/yr [0.021 mrem/yr (see Figure 2)]. For the base case, most dose came from the isotopes Np-237, I-129, and Tc-99. The biggest factors in the dominance of these radionuclides is their low retardations, long half-lives, abundance, and high dose conversion factors.

Waste package failure caused by rock fall is considered part of the base case scenario. For the base case, no waste package failed from corrosion or rock fall (other than juvenile failures) in 10,000 years. Compared to the basecase, faulting contributed to an increase up to a factor of two in peak dose until waste packages start to fail from general corrosion after about 50,000 years. Compared to the basecase, faulting does not increase risk significantly for the 10,000-year simulation period because of its low probability of occurrence (5×10^{-6} per year).

IV.C. Barrier Capability Analyses

These analyses show that for the base case conceptual models (i) the majority of waste packages will remain intact for greater than 10,000 years, (ii) the drip shield will delay the onset of dripping from the drift wall reaching the waste packages for a large fraction of 10,000 years, (iii) more than 90 percent of meteoric water will be diverted by the unsaturated zone above the engineered barrier, (iv) the properties of the unsaturated zone in conjunction with the drifts will act to divert water from many of the waste packages, (v) the properties of the waste form itself will cause radionuclides to be released slowly once other barriers have failed; and (vi) the unsaturated and saturated zones below the repository will retard and retain many of the radionuclides released from the engineered barrier subsystem for greater than 10,000 years.

In the barrier capacity analyses, the vast majority of retarded radionuclides (e.g., plutonium, americium) do not arrive at the downgradient wells in 10,000 years. For the 10,000-year simulation period, the isotope Np-237 was retarded enough in the geosphere that it just began to arrive at the downgradient well near 10,000 years. Np-237 arrived at the well for the 100,000-year simulation period to the extent retardation in the geosphere was no longer a major barrier.

IV.D. Parametric Sensitivity Analyses

The most influential parameters in the 10,000-year simulation period include (i) the mean annual infiltration at the start of the simulation (ii) the drip shield failure time, (iii) the pre-exponential term for spent nuclear fuel-dissolution, (iv) the fraction of the waste packages that are wet, (v) the focusing factor for water diverted to the wet waste packages, (vi) the well pumping rate at the user's location, (vii) the retardation factor for Np-237 in the alluvium, (viii) the distance to the tuff-alluvium interface, (ix) the fraction of condensate that flows toward the repository, and (x) the fraction of waste packages initially defective. The parameters found most influential for the igneous activity scenarios are (i) the airborne mass load above the fresh ash blanket, (ii) the wind speed, (iii) the diameter of the volcanic cone, (iv) the volcanic event power, (v) the volcanic event duration, (vi) the time of next volcanic event in the region of interest, (vii) the mean particle diameter of the ash, (viii) whether the event is extrusive or intrusive, (ix) the fraction of wet fuel for intrusive igneous activity, and (x) the pre-exponential term for spent nuclear fuel dissolution. Staff were able to validate that the choice of the sensitive parameters by the various methods was correct by calculating and comparing Monte Carlo runs with all 330 parameters in the base case sampled, against new Monte Carlo runs for which only the reduced set of sensitive variables determined by the consensus method were either included or eliminated. Results show that keeping only the most sensitive variables gives results similar to the base case. Removing the most sensitive variables from sampling greatly reduced the variance of the results. Both these observations demonstrate that the correct variables have been identified as being most sensitive.

IV.E. Distributional Sensitivity Analysis

Two parameters (out of the top ten identified by the parametric sensitivity analysis) cause the greatest change to dose when their distributions were changed: (i) the flow multiplication factor that determines the quantity of water entering the waste packages and (ii) the pre-exponential term for the spent nuclear fuel-dissolution model.

IV.F. Barrier Component Sensitivity Analysis

From the one-off analysis (see Section III.F), the largest decrease in performance came from suppression of the waste package, followed by unsaturated zone, saturated zone, waste form, drip shield, and invert. The relatively large impact of the unsaturated zone resulted from its role above the repository in diverting water away from the waste package and fuel, thereby reducing the mobilization and transport of radionuclides. One-on analysis, as illustrated in Fig. 3, ranks the contribution to repository performance in the same order as the one-off analysis, but in some respects, the contribution to performance of a single barrier is clearer. For example, the one-on analysis shows that the unsaturated zone alone would reduce the peak dose by more than 95 percent of the value with none of the other barriers effective, thereby demonstrating the capability of multiple barriers likely to be present.

Suppression of multiple barrier components shows some interesting interactions. For example, when both the drip shield and waste package components were off, the increase in dose exceeds the sum of either component individually, revealing the sensitivity to the drip shield that is otherwise masked in the one-off analysis. In this case, the drip shield and waste package can be seen to be redundant (i.e., the function of the drip shield in shedding water could be assumed by the waste package if the former failed).

IV.G. Alternative Conceptual Model Sensitivity Analysis

The range of the expected doses from the alternative conceptual models evaluated in this study spanned four orders of magnitude, from 2×10^{-6} mSv/yr (2×10^{-4} mrem/yr) to 0.02 mSv/yr (2 mrem/yr).

Figure 4 shows the effect of alternative conceptual models on the dose for the 10,000-year compliance period. The largest increase in dose resulted from the assumption of zero retardation of plutonium, americium, and thorium. These elements are normally highly retarded, and assuming they are easily transported in the geosphere is a conservative bounding analysis that could only be contemplated if mechanisms such as colloidal transport or fracture flow transport were highly effective.

Solubility limits of the radionuclides also appear to play an important note in reducing dose. Twelve out of 19 groundwater radionuclides show solubility limited release over a portion of the 10,000-year simulation period. Removing solubility constraints therefore increased dose, but the increase was larger for the flow-through model and less for the bathtub model.

Choice of the fuel wetting assumptions has a significant effect on the calculated peak expected dose. The default fuel wetting model is the bathtub, for which water must first fill the waste package and then overflow to release radionuclides. The flow-through model assumes that water flowing into the waste package is released immediately. Assuming there is a focusing effect for infiltrating water so that fewer waste packages get proportionally more infiltrating water allows faster filling of the bathtub and greater release of solubility limited radionuclides. Flow focusing led to a decrease in dose. This can be explained by the fact that the most important radionuclides released from the waste packages were controlled by the dissolution rate of the fuel rather than solubility limits of the released elements.

Matrix diffusion in fractured rock of the unsaturated zone may be important. The peak expected dose for the no matrix diffusion case is 50 percent higher than the base case dose. However, matrix diffusion in the saturated zone appeared to have virtually no effect on the peak dose.

Models for the release rate of radionuclides from the spent fuel waste form showed a large effect. Model 1, which is based on fuel-dissolution experiments where carbonate ions are present, gives the highest release rate and, therefore, the highest dose, which are approximately three to seven times the base case results. Model 2 is the default model. Model 3 is based on measured release rates of uranium from the Pena Blanca ore deposit[11]. Release rates for this case were significantly smaller than those for Models 1 and 2. Model 4 assumes that the release of all important radionuclide species from the fuel is controlled by dissolution of the secondary uranium mineral schoepite[11], and model 4 has the smallest release rates and doses. Credit for the protection of the fuel by cladding leads to peak doses that are approximately proportional to the degree of protection. The alternative conceptual models for release rate derived from natural analog and schoepite dissolution showed large decreases in dose, as did the model assuming that cladding was effective.

Assuming the fuel has a surface area equivalent to the size of uranium grains (microns to tens of microns) leads to doses 2 to 12 times higher than the default model, which assumes the fuel surface area is based on larger fuel particles.

IV.H. Supplemental Analyses

Criticality-A conservative consequence analysis showed that the conditional occurrence of a steady-state or transient criticality would increase doses by an order of magnitude above the base case, but the probability of conditions leading to this event appears to be low based on available information. Therefore, the risk significance of in-package criticality is not expected to be great.

Human Intrusion-This was a bounding analysis, which is not part of the risk calculations from other disruptive scenarios. Human intrusion is based on drilling a borehole through a waste package and subsequent releases of waste to the groundwater. The borehole acts as a fast groundwater pathway from the Earth's surface to the water table. Modeling this scenario gave a conditional dose of 0.001 mSv/yr [0.1 mrem/yr].

V. CONCLUSIONS

The TPA Version 4.1 code and other performance assessment tools have been used to obtain independent risk insights by investigating the estimated performance of the proposed Yucca Mountain repository and the sensitivity of this performance to repository subsystems and parameters on which they rely. The extrusive igneous scenario was found to dominate the risk to the exposed individual. Two scenarios for human intrusion and in-package nuclear criticality (with no assignment of probabilities) produced maximum conditional dose values that exceed the base case dose by a factor of approximately 5 and 10, respectively.

Barrier component analysis pointed out important features of the repository such as (i) the redundancy of the drip shield and waste package to shed dripping water, (ii) the capabilities of the unsaturated and saturated zones independent of the waste package, and (iii) the relative unimportance of the invert as a barrier. Multiple barriers were also identified based only on their capabilities (i.e., limiting flow of water and transport of radionuclides) rather than their direct bearing on dose or risk. In identifying these barriers, the staff demonstrated that for the conceptual models included in the TPA Version 4.1 code, the drip shield, waste package, waste form, unsaturated zone, and saturated zone all contributed to waste isolation.

Parametric, distributional, barrier component, and alternative conceptual model sensitivity analyses identified parameters that drive performance, parameters that significantly affect performance uncertainty, the parameters for which choice of distribution function significantly affects the dose responses, and alternative conceptual models to which the dose response is sensitive.

The performance assessment presented in this paper helps the NRC staff to risk-inform the review of the DOE post-closure analyses during the pre-licensing interactions. The results presented in this paper are based on numerous simplifying assumptions and use only limited site-specific data. Consequently, the numerical results should not be taken as representative of the performance of the proposed repository at Yucca Mountain, Nevada.

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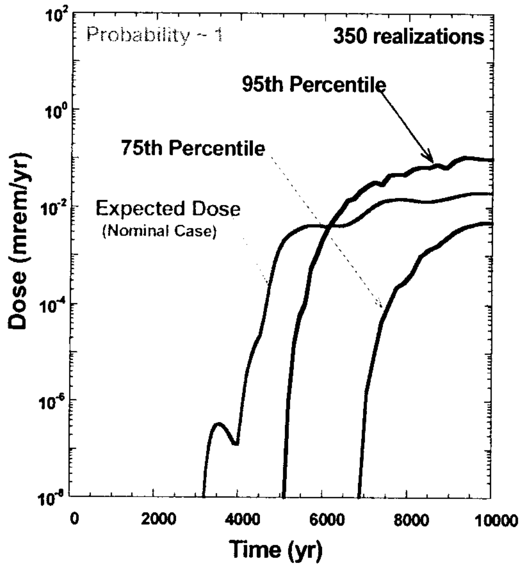

Figure 1. Nominal Scenario Results

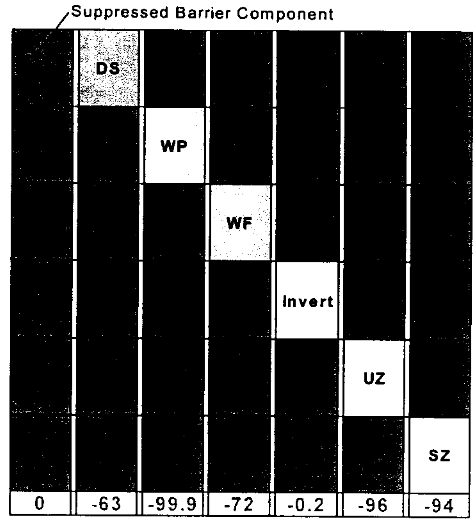

Figure 3 - Barrier Component Sensitivity, One-On Analysis for 10,000 Years. Bottom line shows percent change when each barrier is activated individually.

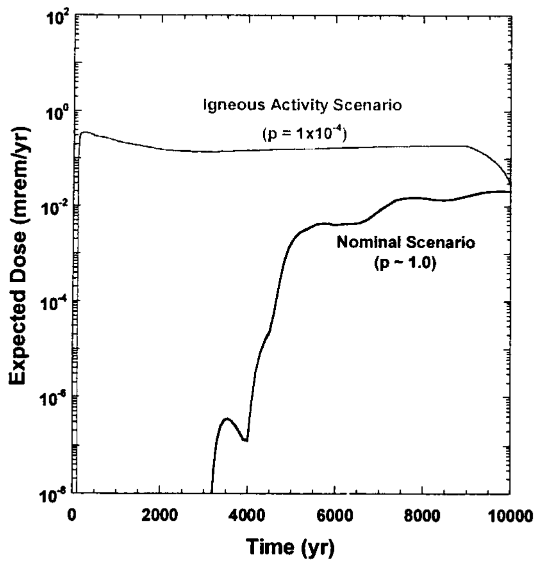

Figure 2. Disruptive Event (Extrusive Volcanism) Scenario Results showing a peak risk 0.35 mrem/years at 245 years.

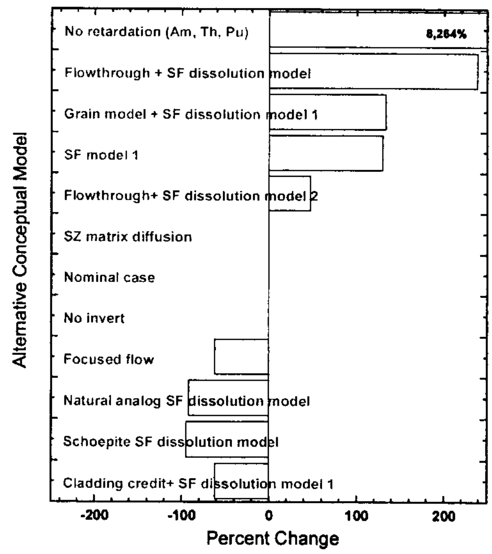

Figure 4 -Bar chart showing the effects of alternative conceptual models at 10,000 years